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Book of Abstracts

Book of the ANIMMA 2021 Abstracts

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#08-2 On the use of pixelated plastic scintillator and silicon photomultipliers array for coded aperture gamma-neutron imaging

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In the nuclear field, the development of instruments for measuring radioactivity and more particularly imaging systems for locating radioactive material is an important issue. This need for localization can be found in many areas of the nuclear industry (decommissioning, waste management and radiation protection) as well as for Homeland Security applications (non-proliferation of nuclear material), for the management of nuclear accidents or for nuclear research (Generation IV and fusion reactors). Gamma imaging is currently the most mature technique and several systems as iPIX, Polaris-H, ASTROCAM 7000HS, RadCam and NuVision are commercially available, meanwhile for neutrons, there are to date no equivalent industrialized systems. Several prototypes were developed over the last years and have demonstrated the feasibility of implementing localization methods applied to neutron imaging.

Nevertheless, significant improvements in either sensitivity or portability still need to be performed to achieve performances that meet the needs of the nuclear industry, which are even more critical when Homeland Security is involved. CEA List presented in 2018 a highly compact (19×14×15 cm 3 , 2.2 kg) fast-neutron/gamma imager based on a modified uniformly redundant array coded aperture and a Timepix detector enhanced with a paraffin layer. In 2019, several studies were carried out to improve and characterize this prototype, but although the compactness requirement was reached, the sensitivity criterion was clearly limited, especially for Homeland Security applications. This limitation was due to the neutron detection approach based on the use of a neutron converter layer. To overcome it while keeping a small footprint, we have investigated the use of plastic scintillators capable of neutron/gamma discrimination coupled with silicon photomultipliers in their matrix form.

As part of these investigations, we present our research on the use of pixelated plastic scintillators and silicon photomultipliers applied to coded aperture gamma-neutron imaging. Specifically, we verified the ability of a multiplexing readout to discriminate and localize neutron interactions. In its intended final configuration, the neutron imager design consists of a coded aperture aligned with a matrix of 12×12 PS each coupled to a silicon photomultipliers. The coded aperture is a rank 7 MURA, composed of tungsten and cadmium, and placed 5 cm away from the detector, with a total thickness of 1.2 cm and a surface area of 100.4 mm×100.4 mm. The pixelated plastic scintillators is composed of polystyrene and standard fluorophores loaded with 6 Li carboxylate [2], which allows the triple discrimination of thermal neutrons, fast neutrons and photons. The dimensions of the pixelated plastic scintillators were chosen to match those of the ArrayC-30035-144P silicon photomultipliers from SensL.

First, this neutron imager design was modeled and its response was simulated using the MCNP6 Monte Carlo code. The encoding capability, field of view, and spatial resolution of the neutron imager was therefore evaluated. Then the expected gain of this concept over the one with Timepix are presented and compared with first promising experimental results. Finally, we detailed the experimental set-up implemented to demonstrate the feasibility of coupling pixelated plastic scintillator to silicon photomultipliers and showed the results obtained.

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#07-3 Neutron coincidence measurements and Monte Carlo modelling of waste drums containing reference nuclear material

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Within the CHANCE project several non-destructive technique are being considered for the assay of waste bearing drums. Such techniques include calorimetry, gamma-ray spectroscopy and neutron coincidence counting. The aim is to reduce uncertainties on the inventory of radionuclides by combining the signatures from different techniques in the data analysis.
In this framework, neutron coincidence measurements were carried out with two slab counters based on 3He detectors coupled to shift register electronics. Such a system consists of two identical slabs with 6 detectors each and is transportable, rather compact and flexible in terms of sizes and geometries that can be measured. With this system three 200 l drums containing certified reference nuclear material and different filling materials were measured. The certified nuclear material was in the form of 21 pellets of mixed oxide of U and Pu with a total mass of about 10.5 grams; in addition, a single pellet of about 10.5 grams was also available. The pellets could be placed in predefined positions within the drum in a reproducible way. The geometry and composition of the three drums was well characterized and consisted of Ethafoam, a mixture of Ethafoam, stainless steel and PVC, a mortar with an inner core of extruded polystyrene. The measurement setup was arranged such that the drum was placed between the two slab counters. The positions of the slab counters relative to the drum were accurately measured before each measurement, and a dedicated system was used to minimize the uncertainty on the detector positioning.
The measurement data were analysed by applying the point model of Hage and the mass of nuclear material in the drum was determined from the totals, reals rate and radionuclide composition. Due to the fact that not all the point model conditions were met, we found that the point model overestimates the mass up to 80%. In addition, a Monte Carlo model of the measurement geometry was developed using the MCNP code. The model was used to determine a calibration factor between the reals rate and the mass of the sample. Measurements with a calibrated 252Cf source were used to verify the model. With a Monte Carlo based approach the mass of the mixed oxide pellets is slightly underestimated up to 10%. The results reveal the importance of an accurate background correction and of accounting for surrounding materials such as walls in the Monte Carlo model.

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#11-4 Response of 4H-SiC Detectors to Ionizing Particles

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We report on response of newly designed 4H-SiC Schottky barrier diode (SBD) detector to alpha, beta and gamma particles. In order to optimize SiC SBD detector's thermal neutron efficiency, it's of particular importance to understand its behavior in various radiation fields. The optimal size of the SBD is limited by degradation of electronic properties, and consequently their charge particle detection. We have manufactured diodes up to 3 x 3 mm active surface area to study those properties in correlation with increasing of detector size. Approximately 25 μm thick epitaxial layer is grown on SiC substrate by chemical vapor deposition, which is sufficient to stop alpha particles up to 6.8 MeV. Different active volume sizes of the detector based have been electrically characterized and exposed to radiation fields of alpha, beta and gamma particles. Extensive studies of the detector response to the various alpha emitters in the 3.27 MeV to 8.79 MeV energy range have been carried out. Results presented here demonstrate not only excellent linear response of the different detector active area to alpha particles, but also shows linear response to gamma particles. The detectors show a linear energy response, high charge collection efficiency and high energy resolution for the alpha particle energies bellow 6.7 MeV which is in a correlation with the range of charged particles in epitaxial layer of our detectors. Electrical characteristics of the detectors were assessed by temperature dependent current-voltage (I-V) and capacitance-voltage (C-V) measurements as well as by Laplace DLTS characterization of the electrically active defects. Ideality factor of the diodes is found to be in the range of 1.01 to 1.02 and free carrier concentration is in the range of 3x10^14 cm^-3 to 4.5x10^14 cm^-3. Co-60 and Cs-137 gamma radiations were carried out in Ruđer Bošković Institute's Secondary Standard Dosimetry Laboratory with the air kerma rates up to 77 mGy/min.

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#07-5 Plastic Scintillators With Tunable Decay Time

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Due to their nature plastic scintillators are widely used in industrial applications as well as in research and scientific fields. In the field of nuclear physics, plastic scintillators show favourable properties such as possibility to be prepared in large volumes and different sizes and shapes; they are cheap, efficient and can be easily modified. Standard plastic scintillators are available with relatively short decay times usualy in the range of 0.7-10 ns and then only with long decay time of 285 ns. For certain applications, however, scintillators with unsual decay times may be required. The work presented proposes a new concept where the scintillation decay time can be finely chosen and tuned in a range starting at 2.5 ns and finishing around 90 ns. Plastic scintillator compositions were modified to include two molecules, both serving as primary fluorophores with strongly different photoluminescence decay times. The relative concentration of the two molecules then led to materials with different decay times. Secondary fluorophore (shifter) had only little impact on the material decay time and it was changed to obtain scintillation material emitting in blue (420 nm), green (500 nm) or red (560 nm) wavelength region. These materials were synthesized, characterized and results are presented.

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#07-6 Monte Carlo calculations of the fluid and tubing effects on the gamma count rate of the NGRS probe

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Natural Gamma Ray Sonde (NGRS) is a gamma-ray logging probe used by ORANO Mining to estimate the uranium content in boreholes by detecting the gamma emissions of the ore. The total gamma count rate recorded with a NaI(Tl) scintillation detector is converted into uranium concentration using a calibration coefficient (in s-1.ppmu-1 units) estimated thanks to different calibration blocks in calibration facility operated by ORANO Mining in Bessines, France. The CEA Nuclear Measurement Laboratory has already performed Monte Carlo simulations with the MCNP computer code to estimate attenuation corrections in the ore (including self-absorption in uranium), in the filling fluids, and in the tubing. In this work, different tubing materials, diameters and thicknesses were simulated with an off-centre tubing. In the present new work, we have increased the number of diameters of the borehole and tubing, we have studied more tubing materials, and we have modelled a tube centred in the wellbore, instead of pressed on the wall, to be more representative of real measurements performed with a centring device. The MCNP model of the probe has been validated through a comparison with calibration experimental data: the calibration coefficient determined by simulation, 5.45 s-1.ppmu-1 with a 10 % uncertainty, is in good agreement with the one measured in Bessines 5.2 s-1.ppmu-1 (uncertainty not provided). The first parametric studies concerns the NGRS probe in a borehole without casing. They have been performed in a larger block than the calibration blocks located in Bessines (70 cm sides), in order to be more representative of a real measurement in an "infinite" ore medium. A small difference in the calibration coefficient is observed between a centered and off-centre probe for drilling hole diameters smaller than 200 mm, because gamma absorption by the drilling fluid is limited. However, for larger diameters, the calibration coefficient significantly decreases and the difference between the centred and off-centred probe positions increases, reaching more than a factor 3 for 800 mm diameter, with small differences depending on the filling fluid density. Then, the probe is simulated in a tubing centred inside the borehole. The calibration coefficients decrease as the borehole diameter increase, and when the density of the annular fluid (between the well wall and the outside of the casing) and drilling fluid in the casing increase. Finally, a probe located in the drilling pipe itself has been simulated, showing as previously that the calibration coefficient decreases with the increase of the rod diameter or with the density of the drilling fluid. This study required nearly 800 simulations carried out with CEA supercomputers.

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#07-7 Low-resolution gamma spectroscopy to estimate the concentration of uranium ores and the radioactive disequilibrium with two energy bands

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The estimation of uranium content in ore samples by high resolution gamma-ray spectroscopy requires long measurement times and expensive high-purity germanium (HPGe) detectors. In this work, we present the possibility to measure uranium by low-resolution gamma-ray spectrometry with an easy-to-operate and cost effective NaI(Tl) detector. This method is based on the analysis of two energy bands of the NaI gamma spectrum, which allows estimating a possible "U/Rn" imbalance between the top (238U and its daughters up to 230Th) and the bottom (226Ra, 222Rn and their daughters) of the 238U decay chain, due to differential leaching in roll-front deposits. In case of secular equilibrium, more than 95 % of gamma rays emitted by uranium ores come from 214Pb and 214Bi isotopes, which are in the back-end of 238U chain. Consequently, a disequilibrium in the chain might produce an overestimation of the uranium concentration if U/Rn < 1, or an underestimation if U/Rn > 1 (activity ratios). Therefore, the estimation of the disequilibrium between the beginning and the end of the chain represents a key objective in uranium mining exploration and exploitation. The disequilibrium is estimated with a count ratio between a low-energy band of the NaI(Tl) gamma spectrum around 100 keV that includes a significant signal coming from the top of the chain (92-keV gamma rays of 234Th and 235U and uranium X-rays), and a higher energy band around 609 keV, characteristic of the back end of the chain (gamma ray of 214Bi). Finally, the uranium concentration is derived from this U/Rn disequilibrium (activity ratio) and from the "Rn" activity of the back-end chain. This last is directly obtained by conventional gamma-ray spectroscopy with the 609 keV peak of 214Bi. The MCNP model of the NaI(Tl) detector has been validated using experimental data measured with a representative ore sample, with known uranium concentration and U/Rn disequilibrium. Then a linear relationship has been established by numerical simulation between the ratio of the two energy bands and the disequilibrium. More than 800 simulations have been performed to establish gamma corrections according to the density of the ore, filling height of the sample, and uranium concentration itself (gamma self-absorption in the sample), resulting in a total relative uncertainty smaller than 30 % on the uranium concentration. Experimental results obtained with a series of 38 ore samples provided by ORANO Mining, with acquisition times ranging from a few dozen seconds to a few minutes (depending on uranium concentration, which ranges from 100 ppm to 10000 ppm) show a very good agreement with ICP-MS analyses performed on real samples. For these samples, the gain in measurement time is between a factor 20 and 50 with respect to HPGe measurements.

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#04-8 Investigation of pulsed fields generated by pulsed ionizing radiation generating equipment, testing of radiation measuring detectors, detection systems and simulations supporting testing

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Devices that generate ionizing radiation that operate with a short pulse time are increasingly used in industry, healthcare, and scientific research. In the case of pulsed operation, these devices create pulsed ionizing fields, which have different properties compared to those experienced at stationary sources. Correct measurement of pulsed spaces is a challenge for manufacturers and users, as well as for the organizations involved in licensing, and should be reviewed.
In pulsed space, currently used radiation measuring detectors typically measure lower-than-true values because they typically do not detect a pulse lasting less than one second or are unable to properly follow sudden, magnitude changes over a short period of time. In the case of short-pulse radiation, it is advisable to use special detectors with a sufficiently good time resolution and a wide measuring range, since in extreme cases the characteristics of the pulsed fields reach or even exceed the operating limits of the detectors.
In recent years, the EK SBL has set up a test track where various detectors could be tested first for stationary spaces and then for dynamic spaces. In 2019, EK BL started testing pulsed fields with equipment that generates ionizing radiation but does not contain radioactive material that produces a pulsed field.
The availability and testing period of the equipment was limited, so the SBL plans to generate a pulsed-field using a constant gamma beam source, in which the shielded-collimated beam is generated using a gamma chopper. With the help of the device to be created in this way, the available time required for the examination of the SBL pulsed space will increase, with the help of which we will be able to perform several examinations and tests in the future. The pulsed-field generated by the device is to be measured optically using a visible light beam and a brightness sensing device. Thus, we can measure the beam characteristic by measurement (frequency, brightness change over time, pulse length, pulse time). In addition to the measurements, we can theoretically determine and then compare the measured and calculated results based on the data of the rotating mechanics. As a result of these tests, the EC will be able to produce pulsed spaces known for their SBL characteristics.
With the help of the new device, there are more testing possibilities than before, in which we want to test several detectors for several characteristics, and then we formulate a recommendation based on the results, which detectors follow well on a given characteristic and are close to the expected value. measures, or if it follows the trend but does not show real value, we try to apply a correction function, with the help of which the device shows the expected value using the characteristic function later when measuring the pulsed space with a given characteristic.

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#05-9 Numerical simulations in support of the design of an ultrasonic device for subassembly identification

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In this paper, it is shown how numerical simulations can help designing an ultrasonic instrument operating in harsh conditions. To prevent fuel handling errors in sodium cooled fast reactors, the identification of fuel subassemblies using ultrasound is being investigated. It is based on the interpretation of a code (aligned notches) engraved on the subassembly head using an emitting/receiving ultrasonic sensor. This reading is performed in liquid sodium with high temperature (up to 600℃) transducers.

A first experiment in liquid sodium demonstrated the feasibility of this method. The reading quality and robustness depend on various parameters related to the ultrasonic beam (spectral response, focal distance, focal spot size), the code geometry (especially the notches' dimensions) and geometrical alignments.

In order to avoid numerous experiments, two numerical models are developed. The first one is a finite element simulation of the sensor providing its radiated field. Unlike well-known analytic one-dimensional models, the finite element model is able to take into account the curved geometry of a focusing sensor. Moreover, it allows to model complex geometries of transducer. Finally, with the continuous growth of computing power, the finite element model allows the calculation of the radiated field with reasonable computational cost. This model is validated with the well-known analytic solution of the Rayleigh integral; then it is applied to the sensor used in the sodium experiment. The focal distance and focal spot diameter are close to the expected values.

The second simulation, using CIVA software, provides the ultrasonic scan of the code. The latter is computed by a ray tracing technique, using the pencil method to derive echoes with their amplitude. It allows computational cost much lower than the finite element modelling. The result is in good agreement with the sodium experiment and a first comparison with a water experiment shows that this numerical tool is relevant for easily taking into account misalignment and misorientation of the scan.

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#08-10 UAV prototype for localization and identification of radioactive contamination and emitters.

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The Dragon (Drone for RAdiation detection of Gammas and Neutrons) prototype aims at designing and developing an unmanned aerial vehicle (UAV) equipped with a detection system able to identify radioactive materials, spread over an area or located in a specific position. The system is focused on the localization of the unknown emitter and its subsequently identification.
The proposed prototype is made up of two easily interchangeable detection systems, one will be used as a counter while the second will be aimed to perform good-resolution gamma spectroscopy. Both solutions have neutron - gamma discrimination capability in order to be suitable for special nuclear materials (SNM) detection in gamma contamination background.
The data acquisition module is made up of a compact digitizer board (RedPitaya, sampling rate of 125 MHz and 14 bits of resolution.), a mini computer (Raspberry, for example). This combination allows to install an embedded operating system (e.g. Linux) that can run the necessary software for the Data Acquisition (DAQ), like the ABCD distributed DAQ.
Our contribution will be aimed to show a comprehensive characterization of the two detection systems, a medium size CLLB scintillation detector, and a large plastic scintillator, EJ-276, in order to assess their potential use in a UAV-based radiation monitoring system.

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#08-11 Design of MICADO passive and active neutron measurement system for radioactive waste drums

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In the frame of the MICADO H2020 project, a passive and active neutron measurement system is being developed to estimate the nuclear material mass inside legacy waste drums of low and intermediate radioactivity levels. Monte-Carlo simulations have been performed to design a modular and transportable neutron system, with the main objective to reach a good tradeoff between the performances in passive mode, i.e. neutron coincidence counting, and in active interrogation mode with the Differential Die-away Technique. Different designs are compared, which mainly differ in their moderation materials, graphite and polyethylene. This parametric study allowed us to define a prototype taking into account practical constraints in view of its final implementation in a wide range of in-situ locations and nuclear facilities. The total neutron detection efficiency of the prototype is 6.75%, as calculated for an empty drum without waste matrix. The detection limit in terms of nuclear material equivalent mass have also been estimated by calculations based on assumptions for a homogeneous distribution of nuclear materials inside the drum, filled with four types of matrices covering the range of nuclear waste drums defined in the frame of the project. The most penalizing matrix is made of polyethylene with an apparent density of 0.7 g.cm-3, which leads to a mass detection limit of 519 mg of 240Pu in passive mode, and 564 mg of 235U or 349 mg of 239Pu in active mode. Measurement time is 30 min for both passive and active modes. The most favorable matrix is made of stainless steel in passive mode and of polyethylene in active mode, with an apparent density of 0.7 g.cm-3 and 0.1 g.cm-3, respectively. The calculated mass detection limits are respectively 68 mg of 240Pu, 62 mg of 235U and 39 mg of 239Pu. Next steps will be a complete investigation of matrix effects based on intensive Monte Carlo calculations and an experimental design to build appropriate corrections. Experiments will also be conducted at CEA Cadarache Nuclear Measurement Laboratory with the neutron system prototype, and mock-up drums filled with different matrices.

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#01-12 Novel neutron detector assembly based on SiPM readout to be coupled with the Active Target for SPES

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The Active Target ATS (Active Target for SPES) is a new time-projection chamber designed for reaction and decay studies with nuclei far from stability. This kind of instrument, initially developed for high-energy physics, has found profitable applications in medium- and low-energy nuclear physics as shown by a successful series of experiments.
The physics cases for the new-generation active target are related to the ongoing developments of facilities for radioactive ion beams. Thanks to its flexibility, this instrument will be capable of taking advantage of the most exotic beams which will become available at the SPES facility under construction at the Legnaro National Laboratories in Italy. Particular attention will also be paid to coupling it with ancillary detectors for both charged and neutral (gamma and neutrons) particles.
In particular, in the present study, we will focus the attention on the neutron ancillary detectors.
The proposed solution takes into account a compact device and the capability of the system to discriminate, by performing pulse shape analysis, neutrons from the high gamma background.
A potential compact system aimed to discriminate neutrons from gamma ray events, using digital Pulse Shape Discrimination (PSD) techniques, can take advantage of recent improvements in silicon photomultiplier (SiPM) technology and the development of new plastic scintillators exhibiting the PSD phenomena.
Our presentation will be focused on the first studies of SiPM coupled with CsI(Tl), EJ-276G and EJ-276 scintillators with different configurations, such as: different SiPM models (Advansid and Ketek), different area coverage (from 3 mm x 3 mm to 17 mm x 17 mm) and different preamplifiers.
Moreover, we will show the comparison of PSD performances of plastic scintillators ranging from 25 mm to 70 mm diameter size, using the best SiPM readout configuration.

04 Research Reactors and Particle Accelerators / 13

#04-13 Experimental study of ISHTAR thermostatic irradiation device for the MARIA research reactor

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Materials and core components for the next generation power reactors technologies require testing that can be performed in existing research reactors. Such experiments employ devices dedicated to reflect the relevant thermal and neutron parameters simulating conditions present in, for example, but not limited to, HTR reactors. A novel thermostatic irradiation device named ISHTAR (**I**rradiation **S**ystem for **H**igh-**T**emper**a**ture **R**eactors) has been designed and constructed in the MARIA research reactor. Its mission is to enable irradiation of samples in controlled, homogeneous temperature field reaching 1000 ℃ and inert gas atmosphere. The high temperature is achieved by a combination of electric and gamma heating, together with carefully designed thermal insulation. Additionally samples holder made of graphite with high thermal conductivity enables the temperature homogenization in all directions. Device will be placed inside the Beryllium matrix of MARIA core and cooled with forced circulation of water from the reactor pool loop. This paper presents the outcome of experiments conducted with the rig prototype in external hydraulic mock-up of the MARIA reactor irradiation channel. The results have proved that the desired conditions for irradiation of the samples were achieved and their comparison against computational data has shown the adequacy of the design process. Finally, the loss of flow scenario was tested in protected and unprotected conditions (meaning with and without the safety system based on temperature feedback), proving the operational safety of the ISHTAR design. Experimental results will be used in the future to validate the numerical models (two and three dimensional) of the irradiation rig, providing an improved understanding of free convection and radiation phenomena modeling.

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#07-14 Boron Coated Straws imaging panel capability for neutron emission tomography for source localization inside radioactive drums

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The evaluation of fissile mass inside radioactive waste drums is essential for radioactive waste management, nuclear safety and criticality issues. However, passive and active neutron measurements can be strongly impacted by the uncertainty on the neutron source position within the drum and by matrix effects.
Therefore, an imaging panel proposed by Proportional Technologies Inc and composed of seven Boron Coated Straw (BCS) detectors has been tested to localize neutron interactions, in view to reduce uncertainties associated to plutonium or uranium position inside radioactive waste drums. In a previous work, a numerical model of the imaging panel has been developed and validated from a comparison with experimental profiles obtained with a 252Cf source.
In the first section, the feasibility of neutron emission tomography by a setup composed of five extended imaging BCS panels is demonstrated by numerical Monte Carlo simulation.
The second section details the experimental validation of the neutron emission tomography. Measurements are carried out with AmBe and 252Cf located inside an empty 118 L drum by rotating the BCS imaging panel around it. Afterwards, deconvolution algorithms are applied to provide 2D neutron source images for each angle. Finally, the 3D images are reconstructed using the RTK circular projection.
The results demonstrate the capability of the BCS imaging to provide the 3D location, i.e. axial and radial positions of one and two neutron sources. Furthermore, the first tests with this passive neutron measurement system show a satisfactory 3D reconstruction for 252Cf and AmBe sources separated by 20 cm.
Consequently, BCS imaging panels open interesting prospects to reduce the uncertainty associated to plutonium or uranium localization in neutron measurements.
Work is undergoing to assess the capability of this system for 118 L drums filled with organic and metallic matrices. Additionally, further prospects concern the performance of other deconvolution and reconstruction algorithms.

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#07-15 Characterization of Inelastic Proton Scattering on Carbon for Active Interrogation Applications

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Passive detection of special nuclear material is challenging because of its inherently low rate of spontaneous emission of penetrating radiation, the relative ease of shielding, and the fluctuating and frequently overwhelming background. Active interrogation, the use of external radiation to increase the emission rate of characteristic radiation from special nuclear material, has long been considered to be a promising method to overcome those challenges. Current active-interrogation systems that incorporate radiography tend to use bremsstrahlung beams, which can deliver high radiation doses. Low-energy ion-driven nuclear reactions that produce multiple monoenergetic photons may be used as an alternative. The inelastic scattering of protons on carbon — specifically, ^{12}C(p,p′)^{12}C — is one such reaction that could produce large yields of highly penetrating 4.4- and 15.1-MeV gamma rays. The gamma-ray energies produced by this reaction would provide higher penetration capability than existing bremsstrahlung systems (photon-energy endpoints of 6 and 9 MeV) while enabling robust material discrimination by means of dual-energy photon radiography. This reaction does not directly produce neutrons below the approximately 19.7-MeV threshold, and the 15.1-MeV gamma-ray line is well matched to the photofission cross-section of ^{235}U and ^{238}U. Rate and spectral measurements were made with liquid organic scintillators and NaI(Tl) scintillators, while the radiation dose was measured using thermoluminescent dosimeters. We use Geant4 simulations to simulate the detector responses and background radiation production in the experimental area, which allows the estimation of the yields of the 4.4- and 15.1-MeV gamma rays from near the reaction threshold up to 30 MeV. The yields in all experimental configurations are greater than in a comparable deuteron-driven reaction that produces the same gamma-ray energies and which has been considered for use in active interrogation — ^{11}B(d,nγ)^{12}C. However, a significant increase of the neutron radiation dose accompanies the proton energy increase from 19.5 to 30 MeV. This source could potentially address some of the key technical specifications required for new active-interrogation systems. By taking advantage of the emission of gamma rays into a large solid angle, even multiple cargo scanning streams may be feasible with a single source, thereby increasing the container throughput. With no direct neutron production at proton energies below 19.5 MeV, the ^{12}C(p,p′)^{12}C reaction should further reduce the neutron-shielding requirements and lower the total radiation dose imparted to cargo.

08 Decommissioning, Dismantling and Remote Handling / 16

#08-16 The Gamma and Neutron Monitor Counters for the MICADO Project

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In the framework of the MICADO EU project, aimed at the full digitization of the low and intermediate level radioactive waste management, we have produced and characterized a full set of radiation counters suitable for monitoring gamma rays and neutrons. In particular the goal of the Work Package 7 is to set up a granular radwaste monitoring system to be used during the final demonstration. The proposed system for the online real-time monitoring consists of an array of many radiation sensors to be deployed all around a number of radioactive waste drums, in order to collect counting-rate data in real time and to make them available to a software platform named DigiWaste. In order to be suitable for mass deployment these detectors have to be small, reasonably inexpensive, robust, easy-to-use and reliable. The sensors developed possess all of these features.
Continuous radiological monitoring of radwaste has to be based on the measurement of gamma and neutron radiation, since these are the penetrating types of radiation more easily detectable out of the drums. This is the reason why the focus was placed on the development of detectors for gamma rays and neutrons. In view of a possible mass deployment the foreseen sensors have to be reasonably low-cost. The overall system must be modular, so that one can easily modify number and placement of the sensors around the drums, and it has to be scalable in order to make it possible to tailor it to small, medium and large scale storage configurations without conceptual limitations.
The proposed system is based on detectors which can be easily installed and/or reassembled in different geometrical configurations, as they are mechanically very simple and are based on commercial electronics (see the attached file with figures). The detection efficiency the SiLiF thermal neutron counter is 4%, whereas the SciFi gamma counter has a ≈2.5% efficiency. Moreover, these detectors do not need special calibration procedures, and their ease of use and installation makes them suitable for short, medium and long term (as long as possible) monitoring.
A continuous automatic monitoring of the radwaste drums after their characterization represents an added value in terms of safety and security, and the availability of continuous streams of counting-rate data around each drum is a comfortable tool toward the transparency, which now more than ever is a hot topic of the nuclear industry with respect to the common people environment-aware.

09 Environmental and Medical Sciences / 17

#09-17 Evaluation of different detector designs for nanodosimetry

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Nanodosimetry is a relatively young research field which could help provide a more thorough understanding of how radiation interacts with cells. The nanodosimeters currently available are unfit for clinical use, due to their large size. The aim of this project is the development and characterization of a portable nanodosimeter.

It is known that how radiation interacts with cells is dependent on the type of radiation. This can be characterized by the Linear Energy Transfer (LET) of the radiation. Alpha radiation passing through tissue will leave a densely ionized path due to its high LET, whereas photons with a low LET will only ionize the tissue sparsely. From different cell experiments it is also known that radiation damages cells which can lead to cell death. Radiation can cause damage to DNA through ionizations in two ways: Indirect damage happens when radiation ionizes water molecules, resulting in free radicals which can then damage the DNA helix. The other form of damage is called direct damage and results from the radiation directly ionizing and damaging part of the DNA. The type of radiation also influences how many cells are inactivated by radiation. Alpha radiation causes more cells to be inactivated than photon radiation, due to its different LET and how it damages the DNA. In order to fully understand the processes on the DNA level a measuring device with a resolution of nanometres would be required. This technology is currently not feasible, but nanodosimetry offers an elegant solution to this problem.

The idea of nanodosimetry is to measure the number of ionizations happening within a small volume. Because DNA is the radiosensitive target of a cell, we are interested in a volume approximating the DNA double helix, for example a cylinder with a diameter of some nanometres. The number of ionizations produced within such a volume is repeatedly measured and called the ionisation cluster size. This is a stochastic quantity, it can therefore be characterized by a probability distribution, the ionisation cluster size distribution (ICSD). The ICSD describes how the radiation interacts with the DNA on a nanometre scale. The problem of observing such events on a nanometric scale can be solved by the equivalence principle, which states that the spatial distribution of ionization events scales linearly with density. This means that instead of measuring microscopic volumes directly, macroscopic volumes of low-pressure gas can be used instead. Different types of low-pressure gas were analysed and compared to liquid water by simulations and measurements by Grosswendt et al. in 2001. Propane gas was shown to be a good approximation of liquid water due to its similar behaviour regarding mean cluster size. In summary: Nanodosimetry measures ionisation cluster size distributions formed in macroscopic volumes of low-pressure gas.

There are different nanodosimeters which have shown this principle to work and which have measured ICSD. However, the Jet Counter from the Soltan Institute for Nuclear Studies, the Ion Counting Nanodosimeter from the Weizmann Institute and the Track-nanodosimetric Counter from Laboratori Nazionali di Lenaro all have one thing in common: They are functioning nanodosimeters, but very complex and bulky. This makes the devices impossible to use in a clinical setting for beam evaluation, where such a device would need to be portable and relatively small. This created the starting point for the development of a novel nanodosimeter at Loma Linda University. This new nanodosimeter was initially developed by Schulte, Bashkirov et al. over the last years.

The working principle of this detector is the detection of ionizations by use of a thick gas electron (THGEM) multiplier of a thickness of 1 cm. An alpha source is placed within a vacuum chamber, filled with low pressure propane gas. An alpha particle will ionize gas molecules along its way from the source to the alpha particle detector. The ions created by this will follow an electric field and drift towards the THGEM hole. Inside the THGEM hole a stronger electric field will cause the ion to be accelerated towards the cathode. During this acceleration electrons will be produced, but they will experience an acceleration away from the cathode and create an electron avalanche. This avalanche is then measured by a read-out pads, resulting in a signal. For each alpha particle a measurement window of a few milliseconds is opened, within which all signals are counted.

The detector can successfully detect individual ionizations caused by alpha particles along their way, however its efficiency remains relatively low, with only few ionizations being counted per alpha particle. The aim of this project is to systematically analyse and characterize the detector for different pressures, THGEM hole diameters, drift voltages, high voltages and many other parameters, thus giving insight in how different parameters influence the efficiency and behaviour of a compact nanodosimeter. Initial results of this first extensive analysis will be presented.

04 Research Reactors and Particle Accelerators / 18

#04-18 C/O logging by using the associated alpha particle method: Proof of principle

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The possibility of using the fast neutron beam with the associated alpha particle technique (APT) by the specially designed tool for Carbon-to-Oxygen (C/O) nuclear well logging was investigated. Measurements were done to show the influence of borehole fluid, iron casing, and the cement around the casing on the C/O analysis. Tests were conducted by using fast neutrons, which is the conventional approach for C/O analysis in well logging, and by using fast neutrons with APT. Analysed samples were made from quartz sand and graphite powder with the different C/O ratio characteristics for the oil-containing formation around the borehole. Diesel fuel was used as a simulant of the borehole fluid. Monte Carlo (MC) simulations were conducted and the MC results were compared to the experimental data. In both tests, the linear relation between the measured and real C/O ratios was obtained, but in the case in which the APT technique was not used, the linear relation was significantly influenced by the presence of the borehole fluid. The use of the APT technique resulted in a significant reduction of a background signal (coming mainly from the borehole fluid) in comparison to the conventional technique.

11 Current Trends in Development of Radiation Detectors / 20

#11-20 Design of a High Energy and High Resolution detector for X-ray computed tomography

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The Nuclear Measurement Laboratory (LMN) at CEA Cadarache in France is developing a high-energy tomograph currently being upgraded to reach energies up to 20 MeV with high dose rates (100 Gy/min). It allows tomographies on massive objects (5 tons, 140 cm diameter) with a millimeter spatial resolution. For the control of absence of cracks, bubbles or defects in the concrete coating of some CEA waste drums, the laboratory needs a "High-Resolution" version of this tomograph. Moreover, it can be used for examination of metal parts produced by additive manufacturing, such as piping parts. Even if several types of detector are available commercially to make high resolution they have drawbacks: some scanners use linear detectors producing one-dimensional images but with high scan times, others use flat panel detectors but there are rapidly damaged by X-ray beam. The purpose of this study is to design an imaging detector able to provide a high spatial resolution on large objects with high dose resistance and scan times of the order of a few hours. This detector is composed of a scintillator, an angle deflecting mirror, and one or more scientific cameras coupled to an optical system. To achieve this design several steps are necessary and will be presented in detail:

- A first step consisted in characterizing the current detection elements of the tomograph such as the camera and its lens via experimental measurements. This allowed us to compare performance of these elements with what exists on the market and to consider a replacement.
- Then we chose the scintillator according to experimental measurements on few scintillators type and a state of the art.
- Next, we optimized the detector configuration to achieve the higher spatial resolution. For example, we can use one camera imaging the whole scintillator, or a coupling of 4 cameras each imaging a quarter of the scintillator.
- The last step was to study the amount and spread of scattered radiation in our design using Monte-Carlo simulations (MCNP6). An optimum to take in account scattered radiation contribution and spatial resolution has been found depending on the configuration and the magnification factor applied to the system.

Through all of these steps, we present the design of a high-resolution detector with a spatial resolution for a contrast at 10% around 250 µm.

04 Research Reactors and Particle Accelerators / 21

#04-21 Characterization of the X-ray spectrum of a linear electron accelerator

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The Nuclear Measurement Laboratory (LMN) at CEA Cadarache in France uses high-energy electron linear accelerators, LINAC (9-20 MeV), to characterize nuclear waste drums or study corium-water interaction. These high-energy accelerators allow to explore new examination modalities, such as active photon interrogation or dual-energy CT to scan large concrete objects with diameters up to 140 cm. These techniques require precise awareness of the photon spectrum emitted by the LINAC. However, direct measure of this photon energy spectrum (for example, with a hyper-pure germanium spectrometer) can not be achieve because of the accelerator pulses causing detector saturation. During the last few years, a large number of indirect methods have been developed. Some employ Monte-Carlo simulation to determine precisely without experimental protocol the accelerator photon energy spectrum. Nevertheless, these methods require a precise knowledge of the accelerator characteristics including electron beam energy spectrum, difficult to assess accurately experimentally or numerically. Other indirect methods are based on an experimental protocol using energy spectrum from the spectroscopy of Compton-scattered photons, or transmission measurements through different thicknesses of a well known material. The latter is the simplest indirect method from an experimental point of view because it can be set up easily and accurately using an ionization chamber as well as an appropriate screen. The obtained transmission curve depends on the photon energy spectrum which can be estimated using inverse models.

In this paper, we present the development of a numerical model to determine energy spectrum from attenuation curve via transmission measurements which combines two types of inverse models: a continue model and a discrete model. The continue model consist in estimating spectrum shape via a continue mathematical function, in our model we use a log-normal function. Photon energy spectrum is first estimated from discrete log-normal equation which gives us the initial spectrum shape and the associated transmission curve. Then, the discrete model consist in tweaking spectrum values channel by channel with an iterative process to minimize cost function between analytical transmission curve of the reconstructed spectrum and the measured curve.

We validate this tool using a test spectrum and its transmission curve obtained via Monte-Carlo simulation (MCNP6). This qualification allowed us to determine its sensitivity (signal-to-noise ratio, SNR) in order to have a good convergence. We show that if the SNR is less than 4%, we have a good estimation of the photon energy spectrum. Then, it was experimentally tested with a transmission curve obtained at the laboratory.

08 Decommissioning, Dismantling and Remote Handling / 22

#08-22 Nondestructive active characterization of large concrete nuclear waste packages using photofission technique and high-resolution delayed gamma spectroscopy

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Nondestructive characterization of large concrete radioactive waste packages is a major challenge. The fissile mass inside has to be estimated very precisely in order to transport and store the wastes in the safest way. Ultimately, knowing the alpha activity of the waste packages could enable to lower their activity category and thus to significantly reduce their final repository cost, e.g. in a surface vs. a geological disposal. The 870 L concrete waste drums, produced and stored at CEA Cadarache in France, represent an issue since they cannot be characterized using Active Neutron Interrogation. Indeed, their large dimensions and poorly known hydrogenous content lead to extremely penalizing uncertainties associated to high attenuation effects. This article focuses on the development of a non-destructive characterization method using Active Photon Interrogation (API) to estimate the fissile mass contained inside these large concrete waste barrels. This technique exploits the photofission reaction of nuclear materials occurring with photons of energy larger than 6 MeV. Since all actinides have similar photofission cross sections, the method cannot directly estimate the fissile mass (for example 235U, 239Pu) or the alpha activity (mainly Pu isotopes), and therefore a discrimination method is needed to identify the main nuclides of interest. This method is based on the difference of photofission product yields in actinides, and consequently on their delayed gamma ray intensities. In a first step, we present here new experimental data of more than 30 photofission product yields for 238U and 235U obtained by irradiating bare uranium samples with a Bremsstrahlung photon flux produced by a 16 MeV electron linear accelerator at the CINPHONIE facility of CEA Cadarache, France. The fission contribution of parasitic neutrons produced by photonuclear reactions in the accelerator conversion target, in the surrounding materials, and in the sample itself, was also calculated and subtracted. These measurements lead to select the most suitable gamma-ray lines for discriminating between 238U and 235U in a dense environment like a concrete package. In a second step, we use these measured photofission yields in our Monte-Carlo (MCNP6) simulations to assess which delayed gamma-rays are detectable in concrete radioactive waste drums. Since matrix effects produce a differential attenuation of these delayed gamma rays, they affect the discrimination of actinides and increase the uncertainty on the fissile mass. Therefore, we also study actinide localization methods via Monte-Carlo simulations, using specific delayed gamma rays from photofission as a probe, which will be qualified experimentally with a mock-up of a concrete waste matrix.

02 Space Sciences and Technology / 23

#02-23 Ultra Fast timing detectors with applications in cosmic ray physics, medical science and other domains

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We were developing a cheap concept of a fast commercial silicon pixel detectors combined with a state of the art readout electronics based on ultrafast sampling and waveform digitizers of the signal giving simultaneously its time and amplitude.
This concept was used originally in High Energy Physiscs and we transpose it
with two applications from cosmic ray physics and particle therapy medicine.
For the astrophysics program, the principle is to perform direct particle identification and energy measurement using a few layers of cheap detectors that will be sent to space in collaboration with NASA. We will discuss the performance of our detectors as well as the results from simulation before launch in few months.
For the medical application in particle therapy, the principle is to measure with high accuracy the beam dose delivery during proton beam treatment individual bunch by bunch to know with high accuracy especially for flash high intensity beam treatment. Results performed at the hospital of Dublin, Ireland, will be presented.

11 Current Trends in Development of Radiation Detectors / 24

#11-24 PLD-grown, isotopically enriched 10B thin films for thermal neutron detection

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Thermal neutron detection is typically carried out by a double-step process involving neutron conversion reactions leading to secondary charged particles and subsequent detection of the reaction products by means of solid state detectors, scintillators or gas chambers. A proper efficiency thermal neutron conversion material should exhibit a high neutron-capture cross section and optimal detection geometries.

In this paper we report on the deposition by Pulsed Laser Deposition (PLD), homogeneity characterization and performances in neutron detection of 10B neutron conversion layers (isotopically enriched at about 96%). High quality 10B films with thickness ranging from 0.5 to 2 m were deposited by using a 1064 nm Nd:YAG pulsed laser on carbon fiber substrates obtaining a thickness uniformity better than 10% over an area of 30 x 30 mm2. Optimal PLD deposition conditions are presented and discussed based on comprehensive characterization of the boron deposits in terms of laser parameters, elemental composition, surface morphology, deposition rate, thickness uniformity and density.

In order to evaluate the detector performances, all the deposited films were coupled to a 30 x 30 mm2 wide silicon solid state detector and exposed to a neutron flux of ca. 5.6 x 106 neutron/s produced by an AmBe neutron source and moderated by means of polyethylene. The neutron flux on the converter film was measured as 267 ± 5 n/cm2/s by a previously calibrated 6LiF detector. The results, as compared with GEANT4 simulation results, clearly showed the functionality of the proposed set-up and allowed the identification of both and 7Li n-capture products. Encouraging results were also obtained in terms of discrimination against the gamma background radiation associated with both the (n,) reaction and the AmBe source. In particular to get rid of the contribution, an experimental approach where the net neutron counts are obtained by difference between two spectra was developed. A first spectrum is obtained with the detector facing the conversion layer; then a second one with the converter turned upside down so that the detector faces the pristine substrateBased on this approach a detection efficiency in the range few percent was assessed. Future developments are discussed in order to improve the detection efficiency.

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04 Research Reactors and Particle Accelerators / 25

#04-25 Characterization of neutron emission during pulse mode of low output electronic neutron generator

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Portable electronic neutron generators are an established technology that can be used for many lab and field applications. Offering significant regulatory and operational benefits over radionuclide neutron sources, they start to be more widely used at universities and research centres. Beside the established industrial applications (e.g., in logging or mining industry), applications are being developed for fields including security, safeguards, or waste characterization. They are becoming more used for E&T in nuclear engineering, nuclear analytical techniques, or nuclear related fields.
Electronic neutron generators can be utilized for many analytical techniques (including NAA, PG-NAA, DNC, DDA). In many cases, the pulsed operation of the generator is used to improve the efficiency of the applied methods. Neutron generators can provide information on neutron production via a TTL signal that can be used for triggering the data acquisition system. However, to improve the interpretation of the measured data, the knowledge on the time profile of neutron production within the pulse duration is important. This can be achieved experimentally by a proper neutron detection system. The response of the neutron detection system is formed by neutrons of different energies for detection systems designed to measure thermal, epithermal, and fast neutrons. For various applications, specific frequencies and pulse durations can be needed. Thus, various frequencies may require or enable various approaches to efficiently relate the neutron detector response to the neutron emission by the generator within the required time resolution. Monte-Carlo particle transport codes are efficient tools to simulate it.
The paper deals with the characterisation of the neutron output profile during neutron pulsing of a neutron generator. First, it provides a validation of time-dependent calculations in MCNP Monte Carlo code showing the role of several model simplifications on the achieved results. Second, it compares the applicability of thermal, epithermal and fast neutron detection systems for pulse characterisation. The study is performed for two cases: first, the generator and detector are placed in a moderated environment; second, the equipment is placed in a free space. Calculations are performed in MCNP6 code. The study includes experimental data measured with the D-D P-385 Thermo Fisher Scientific neutron generator with a maximal neutron output of 7 x 10^6 n/s for several frequencies.

04 Research Reactors and Particle Accelerators / 26

#04-26 JHR Irradiation Devices. Inspection Methods proposal

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JHR irradiation test devices must undergo a periodic inspection every 40 months (French Regulation rule). The first step of inspection proposal for application of non-destructive techniques (NDT) for these devices is presented, through examining the possible methods and locations that could be utilized in the reactor.

The selection of NDT methods depends mainly on the device geometry, material and access: main body of the device consists of two vertical nested tubes to be inspected. Once in use, due to deformations of thermal and radiation origins, these tubes are likely impossible to be separated (removal and re-assembly in a hot cell during inspection phase). Material is Zircaloy-4. The length of the tubes is about 3700 mm, with tubes diameter ranging from about 40 mm to 110 mm and thickness ranging from about 4 to 10 mm.
Some narrow electron beam welds will have to be controlled according to French nuclear pressure equipment regulation rules: thus, the inspection must respect the requirements of RCC-MRx code.
While ultrasonic volumetric inspection is selected for inspecting the tube bodies for checking no crack type of defects, Eddy current examination should be very efficient in finding surface defects.

Simulations are performed with CIVA software platform in order to select the best conditions for Ultrasonic testing (UT) and Eddy current (EC) inspection. Suitable sensors are especially selected to optimize the propagation of ultrasonic waves in the tube. Ultrasonic pulse-waves with center frequencies ranging from 0.1-15 MHz are transmitted to detect internal flaws: the beam-defect interaction allows predicting the amplitude and the time of flight of various echoes: direct echo, corner effect, tip diffraction echo, etc. and also echoes scattered back by the geometry (backwall, entry surface and interior specular interfaces echoes) with acoustic mode conversions.

The initial conclusions of this work are the following:
- Two inspection methods are likely needed.
- Eddy current is likely suitable for surface examination while volumetric ultrasonic inspection can be used for tube bodies.
- Inspection under-water in a storage pool is likely the best option for location in the JHR.
- Some automation and remote controls will be needed during the inspection process.

The next step for experimental reference will be a critical step of the work to assess and qualify the inspection methods: a qualification phase will be launched, performing inspection of representative dedicated mock-ups made of Zircaloy-4 material. For ultrasonic reference, a representative mock-up of the cylinder with a weld and artificial defect(s) will be used. The mock-up testing will provide an experimental reference to CIVA software platform simulations.

This study has been performed in collaboration between VTT Technical Research Centre of Finland Ltd and French CEA Alternative Energies and Atomic Energy Commission.

05 Nuclear Power Reactors Monitoring and Control / 27

#05-27 Multi-parameter discrimination of partial-discharge-induced pulses in fission chambers designed for sodium-cooled fast reactors

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Fission chamber technology has been identified as the most suitable method for neutron detection to be used in the vessel of a sodium fast reactor and may possibly also be used for neutron diagnostics in fusion power (ITER, DEMO). This type of detector, namely High Temperature Fission Chamber (HTFC), must be able to operate under high irradiation of up to 10^10 n/cm².s, must have high sensitivity (~1) to detect fast neutrons, and must operate at high temperatures, up to 650°C in sodium fast reactors and probably at even higher temperatures in fusion power reactors. It has been observed that an effect of the high temperature environment is an additional signal, ascribed to partial electrical discharges, which is of similar amplitude and duration as the useful signal caused by neutron interaction with the fissile layer inside the fission chamber. Partial discharges (PD) in gases are often referred to as "Corona discharges", and are associated with the phenomenon of an electronic avalanche in the presence of strong electric field. Previous work by Hamrita et al. demonstrated how mono-parameter discrimination based on signal full width at half maximum values can discriminate neutron pulses from PD pluses, but only for small diameter fission chambers.

In the presented work, neutron and PD pulses are collected using two large diameter HTFCs, designed to be used in the vessel of a sodium fast reactor, tested at high temperature and under neutron beam irradiation and off beam. Firstly, starting from preliminary results of Hamrita et al., the collected pulses (neutron and PD) are separated using a mono-parameter discrimination method based on full width at half maximum. Then the same data are separated using the proposed multi-parameter discrimination method by means of KNN and SVM algorithms.

Finally, the results obtained from the tested discrimination methods were compared and a PD-neutron discrimination process for HTFC was proposed.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 28

#07-28 Characterization of nuclear waste packages by active photon interrogation with mobile or stationary systems based respectively on 7 or 9 MeV linacs

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Decommissioning and dismantling (D&D) operations conducted in former nuclear facilities generate a large amount of nuclear waste packages. Characterization of the latter is an important step in the management process. Active photon interrogation, based on the photofission reaction, is a well-adapted method to characterize nuclear waste packages containing concrete matrices. Indeed, with such matrices, both passive methods and the active neutron interrogation method reach their limits whereas the high-energy photon beam produced thanks to a linear electron accelerator (linac) could enable to interrogate the center of a concrete matrix package. The photofission reaction is based on two steps. First, high-energy photons induce fission reactions on actinides such as uranium and plutonium isotopes. Secondly, prompt and delayed particles are emitted. The energy threshold of the photofission reaction is close to 6 MeV for most actinides (²³⁵U, ²³⁸U, ²³⁹Pu, etc.). In the frame of the MICADO project, which is part of the European Union's Horizon 2020 research programme, the active photon interrogation method is assessed to characterize packages containing concrete matrices. Two different setups are studied: on the one hand a stationary system based on a 9 MeV linac; on the other hand a mobile system based on a 7 MeV linac. However, characterization of nuclear waste packages using a linac operated at an energy below 10 MeV is not straightforward. Undoubtedly, compared to 9 MeV, the challenge is even higher at 7 MeV, and will require the most effective measurement protocol and sensitive detection systems. Moreover, in both cases, the high-energy photon beam delivered by a linac would enable to gather more information on the package matrices thanks to the high-energy imaging technique. The aim of this paper is to present the recent developments in the field of active photon interrogation for nuclear waste package characterization with mobile or stationary systems based respectively on 7 or 9 MeV linacs. First, performances of the photofission technique with a 7 MeV linac will be assessed by Monte Carlo simulation using both PHITS and MCNP6 codes. Accuracy of cross-sections around the energy threshold of the photofission reaction will be discussed. Secondly, performances of the photofission technique with a 9 MeV linac will be assessed experimentally at the SAPHIR platform (CEA Paris-Saclay, France) using mock-up packages with different matrices including concrete and containing either uranium or plutonium samples. Delayed neutrons from photofission will be detected using ³He-filled gas proportional counters, and high-energy delayed gamma-rays will be detected using large size plastic scintillators. Comparison between simulation and experimental results will enable to validate the MCNP6 simulation model. The latter will then be used to evaluate performances obtained with the photofission technique and a 9 MeV linac on larger packages and more complex package geometries.

04 Research Reactors and Particle Accelerators / 29

#04-29 Development of an LVDT Conditioning unit for use on LVDTs in a research reactor

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Research into materials and fuels for nuclear power reactors is carried out in so-called research reactors where various types of fuels and materials can be monitored online by in-core instruments. At the Halden reactor in Norway, online measurements such as cladding elongation, inner fuel rod pressure, fuel swelling, material creep and stress relaxation were based on Linear Variable Displacement transducers (LVDT). These LVDTs require custom made conditioning units, with some particular requirements such as low driving frequency, ac constant current through the primary coil and high stability over a long period of time. In relation to the closure of the Halden reactor (June 2018) and the ensuing reorganizations at the Institute for Energy technology (IFE), the required LVDT conditioning units are no longer manufactured by IFE in Halden and the production of the LVDTs is almost terminated. Therefore, a project was started at SCK CEN to develop both LVDTs and the required conditioning units, having similar characteristics as those used at Halden. While the development of the LVDTs is still ongoing, the development and testing of the conditioning unit for a 4-wire LVDT has been completely finished and tested. In contrast to the conditioning unit developed at Halden, the system developed by SCK CEN is based on a digital system, using a commercially available chip by TEXAS Instruments. The device implements a digital demodulation algorithm that not only extracts the amplitude but also the phase of the signal with respect to the primary excitation. Additional circuitry was needed to reverse the polarity of the DC output signal at the zero crossing of the LVDT, based on the phase of the signal.

The system is set to drive the primary coil of the LVDT in constant current mode, at an operating frequency of 1 kHz. This frequency is a factor 2.5 higher than the frequency which was used at Halden while still being compatible with long signal cables and in-core use. As a result, for the same LVDT, the sensitivity of the LVDT is increased by a factor 2.5 as well. The external circuitry allows obtaining 50 mA current over a load resistance of 400 Ohm. This high resistance is required because of the combined resistance of coils and signal cables at high temperature. The overall performance was tested on existing LVDTs produced at Halden and was found to be excellent.

The present system is made to fit in a 19-inch rack, containing 6 independent LVDT conditioning units, each with their own frequency and current generator. The unit has been CE and EMC approved.

09 Environmental and Medical Sciences / 31

#09-31 Active dosimetry with the ability to distinguish pulsed and non-pulsed dose rate contributions

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In this work, the concept of an area monitoring dosimeter and its operational regime for pulsed radiation dose rate measurements is presented.
A fast tissue equivalent plastic scintillator EJ-200 (Eljien Technology) is exploited as a detector material. In the energy range from 200 keV up to 4 MeV, the scintillator is tissue equivalent. This minimizes the influence of pile-up on absorbed energy measurements as simultaneous energy depositions in the detector material from multiple photons lead to the proportional value of the energy deposition as from summing up contributions from separately coming photons for a soft tissue. This means that for the dose rate measurements the higher weighting of pile-up events is achieved through their higher energy values. Fast response of the scintillator (in the range of ns) can provide time-resolved dosimetry if necessary.
The detector is connected to a fully digital signal processing board, which creates an active system with adjustable parameters. Signal losses due to pile-up, readout, and dead-time during the data acquisition can be estimated by analyzing generated listmode files and from the knowledge of the dead-time behavior of the system (completely non-paralyzable).
This system was used to measure the absorbed dose rate in conditions imitating radiation protection measurements outside the treatment room of a clinical linear accelerator for percutaneous cancer treatment. The detector was placed in a scattered photon field with a time structure mimicking a radiotherapeutic beam, but also in the presence of a constant radiation field. The LINAC at the γELBE (HZDR, Germany) operated in the macropulses mode with their duration of Δt of 5 µs and the period T of 5 ms. An additional source of continuous radiation (^{22}Na) was used. For all measurements, the source was attached to the PMMA phantom opposite to the detector. The arrangement of the incident beam, PMMA phantom and the detector provides the scattered photon field with the maximum energy no more than 777 keV which is acceptable to conform the energy range where the scintillator is completely tissue equivalent.
The real-time distinction of pulsed and non-pulsed contributions is based on the time structure of a single interaction. Then, the pulsed radiation from the different macropulses of the accelerator has the time difference that belongs to intervals of
$[kT - \Delta t; kT + \Delta t]$,
where k is used for positive integers. If events come from the same macropulse, the time difference between them belongs to the interval of
[long gate; Δt],
where the "long gate" is the integration gate of the data acquisition system. This algorithm can be easily implemented in the software of any active detector.
For the investigated dose rate range up to 8 µGy/h, the pulsed radiation dose rate estimated according to the presented algorithm shows a linear dependence on the accelerator current. Thus, the increase of the accelerator current by 1 µA leads to the increase of the pulsed radiation dose rate by (26.2±0.9) nGy/h.
If the accelerator power continues increasing, greater and greater part of pile-up pulses will be lost due to the limited integration gate. One of the possible decisions is to expand the long integration gate up to the macropulse duration when detector will integrate all events within the macropulse, and pile-up events will be accounted for completely. This approach is going to be tested during the next experiment.
The behavior of the non-pulsed radiation dose rate is not constant and approaches the reference value (measured with the ^{22}Na source only) while increasing the accelerator current. This can be explained by underestimating the pulsed radiation dose rate at low accelerator current values when mostly one event per macropulse is detected, and this event cannot be definitely related to the accelerator or the background and the ^{22}Na radiation. This means that the proposed algorithm will overestimate the pulsed radiation dose rate contribution in cases with very low accelerator current, while in high pulsed dose rate scenarios the accuracy of the approach should improve.

There are some challenges that apply to the improvement of the presented detector. The first one is to adjust it for measurements of operational dose quantities. The specific calibration that relates indications of the detector to the operational quantities can be performed for a number of standard calibration sources. But in the case of mixed radiation field, the deconvolution of spectrum will be required which is not a trivial task as no full absorption peaks are observed in low-Z organic scintillators.
The second challenge is the dosimetry of the low-penetrating radiation like laser-induced X-rays. Here, one deals with the energy range of about tens keV, which means that the housing of the detector will absorb a great part of radiation. Also, the behavior of the detector material in this energy region in comparison with a soft tissue is hardly correctable in the online regime without knowing the energy spectrum of the incident radiation.

05 Nuclear Power Reactors Monitoring and Control / 32

#05-32 The analysis of different physical mechanisms during cladding failure evolution and detection in sodium cooled fast reactors

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To keep the dependability of Sodium Cooled Fast Reactor, the ''clean sodium concept'' is demanded, which means that the sodium is free from contamination. The release of fission products is searched for by a contamination measuring system. We need to have a comprehensive description of cladding failures and the detection of contamination, including the failure occurrence on the fuel pin, the transfer process through the sodium and cover gas, the measurement efficiency, etc. We aim to identify the important parameters of physical phenomena, with modelling and simulations based on the return of experiments from past reactors such as Phenix. There have been a total of 15 open pin failures in Phenix reactor. Through studying these detected signals, we can get a better physical explanation and description of the evolution of failures.

The detection system is related to different stages of the evolution of fuel pin, with different types of fission products, various release mechanism and physical properties. During the evolution of the failed fuel pin, gaseous fission products is released on the first stage of failure followed with other fission fragments including delayed neutron precursors. There are two main parts of the whole detection system. DRG (Détection des Ruptures de Gaine) system is functioned for detecting the crack of the cladding, with only gaseous fission products release. DND (Détection des Neutrons Différés) system is functioned for ruptures with solid fission products leakage.

We propose a block diagram that identify the physical phenomena occurring at each stage of the detection system, and have a focus on two parts for illustration. For gas signal, we propose a qualitative modeling of transfer function to describe the time broadening of the gas release from the fuel pin to the detector. The result matches well with Phenix experiment data, with the same order of magnitude of the time broadening and the same shape of exponential decreasing. For delayed neutron measurement, we study the complex evolution inside the fuel pin to interpret the different release mechanism and neutron signal variation.

08 Decommissioning, Dismantling and Remote Handling / 33

#08-33 Novel thermometry approaches to facilitate safe and effective monitoring of nuclear material containers

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Many established nuclear power producing countries are currently decommissioning first and increasingly second-generation power producing plants and fuel processing facilities. This has led to a growing inventory of different containers and packages containing radioactive waste and other nuclear materials, as well as storage of spent fuel. Here we describe establishment of in-situ yet remote health monitoring techniques based on novel temperature measurement methods for different containers and stores used to hold different radioactive waste forms, such as intermediate level waste (ILW), special nuclear materials (SNM) and spent fuel.
ILW and SNMs are usually held in high quality stainless steel containers. In the case of the containers used by Sellafield Ltd the ILW suspended in cementitious grout and held in 500 litre containers (drums), whilst in the future heterogenous ILW will be stored wet in approximately 3 m3 packages (i.e. covered with water inside the container). SNMs are generally held in nested stainless-steel containers. Advanced gas cooled reactor (AGR) spent fuel will be held in ponds awaiting final disposal. Each of these container waste forms and spent fuel storage has particular monitoring challenges to ensure the on-going integrity of the storage. Particular challenges are; the high level of gamma radiation emitted by the ILW and spent fuel, the gamma and neutrons emitted by the SNM, and the general inaccessibility of the store and pond configurations.
Here we report the use of quantitative remote temperature measurement techniques to assess the condition of waste package or temperature of spent fuel racks. Specifically the following measurement requirements are addressed; ILW container health and internal temperatures, determination of the surface temperature of SNM containers in-situ with the future prospect of identifying surface and sub-surface defects, development of surface thermometry approaches for a new long term storage container of SNM, development of a method to determine the temperature of racks holding spent AGR fuel and a remote thermal method to identify anomalously high temperature 3 m3 packages in stores.
Key to the success of this work is that the thermometry approaches, where needed, are traceable to the current temperature scale the International Temperature Scale of 1990 (ITS-90). Robust traceability to ITS-90 gives a stable baseline against which to compare all subsequent measurements and confidence in the actual measured temperatures. Temperature measurement uncertainties are evaluated according to the internationally accepted Guide to the Expression of Uncertainties (GUM).

11 Current Trends in Development of Radiation Detectors / 34

#11-34 Multitube detectors as an alternative to Uranium fission chambers for neutron beam monitoring

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Neutron beam monitors are key instrumental components in neutron scattering science; they are used to measure the instantaneous neutron flux upstream the sample in order to normalize the experimental data. The measurement consists in converting a small fraction of the neutron beam into individual pulses or into a current signal. Among the different monitoring systems used, Uranium fission chambers operated in pulse mode are considered as the state of the art, thanks to their robustness, high stability and low sensitivity to gamma. Their sensitivity to thermal neutrons is proportional to the density of U235 atoms in the thin convertor film deposited inside the chamber. This parameter being fixed by fabrication, it must be specified according to the range of counting rate the monitor must deal with; this range is defined on one hand by the limit of statistical fluctuation imposed by the experiments performed on the instrument, and on the other hand by the limit of counting rate before monitor saturation becomes an issue. The optimal sensitivity of a beam monitor is the one corresponding to this second limit. When purchasing a fission chamber, the requested sensitivity is generally specified in a conservative way, at a value lower than the optimal sensitivity. This is done to take into account 2 types of uncertainty: first a factor of 2 discrepancy between the sensitivity specified and the sensitivity measured is not rare. Second, the precise value of the maximum flux deliverable on an instrument is often lacking, in particular for new instruments. This reduction of the monitor sensitivity results in a degraded normalization procedure at low flux, due to poorer statistics. To this respect, a beam monitor with adjustable sensitivity would be of considerable interest. The MultiTube beam monitor presented in this paper allows changing the neutron sensitivity parameter easily. Its principle consists of proportional counters machined side by side by spark erosion in a block of Aluminum. The gas mixture contains a small fraction of a neutron sensitive component, generally 3He or N2, and a valve allows changing the gas mixture. Even though the wall of the tubes is 0.5 mm thick, this beam monitor can be operated in a vacuum environment without deformation of its mechanical structure. Furthermore, the MultiTube does not emit fast neutrons like in U235 fission chambers.

For all those aspects, the gas-filled MultiTube monitor is an attractive alternative as it allows adjusting the detection efficiency with a very good accuracy – around the percent level – and covering a broad range of efficiencies, at least comparable to Uranium fission chambers. Experimental results show that it provides a more uniform response, and a better transparency to neutrons, close to 97.5 %, thanks to the 0.5-mm thick aluminum windows and to the absence of organic materials. Its intrinsic geometry guarantees a fast charge collection and the possibility to operate the detector at low pressure enhances further its high counting rate capability. So far, the monitor was tested up to 800 kHz with reasonable count losses (< 10 %). Number of other options, available thanks to the "Multitube" geometry, will be presented, such as absolute flux determination of monochromatic beams, beam profile in 1D, and beam mapping in 2D.

08 Decommissioning, Dismantling and Remote Handling / 35

#08-35 Passive isotope specific gamma ray tomography of a nuclear waste drum using a CeBr3 Compton camera

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In various situations such as legacy wastes or optimising waste storage and decontamination costs, it would be very helpful to know the 3D distribution of various contaminants inside an object prior to decision making. Compton camera with their wide filed of view could allow such tomography in a cost effective way, but so far their spatial resolution and sensitivity was too limited for precision 3D imaging.

We have been imaging a 220 l nuclear waste drum at ANDRA using a temporal imaging CeBr3 camera. Three views of the drum defining an orthogonal trihedral in the referential of the drum center were acquired using a mechanical fixture in order to insure precise relative position and orientation between the views. The drum contained a total activity of 14 MBq with 3 main isotopes detected in the Compton spectrum: 60Co, 110Ag and 137Cs. Each of the 3 acquisitions lasted 20 minutes. As the object imaged is relatively small and low density, one view seems sufficient for each trihedral axis.

An image reconstruction algorithm has been developed based on cone intersection in 3D space. The image was then treated using LM/MLEM to increase contrast. We then create images of activity distribution for each of the isotopes detected. For this drum, we had also a 2D X-ray image showing structures inside the object. We have thus cut our 3D gamma images in planar slices each 2 cm and project them as a movie with the x-ray image as a background in order to understand the organisation of contamination inside the drum.

04 Research Reactors and Particle Accelerators / 36

#04-36 Some Considerations on The Energy Deposition During a Ria Transient Based on Monte Carlo Simulations

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Specific research reactors are capable of reproducing reactivity injection accidents in order to study the behavior of the nuclear fuel pins in accidental situations. The CABRI experimental pulse reactor, funded by IRSN (French Institute for Radioprotection and Nuclear Safety) and located at the Cadarache research center, is used to simulate power transients typical of reactivity initiated accidents. The fuel pin (test pin) to be examined is placed in the center of the core in a dedicated test loop. The pin is then subjected to a power transient, obtained by the fast depressurization of the 3He neutron absorber gas from the transient rods located in the core.

One of the central parameters of the experiment is the energy deposition in the test pin. This parameter however cannot be measured experimentally during a transient. Instead, it is assumed that the relative energy distribution between the core and the test pin is constant regardless the operational state of the reactor. Currently, this correlation between the energy deposition in the core and the test pin is measured and calculated in steady state and it is then applied to the transient. As such, the impact of the variations in the neutron flux, fuel and moderator temperature during the transient is assumed equivalent on the energy deposition in the core and in the test pin.

The aim of this paper is to present a methodological approach for the energy deposition calculation during a CABRI transient. The goal of this work is to improve our knowledge on the mechanisms involved in the transient energy deposition in the test pin. To achieve this goal, the first step is to determine the main parameters to which it is sensitive based on static Monte Carlo calculations. The results show that the transient energy deposition rate is mainly dependent on the helium pressure and the Doppler feedback.

04 Research Reactors and Particle Accelerators / 37

#04-37 Numerical and experimental characterization of the neutron flux and spectra in the core of the CNESTEN's TRIGA Mark II research reactor

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Education, training and isotopes production are the most important uses of the Moroccan 2 MW TRIGA Mark II reactor situated at the National Center for Energy Sciences and Nuclear Techniques (CNESTEN, Morocco). To develop new R&D projects in research reactors, the particular and advanced knowledge of neutron and photon flux distribution, within and around the reactor core, is crucial.

In order to precisely prepare the forthcoming experiments in the reactor, a detailed model of the reactor core was developed using the 3D continuous energy Monte Carlo code TRIPOLI-4 [1] and the continuous energy cross-section data from the JEFF3.1.1 nuclear data library. In particular, all the geometries and compositions of materials were described in the TRIPOLI-4 simulation model with good details. This new model was used to carry out preliminary neutron and photon calculations to estimate flux levels in the irradiation channels as well as to calculate kinetic parameters of the reactor, core excess reactivity, integral control rods worth and power peaking factors. As a first step of the validation of the model, the obtained results were compared with the experimental ones available in the Final Safety Analysis Report (FSAR) of the TRIGA reactor [2]. Fairly good agreement was found, which indicates that the Monte Carlo model is accurate enough to perform criticality calculations of TRIGA reactor.

Furthermore, this work aims at experimentally characterizing the neutron flux and energy spectra in various irradiation channels inside and outside the reactor core. Absolute flux measurements will be carried out using the neutron activation technique [3] [4]. To set up the experimental design for the activation experiments a series of preliminary calculations were performed using the TRIPOLI-4 model to calculate the expected gamma flux/intensity levels of various materials after irradiations in different positions in the core and reflector. Different activation foils with known characteristics will be then irradiated and the activity of several isotopes will be measured with the Gamma Spectrometry Method. During the irradiation process, the reactor power will be monitored via two fission chambers (FCs) adjacently fixed in the reactor cavity. The count rate is proportional to the neutron flux, which is proportional to the reactor power in the operation range. Sources of uncertainty in reactor power can therefore be either the calibration of the FCs, or the uncertainty in the count rate. Along with FCs, gold monitors will be irradiated in an ex-core position in order to monitor the neutron flux leading to estimate the reactor power during each irradiation. Experimental measurements will be compared with calculated neutron flux evaluated with the new TRIPOLI-4 model of the reactor, as well as against the reference calculation scheme using MCNP5 [5] and ENDFB-VII, developed by the CNESTEN [2] [6].

The final results of the above-mentioned experimental campaign will be presented in this study only after the betterment of the current situation of the Covid-19 pandemic. With the gradual opening of the airports and boundaries, we will be able to carry out our experiments with full concentration and in good conditions.

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09 Environmental and Medical Sciences / 38

#09-38 Dose rate measurements in pulsed radiation fields by means of an organic scintillator

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Many dosimetric measurement systems are not suitable for the application in pulsed radiation fields regarding radiation protection scenarios. Even in a low mean dose rate in the range of 1 µSv/h. A main challenge is to process high detector loads within short time periods, while an appropriate dead time behavior and the suppression of pile up effects must be ensured.

A promising approach for an active dosimetric system fulfilling these requirements is the combination of a fast tissue equivalent scintillation detector coupled to a full digital signal processing unit. Such a system could allow real time dosimetry by measuring the deposited energy in the detector, while a discrimination between pulsed and non-pulsed events is realized by comparing the individual time stamps of the measured events. Additionally, pile up events can be identified by analyzing the pulse shape of the individual events. Therefore, a pulse shape parameter for each detected event is calculated by considering two individual integration gates (short gate and long gate) of the detected signal. If the long integration gate is in the range of the macropulse duration of the external beam, the deposited energy in the detector is proportional to the number of detected "pile-up events" and can be additionally considered in the dose rate analysis. Furthermore, due to the completely non-paralyzed dead time behavior of the detection system, it is possible to correct signal losses for the respective measurement.

A potential detector system based on a plastic scintillator and a digital data acquisition board was tested at the Bremstrahlungsfacility (γELBE, HZDR, Dresden, Germany) under various pulse frequencies (up to 10 kHz) and a macro pulse duration between 4 µs and 40 µs. The detection system was placed next to a polymethylmethacrylat phantom, which was irradiated with the bremsstrahlungs-beam. Additionally, dose rate measurements at the clinical TrueBeam therapeutic system (Varian) were performed, where the detector was placed outside the treatment room. For both measurements, it was possible to reconstruct the characteristic structure of the pulsed beam, which comprises the identification of the pulse length and repetition rate. Based on these measurements an appropriate analyzing algorithm for dose rate measurements was developed and will be presented.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 40

#07-40 Contamination Tests of New Silicone-Based Detectors for Beta-Alpha Radiation in Water

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We present the results of a series of radioactivity contamination tests on a novel contamination-safe scintillation detector to be used for alpha and beta radiation detection in water. Due to the short path-length of alpha and beta particles in water and the low detection limits needed to be compliant with the international legislations in matter of radiation safety for water intended for human consumption, this kind of detectors must have large area, very low intrinsic background and avoid, possibly, any kind of window between the monitored water and the detector active volume. When water is in direct contact with the detector surface even an extremely low contamination can therefore destroy the performance, while protecting the detector with a layer of passive material will reduce the detection efficiency, in particular for alpha particles, and the passive layer can get contaminated itself, making necessary its substitution and representing an important limitation to the realization of a radioactivity monitor that should work continuously for years.

The novel detectors are large-area silicone-based scintillators with functionalized surface, developed and produced by our research group in our laboratories. The technology we propose is a significant step forward in the direction of the realization of radioactivity monitors for water with high sensitivity and reasonable costs, to be used to improve security and quality of the water distributed to European and worldwide citizens.

A number of tests were carried out at the ENEA-INMRI laboratories with the aim to characterize the contamination and decontamination properties of the newly developed detector. Basically the initial activity of each detector foil was checked by low background radioactivity measurements. Further, detectors were immersed for fixed times and constant procedure in a variety of radioactive reference liquid solutions with different radionuclides in different chemical forms. To this purpose an automatic devise for controlling the immersion cycle was developed and used. After rinsing with distilled water the detector foils were again measured for residual radioactivity adsorbed on the foils surface.

Radioactivity measurements were based on HPGe gamma spectrometry and low-background alpha/beta counters. Characteristic limits were determined following the ISO-11929 standard.

Results are presented and discussed with reference to the various detector and functionalized surfaces developed.

08 Decommissioning, Dismantling and Remote Handling / 41

#08-41 Study of neutron background in order to improve radioactive waste drum characterization

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A usual way of radioactive waste drums characterization combines gamma spectrometry measurements with passive neutron measurements. This method is well adapted in case of alpha spectra, for which both measurements provide different pieces of information and then enable to measure actinides for a large range of waste densities. But some difficulties are encountered when alpha radionuclides activities are so low that they cannot be measured in reasonable measurement time durations. In high density waste which prevents gamma signal from reaching detector cells, characterization with a high confidence level becomes a major issue.
The CEA DIF facility of waste drum characterization has been operated for more than 30 years. In this framework, a large variety of waste drums has been characterized in terms of spectra, densities, materials and radioactivity levels. As the facility was first dedicated to measure Intermediate-Level Long-lived Waste (ILW-LL), the neutron spallation background was not significant compared to expected neutron emitters from waste packages. These last years, Dismantling and Decommissioning operations have been well advanced at the CEA DIF and are now associated with mostly Low Level Waste production. Therefore, neutron spallation background is becoming significant.
Using the large variety of past characterized drums brings the opportunity to study this background. The present study has been led over a sample of almost 1500 drums of a wide density range. These drums have been selected during the last 20 years by taking into account only one criterion which is any expected neutron emitters from the waste itself. This work first presents the technical settings of the measurement facility before describing raw data of their measurements. Next, following a statistical study over raw data enables to better acknowledge the neutron spallation background behavior in terms of time, density and materials. Finally, ensues a way of using this new knowledge in order to improve how to take into account neutron spallation background in passive neutron measurements of packages of low actinides activities and high densities.

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#07-42 Peak Area Consistency Evaluation in Gamma Spectrometry

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Gamma spectrometry is a non-destructive method used to identify and quantify the activity of gamma emitting radionuclides in a wide variety of samples, including environmental, waste, and radio pharmaceutical. Quantification of the activity of radionuclides in gamma spectrometry depends mostly on three inputs: efficiency calibration, peak area calculation and nuclide decay data. All three can present significant challenges to an accurate analysis. Self-consistency checks based on calculating the activity of a single radionuclide using more than one gamma emission energy are a powerful tool to reveal problems with the inputs. Traditionally, this self-consistency check has relied on calculating the line activity for a radionuclide using the assumption that all the counts in the peak at the gamma emission energy of interest originates from the radionuclide. This approach breaks down when two or more radionuclides contribute to the same peak in the spectrum.

A new approach has been developed based on the consistency of the measured peak area and the peak area that is accounted for from the activities of all identified radionuclides in the spectrum. Using the consistency of the peak areas instead of the line activities makes it possible to use self-consistency for samples containing more than one radionuclide contributing to the same peak. This self-consistency check can reveal incorrect shape of the efficiency calibration, missing interferences in the nuclide decay data, and point to peaks where the peak area calculation needs to be optimized. Because this method relies on the consistency of quantities calculated using multiple inputs, it can be applied to samples with unknown activities. During this presentation the peak area consistency evaluation (PACE) method will be presented and examples will be shown that demonstrates how it can be used to find inconsistencies in the gamma spectrometry analysis. This algorithm will be implemented as a report deployed in a future release of the Genie gamma spectrometry software, which can be used to identify discrepancies in activities and demonstrate how the impact on reported results may range from minimal to significant. Applying this method can identify and rectify sub-optimal inputs, obtaining more accurate results and ensuring quality data.

11 Current Trends in Development of Radiation Detectors / 45

#11-45 Development of new acquisition techniques for portable gamma-ray imaging

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Portable gamma-ray imaging is an emerging field in nuclear instrumentation, with applications in radiation safety, waste management, decommissioning, environmental and security applications. It provides users with a single portable instrument to detect, identify and localize radioactive material. Most imagers are based on position sensitive spectrometric detectors, including scintillators (NaI, CsI, GaGG) associated with small photodetectors (SiPM, APD) or semiconductors (HPGe, CdTe, CdZnTe). Intrinsic qualities of a detector are its sensitivity and its spatial and energy resolution. Compactness and robustness are also key criteria for achieving portability. In our system, named NuVISION, we chose to use CdZnTe (CZT) detectors, a room-temperature semiconductor whose properties are well suited for medium energy photons (few hundreds of keV).
Existing gamma imagers are also based on various imaging techniques. Among them, most popular are pinhole collimators, coded aperture masks and Compton imaging. Each kind of imaging technique has its own strengths and weaknesses.
• Pinhole imaging has low efficiency and narrow field of view but good contrast, good angular resolution and ability to work at very high fluxes.
• Coded aperture imaging has a high efficiency and good angular resolution but a limited field of view, poor contrast and less tolerance to high fluxes.
• Compton imaging is efficient for high-energy photons and provides a large field of view but often has a poor angular resolution and low tolerance to high fluxes. For designing a gamma-ray system, we face many tradeoffs as the various applications addressed have conflicting requirements in dose rate (from nSv/h to Sv/h), field of view (up to 4pi steradians), angular resolution (down to few degrees), type of object (point source or complex shapes). The solution we have investigated is to combine several techniques i.e. use coded aperture and Compton imaging simultaneously. Their good complementarity is a way to get more flexibility. To cover high dynamic ranges in count rate, gamma-ray imagers can switch between several acquisition modes: event-by-event list mode, photon-counting image mode acquisitions with or without spectrometry or even integration mode imaging for dose rate above Sv/h.
Beyond instrument specification, the concept of usage may depend on application. The most common way of working with gamma ray imagers is to perform a long exposure static acquisition on a tripod. However, other needs have emerged: surveillance or monitoring of an area with moving vehicles or pedestrians, embedding on a ground robot or drone, hand held source search and area survey.
To fulfill such requirements we have developed for the NuVISION imager a toolkit of acquisition techniques that we will present.
This gamma imager is composed of a 10 to 16-cm³ CZT
3D position sensitive segmented detector, a highly integrated readout and a FPGA/CPU based processing and control platform, peripherals (inertial sensors, telemeter, visible camera, etc.). The operating energy range is from 20 to 1400 keV. Coded aperture works on the full range. Angular resolution is typically 2.5° and results mainly from mask pattern used. Compton imaging can only work above about 300 keV. Its angular resolution improves with energy and is around 10° at 662 keV. It results from the combination of Doppler broadening effect with detector spatial resolution and energy resolution.
The coded aperture and Compton imaging operate simultaneously. In particular, the Compton imaging, that is capable of detecting a source from any direction is used to constraint coded aperture reconstruction and avoid artefacts due to out-of-field sources.
Our imager works in three main modes:
• Static acquisition, using a fixed acquisition set-up.
• Scan mode, using a pan/tilt mount to acquire data on a field larger than actual mask field of view.
• Dynamic mode, for mobile set-ups.
In particular, we have developed a motion compensated dynamic imaging technique that enable to use the camera as an handheld device, embedded on a vehicle or conversely, to follow activity of

mobile objects, even with weak sources requiring 10 seconds to be localized. We applied this technique to area monitoring for security, by integration inside a videosurveillance system.
Real life examples from the field will be presented for applications in radiation safety, decommissioning and security.
These evaluations show the potential of having in a single hardware platform combining imaging and spectrometry, enabling with appropriate software to build a smart and versatile tool for radiation instrumentation.

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#06-47 Analysis of the Signal over Noise Ratio of the hodoscope determined by Monte Carlo calculation

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The CABRI experimental pulse reactor, located at the Cadarache nuclear research center, southern France, is devoted to the study of Reactivity Initiated Accidents (RIA) for the purpose of the CABRI International Program (CIP), managed by IRSN in the framework of an OECD/NEA agreement. The hodoscope equipment installed in the CABRI reactor is a unique online fuel motion monitoring system, thanks to the measurement of the fast neutrons emitted during a power pulse by a tested rod positioned inside a dedicated test loop reproducing PWR conditions. This system is dedicated to the analysis of fuel displacement and degradation of the tested rod during the power excursion. Hence, one of the most important parameter measured by the hodoscope detectors is the Signal over Noise Ratio (SNR), characterizing the fraction of neutrons directly coming from the test rod ("signal") over neutrons coming from the core ("noise"). Nowadays, this value is measured during a power plateau of the CABRI reactor, a few days before the experiment.
It is interesting to calculate the SNR in order to define some quantitative criterions to improve hodoscope measurements and to understand if any modification linked to the test loop may significantly change this essential parameter.
In this article, the method used to calculate the SNR using MCNP6.2 Monte Carlo code will be detailed. Because the hodoscope detectors are located far away from the test rod (up to 4 meters), a 2D model of CABRI core and instrumentation has been implemented. No variance reduction techniques have been used to solve this problem in order to record the place of birth of neutron which contributes to the different scores with the goal to perform a detailed analysis of the SNR.
Another parameter of interest which will be evaluated by means of this new model is the so-called "scattering coefficient", which corresponds to the fraction of neutrons coming from the test rod and being scattered between their birth and their detection. This parameter is used to enhance the analysis of the fuel displacement which may happen during the power transient.
Finally, the comparison between calculated and measured SNR for a case study will be carried out. A quite good agreement between the 2D simulations and experiments recently performed in the CABRI reactor has been obtained.

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#06-49 MERARG Experimental Loop: A new deconvolution Method for FGR analysis

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MERARG experimental loop hosted at the LECA-STAR Hot Laboratory (CEA Cadarache) allows characterizing nuclear fuels with respect to the behavior of fission gases during thermal transients representative of accident conditions such as for example loss of coolant accident (LOCA). MERARG loop consists of three main parts: 1- an induction furnace 2- a gamma spectrometry detector 3- a glove box containing a micro gas chromatograph (µ-GC). The furnace is located in a high activity cell while gamma detector and µ-GC are both located in the back zone of the hot cell. The fuel pellet to be characterized (generally with its cladding) is introduced into a metal crucible. The sealed enclosure of the furnace is swept by the circulation of a carrier gas (argon or dry air) with a low flow rate (about 60 cm3 /min). As the temperature rises, the released gases are driven to the back zone to the gamma spectrometry station measuring their activities on-line. After passing in front of this on-line gamma station, the gases are then driven to a glove box where they are analyzed by µ-GC and then collected in storage capacities. In order to simulate the thermal characteristics of a loss of coolant accident, the annealing test consists in a first stage of thermalization at 300℃ followed by an increase in temperature up to 1200℃ (according to a ramp of 0.2℃/s or 20℃/s).

Consequently, fission gas release measurement is carried out at a distance of several meters from the sample where the release actually occurs, in a dedicated counting chamber located outside the hot cell itself (i.e. at the so-called back zone). It is therefore necessary to deconvolve the acquired experimental data in order to “rebuild” the release rate data at the sample position. This requires taking into account fission gas flow out of the furnace to the counting chamber and its dilution in the carrier gas. Up to now, a previous deconvolution procedure ran well for the major part of the experiments performed but showed, in very few cases, some drawbacks such as “negative release” or “unappropriate noise” yielding in unusable-reconstructed spectra.
This paper aims to expose a new mathematical deconvolution method for re-processing the experimental data obtained using Laplace transformation. This new method is based on a simulation of the measurement made during the heating of a glass capsule containing 85Kr that opened abruptly during a heat treatment. The release of the 85 Kr during capsule opening was considered as a Dirac function, which allows the fitting of the 85Kr measured at the gamma counter position by an analytically calculated transfer function. The obtained numerical transfer function is used to build elementary functions whose combination allows to describe the experimental release in a satisfactory manner while overcoming numerical instability issues. This deconvolution method provides an accurate modelling of the measured release and allows the analysis of the annealing test data making it possible to describe physical phenomena directly related to the gas release of the samples such as the release of Kr85 by bursts at different temperature levels.

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#11-51 Extended sources reconstructions by means of Coded mask aperture systems and Deep learning algorithm

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The localization of radioactive sources is a fundamental information in the scope of radiative environment analysis, for nuclear safety or Decommissioning and Dismantling applications. However, performing this localization is challenging since conventional optics cannot be used for high-energy photons. One main method consists of using a coded mask aperture, placed in front of a position sensitive detector for X and gamma photons. This is an indirect imagery method, which necessitates inversion algorithms in order to recover the positions of the sources from the hit map observed on the detector and from the knowledge of the projections of the mask on the detector, also called shadowgrams. Classical algorithms of deconvolution and iterative MLEM (Maximum Likelihood Expectation Maximisation) are usually used to perform this inversion. But since the problem is non injective and these classical algorithms are not intrinsically associated to regularisation methods, they are not able to reconstruct extended sources, neither their shape nor their positions especially when the number of pixels to record a shadowgram is limited to few hundreds. In this paper, we propose to address this problem from the prism of Deep learning algorithms, based on Convolutional Neural Networks (CNN), with an application to Caliste detector. Caliste is a CdTe miniature detector of 16x16 pixels, with a 625 µm pixel pitch, and its spectroscopic capabilities allow to select gamma ray events in narrow energy bands enable hyper-spectral imaging at high energy. We generate synthetic learning data of extended sources in order to train our CNN to localize radioactive sources from simulated hit maps representing the Caliste detection matrix. Once trained, we test the CNN on real data acquired with a gamma camera prototype, equipped of a Caliste detector and a coded mask aperture. We demonstrate the ability of this algorithm to reconstruct real extended sources at the 60 keV emission line of 241Am.

08 Decommissioning, Dismantling and Remote Handling / 53

#08-53 The Euratom project MICADO and its innovative characterization process of the Nuclear Waste Packages

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In 2019 the MICADO (Measurement and Instrumentation for Cleaning And Decommissioning Operations) project started under the H2020 Euratom call aiming to become a reference in the nuclear waste characterisation field. It can be possible by the system under development and to the partners involved representing some of the most important actors of the sector. They are covering the roles of academia, industry, and the end-users.

All over the world the nuclear waste management sector is always considered by the population a scaring problem and it is always part of the public debate, mainly due to complexity and costs coming from the storage and management of the waste produced. It is important to underlying that this waste is not only produced by the nuclear power industry, but also by hospitals, universities and non-nuclear industries like oil and gas companies with the production of NORM (Naturally Occurring Radioactive Materials) and TENORM (Technologically Enhanced Naturally Occurring Radioactive Materials) waste. Independently from their origin, the main concern is the radiation emission, which makes it a particular hazard for human health and the environment. It must therefore be managed with special care, from production to final disposal.

The situation gets even more complicated when considering country dependent legislation, storage and final disposals sites. This means dealing with different definition of waste categories and activities (i.e. free release), the necessity to use multiple radiological sensors with not compatible outputs inducing the need to reprocess the characterization at each site, to analyse a large amount of off-line data, process manual reports of the operators, etc.

MICADO wants to show a way to improve the characterization of nuclear waste packages and change current manual operations applying a digital analysis procedure, waste-type dependent, and combining information from different detectors to better qualify the waste package under investigation.

MICADO has established a characterization process, data analysis and information storage able to cope with different types of waste activities (VLLW, LLW, ILW, legacy waste), types (Metallic & concrete filling) and drum dimensions.

This is done with a toolbox with up to date and novel gamma and neutron detection technologies (two other abstracts submitted on the monitoring grid and on the photofission system), working as modular elements, and a digital software platform used as a base for the digitalization of waste information and the off-line analysis for the uncertainties assessment. The procedure was defined to reduce the measurement time in each step and being able to select the required detection technology avoiding multiple identical measurements of the same waste package. The combined data analysis, like other big-data studies, fuses different measurements results to extract information not available by the individual system and reduces the individual uncertainty. This aspect is extremely important as a possible solution to the problem of having a satisfying and reliable categorization of the waste package activity of complex cases as high density waste drums or the request for the free release. The software platform also aims at reducing operator costs and improving the ALARA principle, decreasing the time spent on field by the operators and promise a simple and easy data control on historical basis of all the already characterized waste packages.

The presentation will start with the project overview but will focus on the status of the overall technologies after one year, from its start and tests performed. It will also be given a look to the future steps toward the end of the project and more till the organization of the final demonstration.

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#08-54 Deep Learning for Compton image reconstruction with Caliste miniature imaging spectrometer: An innovative approach to enhance detection sensitivity

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Caliste is a miniature CdTe-based pixelated imaging spectrometer. The detector is a single plane 1 cm² crystal readout by 8 low noise full custom ASICs enabling high energy resolution. Thanks to its position sensitivity and its spectrometric performances, this detector is able to perform Compton imaging in order to localize radioactive point sources, emitting high-energy gamma-ray photons, and its compactness is advantageous to be the detection component of a portable Compton gamma camera. However, our current prototype is equipped with a small detection volume of 100 mm3 (1 mm thick). Because our detector is thin and the photon interaction positions are determined in 2D, Compton reconstruction in such a miniature single plane detector is challenging. More specifically, the 3D interaction position uncertainty in the crystal and the small detector volume are both critical limitations for efficiency and sensitivity of the detector for Compton reconstruction. Nonetheless, Compton reconstruction is possible and reported in the paper. Classical backprojection and more advanced algorithms in the literature, such as LM-MLEM (List-Mode Maximum Likelihood Expectation Maximisation) and SOE-RR (Stochastic Origin Ensemble with Resolution Recovery), require to detect at least an order of magnitude of 2000 Compton events to reconstruct a radioactive point source. With Caliste geometry, this necessitates at least 7 hours of acquisition to localize a 137Cs point source of an activity of 1.6 MBq at 30 cm from the detector, with a corresponding dose rate of 1.4 μSv/h at the detector level. In order to enhance the sensitivity of our tiny system, we develop in this work a Deep Learning approach to reconstruct Compton images from spectroscopic data and identify the position of point sources over 2π steradians field of view and we demonstrate the feasibility of this new algorithm for high-energy imaging. We train a Convolutional Neural Network (CNN) with a synthetic database to localize a point source from direct backprojections, computed by Monte-Carlo simulations. We apply this CNN to real data at the 662 keV line of 137Cs, acquired with Caliste and we show that we are able to reconstruct point sources with an order of magnitude of 200 Compton events. This corresponds to a sensitivity of about 150 nSv/h with only a 0.1 cm3 detector, which surpasses the performances of classical algorithms.

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#09-55 The CORSAIR Project. Characterization of a portable instruments for NORM characterization of stone blocks

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The CORSAIR (Cloud Oriented Radiation Sensor for Advanced Investigation of Rocks) project was born to meet the EU guidelines 2013/59/EURATOM and now the Italian legislation decree D.Lgs.101/2020 on safety standards for protection against ionizing radiations.
With this project there is a specific focus on the detection of the NORM (Naturally Occurrence Radioactive Materials) contribute coming from stone blocks used in the engineering field.
As a matter of fact, radiological measurements and traceability of materials are two aspects becoming more and more important in this and for the green building sectors.
In Europe exist regulatory restrictions on the maximum level of radiological emissions for materials used in public and private building structures. A similar approach is also adopted in other countries all over the world. These legislations not only are fixing the radiological limits, but forbid purchasing and incoming of not controlled or above limits materials in the country. These legislations are acting on import/export of the stone market, one of the Italian sectors already affected by the incoming of low cost not natural (Okite, high pressure laminate or ceramic stones) and not certified natural stones.
Alle these aspects drove the CORSAIR project to design an automated cyber-physical system capable of providing a real-time measurement of the radioactive activity concentration index for building materials according to regulations of more than 20 different countries. It quantifies activities, abundances and related effective dose-rates of natural radionuclides (^{40}K, ^{232}Th, ^{238}U) and their progenies).
The system is empowered by cloud-based technology consisting of sensing nodes, data collection gateways and a centralized cloud application. These components are interconnected in a star-of-stars topology, exploiting GPS, Bluetooth, LoRaWAN, WIFI network connection and providing specific user interfaces with an Android app running on smartphones.
Measurements are conducted through in situ γ-ray spectroscopy techniques on blocks of rock at quarries or processing centres.
The system is designed for providing an autonomous, fast, repeatable, real-time and non-destructive method to measure the radiometric indexes with a 30 min measurement.
The innovative aspects of the system are due to the integration of several elements: its autonomous operation that does not require expert operator to provide results, the easy fruition of the results of the material characterization thanks to the cloud database, the presence of the RFID tags used to uniquely identify each analysed block, the energy calibration and the fully-automated results computation.
All system outputs are finally synced to a cloud database, where they can be easily accessed by operators and clients, enabling to trace the materials along the full market chain, from extraction to the final customer, with modern technologies preventing the placing on the market of blocks hazardous to public health and as a warranty of the origin of the block thanks to the RFID tag used.
This presentation after describing the system will provide tests results performed in-situ on real blocks and the comparison with laboratory measurement performed to characterize the radiological system.

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#08-56 Spid-X: A Gamma camera with spectro-identification and dosimetry embedded functions

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A gamma camera, allowing the localization of radioactive sources, is a very useful device in various fields of the nuclear industry: monitoring, D&D or waste management are some examples. However, the information of the source position is not enough and should come with the sources identification and dosimetry information, which are provided by additional devices, such as radiameters. In this paper, we present the various functions and performances of the Spid-X Gamma camera, which overcomes this problem by performing in real time the 4 following functions: automatic identification of radioactive sources, proportion measurement of each detected sources, imaging of sources and dosimetry at the camera level. This is possible thanks to the embedded Caliste-O technology, which is a CdTe-based pixelated imaging spectrometer using a single plane 1.4 x 1.4 cm² crystal and 8 low noise ASICs as a readout enabling high energy resolution. Thus, Caliste-O allows position sensitivity and fine spectroscopic performances that are a perfect fit to develop the Spid-X camera. The spectro-identification is performed thanks to advanced Convolutional Neural Network (CNN) trained on synthetic data, which can determine from a measured spectrum made of a few photons which radioisotopes created the signal, even in the case of multi-sources detection. In this last case, it also provides the information of the relative proportions of the different sources. This identification capability is coupled to imaging algorithms, allowing precise Coded Mask and Compton imaging on a wide energy range from 2 keV to 2 MeV and localization of each point and extended detected sources. Finally, dosimetry measurement at the device position is automatically calculated in order to provide to the used the answers to the three following questions: what do we detect, where is it, and how much there is, enabling efficient and safe use in the nuclear industry.

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#08-57 Characterization via the RadHand device integrated into the REACh system for a low-cost in-situ waste characterization of nuclear waste

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A key aspect involving both operational activities as well as radioprotection in NPP (Nuclear Power Plant) is the management of nuclear waste. Providing efficient and reliable real-time radionuclide concentrations contributes invaluable information while processing nuclear waste as it can in turn reduce costs with packaging, transportation, and disposition for NPPs.
Typical NPPs procedures can involve characterizing nuclear waste by using an HPGe detector to determine the radionuclide concentrations and the individual isotope abundances. In most instances, this requires NPPs to be equipped with a facility on site (e.g., laboratory) which can perform these measurements. These analyses require a radiological expert to analyse the results. Additionally, this process often taken an immense amount of time and effort to obtain the results for well-defined gamma-emitting radionuclides which are easily detectable.
Due to the integration of software and hardware technologies in the smart era, we are able to provide a faster and more user-friendly waste package characterization system than what current methods are offering; this saves time, improve flexibility, movable, and is a more cost effective solution.
Current characterization practices involve obtaining waste stream specific distributions which are then applied to all waste packages of the same waste stream type (e.g., DAW (Dry Active Waste) waste).
It is important to remember that this type of analysis is typically performed by HPGe detector. The main drawbacks for a system such as this are that it typically involves long measurement times, is a very expensive solution, and requires post process analysis to be performed to determine the radionuclide information. Equipping a laboratory with this type of instrument requires also a supporting structure for the shielding weight of the HPGe, a liquid nitrogen filling for the operations, and a budget on the order of 100 kEuro for buying it.
The REACh system represents a step change in the way low-level radioactive waste is characterized and classified. The system is intended for all waste packages for open geometry measurements by providing a low-cost detection technology with an easily transportable device.
The Reach system directly measures gamma dose rates and gamma emitting activity by radionuclide for packaged radioactive material to perform characterization and classification. To obtain an effective and complete characterization and management of the waste packages (WP) the REACh system is equipped with the RadHand detection system used for the gamma characterization, a software database for data storage and handling and it has the capability to use the RFID technology. The system can track in every moment the uniquely identify WP using the attached RFID tag and restore in every moment the WP characterization history using the developed database.
The RadHand system was specifically re-designed to perform static measurement using a tripod and to work in an open geometry in front of the waste package. The system allows users to work with LLW (Low Level Waste) activities up to which can read up to 1 mSv/h at contact. Additionally, the system is equipped with external shielding to reduce the intensity at the detector surface should it be required.
This presentation will focus on the laboratory test made with the RadHand system to evaluate its performances in the two fixed configurations. The feasibility to substitute the old typical procedure based on HPGe characterization with this new system will be analysed.

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#09-58 Performances of a very high efficiency Imaging Camera for NORM radioactivity detection

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Naturally occurring radioactive materials NORM are materials which may contain any of the primordial radionuclides or radioactive elements as they occur in nature, such as radium, uranium, thorium, potassium and their radioactive decay products, that are undisturbed as a result of human activities. furthermore, the technologically enhanced NORM, TENORM are generated in the form of by-products, residues and wastes, from industrial processes that exploit natural resources such as coal combustion, fertilizers production, processing of metal, oil mineral ores extraction and, generally, many other industrial processing. The management of these materials is receiving more attention compared to the past due to large volumes of generated NORM, low specific activities and very long-lived radionuclides. NORM and TENORM should be evaluated as a pressing environmental hazard and should be monitored and treated with new specific techniques.
We designed a camera for gamma imaging and radionuclide identification based on the coded mask technique. The camera proposed is a compact, self-consistent, with high detection efficiency and good energy resolution in the full range from ^{241}Am up to ^{60}Co, ideal for real-time analysis on the go, with a low power consumption, suitable for industrial process control and ambient monitoring. We built a prototype consisting in 16 CsI(Tl) (3x3x10 cm^3 each) scintillators coupled to photo-multiplier tubes (PMTs) with a digital readout (CAEN digitizer V1725). The scintillators are arranged in a 4x4 matrix and packaged in a metallic frame. We used a 7x7 mask composed by transparent and opaque tiles (3x3 cm^2 for a total size of 21x21 cm^2) to encode radioactive gamma-rays sources image and used a reconstruction algorithm for decoding.
The system was tested in laboratory using free gamma-ray radioactive sources placed at a fixed distance from the mask to measure the camera point spread function (PSF) as function of the acquisition time. we also present the results with a NORM igneous rock sample and a waste drum (typically used in industrial0 nuclear waste management) and we identify the natural radioactive line from its spectrum, and we show the imaging results. We also present the camera minimum detectable activity (MDA) calculated for a source at 15 cm from the camera in a partial lead shielded configuration for different acquisition times, to point out the suitability of the camera in industrial waste monitoring.

11 Current Trends in Development of Radiation Detectors / 60

#11-60 Pulse shape simulation for organic scintillation detectors using GEANT4

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Plastic scintillators exhibiting pulse shape discrimination properties represent a promising, solid-state alternative to the use of organic liquids and crystals for the detection of neutron and gamma radiation. They are robust, inexpensive and can be fabricated in a variety of shapes and sizes. The time-dependent pulse shapes derived from plastic scintillation detectors can be characterised by a rising edge and multiple decay time constants. These time constants relate to the scintillation mechanism following particle interaction and differ depending on the type of interacting particle. The technique of pulse shape discrimination (PSD) enables the separation of neutron and gamma ray induced signals based on subtle differences in their pulse shape. The objective of this research is the development of accurate pulse shape simulations that are capable of reproducing the pulses measured experimentally for organic scintillation detectors. The ability to accurately simulate pulse shapes presents the opportunity to assess the PSD performance of scintillation detectors prior to fabrication, enabling this to be optimised in the initial stages of detector design. This is particularly important for detectors which utilise plastic scintillators as the PSD performance of these materials varies as a result of numerous factors affecting the shape of the pulses. This includes the scintillator geometry, where the ability to separate out neutron and gamma ray induced signals becomes degraded as the size of the scintillator is increased. Work presented demonstrates the use of the Monte Carlo toolkit GEANT4 to simulate the time-dependent pulse shapes from EJ-276, a pulse shape discriminating plastic scintillator developed by Eljen Technologies, coupled to an ET-Enterprises 9214 photomultiplier tube (PMT). GEANT4 has been used to simulate the generation and transportation of scintillation photons up to their detection at the photocathode for commonly used scintillator geometries. These represent the scintillation detector pulse shapes when the response of the PMT is excluded. Since the temporal response of the photodetector impacts on the overall shape of the pulses, future work will focus on experimental work to precisely measure the response of the PMT and integrate this with existing simulations.

04 Research Reactors and Particle Accelerators / 61

#04-61 PISTIL, a reactivity modulation device to probe the transfer function of the research nuclear reactor CROCUS

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The present article summarizes the development and testing of a reactivity modulation device developed by the French Atomic Energy Commission (CEA). It was installed in the CROCUS reactor of the Swiss Federal Institute of Technology in Lausanne (EPFL). Experimental tests were performed in the framework of a collaboration between CEA and EPFL.

The so-called PISTIL reactivity modulator (Periodic reactivity Injection System Transients Induced Locally) aims at measuring the nuclear reactor transfer function in the frequency range of interest, between 1 mHz and 200 Hz. The device is made up of several aluminum tubes (outer diameter 10 mm), two inner ones holding Cadmium foils. The center tube can be rotated by means of a brushless motor, while the outer one remains static. During the rotation, the radiative capture reaction rate of Cadmium can be varied. This gives rise to a controlled reactivity modulation when the device is within or close to a nuclear reactor core, as the in-core neutron population is varied. The height of the ensemble, as well as the relative position between rotary and static parts are adjustable, offering flexibility on reactivity worth and reactivity amplitude.

Thanks to its compactness, PISTIL can be inserted at the center of the fuel lattice of CROCUS. In the experimental setup, the maximum reactivity worth, as estimated using TRIPOLI-4 Monte Carlo code, was less than 0.15 $ protect and the maximum amplitude of the reactivity modulation was about 0.02 $.

Out-of-pile tests inside a dummy mock-up of CROCUS were conducted for the mechanical characterization of PISTIL, while the device was driven periodically to rotate up to 100 Hz. Recorded motion profiles and induced vibrations were analyzed for performance and safety studies. In-core reactivity calibration was then performed for several reference configurations of PISTIL. The measured reactivity worth and modulation amplitude were consistent as compared to TRIPOLI-4 estimations.

06 Severe Accident Monitoring / 62

#06-62 Design and characterization of a gamma imaging system for fuel rod deformations

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The study of accidental situations is one of the major asset for nuclear industry in guaranteeing the so-called regulated safety of its electricity production facilities. The Loss Of Coolant Accident (LOCA) is tested in experimental reactor and hot cell to verify the mechanical properties of the fuel rod, the safety criteria and the associated calculation codes. The VINON-LOCA experimental set-up, currently under development in CEA Cadarache, aims at observing, online, the Fuel Fragmentation, Relocation and Dispersal (FFRD) during a LOCA sequence. The tests will consist in placing a fuel rod in a shielded cell, which will be heated by magnetic induction in order to reproduce the temperature conditions of the first phase of a LOCA transient. Instrumentation positioned in the cell will be placed close to the fuel rod in order to monitor its behaviour during heating. A proposed technique to quantify the fuel rod deformation is the use of gamma imaging. In this context, we have designed and developed, in partnership with CEA Cadarache, EDF and Framatome, a gamma imager to monitor the deformation of the fuel rod. Preliminary experimental tests carried out at LAMIR Laboratory of CEA Cadarache have estimated that the variation in the diameter of the fuel rod during heating is 2 to 3 mm. The gamma emitters, which will enable the swelling to be monitored, are derived from the fission products of the pre-irradiated rod. Only the most energetic gamma rays passing through the cladding of the rod, and then through the experiment's containment, will be imaged.

The work presented therefore concerns, firstly, the design of a collimator offering the best compromise between mechanical feasibility, cost, signal-to-noise ratio and angular resolution, thus making it possible to visualise the deformations of the fuel rod, and secondly, a preliminary evaluation of the experimental performance of the proposed gamma imager.

The chosen detector is the Widepix pixel detector developed by ADVACAM s.r.o. The Widepix detector is based on the Timepix technology resulting from the Medipix international collaboration founded at CERN. It consists in 512 × 1280 square pixels of 55 µm-pitch, hybridised to a 1 mm-thick semiconductor, i.e. a pixelated detection surface of 20 cm². In addition to positions of interactions, the detector also provides their energies, which can be used to reduce the intrinsic resolution of the pinholes in the collimator.

The collimator was dimensioned using Monte-Carlo simulations with MCNP6.2. During the design of the collimator, the material (tungsten), the pattern (with parallel holes) and the geometry (thickness, septa and pinhole diameter) were studied. Among the strong hypotheses, we chose to design the collimator with the most energetic gamma rays emitted by the radioisotope with the highest photon flux, namely Cs-137 emitting gamma rays of 661.7 keV. We finally ended up with a collimator of 6.2 cm thickness, pierced with an arrangement of pinholes of 1 mm diameter and 1 mm septa. This design results in an angular resolution of 0.48° for a field of view of 2.8 × 7 cm² at 25 cm.

Preliminary experimental measurements were then carried out on a Cs-137 irradiator located at the LNHB of CEA Saclay, and validated the possibility of identifying a variation in the incoming gamma rays flux diameter of the order of 2 mm using the proposed imager.

04 Research Reactors and Particle Accelerators / 63

#04-63 CAREDAS: a Comprehensive Architecture for a Redondant and Evolutive Data Acquisition System for JHR reactor and measuring bench for CEA Cadarache facilities

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A new material testing reactor Jules Horowitz Reactor (JHR) is under construction at CEA Cadarache. The materials to be irradiated will be placed into experimental devices around the reactor. Process and measurements of experimental devices will be carried out by command control. Programmable Logic Controller (PLC) dedicated on each experimental device will realize this function. Some experiment devices will need to achieve complex real time processing that PLC cannot reach. A data acquisition system (DAS) having these processing performances will be associated to the PLC. This system will realize measurements and processing. The challenge is to design and realize for twenty experiment devices a high availability data acquisition system architecture for 50 years of sustainability. The real time target will achieve 24/7 data acquisition and real time processing. The other components will be in charge of data storage, online and offline data visualizations, experiment setting and processing without modifying software core.
A proof of concept have been realized by the end of 2019. This scalable architecture could be use as well for JHR experimental devices with high availability as for testbed. This architecture could be run on a standalone station for a measuring bench or deployed on cluster with redundant data acquisition system for high availability. CAREDAS's design is modular and use proven widely used open source solutions of IIoT world (Industrial Internet of Things). All parts are independent from each other and can be replaced with another technology with the same functionalities. This ensures sustainability and control of software sources.
The heart of CAREDAS is the communication between functions. It is based on MQTT (Message Queuing Telemetry Transport) for message transmission and Google Protocol Buffer for data serialization. Data are stored in InfluxDB time series database. Offline data visualization is achieve with Grafana to compose dashboards that query directly InfluxDB databases. Only three software in C++ language are homemade to ensure sustainability: real time data acquisition, writing data into InfluxDB, tools for experiment setting and supervision. All software run on station or cluster nodes on the Linux operating system (Red Hat distribution).
Real time data acquisition software is modular and technology independent. The operating system is Linux with real time functionality. For instance, technologies implemented are PXIe bus, CompactRIO/CompactDaq from National Instruments, Ethercat, various measurement units over Ethernet.
Now, CAREDAS is under development and a first stable version is planning for September of 2021. Meanwhile, a beta version was deployed on five measuring bench on some CEA Cadarache facilities.

09 Environmental and Medical Sciences / 64

#09-64 Deconvolution methods used for the development of a neutron spectrometer.

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This presentation stems from a thesis project in progress, leading to a unique multi detectors Bonner sphere. Neutron spectrometers like Bonner spheres have been studied and developed for more than 60 years for astrophysics and radiological protection applications. In particular, in radiological protection, it is a crucial challenge to determine the initial energy of the incident neutrons with a sufficient accuracy, to evaluate their damage to human body.

During the simulation part of the development, GEANT4 simulations were performed to study the response of Bonner spheres instrumented with 30 to 50 active detectors. The aim of these studies is the optimization of the number and size of the active detectors and the diameter of the sphere. One of the most significant issues was the unfolding of the different simulated "measures" obtained by the active detectors to get the original energy spectra with neutron energies between 0 and 20 MeV and to evaluate the errors related to these spectra. This unfolding is a key point for the optimization of the multi detector Bonner sphere.

In particular, three deconvolution methods have been developed and tested : the least-squares, the maximum entropy coupled to maximum likelihood and the maximum likelihood methods, which require a certain number of parametrical adjustments to become optimal. In general, every method needs a converging algorithm to be accurate.

• For instance, in the maximum likelihood methods, where the likelihood function is based on Poisson distribution, it is due to the nonlinearity of the involved equations, which cannot be analytically solved. However, these methods work for underdetermined – in case of an insufficient number of detectors in comparison to the number of groups of energies - as well as overdetermined problems, and can be coupled to the maximum entropy method ;

• In the least-squares methods, it is due to the important sensitivity to perturbations related to the matrix inversion and to non-positive solutions. To resolve this, the iterative Levenberg-Marquardt's method can be used. It implies nonlinear equations as well but leads in particular to only positive solutions, by inserting exponential functions in the solution and a Levenberg-Marquardt's parameter λ which has to be modified after each iteration. But this method does not work for underdetermined problems.

The results obtained with these methods have been compared to the unfolding package distributed by the Nuclear Energy Agency, UMG 3.3 - Unfolding with MAXED and GRAVEL. This package consists of two unfolding programs based on maximum entropy (MAXED) and least-square methods (GRAVEL), one spectrum analysis program (IQU) and one graphical display program (UMG-plot).

The different methods will be introduced, then some examples of unfolding spectra will be presented and the performances of the different methods discussed. Finally, firsts results of optimizations of the dimensions of the sphere and the dimensions and number of active detectors will be exposed. First developments and tests of neutron detectors dedicated to the multi detector Bonner sphere will also be presented.

05 Nuclear Power Reactors Monitoring and Control / 65

#05-65 Iterative method in modelization of gamma radiotracer measurements on the Colentec loop in Cadarache for the quantified analysis of the clogging phenomena in Steam Generators

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The Tube Support Plate blockage, also named clogging, is a complex phenomenon that can occur in the steam generator of Pressurized Water Reactors. This deposit mainly composed of iron oxides, may induce several consequences (thermohydraulic flows changes inside SGs, vibrational or oscillatory risks, mechanical resistance of SG tubes and internals, impact on water inventory …). .
Since 2014, the representative dedicated equipment, named COLENTEC loop at Cadarache had already provide a large number of results on the physico-chemical properties of the deposit formed in specifics thermohydraulic and chemical stable conditions. A major improvement was the switch to an active configuration allowing the injection of a gamma radiotracer 59Fe inside the circuit. Thanks to three dedicated gamma-measuring stations specifically designed by our laboratory, a first test showed that it was possible to determine on line the influence on clogging of parameters such as the chemistry, the temperature, the pressure without waiting for the opening and the dismounting of the test section.
This paper presents the iterative method of modelling performed for the analysis of the results obtained during this first test on the three gamma detectors.
The analysis starts with the simplest low background gamma measuring station in place and allows quantification of the 59Fe measurements of the fluid inside the loop.
The modelling, with a Monte Carlo code MCNP, of the second on-line measuring station positioned on a section of the loop provides quantitative results on the residual 59Fe deposit along tubes.
Finally, the sophisticated modelling of the geometrical configuration of the main on line measuring station on the Tube Support Plate allows determining the extent of the clogging.
The presented method is then able to discriminate but also quantify primary and secondary phenomena taking place in the specific selected thermohydraulic and chemical conditions. It therefore improves the understanding of clogging and could lead to a better management of the maintenance of steam generators in power plants.
These works are carried out within the framework of a scientific and financial collaboration between CEA and EDF (France) that the authors gratefully thank.

06 Severe Accident Monitoring / 66

#06-66 The VINON-LOCA test facility: exploring the LOCA phenomenology through an out-of-pile thermal sequence on irradiated pressurized fuel rod

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Since the Halden loss-of-coolant accident (LOCA) test series IFA-650, a major safety interest has raised for Fuel Fragmentation, Relocation and Dispersal (FFRD) during a LOCA sequence. Besides the characteristics and the behavior of the fuel ejected from the rod after the clad burst, the occurrence of the ballooning, of the fuel fragmentation and of the possible fuel relocation within the rod before the clad rupture is still to be investigated. For this purpose, the VINON-LOCA program is devoted to perform Out-Of-Pile heating tests on irradiated repressurized fuel rod, in the framework of a trilateral agreement between EDF, Framatome and CEA.

The experimental set-up is implemented in a hot cell of the LECA-STAR facility of the IRESNE Institute of CEA, at the Cadarache research center. After a thermal reconditioning phase at 300℃, the thermal transient sequence consists in a ramp test (typically 5℃/s from 300℃ to 1000℃) performed in an induction furnace under an inert gas circulation. The tested rodlet is initially pressurized at around 50 bar (ambient conditions) and equipped with thermocouples attached to cladding (rather than welded) for monitoring the clad surface temperature.

One of the specificity of the VINON-LOCA experiment is the online gamma measurement systems: the expected ballooned region is examined both with a classical collimated gamma spectrometry station and with an innovative 2D fast gamma imaging system to have on-line measurement of fission product activity (specifically developed for this application with CEA/DRT). Another gamma spectrometry station targets the experimental rodlet plenum region in order to estimate Fission Gas Release (FGR). In this region, the gas composition is also measured online with an acoustic sensor (developed by the IES Electronic Institute at Montpellier) coupled to pressure and temperature on-line measurements. . In addition, FGR after clad failure is addressed thanks to both online and post-test gas analyses (microchromatography, online gamma spectrometry, gas sampling). Pre and post-test characterizations of the rod include in situ gamma scanning (i.e. with the rod still in the furnace) using the capabilities of the collimated spectrometry gamma station. Additional measurements are performed on a dedicated instrumented bench: gamma scanning performed with a better accuracy, possibly gamma emission tomography, and laser diameter measurement. The fuel fragments ejected from the rod are collected, weighted and sieved after the test in order to evaluate their size distribution.

An extensive and substantial qualification campaign has been performed for validating the test protocol and conditions and for qualifying the instrumentation. It has included tests on an out-of-cell twin mockup, tests on dummy inactive rods in the hot cell, and was completed by some preliminary modellings and calculations. This allowed achieving successfully the first experimental qualification test of the program end of 2019 on irradiated UO2. A second qualification test is planned in 2021.

This contribution will highlight the experimental set-up and its associated instrumentation, illustrated by some results issued from the qualification campaign and preliminary tests.

08 Decommissioning, Dismantling and Remote Handling / 68

#08-68 1D OSL/FO dosimeter array for remote radiological investigations in hard-to-access zones

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Introduction

Remote dosimetry (RD) is an essential tool for Decommissioning and Dismantling (D&D) as it enables to predict the impact of decontamination procedures in terms of worker exposition and to set up a cost-effective dismantling scenario associated with a risk analysis and waste management strategy. However, long-range RD in hard-to-access zones (e.g. tanks, reactors, storage ponds etc.) is still challenging. A critical case concerns inspections through small-diameter pipes (< 1 cm) of small radius of curvature that require miniaturized dosimeters. The miniaturization of conventional active dosimeters leads to a reduction in detection volume that in turn degrades the limit of detection (LOD) in dose rate. Furthermore, miniaturization is limited by electromagnetic (EM) shielding and embedded electronics (power supply, signal conditioning, thermal regulation) and signal degradation arises due to cable length as well. Geiger-Mueller detectors are often used to this purpose but they are still too bulky and fragile and require a modeling to retrieve dose rate from counting data. During the past 20 years, CEA investigated an alternative RD technique based on OSL/FO probes (Optically Stimulated Luminescence/Fiber Optics), affixed at the extremity of armored optical cables, robust enough to be pushed and removed within pipes. OSL/FO is a passive online dosimetry technique which does not require local electronics to operate (the readout unit is out of radiological zone). The LOD may be as low as several µGy/h (for daylong integration times), therefore making OSL/FO useful for the assessment of cleaning processes for instance. This technique provides wide range in dose rate detection (6 decades) by a suitable combination of integration time and dose. Other main advantages are EM and Cerenkov immunities, long-range remote operation, high miniaturization (sensor head Ø = 5 mm), high radiation resistance and very low fading (the OSL signal changes linearly with the integration time at constant dose rate). Finally, OSL/FO sensors are waterproof and easy to decontaminate.

The INSPECT Project

The INSPECT project started 11/2016 for 54 months. It aims at developing two multichannel OSL/FO readout units, a human-machine interface, and innovative 1D OSL/FO detectors (10- and 16-point detectors) for the monitoring of dose rate profiles within the range [µGy/h – Gy/h]. They are connected to a readout unit through a MIL-38999 hybrid (electrical + FO) connector which also incorporates an EEPROM for immediate data transfer of manufacturing and calibration data. Finally, field tests are planned on several French pilot sites (Orano La Hague, CEA Marcoule, CEA Cadarache).

Conclusion

OSL/FO dosimetry provides radiological investigations in hard-to-access zones in the range [µGy/h - Gy/h]. It is now a proven RD technique for D&D, sparing heavy duty that would otherwise be necessary to provide access to conventional dosimetric means. Dose rate data coupled with topographic modeling of the inner infrastructure enable activity reconstructions with the help of Monte-Carlo modeling. The use of 1D detectors speeds up the investigation process and improves localization uncertainty in comparison with point dosimeters (0D). A performance assessment of the INSPECT device will be presented during the conference.

INSPECT is supported by Andra under the French "investing for the future" Program (Programme Investissements d'Avenir), organized in cooperation with the French National Research Agency (ANR).

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 70

#07-70 New Neutron Signal Acquisition and Processing Platform for Nuclear Safeguards

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The neutron coincidence counting is one of the fundamental techniques for non-destructive assay of Nuclear Safeguards. Several types of data acquisition modules are currently in use for this technique that is used for nuclear material monitoring and verification. However, some of those modules are obsolete or approaching the production discontinuity, and this aspect represents a serious issue for the Safeguards activity. The subject of this paper is a new platform of devices that addresses the current situation of limited availability of maintainable instrumentation and reflects the recent advances in microelectronics. Those devices shall satisfy the needs of attended and unattended measurements, bringing improvements from all perspectives: guaranteeing nominal performances in a wider range of measurement conditions, combining bigger storage capability, modern user interfaces and low power consumption. The main functions of the new devices of the platform are to collect the signals from neutron detectors, process and convert them into digital information and store them in data files. The devices share a common programmable hardware with industrial grade components for high reliability even in case of high temperature operational conditions. They embed a Field Programmable Gate Array for generating time stamped list of neutron signals for up to 32 input channels. The internal multithreaded microprocessor records the data, reproduces the functionality of shift register, multiplicity register and pulse train recorder. It provides at the same time a web server for a multiplatform graphical user interface. All the above-mentioned processing capability, the high and low voltage power supplies are integrated in a module with a power consumption of less than 10 W, which can be easily dissipated in sealed enclosures of unattended monitoring systems as well. The systems of the the platform in subject have been verified against the reference instruments commonly used the Safeguards accountancy and this paper shows the results of those comparisons.

11 Current Trends in Development of Radiation Detectors / 71

#11-71 Silicon Carbide Neutron Detectors for Harsh Nuclear Environments: A Review of the State of the Art

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Silicon carbide (SiC) semiconductor is an ideal material for solid-state nuclear radiation detectors to be used in high-temperature, high-radiation environments. Such harsh environments are typically encountered in nuclear reactor measurement locations as well as high-level radioactive waste and/or "hot" dismantling-decommissioning operations. In the present fleet of commercial nuclear reactors, temperatures in excess of 300 ℃ are often encountered, and temperatures up to 800 ℃ are anticipated in advanced reactor designs. The wide bandgap for SiC (3.27 eV) compared to more widely used semiconductors such as silicon (1.12 eV at room temperature) has allowed low-noise measurements to be carried out at temperatures up to 600 ℃. The concentration of thermally induced charge carriers in SiC at 600 ℃ is about four orders of magnitude less than that of silicon at room temperature.

Furthermore, SiC radiation detectors have been demonstrated to be much more resistant to the effects of radiation-induced damage than more conventional semiconductors such as silicon, germanium or cadmium zinc telluride (CZT), and have been demonstrated to be operational after extremely high gamma-ray, neutron and charged-particle doses.

Other factors that are advantageous for SiC include:

• high thermal conductivity (10-22 W/cm-K)

• a maximum breakdown field that is eight times that of silicon allowing higher biases to be applied resulting in higher drift velocities and more efficient charge collection

• a high saturated drift velocity (nearly twice that of silicon) leading to low charge trapping

The purpose of the present review is to provide an updated state of the art for SiC neutron detectors and to explore their applications in harsh high-temperature, high-radiation nuclear reactor applications. Specifically, the following will be reviewed:

• Designs of SiC thermal- and fast-neutron detectors

• SiC detector neutron-response measurements

• Radiation damage effects in SiC neutron detectors

• Applications of SiC neutron detectors in harsh nuclear measurement environments

Conclusions related to the current state-of-the-art of SiC neutron detectors will be presented, and specific ideal applications will be discussed.

02 Space Sciences and Technology / 73

#02-73 A test-bench for characterizing SDD-scintillator coupled detectors within the context of HERMES TP/SP nanosatellite mission.

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The main objective of the HERMES-Technologic and Scientific Pathfinder (HERMES-TP/SP) mission is to develop a cheap and scalable network of 3U nanosatellites to promptly detect, localize and probe high-energy astronomical transients such as Gamma-Ray Bursts and electromagnetic counterparts to gravitational waves events.

HERMES will be able to detect GRBs prompt emission over a broad energy band ranging from a few keV to MeV with high temporal resolution and localization accuracy. The key to achieve these ambitious goals is a simple yet innovative miniaturized detector design, in which Silicon Drift Detectors (SDD) play the double role of sensor for scintillation light and independent detector for low energy X-Ray.

In the present we illustrate the HERMES payload architecture and discuss the implementation and performances of a test-bench detector developed for characterizing GAGG:Ce scintillators - SDD coupling. This test-bench allow us to perform X and gamma-ray measurements using HERMES-like detector.

04 Research Reactors and Particle Accelerators / 75

#04-75 3D thermal and radiation-matter interaction simulations of a SiC solid-state detector for neutron flux measurements in JSI TRIGA Mark II research reactor

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In the nuclear reactor, it is crucial to measure key parameters such as neutron and photon fluxes or nuclear heating rate for a better understanding of the behavior of nuclear fuels or materials subjected to nuclear radiation. The coupling of different on-line measurement sensors is at the heart of these scientific objectives. The LIMMEX laboratory (Laboratory of Instrumentation and Measurement Methods in EXtreme media, a joint laboratory between Aix-Marseille University and the CEA) has been carrying out experimental and numerical works (innovation, design, characterization, calibration and optimization for laboratory and nuclear conditions) for accurate on-line measurements by means of differential calorimeters and semi-conductor detectors for many years. A new project, called SiC-CALO, aims to develop a multi-sensor device by coupling a Silicon Carbide (SiC) – solid-state based detector and a calorimeter for measuring simultaneously neutron fluxes and nuclear heating rates. Indeed, one objective is to know accurately the contribution of each radiation on nuclear heating rate thanks to the coupling of these two kinds of sensor and their combined interpretation. The finality of this project is to use this device in the core of the Jules Horowitz Reactor (JHR, under construction at the CEA Cadarache research center). The JHR core characteristics are, for a nominal power of 100 MWth, a high fast neutron flux (5.5 ×10^14 n·cm^-2·s^-1 for an energy > 1 MeV) leading to a high nuclear heating rate (up to 20 W·g^-1 in Aluminum). To be able to endure all of these harsh conditions, a SiC-based detector has been studied within the framework of a previous joint European project, called I_SMART (Innovative Sensor for Material Ageing and Radiation Testing). SiC material was chosen, among other wide band-gap semiconductors, thanks to its several advantages like its fast response (a few ns), its stability under radiations thanks to its wide bandgap energy, its low leakage current and its low thermal resistance and temperature gradient. The I_SMART project was intended to develop an innovative system made of SiC for radiation detection at very high temperature (> 500 ℃) and under high integral neutron flux (about 10^14 n.cm^-2.s^-1). The main result of this project was the development of SiC p+n junction diodes with implanted 10-B thermal neutron converter layer to measure thermal and fast neutrons fluxes. With regard to the short-term outlooks, the SiC-based detector performances can be improved through the using of a 6-Li converter for a better neutron/gamma discrimination, by decreasing the metal contact thickness or by coupling the detector with a suited current amplifier and a treatment process of pulses. The middle-term outlook is to adapt the SiC-diodes for fast neutrons measurements inside tokamaks (high temperature about 400 ℃ - 500 ℃ and extreme magnetic fields about 4 T). The last outlook is to extend the measurement range to values higher than those which was measured previously during the I_SMART project, i.e. 9.4 ×10^8 n·cm^-2·s^-1. Consequently, in 2021, an irradiation campaign at the Jožef Stefan Institute TRIGA Mark II research reactor in Slovenia is scheduled to measure higher neutron fluxes inside an in-core dry air triangular irradiation channel with a SiC detector and a commercial pCVD (polycrystalline Chemical Vapor Deposition) diamond detector (CIVIDEC Instrumentation). Main feature of this channel is its size, as it measures almost 6 cm in diameter. Because of the harsh conditions within this channel: a maximum integral neutron flux of about 1.2 ×10^13 n·cm^-2·s^-1, a nuclear heating rate lower than 0.1 W·g^-1 and a low heat transfer coefficient due to air natural convection, it is essential to predict the most precisely the behavior of the SiC detector for this environment.
Consequently, this paper will focus on 3D numerical studies dedicated to the definition of the most optimized device for the measurement campaign.
In a first part, a state of the art of the SiC detectors will be realized.
In a second part, 3D simulations dedicated to the interactions between nuclear radiations and matter will be performed by means of MCNP calculation code (Monte-Carlo N-Particle transport code). The goal of these simulations is to estimate the flux attenuation and the nuclear heating rate in each part

of the studied system for different axial positions by considering input source spectra, which represent the irradiation conditions of the triangular dry-air channel of the JSI TRIGA Mark II reactor core. The studied system will contain all of the components of the detector such as the housing, the duralumin support, the SiC diode putted on its alumina support, the connectors, the screws and the instrumentation cables.

In the last part, 3D thermal simulation results will be presented. A 3D thermal model built and solved with COMSOL Multiphysics finite element code will be detailed for the same system as that used for MCNP code. The thermal model will take into account heat sources estimated by MCNP simulations, conductive, convective and radiative heat transfers inside the system and natural convection boundary conditions by applying specific heat transfer coefficient values determined thanks to a correlation formula depending on the heat sources. The aim of these thermal simulations is to determine the temperature field inside the system and in particular in the SiC diode. A parametrical study will be given and discussed in order to define the optimized detector design.

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#07-76 ENTRANCE project on EfficieNT Risk-bAsed iNspection of freight Crossing bordErs

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As gatekeepers of EU borders, customs authorities have seen their mission to detect and seize maximum amounts of illicit goods on their way to enter the EU become increasingly challenging. Partly due to the growing number of customs declarations and limited customs staff members, this situation can further be explained by the expanding range of threats EU customs have had to face. Next to the "classic cross-border threats" such as drugs trafficking, cigarette smuggling, weapons trafficking, and duty and other border tax fraud, EU customs have indeed been required to deal with a new and complex risk landscape ranging from counterfeit products and new psychoactive substances to sensitive dual-use technologies, and nuclear and radioactive materials.
Addressing these new threats requires innovative and reliable technologies, combined with information sharing and collaboration mechanisms, to enhance border customs staff's capabilities and allow them to focus on detecting and inspecting high-risk shipments without preventing legitimate trade from circulating as quickly and freely as possible.
It is this challenge that the 3-year EU funded "EfficieNT Risk-bAsed iNspection of freight Crossing bordErs without disrupting business", i.e. the ENTRANCE project, launched on the 1st of October 2020, aims to address.
Concerned with developing, implementing and testing through field-trials a comprehensive user-based Toolbox for risk-based non-intrusive inspection of cross-border freight movements, the ENTRANCE project seeks to deliver five key outputs:
1. Automated Risk Assessment, Threat Recognition and Information Sharing Platform (ENARTIS)
2. Suite of Non-Intrusive Inspection (NII) technologies for detecting contraband hidden in high-density cargo
3. Enhanced relocatable unit for non-intrusive detection of wide number of threats including explosives, illicit drugs, chemical warfare agents, nuclear and radioactive materials and special nuclear materials such as enriched uranium and plutonium
4. Trans-European network of Radiation Portal Monitors (RPM) for passive detection of illicit nuclear and radioactive material combining detection facilities of different types and technologies
5. Novel high-speed RPM detection technology for passive detection of nuclear and radioactive with minimal disturbance of flow

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#04-77 Assessment of irradiation performance in the Jules Horowitz Reactor (JHR) using the CARMEN measuring device

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This paper deals with the assessment of irradiation conditions in various experimental positions in the Jules Horowitz Reactor core and reflector. For this purpose, the CARMEN device is foreseen to measure neutron and gamma flux and nuclear heating in the experimental channels of the JHR [1]. CARMEN is an axially mobile measuring system composed of two fission chambers sensitive to both fast and thermal neutrons, an ionisation chamber sensitive to gamma flux and an aluminium-based calorimeter for nuclear heating measurement. This allows for very detailed axial characterization of irradiation conditions in various radial experimental positions. Actually, the measurements performed with this device can serve as a preliminary assessment of "unperturbed irradiation conditions", before the introduction of the real experimental device at that location. The present study aims at reproducing CARMEN measurements by simulation and connecting them to the irradiation conditions expected in MICA and OCCITANE devices - which are respectively devoted to in-core and out-core experiments for the irradiation of steel materials.
The OCCITANE device provides spectral tailoring capabilities in order to be representative of the neutron spectrum in Reactor Pressure Vessels, and offers regulated temperature conditions by limiting nuclear heating, thanks to neutron and gamma screens. The present study evaluates the correlation between the measurements obtained by the CARMEN device, in unperturbed conditions, and specific irradiation characteristics occurring locally, inside the experimental loading of the device. OCCITANE capsule must be able to reproduce a subset of irradiation conditions identical to those of OSIRIS to pursue irradiation programs, in connection with already existing experimental data. As a preliminary step before examining JHR experimental positions, the present work first evaluates the transfer function between the measurements of nuclear heating in OSIRIS reactor - performed inside a water channel in the reflector, thanks to a graphite sample calorimeter [2] - and the real irradiation conditions finally obtained in the IRMA capsule.
Neutron and photon transport calculations were carried out in critical mode by means of Monte-Carlo simulations, using the TRIPOLI-4® code. The advanced modelling of nuclear heating in the calorimeter requires the use of four-particle-type Monte Carlo simulations (involving neutron, photon, electron and positon). Regarding nuclear heating in steel materials, sensitivity studies were conducted to discriminate the respective contribution of gammas originating from the core and gammas resulting from the radiative capture in iron.
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07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 78

#07-78 Characterization of a Special Nuclear Material fast identifier in respect to ANSI and IEC standards to be used for nuclear interdiction to detect the presence of threat items

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The increase of concerns in global nuclear security has led to the development of advanced systems for the measurement and identification of radioactive materials. Nowadays different radioisotope identification devices (RIIDs) are commercially available, aiming at deterring and combating the illicit trafficking of radioactive materials and their possible misuse in criminal acts, with catastrophic scenarios if used in public areas. For this reason, the most efficient and rapid instruments possible are highly required in homeland security applications in order to detect hidden radioactive materials and identify them, in case of alarm.
This paper describes the latest tests performed with Sniper-GN, a novel portable RIID. This instrument has been designed as a mobile backpack-based device for the detection and identification of both gamma and neutron sources, even multiple ones. Its singular feature is the capability to identify sources through the detection of neutrons, discriminating spontaneous fission sources (like 252Cf), gamma-n type sources (like AmBe and AmLi), and nuclear material containing plutonium or uranium, in different enrichment grade. The entire system is a rugged, battery-powered device, and it has been optimized to fit in a backpack, with a total weight smaller than 10 kg. It contains the detectors and all the electronics necessary for their operation and the consequent data processing.
The presence of only passive detection units considerably simplifies the system, if compared with the common methods of special nuclear materials investigation. Nevertheless, Sniper-GN has the capability of identifying neutron sources thanks to the novel specifically created algorithm, based on the simultaneous detection of fast neutrons and gamma rays coming from the target sources. Currently, this device is the only one in the market showing this unique feature. Several measurement campaigns have been carried out in different laboratories to test the device for the identification of nuclear material. In particular, plutonium and uranium sources as well as ^{252}Cf, Am-Be and Am-Li samples have been measured with 1 minute long identifications. Thanks to its novel algorithm, Sniper-GN has proved to outreach the requirements of international standards. In particular, plutonium samples have been identified up to 10 and 2.5 times the reference source-to-device distance, according to ANSI N42.34-2015, IEC62327:2017, and the more restrictive ANSI N42.43, respectively. Nuclear material has also been measured by applying different shielding materials (both lead shielding and neutron moderators) or gamma masking sources. With a novel patent-pending procedure, Sniper-GN is able to distinguish the different kinds of shielding scenarios surrounding the target source. It has succeeded when gamma spectroscopy has failed due to the presence of the shielding materials. For example, plutonium sources, shielded with 5 cm of lead and 5 cm of polyethylene, have been identified up to 5 times the reference distance for neutron sources (IEC62327:2017). Furthermore, Sniper-GN warns the user of the possible presence of gamma sources masking the nuclear material and provides prompt information regarding the enrichment grade of uranium or plutonium sources, with particular concern on weapon grade uranium and plutonium.

03 Fusion Diagnostics and Technology / 79

#03-79 Metrology of acquisition chains and signal processing of LMJ experiments

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Since the first experiment in 2014, more and more plasma diagnostics are being deployed on the Laser MégaJoule (LMJ) facility manufactured by CEA/DAM. These diagnostics aim at measuring radiations or particles emitted during laser experiments to study high-energy physics, especially inertial confinement fusion (ICF). Up to 176 laser beams, converging on a millimeter-sized target placed at the center of a 10-meter diameter chamber will be installed. Different types of sensors surround the LMJ target chamber (coaxial diodes for soft x-rays, photomultipliers and scintillators for neutrons, etc.…) and realize the conversion of the quantities of interest to an electric signal. The signal is then transmitted via coaxial cables, acquired by a broadband oscilloscope, and digitally post-processed. Each step of this typical acquisition chain adds measurement errors and increases the global uncertainty. In this paper, a study of the whole acquisition chain, including signal processing algorithms, will be presented. First, a numerical model of the digitizer alongside a specific hardware system designed to perform its metrology in situ will be presented. This system has been customized to be included inside an acquisition rack, and it allows quantifying several parameters before each laser experiment. It computes errors sources such as offset, gain and skew, and provides a measurement of the effective number of bits (ENOB) of the digitizer. This number gives insights concerning thermal noise, phase noise and non-linearity, and the ENOB is measured at different frequency values in the range of interest. This range can go up to a few GHz for the measurement of a hohlraum radiation temperature for example. The experimental characterization of the electrical chain via its transfer function measurement will also be detailed. Finally, the numerical methods deployed to handle the inverse problem, based on deconvolution processes, will be introduced, including future developments exploiting Bayesian inferences and statistical approaches.

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#04-80 Sensitivity Analysis of an Advanced Measurement Method for Thermal Neutrons Absorbers Detection in Irradiated Beryllium

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Operation experience in many Material Testing Reactors, where beryllium moderator or reflector is used, has shown that accumulation of neutron absorbers, so-called poisons, has a non-negligible impact on both core performance and safety aspects due to n-Be interactions, and needs to be addressed properly in calculations. As a result, depletion of beryllium, called also beryllium poisoning, was implemented in core follow-up deterministic calculations in several facilities. However, Material Testing Reactors are designed mainly for performing irradiation tests of material behaviour in severe conditions, resulting in frequent repositioning of the core elements in order to host experiments. This specificity makes almost impossible to follow the exact irradiation history of each beryllium elements and consequently inferring the exact accumulation of poisons. Thus, there is a strong need for developing a methodology enabling measurement and assessment of beryllium poisoning.
We addressed the problem of quantifying poisons accumulation using neutron transmission method. It is based on measuring transmission and absorption of thermal neutrons in beryllium elements. The measurement set is composed of a neutron source, a neutron moderator and a neutron detector, mounted on a bench, with beryllium sample placed between detector and moderator. In the study, we used the MARIA reactor's beryllium moderator blocks as material samples. Their characteristic truncate shape contributes to uneven axial neutron flux distribution in the core. To enable evaluation of axial distribution of poisons, both neutron source and detector move simultaneously along the block. A non-irradiated beryllium block has been measured and constitutes a reference point. We then performed series of irradiations in the NCBJ's MARIA reactor on several beryllium blocks delivered in the same batch. Comparing measurement results of irradiated blocks to the reference results, allows evaluating thermal neutrons absorbed in irradiated beryllium. Conducting such experiments requires taking into account several effects that might affect the measurement accuracy. In order to evaluate these effects, we performed a sensitivity analysis and all the above steps were simulated with the SERPENT2 Monte-Carlo code, using JEFF3.1.1 nuclear data library. We focus on analysis of expected uncertainties related chosen detector, impurities in beryllium, nuclear data uncertainty and gamma activation of irradiated beryllium.
In our study, we analysed the possibility of using following detectors: fission chamber with 235U deposit, 3He detector, 10B lined-detector. As no pure thermal neutron source exists, polyethylene-moderated neutron sources have been tested to be used for measurements, taking into account characteristics of the system. Neutron sources considered in this study include Am-Be, Pu-Be, Sb-Be, 252Cf. For each of these sources, we studied the optimal polyethylene thickness.
In this paper, we present the newly developed method for measurement of these neutron absorbers' concentration, we discuss calibration and sensitivity study of the above described measurement setup. The methodology and codes used for simulations are described. The results of the optimization study are detailed, and choice of specific type of detector and neutron sources are justified. Summarizing the results, we also give some recommendations.

11 Current Trends in Development of Radiation Detectors / 81

#11-81 Optimal Design of Scintillator Neutron Detector

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Neutron radiation detectors are essential in various fields, such as the nuclear power industry, nuclear medicine, scientific experimental systems and homeland security. However, the world is experiencing a shortage of 3He, which has been traditionally used as a key element for the detectors. LiF:ZnS(Ag) scintillator is an optional alternative for these detectors. This alternative main obstacle is the opaque nature of the scintillator. The opacity limits the detector width and hence its sensitivity.

This work presents an optimal design configuration for a sensitive neutron detector. This detector is based on LiF:ZnS(Ag) particles spread in a configuration that offers maximal sensitivity and approximation of the neutron energy. Typically the grain sizes in the mixture are about 2-10μm for the ZnS(Ag) and 1-4μm for the LiF. Our study presents that theoretically the sensitivity could be improved by selecting different sizes for the grains and without changing the mixture weight ratio between the two compounds.

This study was made by an advanced simulation tool. The optimization considerate the neutron capture sensitivity, maximal excitation energy transfer, the light transport and the moderator, see the graph.

We have investigated the received excitation energy for several mixtures with the same weight ratio but different grain sizes. Our simulations have shown the potential for vast sensitivity improvement when consideration is given to the mixture arrangement. We have improved both the probability for the produced alpha and Triton to transfer energy for the ZnS(Ag) excitation and also to increase the probability to have higher excitation energy spectra.

We estimate that the present results to improve current LiF:ZnS(Ag) based detectors and that this method has the potential to offer an available and affordable alternative for 3He based neutron detector and the feasibility of this configuration for approximate neutron spectroscopy.

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#04-82 Design of an acoustic sensor for fission gas release characterization devoted to JHR environment measurements.

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Among numerous research projects devoted to the improvement of the nuclear fuel behavior knowledge, the development of advanced instrumentation for in-pile experiments in Material Testing Reactor is of great interest. In the frame of JHR reactor, new requirements have arisen creating new constraints. This research is carried out within the framework of a long-standing partnership between CEA, CNRS and the University of Montpellier.

An acoustic method was tested with success during a first experiment called REMORA 3 in 2010 and 2011, and the results were used to differentiate helium and fission gas release kinetics under transient operating conditions. This experiment was lead at OSIRIS reactor (CEA Saclay, France). The maximal temperature during the irradiation test was about 150 ℃. In spite of the success of the experiment, it appeared necessary to optimize this type of probe especially the acoustic coupling. To overcome the problems encountered during the REMORA experience, we have developed thick film transducers produced by screen-printing process. They offered a wide range of possible application for the development of acoustic sensors and piezoelectric structure for harsh temperature environment measurements. We proposed a screen-printed modified Bismuth Titanate piezoelectric element on alumina substrate allowing acoustic measurements until high temperature.

JHR environment imposes a device maximal size and a working temperature of up to 350℃. This drives design choices.In this paper we will focus on the mechanical design of the new sensor. This acoustic sensor is composed of an acoustic element for generation and detection of acoustic waves propagating into a cavity filled with gaz. We will detail the choice of piezoelectric materials, the thickness of the different layers, the cavity shapes, the electrical connections, the means of assembly of the different parts. Theoretical and experimental results will be given. The evolution of the impedance response and the piezoelectric parameters of screen printed piezoelectric structures on alumina will be studied.

All that points will be discussed in term of acoustic sensor sensitivity versus dimensional constraints, in the case of a high temperature range working.

Keywords: Material Testing Reactor / Acoustic sensor / Gaz release / In-pile experimentation / High temperature

01 Fundamental Physics / 84

#01-84 Measurement of anomalies in angular correlation of electron and positron internally produced in excited 8Be and 4He

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Theoretical prediction for the distribution of the angle between electrons and positrons originating in internal pair creations is a monotonic featureless decrease with the opening angle. Recent studies on excited states of 8Be and 4He nuclei, made in ATOMKI, Hungary, however, revealed deviations from this expectation. If true, such a result may have a fundamental impact: the anomaly can be explained by introducing a new short-lived neutral boson that can still fit into known experimental and theoretical constraints. Although serious work has been done on the theoretical side, an independent laboratory has not yet verified these results yet, although related experiments are being prepared worldwide. We present the ongoing construction of a suitable time-projection-chamber-based (TPC) spectrometer for light charged particles, utilizing magnetic field as a means for energy measurement and also Multiwire Projection Chambers (MWPC) together with Timepix3 pixel detectors, for unprecedented spatial and angular resolution. The experimental effort will be operated at the Institute of Experimental and Applied Physics (IEAP) Van-de-Graaff accelerator facility in order to confirm or either refute the above-mentioned anomaly. Details of the detectors will be shown, together with simulations that provide information on the expected performance of this system.

11 Current Trends in Development of Radiation Detectors / 85

#11-85 The effect of the aging of liquid organic scintillators used for gamma-neutron separation

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Since the beginning of using liquid scintillators for gamma-neutron separation, there have been many articles dealing with long-term degradation especially due to oxygen presented during scintillator encapsulation. The effect of aging of liquid organic scintillators namely EJ 301, EJ309 (both Eljen Technology), and new house-made cocktails based on 1-Phenyl-3-(2,4,6-trimethylphenyl)-2-pyrazoline and 2,5-Bis(5-tert-butyl-benzoxazol-2-yl)thiophene fluors were investigated for more than half a year. The research was focused on the Compton edge shifting of gamma particles since the position is proportional to the light yield of the selected scintillator. Furthermore, the gamma-neutron separation was observed and quantified using FOM (Figure Of Merit) for samples prepared and filled under normal and nitrogen atmosphere during the mentioned period. All stated parameters of liquid scintillator NE 213 manufactured by Nuclear Enterprises Limited opened more than three decades ago were measured and used for comparison.

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#07-86 Design and Optimization of 4π Directional Radiation Detector based on Compton Effect

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Detection of gamma ray sources is a challenging task in many applications, where monitoring and mapping of radiological and nuclear materials is required. Obtaining directional information is required for nuclear homeland security (HLS) and safety, mapping for post-accident decontamination of nuclear incident or radiological event, etc. Many of directional radiation detectors are based on directional shielding, made of lead or tungsten collimators, which introduces two main drawbacks. The first is the size and weight, that makes those detectors too heavy and irrelevant for utilization in HLS handheld devices, drone mapping or space applications requiring cosmic gamma ray directional measurements. The second drawback is the small field of view (FOV), which requires multiple detectors to cover all the required FOV or machinery to rotate the limited FOV detector.

We propose a novel 4π directional detector based on Compton Effect interactions. Instead of using various shielding methods for directional information, we use the Compton scattering event, with two or more interactions within the detector. Based on the interactions locations of a single gamma ray event, we can back-project a probability cone of source locations. Using enough events, a collection of such cones converges into a single point source. One of the main advantages of this method is the ability to see more than just one source, just like a Compton Camera (CC). Whereas collimated detectors are erroneously calculated the mean location of a few sources, missing most of them. Additional advantage of the proposed detector is the 4π directional ability, which is missing in both the shielding collimated detectors and the CC devices. Moreover, shielded detectors undergo decrease in signal to noise (SNR) at high background radiation. Nevertheless, the proposed solution is not affected severely, since non relevant background events are identified as of incorrect direction events and rejected, without entering the calculation algorithms.

In order for the defector to be accurate, we need small voxels for accurate localization of an interaction location and a large distance between the interactions of a single event for high directional angle accuracy. We suggest an efficient geometrical solution that spreads the radiation detection voxels as far as possible in the detector volume, on the faces of the cubical directional detector. Doing so, we allow the Compton scattered gamma ray photon to travel from one side to the other side of the cube, obtaining maximal distance between the voxels. Further geometrical optimization is required for an optimal voxel size. Each voxel has an impact area and an interaction depth. Maintaining the total detector size, a small voxel area provides high interaction accuracy, but requires more amplification and sampling ADC channels which introduces complexity of the electronics. For high detection efficiency an increased voxel depth is required. On the other hand, this causes an interaction location uncertainty along the depth axis and Compton interaction probability to decrease, lowering the detection efficiency.

The 4π directional radiation detector provides significant advantages over the traditional directional detectors. It presents compact and lightweight structure without heavy collimators enabling the usage in drones and satellites; Introduces a radiation background rejection abilities; Detects gamma rays in a widest FOV of 4π with the ability to detect multiple radioactive sources, simultaneously, in a single measurement without mistakenly combining them into one. Such abilities enable utilization of such detectors for HLS and mapping of post-accident decontamination.

09 Environmental and Medical Sciences / 87

#09-87 ISOLPHARM_EIRA: a new approach to create high purity radionuclides for nuclear medicine applications

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Radiopharmaceuticals are drugs containing radionuclides and are routinely used in nuclear medicine for diagnosis or therapy of different diseases, mainly cancer. The main difficulties for nuclear medicine to assume a wider role in clinical practice are the availability of diagnostic/therapeutic isotopes and a technique for their specific localization in diseased sites.
The radionuclides of interest are currently produced in cyclotrons or nuclear reactors, with associated issues such as highly enriched target costs, low reaction cross-section and consequently low produced activity, production of undesired long-lived radioactive wastes and contaminants, long and expensive chemical separation routes.
In this context, the ISOLPHARM project at INFN-LNL (Istituto Nazionale di Fisica Nucleare-Laboratori Nazionali di Legnaro) has the aim of producing high purity (no-carrier added) radionuclides for nuclear medicine applications. By means of the Isotope Separation On-Line (ISOL) technique, both traditional and innovative radionuclides from many different regions of the nuclide chart will be produced with high specific activities, going beyond the state of art of the radiopharmaceuticals research.
The availability of such isotopes can potentially open the possibility of developing a new generation of radiopharmaceuticals, based on nuclides never studied so far, because of their production difficulties with traditional techniques. One of such nuclides is certainly Ag-111, that is regarded as a very promising radionuclide for therapy. Its decay properties make it, without any doubt, a very good candidate for internal radiotherapy. It is a β- emitter with medium half-life (7.45 d), convenient β-energy and medium tissue penetration (average β energy 360 keV and average tissue penetration 1.8 mm) and low percentage of associated γ-emission.
The ISOLPHARM_EIRA project have three main goals, based on the application of the ISOLPHARM method to the production of Ag-111 radionuclides as radiopharmaceuticals precursors: (i) test production of Ag-111 using standard techniques (thermal neutron irradiation of a sample of natural Pd) and its quality control, (ii) synthesis and characterization of chelators, linkers and targeting agents, (iii) biological characterization on cells, scaffold production and 3D cell cultures. The final goal of the project is the first in vitro and in vivo studies.
It this contribution I will present the first results of the Ag-111 production studies at the LENA reactor labs and its quality control system.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 88

#07-88 Neutron time-of-flight technique for nuclear material localization using plastic scintillators

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Reprocessing, nuclear fuel fabrication, or uranium enrichment require large facilities that contain many glove-boxes, tanks and pipes where uranium and/or plutonium can build up. During the cleaning and dismantling of these facilities, the quantity of fissile materials need to be evaluated for nuclear material accounting and control, criticality safety, radiation safety and waste management. Holdup measurements are usually performed using gamma-ray techniques, although neutron measurements with 3-He detectors are also used. During the measurement, many factors can influence the result, such as the location of the source, its distribution and isotopic composition.
Neutron time-of-flight event detection and fast coincidence counting are powerful techniques that can be used to discriminate neutron from gamma and establish the location of nuclear materials. MCNPX-PoliMi is used to model Pu-240 source in a glove-box and an array of plastic scintillators around the inspected object. A post-processing macro based on ROOT simulates the response of the acquisition system (energy threshold, energy resolution, time jitter, etc.). The neutron time-of-flight technique is performed to identify fission signatures. Afterwards, using time-of-flight distribution, the location is established and activity calculated.

08 Decommissioning, Dismantling and Remote Handling / 89

#08-89 Efficient System for Small Waste Containers Activity Estimation

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A new system design and a method to estimate and classify small waste containers is presented. A homogenous radioactive waste, placed in these small containers. Each container activity needs to be assessed in order to classify its activity level, e.g. Very Low-Level Waste etc. We propose a measurement system in the shape of a rectangular box made of several plates of scintillator detectors. The waste-container is placed inside the scintillators box to measure the emitted Gamma-photons. This detection configuration is the most efficient due to the full 4π coverage of the radioactive material with detectors. In this configuration the photons count rate is not affected by the 1/R2, since all photons emitted from the radioactive material and not absorbed by the container matrix reach the detectors. These photons are counted with high efficiency if the scintillator is thick enough relative to the emitted photons energies. The total count rate is predominantly affected by the matrix attenuation due to absorption and scattering.
To estimate the activity accurately, one needs to know the matrix attenuation and the spatial distribution of the radioactive material or 'hotspots' position. For known homogenous materials the attenuation can be estimated by weighting the container and calculating its density. Otherwise, we validate the homogeneity by measuring the attenuation along the three container axes using a collimated external source on one side and collimated detector on the other side.
When all the activity is emitted from a single concentrated source, in the center of the container, then the count rate of the detector is minimal due to the maximal attenuation. Any other source with the same activity in a different position or spatial distribution (such as several sources with the same total activity), will give a higher count rate. Thus, for a measured count rate and attenuation, if we assume that all photons were emitted from the center of the container, the estimate of the total container activity is the highest possible activity. In other words, the upper bound of the activity in the container is calculated by assuming a single source in the center of the container. Hence, a better estimation of the source spatial distribution will lower the upper bound of the activity estimation and will make it more accurate.
For a better spatial distribution estimation, we 'break' the box detector into many smaller detectors (pixels). For example, when each side is measured independently, it gives six detectors readings. If each face is divided into several pixels, then the fuller data will provide a better spatial estimation, and hence more accurate activity estimation. However, this leads to a more complex and expensive system. The purpose of this work is to find, using simulations, the relation between the number of pixels in the detection box, to the total activity estimation accuracy, for different attenuations (the medium type, density and energy emitted). The results will be used in the system prototype design in order to meet the requirement for the allowed activity estimation error.

04 Research Reactors and Particle Accelerators / 90

#04-90 New Neutron Imaging Facility development at the Penn State Breazeale Nuclear Reactor

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Neutron imaging is a powerful tool in the field of non-destructive testing that utilizes the unique attenuation properties of neutrons to image high-density objects. The Penn State Breazeale Reactor (PSBR) at the Radiation Science and Engineering Center (RSEC) has had a neutron radiography facility for the last several decades. With the installation of a new core moderator assembly and new beam ports, a dedicated neutron beam port has become available for a new neutron imaging facility (NIF) at RSEC. The new RSEC NIF will have collimators with variable apertures and will utilize state-of-the-art equipment and software for conventional neutron radiography and tomography. The centerpiece of the RSEC facilities is the PSBR. The PSBR, which first went critical in 1955, is the nation's longest continuously operating university research reactor. The PSBR is a 1 MW, TRIGA with moveable core in a large pool and with pulsing capabilities.

Although the final design for the RSEC-NIF system is under development, it has been decided to characterize the initial Open Beam (OB) configuration for the NIF beam port by the universal standards to access the current capabilities of the facility and to set the starting point of system development. Since all current Neutron Radiography (NR) facilities around the world characterize their system capabilities using American Society for Testing and Materials (ASTM) standards dedicated to neutron imaging techniques, the same approach was taken in the categorization process of OB. Thermal neutron flux measurements with bare and cadmium covered gold foils are performed at the beginning and the exit surface at the biological side of the OB port are measured to characterize the beam. Effective collimation ratio (L/D), Beam Purity Indicator (PBI) and Sensitivity Indicator were determined for the OB in order to have an initial indication of the beam. Based on these finding we are now finalizing the aperture, filter and collimator designs.

The RSEC-NIF with an open beam and without any apertures and collimators can produce images of medium quality, being Category IV facility by ASTM designation of quality and having the effective L/D ratio between 34.6 and 42.5. However, there is plenty of room for the development of the system in improving its resolution and uniformity by the means of lead/borated aluminum collimator steps (convergent and divergent parts), a Boral primary aperture with a cadmium lining, and filters for gammas (bismuth), and fast neutrons (sapphire). In addition to that, recent flux measurement via gold foil activation in multiple places across the exit surface of the beam resulted an average thermal flux equal to 3.5×10^8 n cm^(-2) s^(-1) at 1 MW power with the fluctuation of a 6%, which indicated that the main problem in uniformity of resulting images is most likely due to direct gammas. In order to tackle the existing problems, the initial MCNP simulations are being conducted with the addition of conceptual collimator within the beam aiming the final goal of obtaining L/D ratio of more than 100, thermal neutron content of more than 65, thermal neutron flux of more than 10^6 n cm^(-2) s^(-1), and with the neutron to gamma ratio of at least 10^5 n cm^(-2) mrem^(-1) at the imaging plate. These features would be sufficient to declare RSEC-NIF as Category I by the ASTM designation of quality.

11 Current Trends in Development of Radiation Detectors / 92

#11-92 Why the standard theory of scintillation spectrometers is needed?

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The main drawback of all current theories of scintillation spectrometers is in introducing various terms into the formula for the energy resolution of scintillation spectrometers, without giving specific formulae for the relationship of these terms with characteristics of scintillation detectors. Such insertion of various contributions by "hands" is not only wrong but also counterproductive, since it does not allow comparing the results obtained by different scientific groups. In this work, the microscopic mathematical model was formulated, which serves as the basis for the standard theory of scintillation spectrometers. The standard theory allows obtaining the formulae for arbitrary moments of the signal distribution function at the output of the scintillation spectrometer. In particular, the formulae for the average value and the variance of the signal at the output of the photodetector are obtained. The structure of the formula for the energy resolution of a scintillation spectrometer reveals the contributions of the processes that take place at converting the energy of a primary particle into the output signal, particularly the contribution associated with the nonlinearity of the scintillator light output. It was shown that in the developed standard theory of scintillation spectrometers there are no drawbacks of the currently existing theories of scintillation spectrometers. Thus, the developed standard theory of scintillation spectrometers creates a solid basis for linking theoretical and experimental researches in this field.

03 Fusion Diagnostics and Technology / 94

#03-94 Designed A Real-time Full-Range Digitizer for Neutron Flux Monitor On EAST

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The neutron flux monitor (NFM) is a standard diagnostic for fusion neutron yield measurement on experimental advanced superconduct tokamak (EAST), and the neutron yield is a most important parameter to research of plasma auxiliary heating. After the upgrade of EAST, higher requirements for neutron flux measurements are put forward. Based on fast ADC (14bit, 500MSPS) and a high-performance FPGA, a four-channel real-time full-range digitizer for NFM was designed. In order to meet the requirement of fusion neutron flux wide-range measurement during high-parameter operation, the advantages of pulse counting modes, Campbell modes and current modes were combined to expand the dynamic range, so that the measuring range could reach 8 orders of magnitude. In addition to reduce the interference of gamma ray backgrounds on neutron flux measurements, a pulse signal processing technology was used to real-time distinguish between neutrons and gammas. Furthermore, to meet the requirements of mass data storage and high-speed transmission in the case of high neutron flux, a DDR3-based controller and PXIe bus technology-based DMA mode were separately designed. Moreover, a series of test were performed on laboratory and EAST device. Those test results have reached the expected design indicators, and the time resolution could reach 1ms, which proved the feasibility of NFM for EAST neutron flux measurements.

04 Research Reactors and Particle Accelerators / 97

#04-97 CABRI test events monitoring through three measurement systems

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The CABRI experimental pulse reactor, located at the Cadarache nuclear research center, southern France, is devoted to the study of Reactivity Initiated Accidents (RIA). When certain conditions (mostly temperature, pressure and flowrate conditions) are obtained, a power excursion, triggered by the 3He reactivity injection system, is performed on a nuclear fuel test rod in order to simulate a control rod ejection accident. The aim of the experiment, designed by the IRSN, is to observe both the fuel and cladding behavior of the test rod placed into the center of the reactor during the power excursion.

Several test rods pre-irradiated in pressurized nuclear power plants and with different characteristics (burn up, cladding material, fuel type, Zirconia thickness) are considered for the programs performed in the CABRI reactor.

Physical phenomena occurring during the power transient are monitored by various measuring systems designed or operated by the IRSN. Each system provides various information in the different phases of the experiment. Three main measuring systems will be considered in this paper.

The first one is the CABRI test device. It holds the test rod and is also equipped with a variety of sensors that allow to record several thermal and hydraulic parameters of the system (such as temperature, pressure and flowrate) in the channel where the test rod is placed, as well as some other more exotic sensors which can measure more complex phenomena (like microphones, transient flowmeters, etc…).

The hodoscope, the second main measuring system coupled with the CABRI reactor, is a unique online fuel motion monitoring system, dedicated to the measurement of the fast neutrons emitted by the tested rod during the power pulse. This system is used to observe the degradation of the fissile column, and is able to measure fuel motion, potentially linked to clad failure, if it arises during the power excursion.

The third equipment is the IRIS (Installation for radiography, imaging and spectrometry) facility dedicated to pre and post non-destructive examinations of the rodlet. This measuring system is equipped by two main parts:

• A gamma spectrometry system which is used to measure gamma rays emitted by the fission products in the test rod.

• An X-ray facility which aims at performing x-ray radiographies and tomography images.

During the experimental sequence, different events are recorded. This paper focuses on the cladding failure that occurred during the TOP (Transient Over Power) and that can be determined by the three above mentioned systems signals. The test device instrumentation allows to determine the timing of the failure and the investigation of its consequences in the vicinity of the test rod from a thermal and hydraulic point of view, while the hodoscope measures fuel elongation and relocation during the power excursion. The IRIS facility, then, helps to confirm the failure, its location and its extent. These three systems are complementary and they allow the analysis of the same event from different perspectives. Their combination will ease the interpretation of the events in the next steps of the test results analysis. The study case taken into account in this article concerns nuclear fuel after three irradiation cycles in a commercial PWR.

01 Fundamental Physics / 98

#01-98 The Data Acquisition System for the ATLAS Phase-II Tile Calorimeter Demonstrator

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he Tile Calorimeter (TileCal) is the central hadronic calorimeter of the ATLAS detector at the Large Hadron Collider (LHC). This subdetector is a sampling calorimeter composed of steel plates as absorber material and plastic scintillators as active material. The complete readout of the detector is done using approximately 10,000 photomultipliers (PMTs).
The LHC will undergo a series of upgrades in 2025 leading to the High-Luminosity LHC (HL-LHC). The HL-LHC will provide an instantaneous luminosity 5 to 7 times larger than the nominal LHC design value. The ATLAS Tile Calorimeter Phase-II Upgrade (2025-2027) will completely replace the readout electronics with a new clock distribution and readout architecture with a full-digital trigger system.
In the upgraded data acquisition architecture, the on-detector readout electronics will transmit digitized signals from the PMTs for every bunch crossing (~25 ns) to the Tile PreProcessor (TilePPr) boards located in the counting rooms. The TilePPrs will store the detector data in pipeline memories until the reception of a trigger acceptance signal activating data transmission to the ATLAS Front End LInk eXchange (FELIX) system. The new readout system will require 32 TilePPr modules to read out the entire detector for a total bandwidth of 40 Tbps. In parallel, the Trigger and Data Acquisition Interface boards will receive reconstructed cell energies from the TilePPrs and will transmit pre-processed trigger objects to the first level trigger with improved precision and granularity.
As part of the Phase-II Upgrade Demonstrator program, a full-size Demonstrator module containing all the upgraded readout electronics was installed into the ATLAS experiment in 2019 with the aim of validating the new clock and readout strategy for the Phase-II Upgrade.
The Demonstrator module consists of four independent mini-drawers capable to operate up to 12 PMT blocks. A mini-drawer is composed of a mechanical aluminum structure that supports one Mainboard, one Daughterboard, one high voltage regulation board, and up to 12 PMT blocks equipped with 3-in-1 cards. The PMT signals are shaped and amplified in two gains by the 3-in-1 cards and digitized in the Mainboard by 12-bit dual ADCs. The Daughterboards transfer the digitized data to the off-detector electronics through high-speed optical links. In addition, the lower gains of the 3-in-1 cards are summed in towers and transmitted to the Level-1 Calorimeter trigger system for trigger decision.
In the off-detector electronics, the Tile PreProcessor prototype receives the detector data for every bunch crossing through optical links with fixed and deterministic latency which provide a total data bandwidth of 160 Gbps. The main FPGA of the PreProcessor prototype processes and buffers the detector data in pipeline memories capable of storing up to 10 μs of samples. Upon the reception of a trigger acceptance signal, the selected events are extracted from the pipeline memories, packed, and transmitted to the legacy Read-Out Drivers (RODs) keeping backward compatibility with the ATLAS DAQ system, and to the FELIX system.
The PreProcessor prototype also distributes the LHC bunch-crossing clock embedded with configuration commands to the on-detector electronics for synchronization with the accelerator. In the current Demonstrator, the Preprocessor prototype interfaces with the legacy Timing, Trigger and Control (TTC) system to receive the LHC clock and configuration commands. The recovered clock is then cleaned with dedicated jitter cleaner chips before driving it to the PreProcessor FPGA transceivers to ensure stable high-speed communication with the Daughterboards.
The Demonstrator module is fully integrated with the ATLAS TDAQ software and Detector Control System through the PreProcessor prototype, which translates the legacy commands into Phase-II commands, and transmits the triggered data to the RODs using the G-Link protocol. Therefore, the PreProcessor prototype enables the operation Demonstrator module for taking calibration and physics runs using the current ATLAS software tools. From the point of view of the ATLAS TDAQ system, the Demonstrator module behaves as a legacy TileCal module.
The Demonstrator module has been operated together with the rest of the TileCal modules since its installation in June 2019. The performance of the upgrade electronics has been studied with Charge

Injection, Laser, and Cosmic runs, showing excellent performance in terms of low noise, signal quality, and timing. One of the goals of the Demonstrator program is to keep the Demonstrator module in ATLAS during Run-3 (2021-2024) to continue studying its performance in real conditions.
This contribution describes in detail all the hardware, firmware, and software components of the clock distribution and data acquisition system for the Demonstrator module, focusing on the Pre-Processor prototype developments, as well as the results of the Demonstrator module operation in the ATLAS experiment and future plans.

11 Current Trends in Development of Radiation Detectors / 100

#11-100 Measurement and simulation of the new liquid organic scintillator response to fast neutrons

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Liquid organic scintillators are important devices for measurements of neutron radiation. This work aims to develop and optimize the composition of liquid organic scintillators so it can be used for fast neutron spectrometry. As the neutron radiation is usually accompanied with γ ray radiation, it is important to have quality γ/n discrimination. The new cocktail for house made liquid organic scintillator is prepared and studied with intention of being able to separate gamma and neutron for neutron energies above 0.5 MeV while keeping lower constraints on practical use (e.g. sealing because of oxygen) than commercial liquid scintillators. In preceding work the composition of liquid scintillators was optimized. Two two-component scintillators were selected for further studies. Solvent DIPN (Di – iso – propyl - naphthalene Mixed Isomers) is selected for both. First is mixed with uminophore PYR (1 – Phenyl -3 -(2,4,6 – trimethyl – phenyl) -2 – pyrazoline) of concentration 5 g/l. Second is mixed with luminophore THIO (2,5 – Bis (5 – tert – butyl – benzoxazol – 2 – yl) thiophene) of concentration 5 g/l.

In this work the response of scintillator to monoenergetic beam of neutrons was measured for multiple neutron energies at PTB in Braunschweig. The two parameter spectrometric system NGA-01 is used to analyze the energy and discrimination characteristics. 137 Cs and 60 Co are used as radiation sources for calibration with pure γ rays. Then the response of scintillator for same neutron energies was simulated using GEANT4. The dissipated energy in the scintillator in response to monoenergetic neutrons is obtained. Both, measured and simulated responses are compared. Functional dependence for yield of recoiled products is estimated. It is seen that main recoil product hydrogen proton is well observed in both. From the edge of proton response one can assume the yield for given neutron energy. The recoiled carbon ion (from elastic collision) is on the other side difficult to observe in measured results but clearly seen in dissipated energy plots. It suggests that yield of carbon ion is very small relatively to proton yield. These results will serve as basis for response function evaluation of scintillator which is necessary for evaluation of unknown neutron spectra from measurements with scintillator.

11 Current Trends in Development of Radiation Detectors / 101

#11-101 Progress in MEMS-based silicon radiation detectors at FBK

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In the past few years, there has been an increasing interest toward micromachined silicon radiation detectors, which use fabrication techniques normally adopted for Micro-Electro-Mechanical Systems (MEMS). In particular, Deep Reactive Ion Etching (DRIE) by the Bosch process allows to obtain vertical electrodes in a variety of shapes and dimensions with high aspect ratio (depth to surface size), making it possible to exploit the 3rd dimension within the silicon substrate, and offering several interesting advantages over more traditional planar detectors.
The Microtechnologies Laboratory of Fondazione Bruno Kessler (FBK) in Trento, Italy, is one of the few processing facilities worldwide able to manage the fabrication challenges of micromachined silicon detectors. R&D activities in this field have started in 2004, mainly in collaboration with the University of Trento and INFN, and significant results have so far been obtained.
The most famous example are 3D pixel detectors, which represent the most radiation-hard solution for charged particle tracking in High Energy Physics (HEP) experiments. After contributing to the production of 3D pixels for the ATLAS Insertable B-Layer, the first application of these devices in a HEP experiment, we have recently developed a new generation of these devices, featuring very small-pitch (25 um x 100 um and 50 um x 50 um) and reduced active thickness (~150 um), able to cope with the severe operational challenges of the innermost tracking layers of the ATLAS and CMS detectors at the High-Luminosity LHC. Besides the traditional version using columnar electrodes, we have also recently fabricated 3D pixel variants using trenched electrodes, which allow for a more uniform electric field and weighting field distribution within the active volume, thus boosting the timing performance. This is a key feature to obtain a high 4D resolution (tracking + timing) as requested by future HEP experiments. We have also developed a variety of planar active-edge sensors oriented to the ATLAS and CMS tracker upgrades at HL-LHC and to X-ray imaging at Free Electron Laser facilities. In these device, deep trenches (either continuous or segmented) are used as field terminations at the sensor periphery, minimizing the edge extension and allowing for full signal sensitivity up to a few microns from the sensor physical edge.
Another example is that of 3D micro-structured silicon sensors for thermal neutron detection, where narrow and deep cavities are etched by DRIE and filled by proper converting materials (e.g., 6LiF, 10B), leading to an increased probability for neutron reaction products to reach the active volume, hence higher detection efficiency with respect to standard planar sensors. We have fabricated both 3D diodes and 3D pixelated sensors compatible with the TIMEPIX readout chip. The first prototypes have been assembled at ADVACAM and are currently under test in collaboration with the Czech Technical University in Prague.

In this contribution, we will address the main design and technological issues relevant to MEMS-based silicon detectors, and report selected results relevant to the above-mentioned case studies.

04 Research Reactors and Particle Accelerators / 102

#04-102 Measurement of prompt gamma field above the VR-1 water level

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The good knowledge of gamma spectrum in vicinity of nuclear reactor core is essential in characterization of radiation field. It is important for ensuring of radiation safety, but well described gamma field can be used also for testing of radiation measuring devices. The field in vicinity of reactor core is interesting for testing because in contrary to commonly use gamma fields formed by common gamma sources, in the reactor fields there are high energy gammas formed by neutron interactions with reactor structural components. Due to this mechanism, the gammas in reactor field are mostly accompanied by presence of neutrons. An interesting situation may occur in deep penetration issues in water, because deep water slab is an excellent fission neutron absorber but week gamma absorber. Behind deep water slab one can expect high energy capture gamma field with negligible share of neutrons. This criterion is well filled above water surface of the VR-1 Czech Technical University research reactor. The water thickness above fuel is more than 300cm. The gamma fluxes were measured 88.6 cm above the water surface, and about 450cm from the center of the reactor core. The measurement was performed using NGA-01 spectrometric system and stilbene scintillation detector of 45 x 45 mm sensitive for both neutrons and gamma radiation. The NGA-01 spectrometric system was used for measurement and data evaluation. This system features two-parameter data processing from scintillation detectors in mixed fields of neutron and gamma radiation. The system works with high-speed ADC converters with 500 MS / s (alternatively 1 GS / s) and a resolution of 12 bits. FPGA (Field Programmable Gate Area) with advanced digital filters and PSD algorithms ensures lossless data processing. The measured gamma spectrum was compared with calculated one. The computational simulation of the prompt gamma field above the VR-1 water level was performed. This deep penetration problem was solved as fixed source gamma model, where the source gamma spectrum and gamma emission density was calculated in critical model. Generally, the calculated spectra are in satisfactory agreement with experiment.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 103

#07-103 Performance assessment of amplification and discrimination electronic devices for active neutron measurements

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The knowledge of the fissile material mass is a key challenge to enhance radioactive waste management and to ensure a high level of safety in nuclear industry. The assessment of plutonium fissile mass by passive coincidence measurements is usually obtained by detecting the neutrons generated by the spontaneous fissions of Pu isotopes. The detection and the quantification of a fissile mass in radioactive waste can also be carried out using active neutron interrogation with a pulsed D-T neutron generator. The 14 MeV neutrons are moderated to induce fissions in the uranium and plutonium contaminated waste. The emerging neutron signal can be analyzed by taking into account their detection time after the generator pulse. Data is analyzed according to the principles of the neutron measurement techniques. Currently, the ACH-NA98 charge amplifier from Mirion Technologies (France) are used as reference electronics in industrial facilities specially dedicated for the reprocessed fuel assembly (ORANO/La Hague, France) and for the radioactive waste characterization(CEA/Cadarache). However, it is primordial to evaluate the performance of the other commercially available electronics in order to maintain and/or improve the performances of the several neutron measurement set-up presently implanted. This paper describes the performance assessment for active neutron measurement of different commercially available electronics from Mirion Technologies (ACH-NA98), Precision Data Technology (PDT10A and PDT10M+PDT12S modules), Mesytec (MRS2000-1 and MRS2000-2 amplifiers), as well as MONACO electronics originally developed by CEA LIST for fission chamber measurements in experimental reactors. The experimental campaign has been carried out in the DANAIDES facility located in the Cadarache Research Center of CEA (French National nuclear energy Commission). The experimental tests consist in comparative active neutron measurements for different neutron emission rates of the SODERN D-T neutron generator. These measurements for neutron emission rates up to 2.3×10^8 n/s give quantitative information on the behaviour of the amplification electronics such as the count loss according to the time.

11 Current Trends in Development of Radiation Detectors / 104

#11-104 Development of a position-sensitive fast scintillator for gamma-ray imaging application

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We have characterized a Cerium doped Lanthanum Bromide (LaBr3(Ce)) crystal coupled with the position-sensitive photo-multiplier system for the γ-ray imaging application. One can use this detector set-up for the scanning of high purity germanium detectors for pulse shape analysis in γ-ray spectroscopy experiments and the image formation of an object by Compton back-scattering [1, 2]. The sensor has been tested for energy, timing and position information of the gamma-rays interacting within the detector crystal. The GEANT4 simulation results are consistent with the experimental results. We have reconstructed the image of irradiation spots in different positions throughout the detector crystal. Position resolution is found to be around 3.5 mm with the 1.5 mm collimated gamma-rays. The 2-d image of hexagonal Bismuth Germanate (BGO) crystal and a cylindrical LaBr3(Ce) crystal have been reconstructed in coincidence technique. The performance of the detector for imaging application has been investigated by coincidence technique in GEANT4 simulation and compared with the experimental data. We have reconstructed the 2-d images of objects with various geometrical shapes by Compton back-scattering of the gamma-rays. We have simulated a Compton camera for the image reconstruction of an extended radioactive source where we have used the LaBr3(Ce)-PSPMT detector as an absorber of the camera. One can also use this kind of set-up in radiation imaging and many other applications where the energy and source position of the γ-ray is the main interest.

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11 Current Trends in Development of Radiation Detectors / 105

#11-105 Optimization of liquid organic scintillator composition for fast neutron spectrometry

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Liquid organic scintillators are important devices for measurements of neutron radiation. This work aims to develop and optimize the composition of liquid organic scintillators so it can be used for fast neutron spectrometry. As the neutron radiation is usually accompanied with γ ray radiation, this work is focused on γ/n discrimination. In our experiments, the 252 Cf is used as a radiation source for a mixed field of γ rays and neutrons, and 137 Cs and 60 Co are used as radiation sources for pure γ rays. The scintillators within 20 ml glass vials are placed above the photomultiplier (PM) RCA 8575. The two parameter spectrometric system NGA-01 is used to analyze the energy and discrimination characteristics. It is shown first that from six selected solvents, the DIPN (Di – iso – propyl - naphthalene Mixed Isomers) and 1-Methylnaphthalene are capable of good γ/n discrimination in combination with luminophore PYR (1 – Phenyl -3 -(2,4,6 – trimethyl – phenyl) -2 - pyrazoline). In second stage the solvent is DIPN and the concentration of luminophores is varied. The concentration of luminophore PYR from 3 g/l to 9 g/l is shown to be best for γ/n discrimination. The energy threshold for good γ/n discrimination is 0.14 MeVee. Luminophore THIO (2,5 – Bis (5 – tert – butyl – benzoxazol – 2 – yl) thiophene) has best γ/n discrimination at concentrations from 3 g/l to 6 g/l and the energy threshold is 0.17 MeVee. Luminophore XAZ (2 - (4 – BiPhenylyl) – 6 - phenylbenzoxazole) has best γ/n discrimination at concentrations from 2.4 g/l to 3 g/l and the energy threshold is 0.20 MeVee. These scintillators at optimal concentrations have lower energy threshold for γ/n discrimination than reference liquid scintillator AquaLight AB. However the stilbene crystal scintillator is better than scintillators from this work. It was shown also that the additives of water and detergent, components used to improve the solubility, deteriorate the γ/n discrimination of the scintillator.

01 Fundamental Physics / 106

#01-106 Calculation of spatial response of a collimated segmented HPGe detector for gamma emission tomography by MCNP simulations

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Segmented HPGe (High Purity Germanium) detectors are commonplace in nuclear structure studies, where position and amount of energy deposited by the photon's interactions inside the detector are fully reconstructed at every event using the pulse shape analysis methods. Recently a similar segmented HPGe detector concept in combination with a multi-slit collimator was proposed for gamma emission tomography. The feasibility of the proposed concept for use in gamma emission tomography and its performance was evaluated by a simulation study. A segmented HPGe can offer high energy resolution and high spatial resolution, and its ability to simultaneously interrogate the fuel object with multiple detection elements can also make it suitable for faster operation requirements in the post-irradiation examinations (PIE). The use of such segmented HPGe detectors can facilitate imaging of nuclear fuel rod internal features like fission product migration, pellet cladding interactions, rod bow and swelling, fuel fragmentation, relocation and dispersal in transient tests etc., as well as for analysis of miniaturized irradiation tested fuel samples.

It is important to quantify the spatial resolution achievable by using the collimated segmented HPGe before the actual detector is manufactured and operated. The spatial response for a segmented HPGe detector concept was evaluated using simulation with Monte Carlo N-particle transport code MCNP6. Full detector and multi-slit collimator system was modelled and for the quantification of the spatial response, the Modulation Transfer Function (MTF)/Contrast Transfer Function (CTF) was chosen as the performance metric. These performance metrics were obtained through the calculation of the Line Spread Function (LSF) and Edge Spread Function (ESF) by analysing the simulation data. In addition, the proportionality between the activity inside the object and the as-measured profile with the detector collimator system was also examined through simulations of a ramp pattern/stair-step pattern.
The results from the simulation study will be presented at the ANIMMA-2021 conference.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 107

#07-107 Toward UO2 micro/macro machining: A Laser processing approach

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Linked to experimental data acquisition and to development of improved models, a better detailed description of the behaviour of the nuclear ceramics as regards the fission gases release during thermal transients representative of accidental conditions such as RIA (Reactivity Initiated Accident) or LOCA (LOss of Coolant Accident) requires access to local information within the fuel pellet, and not only averaged over the whole pellet. One of the major challenges in this context is the sample size, which depends on the main objective of the study, typically from the order of a few hundred microns to millimeters. Few techniques allow this scale dynamic while being compatible with irradiated fuel constraints. Laser micromachining is a high precision non-contact material removal process that would be adapted to this dynamic.

During a laser micromachining, the interaction between the laser beam (in our case infrared) with the material involves several thermal processes: absorption of the deposed energy, heating, thermal diffusion, fusion, vaporization. Depending on the experimental parameters, we can achieve material removal (ablation) with precise cutting edges. The lack of contact between the tool and the part allows micromachining of fragile samples, such as ceramics. However, the control of the Heat Affected Zone (HAZ) is generally critical and requires optimization of the operating parameters and a perfect control of the laser-material interactions, as well as the other induced effects (plasma generation, delamination, micro cracks, and denaturation of the target).

The present paper presents experimental and numerical studies, carried out in order to evaluate the possibility to apply this process for the preparation of irradiated UO2 samples of various dimensions. First, preliminary works conducted on materials which have comparable properties (in particular their behaviour under laser irradiation and their high melting point) in order to validate the feasibility of the process will be detailed. Afterwards, optimization of the global process will be presented. This has been done:

• by predicting the amount of material ablated for each laser pulse in order to perfectly control the micromachining,
• by controlling the dynamics of the laser spot displacement on the target, through a galvanometric head,
• and by multiplying the laser passes in order to obtain the desired depth.

Finally, numerical simulations performed by a finite element method to strengthen the experimental results in order to transfer the technique to non-irradiated UO2, and then to the irradiated material will be highlighted.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 108

#07-108 UO2 thermal diffusivity measurement with laser techniques

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The knowledge of the thermal conductivity of nuclear fuel and its evolution as a function of temperature and burn up is a major challenge in the context of the evaluation and understanding of irradiated fuel performances in current reactors. It is also the case for the development and qualification of fuel for future reactors. Indeed, numerical simulations of the fuel behaviour under various conditions require the accurate knowledge of thermal conductivity over a wide range of temperature (from 293 K to melting point temperature) but also at the scale of few tens of micrometers to take into account the microstructural effects on the thermomechanical evolution of the fuel in normal or incidental irradiation conditions. For example, the decrease of thermal conductivity involves an increase of the fuel pellet temperature, which can generate thermal expansions as well as cracking of the fuel due to internal stresses resulting from the radial thermal gradient. Thermochemical equilibria within the pellet, and hence fission product release, may also be affected.

The laboratory reference method can deduce the thermal conductivity from a thermal diffusivity measurement. One of the most common and non-intrusive method is the laser flash method. This technique is based on a laser-flash excitation and infrared thermography non-contact measurement of the thermal response of a studied sample. The sample can be heated to different temperatures (for instance with lasers) to determine the dependence of thermal conductivity on temperature. The thermal diffusivity measurements obtained with this technique present typically millimeter spatial resolution. Another method is the photoreflectance microscopy, based on the measurement and analysis of the periodic temperature increase induced by the absorption of an intensity modulated laser beam ("pump" beam). By detecting the thermally induced reflection coefficient variations with the help of a secondary continuous laser beam ("probe" beam), the temperature increase at the sample surface is measured as a function of time. Unlike laser flash method, this technique has a micrometric spatial resolution. A laser method with intermediate spatial resolution (few tens of microns) is the Infrared Microscopy technique, based on the detection, by means of infrared thermography, of the surface sample temperature rise distribution induced by the absorption of an intensity-modulated focused laser beam. All of these approaches, using laser, allow reaching thermal diffusivity measurements for temperatures up to at least 3000 K.

The potential of each of these techniques in the context of the determination of the fuel thermal conductivity will be discussed. Furthermore, the experimental bench being developed at CEA Cadarache, implementing laser techniques to obtain thermal conductivity, from thermal diffusivity measurements, at different scales of interest and for different range of temperature will be presented. The aims of the development of this new experimental set-up in a "test" glove box and the associated methodology are to assess the potential of these techniques by applying them first on non-irradiated UO2 or model materials before its transposition in a hot cell for measurements on irradiated fuels.

This work is also supported by EDF and Framatome in the frame of Tripartite Institute (CEA/EDF/FRA).

04 Research Reactors and Particle Accelerators / 109

#04-109 Development of a wideband current amplifier dedicated to Fission Chamber measurement

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Fission chambers are widely used in nuclear reactors, either to occasionally probe the neutron flux maps in nuclear power plants or as part of the nuclear instrumentation used in research reactors to characterize neutron flux levels in irradiation locations. In the frame of the Jules Horowitz Reactor (JHR) under construction at the CEA CADARACHE site, the Instrumentation Sensors and Dosimetry Lab (LDCI) is designing and manufacturing instrumentation (electronics and detectors) for the monitoring of experiments as well as for the reactor start-up protocol. For this purpose, instrumentation capable of high dynamics is required.

Wide dynamic range acquisition systems are very limited on the nuclear instrumentation market. Thus, since 2012, the LDCI is developing the MONACO system, which allows a measurement in pulse, fluctuation and current mode running simultaneously on the same channel. The latest prototype of this measurement system has reached TRL7 [1-2] and is now being industrialised. [1-2] considered now at a validated TRL7 level.

The simultaneous use of three measurement modes places high requirements on the amplification stage. Indeed, the current mode exploits a bandwidth between 0 and a few kHz, while the fluctuation mode is based on the frequency range 100 kHz - 1MHz. Finally, for the pulse mode, it is necessary to identify current pulses of a few uA in amplitude and with a duration of about ten nanoseconds, which implies a bandwidth that extends up to 10 or 20 MHz.

LDCI is then pursuing the industrialization process of the MONACO system in collaboration with the INSTRUMENTATION TECHNOLOGIES (I-TECH) company. In this paper, the first step, development an industrialized current preamplifier, is presented. The main goal of this redesign is to to improve the measurement of the DC current and extend its bandwidth. CEA miniature fission chambers require a wideband current transimpedance preamplifier to retrieve the current signals from detectors and convert them into a voltage signal compatible with a digitizer. The I-TECH designed preamplifier contains two modules, the first one corresponds to a trans-impedance amplifier built with operational amplifiers, which allows to measure AC current in the frequency band 10 kHz - 50 MHz. It is suitable both for negative and positive pulses, specific solutions and shieldings were implemented to reduce the parasitic cable capacitance as well as to increase its immunity to electromagnetic noise: The output noise level is ±25 mVpp The second module measures the DC part of the current by using a shunt and an isolated circuit operating at the potential of the fission chamber.

LDCI and I-TECH performed very promising bench tests in October 2020. Tests with synthetic signals corresponding to pulse and fluctuation mode showed very good performance of the electronics. The positive and negative current pulses are correctly amplified and the measurement noise was estimated at 0.1 uA pp thanks to a proper electronic shielding. The DC current measurement is also satisfactory in the tested range with an accuracy of less than nA and a bandwidth of 1kHz. Preamplifier qualifications are planned within realistic experiments using a fission chamber placed in a neutron field.

This preamplifier will be integrated as frontend in a complete new industrialized version of the MONACO system developed jointly by CEA and I-TECH (foreseen in 2022).

03 Fusion Diagnostics and Technology / 110

#03-110 Development of a real-time signal processing unit for diamond detectors of ITER Vertical Neutron Camera

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Diamond detectors are going to be used in ITER neutron diagnostics, including the Vertical Neutron Camera (VNC). They are meant for neutron flux measurements and measurements of the energy spectrum of fast neutrons.
One of the main functions of VNC is measurement of the neutron source profile. In accordance with ITER requirements VNC has to calculate an updated neutron source profile each millisecond (at a frequency of 1kHz), as well as provide neutron spectrum updates each 100 ms. The required measurement accuracy is 10%. This necessitates development of a high-speed real-time signal processing unit.
Nuclear reactions between fast neutrons and carbon atom nuclei lead to ionization in the diamond detector's volume. If voltage is applied to the crystal, a pulse of current is generated on each interaction between a neutron and a carbon atom nucleus. The amplitude of the resulting signal depends on the amount of energy E_p absorbed on each interaction. Typical width at base for pulses registered at the ADC input is around 30 ns. Due to this we have to use an ADC with a sample rate of at least 500 MHz.
This study describes the development process of a diamond detector signal processing unit, based on a heterogeneous computing device, consisting of an industrial computer and a fast ADC coupled to a high-speed I/O board with data processing capabilities provided by an on-board reconfigurable FPGA. In it we describe techniques used for signal filtration and pulse detection.
An algorithm has been developed to identify pulses in real-time and measure their parameters: amplitude, width at base. It is able to reject pulses based on their duration to avoid pile-ups, which has an unpredictable influence on the resulting statistics. This algorithm is able to mitigate a constant non-zero shift of the data baseline (from which amplitudes of pulses are measured), as well as baseline fluctuations.
This algorithm has been successfully applied in an experiment on the NG24M neutron generator. Diamond detector measurements have been performed in a neutron field with a flux of 14 MeV neutrons of up to $10^9\ \frac{n}{cm^2 \cdot s}$. In this experiment the pulse count-rate of the diamond detector exceeded 300 kHz. Results of this experiment show that this algorithm is suitable for use in the VNC diagnostics.
This work is being carried out in accordance with the state contract dated April 21, 2020 No. H.4a.241.19.20.1042 "Development, pilot production, testing and preparation for the supply of special equipment to ensure the fulfillment of Russia's obligations under the ITER project in 2020".

11 Current Trends in Development of Radiation Detectors / 111

#11-111 Neutron-Gamma Discrimination of Stilbene Crystal, 6Li-doped Plastic and BC501A Liquid Scintillators-based Machine Learning and Signal Processing Techniques

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This paper presents a comparative study of neutron-gamma discrimination performance with stilbene crystal, 6Li-doped plastic and BC501A liquid scintillators coupled to photomultiplier tubes. Neutron-gamma discrimination itself relies upon supervised and unsupervised machine learning algorithms. The method, which is based on blind non-negative matrix factorization (NMF) as an unsupervised model-based source separation and support vector Machines (SVM) as a supervised learning model, aims to achieve separation of neutron and gamma-ray pulses generated from scintillation detectors with high specificity, and high sensitivity as well. The NMF is used for blind source separation when there is no prior information about the mixing process and source signals. It is applied to reconstruct a set of statistically independent original sources from a mixture of output signals induced from radioactive decay of Cf-252 source. A factor of merit, namely the signal-to-interference ratio, is used to validate the separation and reconstruction quality of original sources. The reconstructed independent sources are then characterized by applying a continuous wavelet transform that converts the one-dimensional into two-dimensional time-scale representation (or scalogram). The latter is a complete analysis of the time and scale of the one-dimensional time series signal and allows to identify the characteristics of the reconstructed original sources more accurately. We then use these scalograms to construct and train a binary classification SVM model devoted to the quantitative recognition of neutrons and gamma-rays in a mixed radiation filed. Before using SVM, the Otsu thresholding method and a principal components analysis are implemented to increase the prediction ability of the SVM model. The performance evaluation of the proposed method using stilbene crystal, 6Li-doped plastic and BC501A liquid scintillators is performed by comparing it with a conventional pulse shape discrimination method, namely the charge comparison method (CCM) that is used to obtain the pure training data set for SVM model. Finally, the SVM model based on NMF and CCM is assessed using proper performance metrics, namely the confusion matrix and precision-recall.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 112

#07-112 A computational methodology for estimating the detector pulse-height spectrum in gamma-ray spectrometry of irradiated nuclear fuel

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Gamma-ray spectrometry using collimated detectors is a well-established method to acquire information about the state of irradiated nuclear fuel. However, the feasibility of examining a particular nuclide of interest is subject to constraints; the peak must be statistically determinable with the desired precision, which is governed by the peak count rate, and the continuum background in the region of interest. Furthermore, the total spectrum count rate in the detector will pose constraints, by potentially causing pile-up or dead time issues, that may paralyze the detector. Therefore, the prediction capability of these spectral parameters is needed.
Methods were assembled for spectrum prediction with the intended usage in the optimization of gamma emission tomography, and to enable a priori feasibility evaluation of determination of the intensity of a peak in an energy spectrum of gamma-rays emitted from an irradiated nuclear fuel rod. The focus was on finding reliable results regarding total spectrum and peak count rates with faster computation time, as compared to exclusively using Monte Carlo. For this purpose, the method is based on depletion calculations with SERPENT2, a point-source kernel method for the collimator response, and a detector response matrix pre-computed with SERPENT2. The computational methodology uses as input the fuel properties (dimensions, materials and power history and cooling time), and the instrumental setup (collimator and detector dimensions and materials).
The prediction method was validated using measured data from a high-burnup, short-cooled test fuel rodlet from the Halden reactor. Absolute count rates and ratios of characteristic peaks were compared between predicted and measured spectra. The comparison showed a total count rate underestimation of 6%. The trend is also confirmed for the single peaks, showing a minimum discrepancy of 4-5% for the characteristic gamma lines of 137Cs and 140La, and a maximum discrepancy of 30% for the peaks present in the energy interval between 400-600 keV.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 113

#07-113 The natIn(n, γ)116mIn reaction cross-sections - An important neutron monitor reaction

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The (n, γ) reaction cross-sections of Indium (In) isotopes are of prime interest for the study of neutron-induced nuclear data. The (n, γ) reaction cross-sections are crucial for upcoming nuclear technologies, like Accelerator Driven Subcritical Systems (ADSs) and Advance Heavy Water Reactors (AHWR) [1, 2]. Indium has extensively been used for the flux measurements in the neutron-induced reaction experiments. There is a need for improvement in the neutron monitor reaction data as the uncertainties in flux directly goes into the measured sample reaction cross-sections. For this purpose, production cross-sections of 116mIn isotope were measured using neutron activation [3] of natIn target following off-line γ-ray spectroscopy by using a pre-calibrated HPGe detector. The 27Al(n, α)24Na reaction is used for neutron flux monitoring purposes. Appropriate energies of proton beams were used from the 14UD BARC-TIFR Pelletron facility, Mumbai, India to generate 10.95 ± 0.67, 13.97 ± 0.0.97, 16.99 ± 0.88, and 20.00 ± 0.94 MeV average energy quasi-monoenergetic neutrons using 7Li(p, n)7Be reaction. Experimentally measured data from the present work have been compared with the existing data libraries such as ENDF/B-VII.1, JENDL-4.0, JEFF-3.2 and CENDL-3.1 [4]. The uncertainty and correlation between the present experimental data have been determined using the ratio technique of covariance analysis [5]. The results were also reproduced and compared with the theoretical nuclear modular codes like TALYS-1.95 [6] and EMPIRE-3.2.3 [7]. The present results show a good agreement with the theoretical as well as with the existing experimental data in different data libraries. The present findings are important for the improvement in the nuclear reaction data, advanced reactor design, dose estimation, and flux measurements.

Keywords — natIn(n,γ)116mIn reaction, natLi(p,n)7Be reaction for neutrons, cross-section, γ-ray spectroscopy, EMPIRE-3.2.3, TALYS-1.95, Covariance analysis.

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05 Nuclear Power Reactors Monitoring and Control / 114

#05-114 Monte-Carlo determination of correction factors for the absolute measurement of the activity of solid-state dosimeters using γ- and X-ray spectrometry

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The MADERE platform (Measurement Applied to DosimEtry for REactors) is a metrology facility whose main purpose is to determine the specific activity of solid-state dosimeters irradiated in nuclear reactors, using γ- and X-ray spectrometry. The platform is accredited by the French accreditation Committee (COFRAC) for specific activity measurement of γ- and X-ray emitters.
Analysis methods used at MADERE platform for deriving absolute activities of a dosimeter from the spectrometry measurements induce to take into account several perturbing inherent phenomena of γ and X-ray spectrometry experiments such as self-attenuation inside the dosimeter or escape peak inside the detector. Thus, in order to perform a reliable and accurate analysis, the usual method to determine these correction factors is to use semi-empirical formula and dedicated software. Associated uncertainties are calculated using a Bayesian approach and can reach up to 10 % of the correction value.

In the framework of the continuous improvement of the MADERE platform, an integral Monte-Carlo simulation approach has been initiated to model all the interactions occurring in the whole measurement system (sample, holder device and semiconductor detector).

In order to improve the accuracy on the correction factor determination, a simulation scheme using the combination of the Monte-Carlo codes TRIPOLI-4® (CEA) and GEANT4 (CERN) is developed. In this work the different method developed to determine the correction factors due to the photon interaction inside the detector and inside the dosimeters are presented. Monte-Carlo correction factors are determined for different dosimeters configurations (matrix, geometry) and measurements setup (detector type, dosimeter-detector distance). To validate the results provided by the Monte-Carlo methods, simulations have been compared with the factors routinely applied at the MADERE facility. Then a qualification process of these simulated correction factors will be set up based on the comparison with results of dedicated experiments.

02 Space Sciences and Technology / 115

#02-115 Gamma-ray Detection and Localization at high angular resolution

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The recent association of gamma-ray bursts with neutron star mergers highlighted the need for sensitive gamma-ray detectors with high localization accuracy. Identification of the host galaxy of the gravitational wave event GW170817 and the gamma-ray burst GRB170817A was only achieved 11 hours after initial detection, due to difficulties in surveying the uncertainty region of the gravitational wave event. Gamma-ray bursts are transient events with a rapidly diminishing afterglow at longer wavelengths. A delay in the precise localization means crucial data were lost until observations at longer wavelengths could begin. This problem can only be mitigated using detector systems with large viewing angles, preferably the entire sky, and far improved angular resolution of a few degrees, which is required for prompt follow-up.

We present a novel gamma-ray detector concept aimed at improved angular localization with respect to the current state of the art. The presented detector system relies on a non-uniform pattern of small scintillators coupled to silicon photomultipliers. The pattern utilizes the mutual occultation between detectors to reconstruct the gamma-ray burst's direction in the sky with good angular accuracy. Our simulations show that the achievable localization accuracy for such a configuration is considerably better than those obtained by larger scintillator assemblies while maintaining a field-of-view of the entire sky. We show that even when the total effective area is decreased, our detector system still shows better angular sensitivity compared to designs based on the current state of the art. We present laboratory experiments on various configurations of 90 small detectors. We show the effects of changing the configuration and of changing the scintillator type. The systems are also experimentally compared with a larger detector system built with a traditional approach. Both simulations and experiments clearly show our novel concept can achieve a considerable improvement in angular sensitivity, without compromising sensitivity or field-of-view. The proposed concept can be easily scaled to fit into small satellites, as well as larger missions.

09 Environmental and Medical Sciences / 117

#09-117 Progress and prospects of MACACO: a multi-layer Compton camera for range verification in hadron therapy

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The detection of prompt gamma-rays exiting the patient during hadron therapy treatments could provide a way to achieve online range verification, which in turn would represent a key step in tumor oncology. However, the desired detector must be able to image gamma-rays from an large source and a broad spectrum in the few MeV region, while dealing with low statistics and low signal to noise ratio. MACACO, the multi-layer Compton Camera currently being developed by the IRIS group at IFIC-Valencia, could be a suitable candidate. It features three continuous LaBr3 scintillator crystals coupled to Silicon Photomultipliers, thus maximizing the efficiency to gamma-rays detection in a compact design.

By using electronic collimation and Compton kinematics, the detector is able to recover the origin position of those incident gamma-rays interacting in any two layer combination, or in all three of them. In addition, the recent developments in software made by the group allow recovering the incident energy of the gamma-rays through spectral reconstruction algorithms, which translates into a four-dimensional imaging capability able to deal with broad gamma-rays spectra where the incident gamma-ray energy is unknown.

The different detection channels can be combined into a joint image reconstruction process, thus having increased efficiencies when compared to conventional Compton cameras. Experimentally, this joint image reconstruction has already proven to yield better quality images of complex structures, such as arrays of radioactive sources. Ongoing simulations indicate that the joint use of all measurement channels could become critical in the correct assessment of the Bragg peak distal fall-off.

The recently assembled MACACO III prototype shows significantly improved performance and is being tested at beam facilities with promising results. An energy resolution of 4.8 % at 662 keV and an angular resolution of 6.0º at 1275 keV have been achieved. In parallel, the MACACOp prototype employing TOFPET2 Petsys ASICs has been also assembled and tested. Such ASIC provides time and energy digitization of signals from each SiPM channel, and preliminary results show an enhanced coincidence time resolution of 2.2 ns (fig. 4). The latter may become a relevant feature in a hadron therapy scenario, where high rates are expected. Further tests of both prototypes in a cyclo-synchrotron facility at clinical intensities are foreseen.

Simulation studies on the background composition expected in a hadron therapy scenario have been driven by the group, and a twofold strategy of background reduction is under investigation. On the one side, a hardware approach through the addition of a silicon layer is under development, which would allow detecting primary background particles as well as escaping Compton electrons that do not fully deposit their energy in the LaBr3 detector. On the other side, a software approach through the development of dedicated neural networks for background rejection has been successfully tested. The latter has already allowed MACACO to detect 3 mm Bragg peak shifts in a proton beam with therapeutic energies, which represents another step towards the final application of the prototype.

09 Environmental and Medical Sciences / 118

#09-118 New probe for the improvement of the Spatial Resolution in total-body PET (PROScRiPT)

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The development of new PET (Positron Emission Tomography) scanners with improved performance is still an important line of research in nuclear medicine. One current line of research in nuclear medicine is the study of a total-body scanner, the development of which would represent an improvement on the current features of PET scanners. Currently, the typical spatial resolution of the reconstructed PET images is between 4 and 6 mm. The limitation in spatial resolution is a result of the detector module design, the positron range and the non-collinearity of the detected photons. Using smaller detector modules in PET scanners would improve the spatial resolution but would worsen the sensitivity. For this reason there have been studies suggesting the use of PET probes. Such probes would improve spatial resolution without compromising the scanner sensitivity.

The use of a PET probe has been studied in several research groups with a view to improving the spatial resolution. A PET probe consists of a small detector that operates in temporal correlation with a PET scanner (by means of the scanner electronics or with rear signal processing). In this way, one of the annihilation photons interacts with the PET scanner and the other one with the probe, which has much better spatial resolution than the scanner. The fact that the PET probe can be positioned in the whole field of view (FOV) of the PET scanner is of particular interest since the probe increases the spatial resolution of a specific region/lesion of the patient's body.

The suitability of a PET probe based on scintillation crystals and silicon photomultipliers (SiPM) has been studied. The functioning of the probe will be tested in a Preclinical Super Argus PET/CT scanner for small animals.

Preliminary GATEv8.2 simulations of the probe and the Preclinical Super Argus PET/CT have been performed showing promising results. The simulated setup consisted of a PET with 4 rings and 24 modules per ring, with each module containing an array of 13x13 scintillation crystals (GSO and LYSO phoswich). The probe, consisting of a continuous scintillation crystal (LYSO) of 25.8x25.8x5 mm3, was positioned at 4 mm from a small Derenzo-like phantom which was in turn positioned in the centre of the field-of-view of the scanner. The small Derenzo-like hot-spot phantom, contains an activity concentration of 0.108 mCi/cc and has several inserts measuring 4.8, 4.0, 3.2, 2.4, 1.6 and 1.2 mm in diameter. The coincidence time window for the simulation was set to 2 ns. Images were reconstructed using a MLEM code and all the reconstructions were stopped at the 5th iteration. A reconstructed image of the Derenzo-like phantom was obtained for three cases: events in which there were temporal coincidences between any detector in the setup, events in which there were temporal coincidences between the scanner detectors and events in which at least one of the detectors involved in the temporal coincidence was the probe. The obtained results show that for the case in which there was coincidence between the scanner and the probe, the spatial resolution improved with respect to the other two cases.

Regarding the hardware involved in the project, the probe is being readout by the PETsys system, a commercial time-of-flight PET system. The system provides the digitalised signal of 64 channels per detector and has a high time resolution (time binning 20 ps). A set of calibration measurements with different monolithic crystals (LaBr3, CeBr3 and LYSO) of 25.8x25.8x5 mm and two Hamamatsu S13361-350AE-08 MPPCs (pixel pitch 25 µm and 50 µm) have been performed. The results show a better linearity of the photo-peak position with respect to the feeding voltage and a more stable energy resolution versus feeding voltage for the MPPC with 25 µm pitch. An energy resolution of 6.3% for the 511 keV energy peak of Na22 was measured at 20oC with the LaBr3 crystal.

In conclusion, the GATEv8.2 simulations show the capability of the probe to improve spatial resolution. Furthermore, tests carried out with the PETsys system show it to be a good candidate for the probe electronics. In future work, further simulations with the probe in a total-body PET scanner will be performed. In addition, measurements in time coincidence with the probe and the scanner will be acquired and reconstructed.

04 Research Reactors and Particle Accelerators / 119

#04-119 Study review of the CALORRE differential calorimeter: definition of designs for different nuclear environments

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The high neutron and gamma fluxes as well as strong displacements per atom characterize conditions of materials testing reactors (MTRs). These research reactors constitute major support research facilities and allow experimental and real-condition studies of the behavior of fuels and inert materials (vessel, reflector, cladding, etc.) in an extreme radiation environment. These studies are important as they lead to progress on accelerated ageing of materials and/or advanced scenarios up to accidental conditions and consequently they bring data for safety issues, the lifetime of existing nuclear power plants and their advancements with new concepts. Therefore, new instrumentation is needed to measure online key parameters both before the experiments for the device design and during the experiments for the result interpretation.
The construction of the Jules Horowitz Reactor, a new 100 MWth nominal power MTR with unequalled performance in Europe (high fast neutron flux of 5.5x10^14 n/(cm².s) (from 1 MeV) leading to a high accelerated ageing (up to 16 dpa/year) and a high nuclear absorbed dose rate (up to 20 W/g in aluminum)), initiated new collaborative research work. Since 2009 Aix-Marseille University and the CEA (within the framework of the joint laboratory LIMMEX - Laboratory for Instrumentation and Measurement in Extreme Environments) and its IN-CORE program - Instrumentation for Nuclear Radiations and Calorimetry online in REactor) have been developing a new research topic. More precisely, they have been focusing on innovation in instrumentation and advanced measurement methods for the quantification of key nuclear parameters such as neutron and photon fluxes and nuclear absorbed dose rate, also called nuclear heating rate. The online measurement of this latter quantity requires specific sensors: non-adiabatic calorimeters. With regard to the state of the art, two distinct sensors are used in MTRs: French differential calorimeters (CALMOS, CARMEN or CALORRE type) and European single-cell calorimeters (such as gamma thermometers or KAROLINA-type calorimeters). These two types of calorimeter allow the quantification of the nuclear absorbed dose rate thanks to temperature measurements and preliminary calibration under non-irradiation conditions from steady thermal states in the case of integrated heating elements or from transient thermal states in other cases.
The paper will present a review of recent work carried out on a new differential calorimeter called CALORRE and patented by AMU and the CEA in 2015. The work allows the design and the characterization of several CALORRE configurations over a wide range of nuclear absorbed dose rate (up to 20 W/g).
The first part of the paper will describe the first design of CALORRE calorimeter fabricated for a range up to 2 W/g and its first qualification under real conditions during an irradiation campaign inside the MARIA reactor (in November 2015).
The second part will be dedicated to the characterization of the response of 6 new configurations of CALORRE calorimetric cell by means of a comprehensive approach coupling experimental, theoretical and numerical work from laboratory conditions to nuclear environments. The experimental metrological characteristics of the 6 configurations, in terms of sensitivity, linearity, range, reproducibility and response time, will be detailed. Their responses will be compared and analyzed thanks to a 1-D theoretical thermal model. Moreover, by means of a predictive model (based on a heat balance and calibration curves under laboratory conditions), their response under real conditions will be predicted. Then, thanks to this panel of configurations, the influence of various parameters such as the sensor geometry, its size and its structural material will be given. The advantages of this kind of calorimeter will be discussed and compared to other calorimeters (differential and single-cell calorimeters). Finally, 3-D thermal simulations allowing the optimization of the calorimeter assembly in term of compactness will be summarized for the chosen JHR configuration (for high nuclear absorbed dose rates) already fabricated, tested up to 60 W under laboratory conditions (electrical power range ten times higher than that of previous calibrations for common differential calorimeters).
The last part will present new results obtained within the framework of a new research program called CALOR-I, funded by Aix-Marseille University, involving the Nuclear Reactor Laboratory of

the MIT and the CEA (2020-2022), and focusing on the mapping of an in-core water loop of the MITR reactor in terms of nuclear heating rate by means of a new CALORRE calorimeter in particular. The criteria for the choice of this new configuration of CALORRE will be detailed (mass, size, temperature, sensitivity). Then, the parametrical study for the definition of a new CALORRE differential calorimeter assembly with three calorimetric cells and two kinds of sample material will be presented.

09 Environmental and Medical Sciences / 121

#09-121 Plutonium and Americium Inventories in Soil Cores from the English Lake District, Cumbria (UK)

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Measurements of trace levels of environmental radioactivity, made with a broad energy germanium (BEGe) detector, for trace actinide (241Am) analysis of soils from the shores of two lakes in the English Lake District, UK, are described. In particular, the need to determine the radioactive concentration (Bq/g) in the soil samples with the intent being to discern natural and anthropogenic contributions, and trends in abundances associated with influences of the landscape, at trace levels in the environment. This has potential benefit for the assessment of 241Am and 241Pu and especially comparisons that might be made with accelerator mass spectrometry assessments.

01 Fundamental Physics / 122

#01-122 The spectrometer MULTI for measurement of pygmy dipole resonances in exotic nuclei

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The construction of new portable spectrometer "MULTI" is designed for the measurement of β-γ-neutron coincidences in search for pygmy dipole resonances in exotic nuclei. It is based on many years of experience with spectrometer for direct measurement of the total reaction cross section with radioactive beams. Experimental technique includes measurement of the ratios between gamma-ray emission and multi-neutron emission, following beta-decay of the implanted radioactive nucleus. The spectrometer consists of an in-beam multi-detector telescope for identification of the secondary beam projectiles and a gamma-ray spectrometer with neutron counters array, surrounding the β-detector. The in-beam part consists of active collimators, silicon dE0 detector for particle identification by dE:TOF (time of flight) method and β-detector for implantation of the nuclei of interest and measurement of electrons emitted in β-decay. Gamma-rays and neutrons are measured by eight CeBr3+NaI(Tl) phoswich scintillation detectors, surrounded by 40 neutron counters in form of 3He tubes in polyethylene blocks. The design was optimized with Monte Carlo method in Geant4. Data acquisition electronics is based on the digital pulse processing modules in VME standard, including pulse-shape analyzing techniques for identification of the particle type. Project and construction of the spectrometer, calibration measurements of its fundamental characteristics and the method of measurement with radioactive beams will be presented.

04 Research Reactors and Particle Accelerators / 123

#04-123 Characterization of calorimeter responses under laboratory conditions thanks to an optimized transient thermal test bench

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The nuclear heating named also "absorbed dose rate" is a quantity that it is essential to predict numerically in order to optimize the design of different elements of a nuclear research reactor and of its experimental irradiation devices for instance for safety, thermal and mechanical aspects and then to measure it accurately in order to analyze results associated to in-pile experiments. Heat flow calorimeters are the sensors that permit real time measurement of this key parameter.

Since 2009, Aix-Marseille University and the CEA have been conducting research dedicated to improving such sensors in terms of their size, range, out-of-pile calibration and in-pile measurement methods. In 2015, a new differential calorimeter (called CALORRE) composed of two compact calorimetric cells with predominantly radial heat transfer was patented and tested successfully in the Polish MARIA reactor. At present, a new joint research project, which also involves the Nuclear Reactor Laboratory of the MIT, is in progress. This project, called CALOR-I, focuses on the characterization of a new reduced-height design of the CALORRE calorimeter and its use for the mapping of the nuclear heating rate inside the MITR water loop. The study will be from laboratory calibration conditions to real conditions inside this loop located in the reactor core (a cylindrical pressure vessel of 38 mm in diameter, forced or natural convection, a fluid flow temperature up to 300 ℃ and a nuclear heating rate up to 2 W/g (in stainless steel) at a thermal power of 6 MWth).

The method of calibration of a calorimeter varies according to the type of calorimeter used. For single-cell or differential calorimeters with an electrical heating system, a steady state thermal calibration is usually applied, whereas for calorimeters without an electrical heating system a transient thermal calibration is used.
Within the framework of the CALOR-I project, the aims of this paper are the optimization of a transient thermal calibration test bench, called BERTRAN, and the study of the response of different calorimeters by using this tool and a 1-D transient thermal model.

First of all, the test bench and its improvement will be detailed. The bench is composed of two temperature-controlled thermostatic baths filled with a heat-transfer fluid which can reach a temperature up to 250℃ without boiling (silicone oil). The hot bath hosts three heating cartridges and the cold bath integrates a heating cartridge and a cooling coil. In addition, inside each bath, there are a cylindrical tube with an internal diameter similar to that of experimental channels and a propeller located at the bottom of the tube in order to create an upward silicone oil flow (adjustable speed). This configuration allows the generation of thermohydraulic conditions close to those existing inside experimental channels of research reactors. Moreover, this bench has a mechanical system used to transfer the calorimeter automatically from the hot cavity (at a temperature from 100℃ to 250℃) to the cold cavity (at a temperature from 20℃ to 50℃) (or vice-versa). The bench improvements concern the implementation of two new K-type thermocouples to measure the temperature around the calorimeter and an infrared sensor to determine precisely the position of the calorimeter (the inlet or outlet time in each bath); and the change of the acquisition device to improve the sample time.
Next, the operating protocol allowing the determination of the transient response of the sensors will be described. In fact, after the thermal stabilization of the two baths previously tuned for two

different temperature set-points, the calorimeter is inserted inside the cold cavity. When a first stationary state is reached inside the calorimeter, the data acquisition is started. After 10 minutes to ensure stabilization, the calorimeter is transferred automatically and quickly (1-2 seconds) from the cold cavity to the hot cavity (heating phase). After a 10 minutes immersion inside the hot cavity, a second stationary state inside the sensor is achieved, then the reverse transfer of the calorimeter is done (from the hot cavity to the cold cavity, cooling phase). Finally, the calorimeter remains for 10 minutes in the cold cavity in order to reach a final thermal equilibrium.

Then the experimental results obtained by applying this procedure for a parametrical campaign carried out in July 2020 for a single-cell calorimeter (called KAROLINA) will be presented. The response curves of this calorimeter, for the heating phase and then the cooling phase, will be shown for various experimental conditions, with consideration of repeatability and reproducibility. The influence of the fluid flow conditions (temperature and velocity) and of the operating phase will be given on the response time, the thermal constant and the deduced sensitivity of the calorimeter.
In addition, a 1-D transient thermal model will be described and applied to predict the temperature inside the calorimeter versus time and to analyze the experimental results.

Finally, the results obtained with the KAROLINA calorimeter will be compared to forthcoming experimental results of other calorimeters (CALORRE calorimeter and a Gamma Thermometer). If the health situation causes difficulties in carrying out these experiments, the experimental results already obtained will be compared to 3D thermal simulation results obtained with COMSOL Multiphysics.

08 Decommissioning, Dismantling and Remote Handling / 125

#08-125 Modelling the response of an in-situ CdTe detector to radionuclides in groundwater.

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This research examines the potential deployment of a cadmium telluride strontium-90 detector in groundwater boreholes at nuclear decommissioning sites. This is represents a novel approach to monitoring strontium-90 contamination at decommissioning sites such as Sellafield, and has the potential to reduce lifetime monitoring costs while providing information on a significantly reduced timescale. A Geant4 simulation was used to model the deployment of the detector in a contaminated groundwater borehole. It was found that the detector was sensitive to strontium-90, yttrium-90, caesium-137 and potassium-40 decay, some of the significant beta emitters found at Sellafield. However, the device showed no sensitivity to carbon-14 decay, due to the inability of the weak beta emission to penetrate both the groundwater and the detector shielding. The limit of detector for such a sensor when looking at solely strontium-90 decay would be 323 BqL^{-1} after a 1 hour measurement and 66 BqL^{-1} after a 24 hour measurement. Existing techniques are capable of examining strontium-90 decay below the World Health Organisations safe drinking water limit of 10 BqL^{-1}. A GaAs sensor with twice the surface area, but 0.3 % of the thickness was modelled for comparison. Using this sensor, sensitivity was increased, such that the limit of detection for strontium-90 was 91 BqL^{-1} after 1 hour and 18 BqL^{-1} after 24 hours. However, this sensor sacrifices the potential to identify the present radionuclides by their end-point energy.

04 Research Reactors and Particle Accelerators / 126

#04-126 CEA-JSI Experimental Benchmark for validation of the modelling of neutron and gamma-ray detection instrumentation used in the JSI TRIGA reactor

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Constant improvements of the computational power and methods as well as demands of precise and reliable measurements for reactor operation and safety require a continuous upgrade of the instrumentation. In particular, nuclear sensors used in nuclear fission reactors (research or power reactors) or in fusion facilities are operated under intense mixed neutron and gamma-ray fields, and need to be calibrated and modelled in order to provide selective and precise neutron and gamma-ray measurements.

With this aim in mind, the French Alternative Energies and Atomic Energy Commission (CEA) and "Jožef Stefan" Institute (JSI) have started an experimental program dedicated to the set up of a detailed experimental benchmark with analysis using Monte Carlo particle transport calculations and uncertainty evaluation for the main neutron and gamma-ray sensor types used in the JSI TRIGA (Training, Research, Isotopes, General Atomics) reactor. This benchmark should also provide recommendations for instrumentation modelling and analysis for mixed gamma-neutron fields.

CEA has setup a simplified TRIPOLI-4 (three-dimensional polykinetic code) modelling scheme of the JSI TRIGA reactor based on the information available in the International Reactor Physics Experiment Evaluation Project (IRPhEP) and International Criticality Safety Benchmark Evaluation Project (ICSBEP) benchmarks in order to simulate the sensor responses for neutrons and gamma-rays. These allow the CEA to perform a TRIPOLI-4 instrumentation calculation scheme taking into account the geometry of the sensor and of the irradiation device as well as the surrounding irradiation conditions.

This paper presents main results of this CEA calculation scheme application and the analysis of their comparison to the JSI results obtained with the Monte-Carlo N-Particle transport code (MCNP) – Evaluated Nuclear Data File B-VII.0 (ENDF/B-VII.0) - International Reactor Dosimetry and Fusion File (IRDFF) calculation scheme. This paper will conclude by giving some information on the second step of the experimental program consisting of a dedicated experimental campaign to be carried out in the following months in the TRIGA reactor core.

11 Current Trends in Development of Radiation Detectors / 127

#11-127 Spectral Resolution Enhancement of a SiPM Array-Based Radiation Detector

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Silicon Photomultipliers (SiPM) are becoming more attractive in radiation detection applications then the traditional photo multiplier tubes due to their low working voltage, compactness and immunity to electromagnetic interference (EMI). However, due to their small size, an array of SiPM components is required in order to cover the whole plane area of a scintillator. On the other hand, since SiPM is a semiconductor, that is biased in a reversed voltage, gain variation and strong temperature dependence are introduced. As a result, SiPM based detectors, particularly an array of SiPMs, undergo spectral resolution reduction.
In our work, we propose an electronic approach to overcome this technological drawback by individually adjusting the bias voltage for each SiPM in the array. This developed technology, provides an adequate temperature dependent, commonly distributed high bias voltage and an individual offset-voltage fine tuning. That enables to adjust all the SiPM components to their optimum operating points. Power-wise it is beneficial to operate SiPM at lower voltages, where undesirably gain variation are more dominant. This presented solution enables lower bias voltages, which provides both lower power consumption and enhanced spectrum resolution.
A group of ten SiPM devices was tested for break-down voltage analysis, by Inverse Logarithmic Derivative (ILD) criteria. The most suitable devices with respect to resolution were selected, to assemble a 2X2 SiPM array that is biased with our individual-offset topology. By means of pulse-height analysis, we investigated the ability to enhance the spectral resolution of an SiPM array coupled to various scintillators. The presented topology provides improved power consumption with enhanced spectral resolution over the traditional biasing methods.

11 Current Trends in Development of Radiation Detectors / 128

#11-128 Analog Pulse Shape Discrimination based on Combination of Time Duration with Pulse Hight

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Pulse Shape Discrimination (PSD) is a useful technique used to detect and distinguish between different types of radiation interactions. PSD methods are frequently adopted for n/γ or α/β discrimination. Over the years, many techniques for performing PSD were presented, both analog and digital implementations. A digital PSD enables implementation of complex algorithms, which analyze various parameters, hence provides advanced discrimination capabilities. However, digital PSD requires high speed acquisition hardware, especially for pulses with fast decay time, e.g. originating from plastic and organic scintillators. Furthermore, sophisticated algorithms require long processing time that limits the count rate and increases the dead time. Moreover, fast samplers require significant power consumption making them less suitable for portable devices. On the other hand, analog PSD methods can be more suitable for high speed scintillators both from rate and power consumption perspectives. Common analog discrimination methods are based on pulse-height and pulse-energy discrimination techniques. Other techniques rely on the time difference in the pulse width such as the Zero-Crossing (ZC) methods. Neither of the above combine both amplitude and time methods.
In degraded light collection conditions, such as long and opaque scintillators, a lower pulse-height is obtained, while the noise level is unaffected. Consequently, lower signal to noise ratio is obtained, causing amplitude-based methods to yield a considerable number of miss-classification. Contrarily to pulse height and energy that decrease proportionally to the light collection efficiency, the pulse shape is less affected. Time and pulse width-based methods are not error proof either. Pulse width at over threshold of fast and high amplitude pulses can have a similar width as slower pulses with lower amplitude.
We present a novel analog PSD topology that overcomes this issue. The topology is based on discrimination according to the pulse duration in time combined with compensation function of the pulse height. Amplitude of the pulse is used as a restraining factor. Subsequently, our topology correctly identifies fast pulses that are prolonged in time due to their high amplitude. The topology was realized on a high speed printed circuit board techniques providing superior discrimination capabilities with an uncertainty gap smaller than 1 ns in the pulse width. The ability to control both the time and the amplitude parameters individually, provides tailored adjustment for various detectors and PSD applications.

10 Education, Training and Outreach / 130

#10-130 On Teaching Experimental Reactor Physics in Times of Pandemic

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Practical exercises or hands-on experiments in Reactor Physics at a research reactor are an essential part of education and training in Nuclear Engineering. They are mostly performed at low power research reactors and/or critical and subcritical assemblies. Institutions that do not have their own research experimental facilities often visit other facilities to perform hands-on experiments. In times of pandemic, when mobility of people is very limited, such visits are practically impossible to perform hence other options has to be considered.
The Jožef Stefan Institute (JSI) TRIGA Mark II research reactor is regularly used for education and training. However, since spring 2020 the access to the reactor has been limited due to worldwide Covid-19 pandemic. In spring 2020 the reactor was completely shut down, practical exercises on experimental reactor physics had to be performed online using videoconference software, cloud services and research reactor simulator. In June 2020 the reactor started normal operations, however due to travel restrictions around the world and in Europe, students could not come to Slovenia. Hence a practical educational curse ''Experimental reactor physics'' for students from Uppsala University was organised remotely.
A five-day practical Educational Course was organised and performed using off the shelf but advanced software and hardware components such as: a remotely controlled dome camera in the control room, portable cameras that the lecturers could take to the reactor, two video-conference setups, remotely controlled laptop used to operate the data acquisition software and the Digital Reactivity Meter, a remotely controlled common whiteboard, a remotely-operated camera showing the reactor core, cloud document system.
The educational course program consisted of the same exercises that are offered during in-person exercises: Introduction to JSI TRIGA reactor, Critical experiment, Reactor response to step reactivity changes, Reactor operation, Void reactivity coefficient, Control rod worth measurements and Temperature reactivity coefficient.
The performance of the course was evaluated using an online anonymous survey taken by all the students and their mentors. The organizers provided both open-ended questions and questions that were answered using a rating scale. The aspects being evaluated included the technical content, quality of material, performance of the individual lecturers and the quality of the remote session. In general, the response was overwhelmingly positive with most questions with a rating scale answered with "excellent". Multiple participants complained about the occasional problems with the sound quality, as they could not clearly hear the speakers when they stepped too far away from the conference microphones. The organizers of the course agree that the first remote exercises on a research reactor in Slovenia were well organized and useful but observed difficulties in achieving the same level of student involvement as the in-person exercises.
In the paper we describe the education course and its implementation. This is followed by evaluation of the course and outlook for future improvements. The paper can serve as demonstration and a guide on how to organise remote hands-on exercises at a research reactor, using relatively cheap and widely available components. It is important to note that the purpose of remote hands-on experiment is not to replace the in-person ones but to provide an alternative in difficult times, such as pandemic or any other situation affecting free movement of people.

01 Fundamental Physics / 132

#01-132 Autonomous Measurements Driven By Machine Learning

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During the last reactor cycle in 2020, a combined team from the Institut Laue-Langevin and Berkeley National Lab has commissioned and tested a self-learning algorithm capable to perform autonomous measurements. For the first time the computer took control of the three axis neutron spectrometer ThALES, without any human intervention. The algorithm was able to explore the reciprocal space and fully reconstruct the signal without any prior knowledge of the physics case under study. Thanks to autonomous learning gpCAM - developed by Marcus Noack of the CAMERA team at Berkeley Lab – estimates the posterior mean and covariance and uses them in a function optimization to calculate the optimal next measurement point. The posterior is based on a prior Gaussian probability density function,

which is repeatedly retrained on previously measured points. The main advantage of such an approach is clearly the possibility to drastically reduce the number of measurements with respect to a classical grid scan and therefore optimize the beam-time usage. In the present paper, the excellent results obtained will be discussed as well as the opportunities for further improve this technique.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 133

#07-133 Unmanned Aircraft Systems Based Radiological Mapping of Buildings and Objects

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This paper is focused on the radiological mapping of buildings and structures/objects by utilizing unmanned aircraft systems (UAS), including multicopters in particular. These platforms make possible measuring data in close distance to studied objects and with adjustable altitudes, in contrary to terrestrial and other aerial assets. Tasks such as inspection of illegal transportation or storage of radioactive nuclear material, search for uncontrolled radioactive sources, detailed survey of buildings and objects for possible contamination, monitoring of nuclear facilities (e.g., nuclear repositories) require accurate and up-to-date topographic information on the area of interest.

Performing the detailed radiological characterization or mapping requires the aircraft to be equipped with a camera and a radiation detection system. The camera is utilized to take a series of aerial pictures of the object in order to reconstruct its three-dimensional (3D) model via photogrammetric techniques. In addition, in the case automatic flight mode is employed, an accurate georeferencing is essential.

The radiation data is preferably acquired following a regular grid around the building/object whenever the circumstances allow it. Knowledge of the model and georeferenced data points enables to project the measured values onto surfaces where they are then subjected to interpolation based on the Delaunay triangulation. The mapping of the gamma dose rate can be superposed to the reconstructed 3D model of the inspected area or building/object. The map may further be converted in the form of isodose areas. Radionuclide identification is also possible when using a detection system with gamma spectroscopy capabilities.

We report in this paper on the development and testing of a radiological mapping system based on the use of a DJI Matrice 210 v2 drone equipped with Zenmuse Z30 camera and a lightweight radiation detection system. The latter device is based on NaI(Tl) and CeBr3 scintillators, coupled with silicon photomultipliers. Its design includes a digital pulse processor, a laser altimeter and a GNSS receiver providing precise synchronization of the dose rate values with position and altitude or distance from the object, respectively. Real-time data processing is also feasible. A built-in RF module and a small battery ensure the complete independence of the detection system from the used aircraft.

The goal of the research is to provide experimental verification of simulation results performed during earlier studies. The utilization of UAS in the field of radiation mapping shows promising results, however, it is still challenging to find an optimal trade-off in the size and weight of the detector. Heavier instrumentation grants better sensitivity, conversely, it reduces the flight time and therefore challenges the data collection process. Poor reception of GNSS signal in the vicinity of walls also remains an open issue.

This work demonstrates the capabilities of using the UAS based radiation detection technology for radiological mapping of buildings and objects with possible applications in areas such as nuclear safety and security, radiation protection and environmental monitoring.

03 Fusion Diagnostics and Technology / 134

#03-134 Fusion fuel ion ratio measurements at JET using neutron spectroscopy

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The fuel ion ratio, n_T/n_D, is an important plasma parameter in magnetically confined fusion, especially for future fusion devices (ITER, DEMO), which plan to achieve fusion through deuterium-tritium (DT) plasmas. In order to maximize the fusion power output, it is necessary to ensure equal number densities of tritium and deuterium (i.e. $n_T/n_D = 1$). Neutron spectroscopy offers the possibility to measure the fuel ion ratio by comparing the contribution to the neutron spectrum from 14 MeV neutrons generated by the T(D, n)$^4_2$He reaction, and 2.5 MeV neutrons from the D(D, n)$^3_2$He reaction. Consequently, neutron spectroscopy has been identified as one of the primary diagnostics for measuring n_T/n_D at ITER. Measurements of the fuel ion ratio is required at ITER over a range of $0.01 < n_T/n_D < 10$ with a precision of 20% and time resolution of 100 ms. In future fusion power plants, the ion ratio will likely be a continuously monitored parameter to ensure the reactor is running at maximum capacity. In this paper we describe a method for measuring the fuel ion ratio for low concentrations of tritium ($n_T/n_D < 1$) at the Joint European Torus (JET) using the time-of-flight (TOF) neutron spectrometer TOFOR. TOFOR has a vertical sight-line of the JET plasma and consists of 37 plastic scintillation detectors which when used in coincidence provide TOF spectra from which information on the energy spectrum of the different neutron components can be inferred. Measurements of the fuel ion temperature are commonly performed using TOFOR data. In this paper, the fuel ion ratio is determined by comparing the TOF peaks corresponding to the two different neutron energies (2.5 MeV & 14 MeV) after removing background and scattered components. Finally, an estimate of the upper limit of n_T/n_D at which the ratio can be accurately determined using a forward scattering TOFOR-like spectrometer is presented.

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#07-135 Qualification Test System for Radiation Detection Devices - QuTeSt

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Measurement equipment for the detection and identification of radioactive and nuclear (RN) material has a wide application area. The main application aspects are monitoring, search, and identification. Measuring systems are divided into different device classes. There are both portable systems like the handheld systems radiation isotope identifiers (RIID) or personal radiation detectors (PRD) and their spectroscopic versions (SPRD) as well as stationary systems like radiation portal monitors (RPM). A common goal is to gain reliable measurement results. In the past, the only way to assess the performance of a measuring device was to rely on the data given by the manufacturer of the device itself. This situation is unsatisfactory, which is also due to the lack of an international seal of approval. Reliable test results from an independent third party are more than welcome. These tests can be performed against consensus standards in order to have reproducible test results, independent of the testing location and the performing laboratory. This is especially relevant for the procurement of new devices, as well as for the comparison between different systems. Fraunhofer INT has conceived and built a test environment to perform dynamic and static test measurements using neutron and gamma sources. These qualification systems were part of a round robin test during Phase II of the project "Illicit Trafficking Radiation Assessment Program (ITRAP+10)", initiated by the European Commission to capacitate several laboratories to conduct qualification tests. Several tests were performed in accordance with the ITRAP+10 test procedures, which are based on ANSI and IEC standards, as well as in accordance with the ANSI standards themselves. In this paper both parts, the static one and the dynamic one, are presented, and exemplary results are shown. This includes qualification tests of truck portal monitors with the dynamic test system. Generally, the effects of one test parameter on other test parameters are not considered in the test procedures. For example, the accuracy of the dose rate may depend on the energy range of the radioactive source used. This will then also affect the over range tests. So far, the latter are only intended for one single nuclide. Besides the overview of the test systems the paper will address restrictions, problems and limitations of the possible qualification measurements as well as potential limitations arising from the given test procedures themselves.

04 Research Reactors and Particle Accelerators / 136

#04-136 Reactor pulse operation for nuclear instrumentation detector testing – preparation of a dedicated experimental campaign at the JSI TRIGA reactor

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The availability of neutron fields with a high neutron flux, suitable for irradiation testing of nuclear instrumentation detectors (such as fission and ionization chambers and self-powered neutron detectors) relevant for applications in nuclear facilities such as material testing reactors (MTRs), nuclear power reactors and future fusion reactors is becoming increasingly limited. Over the last several years there has been increased interest in the experimental capabilities of the 250 kW Jožef Stefan Institute (JSI) TRIGA research reactor for such applications. This is thanks to extensive experimental and computational efforts in the past to characterize the experimental conditions in the reactor, in large part in collaboration with the Instrumentation, Sensors and Dosimetry Laboratory of the French Atomic and Alternative Energy Commission (CEA) – Cadarache. The maximal achievable neutron flux in the reactor in steady-state operation mode is approximately 2×10^{13} n cm^{-2} s^{-1}, in contrast to the MTR-relevant neutron flux range 10^{14} – 10^{15} n cm^{-2} s^{-1}. The JSI TRIGA reactor can also operate in pulse mode, where one of the reactor control rods is ejected from the reactor core, thus introducing a sufficiently large reactivity to achieve prompt supercriticality. This operation mode is made possible by the TRIGA fuel, in the form of a dispersion of uranium and zirconium hydride, which gives rise to a large, prompt and negative temperature coefficient of reactivity. The resulting time dependence of the reactor power is a pulse, the peak power and duration depending on the initial inserted reactivity. In pulse mode, the maximal achievable peak power is approximately 1 GW, corresponding to a peak neutron flux level of the order of 10^{17} n cm^{-2} s^{-1}, for a duration of a few milliseconds.

Pulse operation mode at the JSI TRIGA reactor has been investigated mainly in the academic context, for the validation and improvement of the Fuchs-Hansen model describing the reactor behaviour during reactor pulses and for educational activities, however its application for irradiation experiments was of limited scope. Recognizing the potential of reactor pulse mode for nuclear instrumentation detector testing, in particular the possibility of extending the useful neutron flux range up to MTR-relevant levels, in 2019, a bilateral collaboration project between the CEA and JSI was initiated. The aim of the project is the performance of absolute neutron flux measurements at very high neutron flux levels in reactor pulse operation. The measurements will be made possible by special CEA-developed miniature fission chambers (fissile deposit mass targeted at 50 ng) and by modern, validated, wide dynamic range data acquisition systems, in particular the CEA-developed MONACO system. In addition to the JSI TRIGA nuclear instrumentation, providing information on the peak power and released energy, activation dosimetry will be employed as a reference for the normalization of the recorded detector signals (neutron flux integral) for reactor pulses with differing characteristics. Finally, as an alternative experimental method enabling the measurement of the reactor power dependence during pulse operation, measurements of the intensity of Cherenkov light are proposed and being investigated.

This paper presents the 2019-2020 preparatory activities for an exhaustive experimental campaign to be carried out at the JSI TRIGA reactor jointly by CEA and JSI researchers during the first semester in 2021. A series of test measurements using not fully appropriate fission chambers in reactor pulse operation was performed using a Keithley electrometer and the MONACO fission chamber acquisition system. Activation dosimetry measurements were performed for several thermal and fast neutron sensitive nuclear reactions. Photodiodes and silicon photomultipliers (SIPMs) have been tested in steady state and pulse operation modes. The presented results provide useful and promising experimental indications relevant for the design of the experimental campaign.

08 Decommissioning, Dismantling and Remote Handling / 137

#08-137 Advanced Sectorial Gamma Scanning of Radioactive Waste Drums with Spatial Reconstruction of Activity Distributions and Quantification of Uncertainties

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The non-destructive assay based on radiation detection techniques is cost-effective measure to characterize radioactive waste and serves to verify the conformity with safety requirements for waste packages. Gamma scanning is a standard widespread measurement technique for the non-destructive assay of radioactive waste drums to determine the nuclide-specific activity content of the waste product. In the past decades, the pre-dominantly used method is segmented gamma scanning (SGS), which is based on simplifying assumption of a uniformly distributed activity and a homogeneous waste matrix. The simplification leads to large model-error for a non-uniformly distributed activity distribution which reduces the accuracy of the measurement. The deviation from the calibration condition of SGS adds to the measurement uncertainty which in turn increases the upper limit of the confidence interval used to quantify the conservative estimate of the activity content. In some cases, this leads to a significant overestimation of the true activity, the difference constituting 'virtual activity' which is not actually present in the waste drum. The virtual activity caused by the large measurement uncertainties results in an excessive and inefficient exhaustion of activity limits for waste packages resulting in higher costs for disposal.
The companies AiNT and Mirion Technologies (Canberra) collaboratively developed a novel system named Advanced Sectorial Gamma Scanning (ASGS) for the gamma-spectrometric waste assay of waste drums with uniform and non-uniform activity distributions. The ASGS system is designed for commercial use allowing for

- high throughput,
- flexibility with respect to size of the waste drum,
- a high dynamic range with respect to the gamma radiation fields of the measured waste drum,
- a high degree of automation of the measurement and analysis procedure.

The analysis method to reconstruct the spatial activity distribution of nuclides within the waste was implemented in a software module named ECIAD (Efficiency Calculation for Inhomogeneous Activity Distributions). The software undertakes the calculation of the photopeak-efficiencies based on a geometric modeling of the measurement configuration, the waste drum, and the active matrix. The activity distribution within the active matrix is approximated with a discrete model of sources on a grid within the drum. The best estimator for the total activity is determined from the measurement data using an optimization method and considering the Poisson statistics of the counting measurement, uncertainty contributions in the self-absorption of the active matrix as well as the model uncertainty in localization of the activity distribution. The ECIAD software reports the measurement results including the uncertainties and the characteristic limits in accordance with the ISO 11929. This feature is highly relevant for the safety assessment and qualification of radioactive waste packages where a confidence level of the activity inventory needs to be declared.
An ASGS system was commissioned at the technical center of AiNT for experimental validation of the gamma scanning method and the analysis software. To this end, reference drums with three different matrix materials consisting of medium density fiberboard, polyethylene, and garnet sand with respective densities of 0.7, 0.9 and 2.3 g/cm^3 were measured with calibration sources located at various locations within the matrix. In addition, measurements using the method of segmented gamma scanning were performed for a direct systematic comparison of the measurement accuracy.

The validation of the ASGS system and the ECIAD software demonstrates the capability to determine the nuclide activities of a 'hot-spot' in 1.5 hours total scanning time with a significantly lower total measurement uncertainty than the standard method of Segmented Gamma Scanning. This paper presents the ASGS system design, the analysis procedure, the treatment of uncertainties and the results from the experimental validation.

10 Education, Training and Outreach / 138

#10-138 A Monte Carlo Study of Radiation Resistant Materials using JA-IPU Code for IV Generation Nuclear Energy Systems

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A Monte Carlo simulation study of four tentative radiation resistant materials (RRMs) at energies corresponding to proposed DT-neutron source up to 15 MeV using the JA-IPU code has been presented. The code incorporates full cascade development of both projectile neutron and energetic knocked on recoiling atoms. The code involves basic processes of elastic collision and ionization lose as well as many other practical features such as escape out, displaced number of
atoms and damage energy cross section, <σTdam>b.keV by an imparted neutron. In the code exclusive shielding features can be developed. From the preliminary results presented in this paper, it is revealed that the data of <σTdam>b.keV estimated for SiC is smaller than Zr3Si2 RRMs. Also, SiC shows better neutron reflection characteristic using simulated results of B/F ratio in comparison to the Zr3Si2 and V-alloy and AlN ceramics. From the DPA study also, SiC and V-alloy are better than Zr3Si2 and AlN. The study is useful for the development of manpower and data development for the IV generation nuclear energy systems.

03 Fusion Diagnostics and Technology / 139

#03-139 Design of Real-Time Electronics System for Two-Dimensional Hard X-Ray Diagnostics with Intensity Imaging and Energy Spectrum in EAST

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Hard X-ray diagnosis is the most important diagnostic in Experimental Advanced Superconducting Tokamak (EAST) for studying low-hybrid wave physics and fast electrons, which helps to improve the operating performance and guarantee the operation safety of EAST Tokamak. The Hard X-Ray diagnostics used 16×16-channels pixel array Cadmium Zinc Telluride (CZT) detector which has the features of high resolution, large absorption coefficient, high detection efficiency, small size, and can work under normal temperature conditions. A charge-sensitive pre-amplifier card is designed which is magnetic shielded and easy to maintain, with low noise and low signal crosstalk. The card has 32 channels and the actual conversion gain can reach up to 1mv/fc. Then, a compacted pre-amplifiers module is built which is composed of 8 preamplifier cards and one main-board (transform signal from detector to pre-amplifier). Moreover, the multi-channel high precision data acquisition and signal process system based on FPGA (made in China) and PXIe bus is designed, which solves the problem of time synchronization during the acquisition of multiple boards. The direct memory access(DMA) mode is used for hardware programming to achieve high-speed data transmission. Meanwhile, a software driver has been developed based on the Linux platform, and through the QT framework. The real-time display of the two-dimensional intensity distribution of hard X-ray radiation over time and the three-dimensional curve of the pixel signal spectrum over time have been realized, and the data is stored quickly to facilitate further offline analysis and processing. The electronics system has been tested and the results show that it has reached the design performance.

09 Environmental and Medical Sciences / 140

#09-140 Microradon project

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Radon is the most important radioactive noble gas existing in nature. It belongs to the radioactive decay chains of 238U and 232Th. As an inert gas, radon, and in particular the long-lived isotope, 222Rn (T1/2 = 3.82 day), has high mobility and can easily escape from materials containing traces of Uranium. For low energy and low counting rate experiment in particle and astroparticle physics (some neutrino experiments or direct dark matter research), the diffusion of radon, from the detector components, to active part of the experimental setup is very often the origin of the most important background. A radon concentration below a few tens of atoms of 222Rn /m3 is often a prerequisite for future experiments.

In this context, the MICRORADON project between three IN2P3 laboratories (CPPM-Marseille, CENBG-Bordeaux and IPHC-Strasbourg) has recently been launched. The aim of this project is to study the physics of radon (emanation, transport and capture) in extreme conditions (low temperature, high pressure, liquid gases, etc.) in connection with the requirements of future very low background experiments. Special attention is paid to the development of radon detectors in accordance with these conditions (low background, high sensitivity, high specificity…)

After the exposure of the basic mechanisms leading to the presence of radon in an experiment, we will present the main results obtained in the MICRORADON project and the perspectives for the future.

08 Decommissioning, Dismantling and Remote Handling / 141

#08-141 Simulation of the physical processes associated to the detection of beta particles with scintillating fibres

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In this study, we present an exhaustive model to simulate the detection of beta particles with scintillating fibres, based on a chronological follow-up of the information carrier. We manufactured a detector composed of a bundle of 100 BCF-10 scintillating fibres, which will be used to compare simulation and experimental results. A Monte-Carlo simulation model of the detector was generated using the MCNP6.2 code. To simulate more accurately the generation of scintillation photons inside the detector, particle tracking was used which allows us to use the incident beta particle stopping power instead of the energy deposition. A small energy deposition from a high-energy beta particle should not be processed as an incident low energy beta particle, as they will not yield the same number of scintillation photons. Using stopping power over energy deposition for beta particles enables to use theoretical scintillation laws, which take into account the ionisation quenching effect. Indeed, for low energy electrons, there is a discrepancy of the scintillation linearity. By integrating the quenching ionisation law into our model, we can evaluate more precisely the expected experimental spectrum at very low incident energies. Based on the literature, a more detailed photon trapping efficiency is calculated, which was found to be underestimated in the scintillating fibre technical sheet. The manufacturer value corresponds to a trapping efficiency based on meridional rays solely, while the calculated value also factors in the skew rays. Self-absorption in the fibre was estimated using an exponential loss at the scintillation wavelength. Interface between the fibre and the photomultiplier was evaluated using Fresnel and transmission equations. Finally, the probability distributions associated with the photocathode quantum efficiency and the photomultiplier gain were both taken into account to generate expected acquisition spectra. To do so, we used the thermionic signal, comparable to the amplification of a single photoelectron, to evaluate the photomultiplier gain distribution. This model will help in evaluating the detector applicability to specific experimental scenarios such as nuclear decommissioning and dismantling operations.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 142

#07-142 Ruggedized High Purity Germanium Detectors for in-situ Gamma Spectroscopy

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This paper presents the design and performance of turnkey and compact HPGe solutions, developed by Mirion Technologies (CANBERRA) for radionuclide identification outdoor and under harsh environmental conditions. Surveys can be undertaken under various weather conditions, in contaminated areas, underground or immersed under water (sea, rivers, pools), with fast on-site deployment and without compromising the performances and reliability experienced with laboratory-grade HPGe instruments.
In situ measurement is a privileged way of detecting radioactive contamination compared to analyzing samples in a distant, specialized laboratory. On the other hand High Purity Germanium (HPGe) spectrometers provide unmatched nuclide identification capability with the lowest minimum detectable activities thanks to its excellent energy resolution and high stopping power. However, HPGe instruments are not always of practical use on the field (because of the liquid nitrogen, weight and bulkiness).
These systems relies on advanced technologies such encapsulating the HPGe crystal under ultra-high
vacuum (UHV), different low vibration electrical cooler adapted to the crystal size, and advanced digital spectroscopy processor. Besides, their design includes hardened pressure housing, minimization of footprint and weight, sealing and water-tightness allowing easy cleaning from dirt or contamination.
Several examples of such ruggedized HPGe detectors will be described, illustrating the wide new range of applications permitted by these technologies: they are respectively designed for borehole measurement, high efficiency spectrometry from an aircraft or other vehicles, in situ sea and river contamination monitoring, as well as an ultra-compact detector for D&D or high count rate environments.
The sealed probe is an assembly consisting of a 80 mm in diameter shock proof and watertight external housing, including a 20% relative efficiency HPGe crystal mounted in an ultra-high vacuum (UHV) cryostat (CANBERRA proprietary technology) along with a compact cryocooler. A detailed view of the probe is shown in Figure 1.
The UHV encapsulation of the HPGe crystal allows partial thermal cycling without harming the crystal and degrading the detector performances, thus extending the life of the detector. The HPGe crystal cooling relies on a new compact inline cryocooler, operated with active vibration reduction in
order to keep the excellent intrinsic crystal energy resolution. This technology allows an increased portability, smaller footprint and safety of operation without the use of any flammable gas. It is also maintenance free and the reliability has also been probed with a large MTBF.
Higher efficiency versions of water tight detectors have also been designed and manufactured for continuous monitoring of contaminants in rivers, lakes or sea water (Figure 2). Such configurations can accept germanium crystals of several kg and relative efficiencies in excess of 100%). As the sealed probe, the heat generated by the electrical cooler is dissipated passively through the outer housing of the detector. A full set of monitoring and readout equipement and software is also provided.
These systems provide solutions to perform high resolution gamma spectroscopy similar to the performance achieved in laboratories with regular High Purity Germanium detectors (HPGe), but where no current products are compact or robust enough to be installed.
They features a FWHM of 2 keV at 1.33 MeV, 1.7 keV at 662 keV and below 1 keV at 122 keV. A complex mixture of nuclides can therefore be analysed in order, for instance, to distinguish anthropogenic radioactive sources from natural ones. The excellent resolution also allows for a significant increase in the minimum detectable activity (MDA), 3 to 5 times higher than scintillator-based detectors of similar sizes. For instance, simulations lead to a MDA of Cs-137 in water of less than 0.5 Bq / L for 600 seconds acquisition time.

05 Nuclear Power Reactors Monitoring and Control / 144

#05-144 Novel Model-Based approach for instrumentation and control of nuclear reactors

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Technological platforms dedicated for digital instrumentation and control of nuclear reactors are quite complex in terms of functionalities and devices. Hence, the design of these platforms requires high-level abstraction layers able to reduce the complexity, to rise the automation and to check the consistency between different development stages. The development of such systems is a challenging task that requires modeling of various components at different levels of abstraction and viewpoints, notably functional, hardware and software levels. In this paper, a new system engineering methodology is proposed to provide high-level models of different components and inter/intra-communication between them. These models are used for system specification, architecture design, performance evaluation or verification and validation. This approach focuses on the internal behavior of different components at different levels of abstraction in order to enable the interoperability of these components and to enhance cooperation between different stakeholders of the development process. An experimental setup has been carried out to validate this approach by customizing an open source model based engineering tool, Eclipse Papyrus, towards a significant reduction of system development cost in terms of engineering resources and equipment devices.

01 Fundamental Physics / 146

#01-146 Measurement of 93Nb(n,2n)92mNb reaction cross-section at different neutron energies for reactor application

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The natural niobium metal is well suited for atomic reactors due to its high temperature resistant, corrosion resistant, low long term induced radioactivity properties. Niobium alloys such as NbTi and Nb3Sn are used as super magnets in fusion reactors due to its super conducting properties. The 93Nb(n,2n)92mNb reaction is also used as monitor reaction. Therefore accurate data of 93Nb(n,2n)92mNb reaction cross-section needed. The cross-section of 93Nb(n,2n)92mNb reaction was determined at the neutron energies of 16.99±0.53 and 20.00±0.58 MeV by using activation and off-line γ-ray spectrometric technique. The neutron beam of different energy was produced by 7Li(p,n)7Be reaction and the neutron flux was determined using the 27Al(n,α)24Na monitor reaction. The measured cross-sections are compared with the theoretically calculated values based on TALYS-1.9 code and also with the evaluated and literature data.

08 Decommissioning, Dismantling and Remote Handling / 147

#08-147 Radiation Stability of Gadolinium Zirconate: A Nuclear Waste Immobilization Matrix

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The ease of formation of defect fluorite structure of Gd2Zr2O7 pyrochlore oxide addresses the remarkable radiation tolerance for the nuclear waste immobilization. The rare earth, particularly Gd, zirconates are effective neutron absorbers, thus advantageous for the disposal of plutonium. Several studies have been conducted on the Gd2Zr2O7 ceramic, the exact nature of grain size dependent structural phase transformation upon irradiation are not well captured. In this report, the grain size dependent radiation effects of microcrystalline Gd2Zr2O7 ceramic upon irradiation of 100 MeV I7+ ion at various fluences are examined and discussed. The grazing incidence X-ray diffraction, Field emission scanning electron microscopy, Raman spectroscopy, and high-resolution transmission electron microscopy are employed to investigate the microcrystalline Gd2Zr2O7 ceramic. FE-SEM results indicate that grain size and grain boundaries are clear in pristine Gd2Zr2O7. GIXRD results demonstrate that the amorphization fraction appears to be grain size and irradiation ion dose dependent. The Raman spectroscopy analysis exhibits that there is a distortion in atomic order/local disorder and increases with enhanced fluence. HRTEM results confirm the partial amorphization upon ion irradiation. We conclude that grain size plays a crucial role in the irradiation resistance of microcrystalline Gd2Zr2O7.

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#09-150 Development of a proton bunch monitor for accurate particle therapy treatment verification

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Treatment verification is expected to improve targeting precision in particle therapy. A promising technique to achieve this goal is the detection of prompt gamma rays emitted along the particle tracks inside the patient. The range of the particle beam can be inferred by determining the time distribution of these gamma-rays relative to the radio frequency of the accelerator, a method commonly referred to as Prompt Gamma-Ray Timing.

However, the translation of this method into a clinical setting is currently hindered by instabilities of the phase relation between the arrival of the proton bunches and the radio frequency of the accelerator. These instabilities include two effects, which have been studied at the clinical treatment facility of the University Proton Therapy Dresden. Firstly, a long-term drift of the proton bunch phase relative to the radio frequency in the order of several hundred picoseconds per hour was observed, which may be caused by small temperature changes in the cyclotron's magnet resulting in magnetization variations in its iron parts. Secondly, strongly damped oscillations in the mean of measured prompt gamma-ray timing spectra with an amplitude in the order of few hundred picoseconds occur for about two seconds after each change of the particle energy during pencil beam scanning. This oscillation is caused by ramping the acceleration voltage back to its nominal value, which is reduced between energy layers to minimize the dark current of the accelerator and the resulting excess dose to the patient.

While the former effect is only of secondary importance for the treatment due to its comparably long time scale, the phase oscillation has a considerable negative impact on the accuracy of the Prompt Gamma-Ray Timing method, which has to detect time shifts in the order of a few picoseconds for the detection of millimeter range changes. Therefore, the development of a method to monitor the arrival time of the proton bunches independently from the accelerator radio frequency, a so-called proton bunch monitor, is crucial.

To this end, a bunch monitor prototype was developed consisting of scintillating fibers placed in the halo of the proton beam. The fibers were read out on both ends by silicon photomultipliers. A thick acrylic glass target with cylindrical air cavities of varying thickness and different tissue-equivalent inserts was irradiated with protons of clinically relevant energies and typical beam currents. The mean proton arrival time, determined from the time spectra of the proton bunch monitor, was used to correct the prompt gamma-ray timing spectra acquired by Ø2"x2"CeBr3 high-resolution scintillation detectors. This correction allowed to resolve differences in the prompt-gamma ray timing spectra acquired with the different cavities and inserts.

In conclusion, the developed proton bunch monitor was successfully integrated to the Prompt Gamma-Ray Timing method and is expected to enable the clinical application of this method for clinical treatment verification in particle therapy.

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#07-151 Tests of various scintillator detectors in selected monoenergetic neutron beams

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The neutron detection using organic scintillators is a perspective technique for online neutron detection independent of the use of 3He detectors. This is an important issue due to the increas-ing need of neutron detection in homeland security, nuclear energy and also in nuclear nonpro-liferation applications. One of the important parameters in organic scintillation spectroscopy is the energy resolution of a scintillation detector, which for example in homeland security can distinguish between fission neutron source and radioisotope source. The measurement of pro-ton recoil spectra in monoenergetic neutron fields can be used for both validations of neutron response function and also for the determination of resolution of new neutron scintillation ma-terials. These responses were compared with the stilbene response. The stilbene was used for comparison because it is the best scintillation material for fast neutron spectral flux measure-ments in neutron gamma mixed fields. A set of monoenergetic neutron sources in the range of 1.5 – 19 MeV in PTB Braunschweig were used in the testing of detectors. Hamamatsu R329-02 photomultiplier was used to collect the scintillation light. It is connected to the preamplifier that uses different amplification to increase the dynamic range. This also allows us to check on the linearity of the output using calibration sources Co-60 and Cs-137. A verified measurement apparatus NGA-01 was used for the evaluation of the signal of the photomultiplier. This sys-tem can work with a high impulse count per second (>10^5) without any dead time. NGA-01 contains integrated PSD circuits based on the integration method. The measurements show that the best resolution was reached with stilbene, a satisfactory resolution with p-terphenyl, why in the case of the EJ299-33A scintillator the resolution is not very good. On the other hand, due to the low price of plastic scintillator EJ299-33A, the resolution is good enough for large area de-tectors applicable in homeland security applications.

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#11-152 Multi-platform software suite for data acquisition and analysis with hybrid pixel detectors of the Timepix family

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Timepix detectors (Timepix, Timepix2 and Timepix3) are hybrid pixel detectors that use a square matrix of 256×256 pixels with 55 $\mu\mathrm{m}$ pitch. Thanks to their versatile design, compact form factor, desirable energy and time resolution, they have found applications in a variety of fields, e.g. radiation field characterization, tracking, dosimetry and imaging. In this contribution, we present a novel suite of software tools designed to conveniently operate and process data from detectors of the Timepix family. The suite includes a parallelized data acquisition program that is compatible with Katherine readouts. Relying on a centralized backend to store device configuration, the program delivers plug-and-play experience to novice users, and allows experienced users to apply frequent data pre-processing approaches online, for instance clustering, cuts, Time-over-Threshold calibration and time-walk correction. This makes the software particularly suitable for operating radiation monitoring networks in applications, where access to real-time information is critical, e.g. in nuclear decommissioning, medicine and others. Producing structured measurement files that follow a well-defined format based on text or binary encoding, the program's outputs are easily interoperable with the rest of the toolkit as well as popular scripting languages like Python. Recorded measurements can be analyzed using a viewer program, which allows to easily aggregate, filter and visualize large data volumes based on complex user-defined rules. To increase compatibility with older tools, the presented suite also includes a converter utility, which facilitates data migration from legacy formats. Based on the C++ Qt framework, all presented programs offer high performance as well as extensibility via pluggable architecture, permitting easy future adoption of new hardware, such as Timepix4. Binaries are available for download upon request for the majority of Linux distributions, Microsoft Windows and macOS.

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#05-157 Radiation Hardness Test of a Silicon Detector under Radiation Dose Rate of Nuclear Power Plant for In-Containment Coolant Leakage Detection System

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An influence evaluation by background radiation on a silicon detector, which will be used to detect a coolant leakage, and be installed in a containment building of a nuclear power plant, was performed and the result was discussed. The detector that consists of a silicon sensor and preamplifier mounted in a shielding structure which composed of a 5 cm lead cylinder will be installed in an annulus zone that is influenced by background radiation (neutron and gamma ray) from an operation of a nuclear reactor. Absorbed dose rates on a silicon sensor and preamplifier were calculated as 4.17 mGy/hr and 2.1 mGy/hr, respectively, by Monte Carlo N-Particle (MCNP) simulation. Data of background radiation had referred to a Final Safety Analysis Report (FSAR) of a nuclear power plant in the Republic of Korea.
A silicon sensor and preamplifier were irradiated by a 60Co gamma radiation source equipped in a facility of Korea Atomic Energy Research Institute Advanced Radiation Technology Institute (KAERI ARTI) of the Republic of Korea. A 210Po alpha source was used as a check source to evaluate a state of a function of the detector during gamma irradiation. Absorbed dose rates were about 22.92 mGy/hr and 6.6 mGy/hr on silicon sensor and preamplifier, respectively. Before and after gamma irradiation, a counts rate from the check source wasnt changed (from 18.4 cps to 18.4±0.2 cps after irradiation), and any degradations of function also werent observed. Even more harsh condition than calculated dose rates referred by the condition of background radiation of in-containment, the silicon detector maintained the ability of function of charged particles detection. Based on the result, it has been demonstrated that a silicon detector is a suitable detector for detecting charged particles from a leaked coolant even during interfered by the background radiation of a primary system of a nuclear power plant.

Keywords: Silicon detector, Irradiation test, Coolant leakage detection system

04 Research Reactors and Particle Accelerators / 158

#04-158 Fragment production in 200MeV proton reaction on 232Th, experimental and simulated data.

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This paper shows the results from the experiment at Joint Institute for Nuclear Research in Dubna and was carried out in 2014. The sample, using a special device, were placed in the accelerator chamber at a radius corresponding to the energy of protons 200 MeV at the current of 0.3 A. The position of the 200 MeV beam was determined by placing an aluminium foil inside the accelerator and perpendicular to the proton beam and evaluating the activity of the foil. For the target, foils of Th232 were used with a thickness of 100 microns and a weight of 149.5 mg placed between two Al foils with thickness of 50mkm. The foil area was 1.5 cm2. After target irradiation, foils where removed from the device and moved to YSNAPP-2 complex where was separately measured the spectra of radiation from the Th and Al foils by HPGe detectors of the CANBERRA company with efficiency 18% and resolution of 1.9 keV in the line 1332 keV. The processing of the gamma spectra was carried out using the DEIMOS32 program to find positions of peaks, their areas and other parameters. The identification of the nuclei formed in Th232 samples as a result of nuclear reactions with protons and their reaction rates was carried out using a set of scripts based on the Ruby programming language. For modelling of the experiment MCNP 6.1 was used. The main purpose was to simulate creation of parent residual nuclei during proton irradiation, their distribution and escaping from the Th foil and calculate the cross-section of the residual nuclei. The experimental data on the fragmentation of the Th232 nucleus under the influence of protons in the energy of 200 MeV have been processed As a result we compared the calculated and simulated results, found residual nuclei and their cross-sections.

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#04-160 Online optical refractive index measurement in research reactor core

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There is a growing interest in fiber optic measurements for applications in radiation environments. They can be used to monitor environmental parameters such as temperature, size, pressure, chemical composition, irradiation doses and dose rates…. Often, the developed systems imply no propagation of the light beam outside the fiber, but for some applications, fiber optic is combined with an optical system that collects or focuses the light beam. The question then arises not only about the impact of the RIA (Radiation Induced Attenuation), which has already been largely studied, but also of the radiation induced change of the refractive index of the used glasses, which is a determining value for the optical function. Studies of refractive index variations under irradiation are already being carried to evaluate their impact on imager performances [1]. Also, for the development of optical sensors for applications in research reactors core, CEA is preparing an irradiation in the BR2 reactor of SCK.CEN in Belgium. We implemented an on-line refractive index measuring device in order to test various glasses that can be good candidate to take place in a hardened optical system. The assembly uses an interferometric measurement system, which is a challenge because of its very small size (diameter of 9 mm), the impossibility of making optical adjustments once installed, and the goal of monitoring these changes at an elevated high temperature. The refractive index variation of silica for example under high fluence – 1019 n/cm² (E>1MeV) and several GGy in gamma- is in the range of some 10-3 [2]. The precision required for the measurement is then very ambitious for such difficult conditions. In addition, the variation in density observed at high neutron fluence on silica causes an elongation of the optical components [3], the targeted online measurement therefore becomes an optical path measurement (product of the length by the refractive index) and no longer just the one of the refractive index. Apart from this online monitoring, we expect to perform post-mortem measurements of irradiated sample dimensions which could help to dissociate the refractive index and length variations. In this work, we will present the planned irradiation and the first results of the index change measurement under the effect of temperature. The index variation between 20℃ and 350 ° C is of the same order of magnitude as the variation of the refractive index of silica expected under reactor core neutron fluence. This work is now going on in the framework of a PHD at the French Alternative Energies and Atomic Energy Commission (CEA) in collaboration with the Laboratoire Hubert Curien (LabHC) of the University of Saint-Etienne and also the STIL company [4].

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#04-161 Confocal chromatic sensor for displacement monitoring in research reactor

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Confocal chromatic microscopy is an optical technique allowing measuring displacement, thickness, and roughness with a sub-micrometric precision [1]. Its operation principle relies on a wavelength encoding of the object position. The light emitted by a polychromatic source is transported by an optical fiber up to an optical system, called optical pen only composed of passive components and presenting a strong chromatism. That means that the various wavelengths will be focused at different positions by the optical system on the optical axis. The surface of the object under test will reflect all the wavelengths but only a small fraction of the reflected spectrum will be focused back on the cleaved end of the optical fiber and then will be able to be guided towards an optical analyzer. The change in the spectral properties of the analyzed signal provides an information about the distance of the interface located within the measuring range of the sensor. Historically, this class of sensors has been first developed in the 90's by the company STIL based in the south of France.

Of course, this sensor can only operate in a sufficiently transparent medium in the considered spectral domain. But it presents the advantage of being contactless which is a crucial advantage for some applications such as the one of measuring displacement in the core of nuclear research reactor: cladding swelling measurement for instance.

The extreme environmental conditions encountered in a nuclear reactor -high temperature, high pressure, high fluency irradiations, strong vibrations and surrounding turbulent flow- can affect the performances of those optical system and mitigation techniques need to be put in place to optimize the sensor response for this specific environment. Another constraint concerns the small space available between the rods to be monitored, implying the challenge to conceive a very small sensor able to operate under these constraints.
This development is made in the framework of a PHD at the Commissariat à l'Energie Atomique (CEA) -in collaboration with the Laboratoire Hubert Curien (LabHC) of the University of Saint-Etienne and also the company STIL.

Among various scientific and technological locks, the first to be considered is the Radiation Induced Attenuation (RIA) phenomenon of the fiber and of the bulk optical components constituting the optical system. Indeed, glasses and fibers tends to darken when exposed at high neutron fluencies decreasing the light transmission efficiency. In the state of the art, it has already been shown that some spectral regions show huge RIA levels under high level of mixed gamma/neutron irradiation. In the visible, the absorption band peaking at 630 nm and caused by the Non-Bridging-Oxygen Hole Centers (NBOHCs) can strongly alter the sensing as well as the growing of the vibration band at 1380 nm of hydroxyls that could appear when NBOHCs are passivated by hydrogen. Basic mechanisms at the origin of the RIA are known [1, 2] but still only a few experimental data at the very high fluency associated with the application (neutron fluency about $10^{19} n/cm^2$ with gamma doses of more than one GGy, and also high temperature) are available, making difficult to really assess the RIA impact of the sensor performances.

However, RIA is not the only scientific lock; elevated temperature can also affects the optical system. Indeed, if the sensor exploits the chromatism, related to the refractive index dependence to the wavelength, the temperature is also able to modify the refractive index causing a direct shift of the probing spectral range but also dilatation of the lenses and optical aberrations. Irradiations too can

alter the well behaviour of the sensor -because of compaction. We also have to stay cautious about water under pressure with turbulent flow and vibrations.

We will present the detailed specifications and a preliminary analysis of the constraints, with a special focus on the RIA at high fluency.

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03 Fusion Diagnostics and Technology / 162

#03-162 Data acquisition system prototype of the ITER diagnostic Divertor Neutron Flux Monitor testing at research nuclear facilities

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The Divertor Neutron Flux Monitor is the one of the neutron diagnostics of the International Thermonuclear Experimental Reactor ITER. This diagnostic consists of three subsystems. Each subsystem consists of the detector module includes six fission chambers and the data acquisition system. Detector modules are placed on the inner shell of ITER Vacuum Vessel under the Divertor Cassette. Expected neutron flux at the Divertor Neutron Flux Monitor detector modules position is from 1E3 to 1E13 n sm-2 s-1. The multidetector module is used to solve the task of the neutron flux measurements at ten orders of magnitude with 1 ms of time resolution. Divertor Neutron Flux Monitor module consists of two detector units. The first detector unit concludes three-section fission chamber with 235U as the fissile material and the second concludes three-section fission chambers with 238U. To demonstrate the possibility of the neutron flux measuring in a wide dynamic range using such a detector module and to evaluate the characteristics of the data acquisition system prototype a number of tests were carried out under conditions of the intense neutron radiation. Detector units, which are similar to those planned for use on the ITER, and the mobile version of the prototype were used for the tests. The tests were carried out at the plasma neutron diagnostic stand based on the NG-24M neutron generator with yield of 1E11 n s-1, and at the IBR-2 pulsed reactor of the Joint Institute for Nuclear Research with neutron flux of 1E9 - 1E12 n cm-2 s-1 depends on monitor position at peak power 1.6-1.85 MW. During tests at the plasma neutron diagnostic stand the mobile version of the data acquisition system prototype functionality was checked. The averaged pulse shapes and pulse-height spectra of all fission chambers were recorded. The data for the calibration of the data acquisition and processing system measuring channels were obtained. During the tests at the IBR-2 pulsed reactor the signals from the measuring channels of the subsystem were recorded while the neutron flux was changed. This paper shows the test results and discusses techniques and methods of the DNFM neutron diagnostic subsystem calibration.

05 Nuclear Power Reactors Monitoring and Control / 163

#05-163 Fiber Bragg Grating based sensors for applications in harsh environments.

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Nowadays, the optical fiber sensors (OFSs) have attracted the interest not only of researchers but also of industries, for their applications in harsh environments characterized by high temperatures and/or radiations. The Fiber Bragg Grating (FBG) based technology is one of the most widely studied under radiation [1]. Even today it presents qualities unbeatable by other OFSs, such as its high acquisition repetition rate, that can reach at least 20 kHz with an adequate acquisition system. Even if it is not a distributed sensor, as the ones based on scattering phenomena, it is easily multiplexable, which means that several point sensors can be placed in series on a same fiber. However, the research on the radiation-response of different types of gratings is still active, since the radiation degrades their performances through two phenomena:
- Radiation-Induced Attenuation (RIA), which reduces the signal transmission,
- Radiation-Induced Bragg Wavelength Shift (RI-BWS), which introduces an error on the sensing parameter measurements.

In this work, we will compare two of the best FBG technologies that are considered for applications in severe environments with the more classical FBG type.
o The Rad-Hard type II gratings (RH-type II-FBGs) are manufactured with the patented procedure developed by Areva and the Laboratoire H. Curien [2]: these type II FBGs were written with a femtosecond laser at 800 nm in radiation-resistant fibres, having a fluorine doped or pure-silica-core. Before the irradiation they were subjected after their writing to a short thermal annealing at 750℃. The chosen manufacturing process parameters combined with this pre-treatment enable to reduce the RI-BWS below 10 pm (error of about 1℃ on the temperature measurements performed by the sensor) after MGy dose of irradiation [3].
o The regenerated gratings (R-FBGs) are obtained by treating at high temperatures (higher than 650℃) a strong type I grating inscribed in a H2-loaded Ge-doped fiber by an UV laser (continuous wave or pulsed in the nanosecond regime) [4]. The seed grating quickly disappears during the annealing and, in its place, another grating, the R-FBG, appears. The new grating is resistant to high temperatures and, preliminary results show that its radiation resistance increases at high temperatures, with a RI-BWS of only 10 pm, at irradiation temperature higher than 150℃ at 0.1 MGy dose [5].
o The type I-UV FBGs are the most classical grating type: the FBGs were written with a cw UV laser (at 244 nm) in a H2-loaded Ge-doped fiber and thermally stabilized at 120℃. Contrary to the two previous gratings, this one does not resist to high temperature or to radiations: the Bragg peak shifts with the dose; moreover, the higher is the temperature, the lower is the BWS [1].

It is worth noticing that both R-FBG and RH-type II FBG undergo a treatment at high temperature, however, the mechanisms at the origin of these gratings are very different and still under investigation. Moreover, whereas the R-FBG has to be written into a photosensitive fiber, as the Ge-doped one, the RH-type II-FBG can be inscribed into a rad-hardened fiber, as the F-doped one, which is less affected by the RIA compared to the Ge-doped one.
We report in Fig. 1 a first example of the Bragg wavelength shift induced by the gamma-rays at room temperature (RT) on the three FBGs. The experiments have been conduced at the IRMA facility of the IRSN (Saclay, France).

Fig. 1. Bragg wavelength shift, corrected for temperature fluctuation (recorded by thermocouples) induced by gamma-rays on the three types of gratings at RT, dose-rate being 1.3 kGy/h, up to TID of 500 kGy.

The BWS of the RH-type II-FBG is only few pm, as expected for this rad-hard solution. The type R and type I-UV FBGs, instead, red-shift of ~30 pm and 50 pm, at the total ionizing dose (TID) of ~500 kGy. The reported curves are a bit noisy, probably because of a non-perfect correction of the nights

and days temperature fluctuations.
In the final paper, these three grating types will be irradiated under gamma- and X-rays at different temperatures up to the MGy dose level, in order to investigate the real performances of the sensors based on these technologies. We will also investigate different pre-loadings of the Ge-doped fiber for the R-FBGs inscription, since our group recently showed the possibility to regenerate FBGs written in D2-loaded Ge-doped fibers [6].

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09 Environmental and Medical Sciences / 164

#09-164 Multi-Feature Treatment Verification in Particle Therapy

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Particle therapy constitutes a promising and rapidly developing method in modern cancer treatment. In order to exploit its full potential, however, it requires detailed dose verification.
Although the applicability of in-beam positron emission tomography and prompt gamma rays has already been demonstrated in patients, range verification is not yet part of the clinical routine in particle therapy. This is due to not only the methodological limitations of previous systems, but also to commercial, clinical and physical boundary conditions.
In pencil beam scanning, the state-of-the-art treatment method in particle therapy, the number of secondary particles (essentially positrons, prompt gamma rays and fast neutrons) available per spot ($\Delta t = 10$ ms to 100 ms) is limited. This leads to statistical accuracy limits for verification systems exploiting these secondary particles as range probes. The development of a clinically useable treatment verification system requires gathering as much information about the local dose, as possible.
The instantaneous fluence rate of prompt gamma rays reaching 5×10^6 cm^{-2}s^{-1} to 10^8 cm^{-2}s^{-1} challenges modern data acquisitions connected to monolithic inorganic scintillators with typical sizes used in present verification systems. In order to reduce the detector load, and also with regard to the ever higher intensities of next generation medical accelerators, future systems have to be more granular.
Multi-Feature Treatment Verification combines and extends established methods (prompt gamma-ray imaging, spectroscopy, timing, etc.) in order to achieve higher reliability and performance. This idea was taken up by the NOVO project and expanded by a multi-particle approach. The NOVO (i.e. *N*eutron and gamma-ray imaging with quasi-monolithic *o*rganic detector arrays – a novel, holistic approach to real-time range assessment-based treatment *v*erificati*o*n in particle therapy) consortium is a large collaboration of medical, nuclear and detector physicists, nuclear engineers, and mathematicians, which aim to develop a holistic real-time treatment verification system in particle therapy.
Elements of a potential multi-feature/multi-particle treatment verification multi-channel system were characterized in a double time-of-flight experiment at the pulsed photo-neutron source *n*ELBE (neutrons @ *E*lectron *L*inac for beams with high *B*rilliance and low *E*mittance). The essential properties (time resolution, light yield, detection efficiency and pulse shape discrimination) of an EJ-276 plastic scintillator were determined.
The very first experimental results show that the time resolution ($\Delta T < 400$ ns) of a $20 \times 20 \times 200$ mm^3 EJ-276 plastic scintillator with double-sided readout will reach the high demands of such a proposed range verification system. However, the determined quality of the pulse-shape discrimination, the energy resolution and the quite high neutron detection threshold of above 200 keV show that the light yield of this type of scintillator is not high enough to be used in a multi-feature-based treatment verification system. Particle transport calculations with MCNP6 and GEANT4 were performed to confirm the experimental results of a single detector element. Furthermore, they also show a promising measurement accuracy of a multi-channel overall system.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 166

#07-166 Simulation and Development of Prototype Simplified Neutron Scatter Camera for Nuclear Safeguards Applications

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Key in the international safeguards regime is the use of radiation detectors to track and characterize nuclear material. A relatively new area of interest in detector design is directional detectors: detectors that can report information on radiation source location and distribution along with data on source strength and identity. Neutron scatter cameras are a type of directional neutron detectors that rely on multiple neutron scatters to generate images that can reveal the direction and distribution of neutron sources. Fast neutron cameras which have recently been developed rely on multiple detector volumes and make use of neutron time-of-flight measurements. These designs, though effective in localizing the source direction, relies on a large amount of detection and electrical equipment, thus increasing size, cost, and complexity of the systems to unreasonable levels for some applications. This project seeks to develop a compact scatter camera that is less expensive than systems relying on multiple detector volumes. Crucially, two components and capabilities are needed to achieve this: fast scintillation detection materials and picosecond electrical pulse timing. Utilizing such electronics, distinguishing between scintillation light pulses generated by the same neutron within one detector volume is possible. An MCNPX-PoliMi model of such a system has been developed to guide prototype designs. A cube of EJ230 fast plastic scintillator and six photomultiplier tubes (PMTs) were used to construct the prototype camera that localizes neutron sources based on the principle of cone back projection. Neutron scattering positions within the detector volume are found by comparing the timing and quantity of light arriving at PMTs mounted to opposing faces of the scintillator volume. The use of a digitizer with a sampling rate of 5 GSPS allows for the identifications of secondary features on light pulses, indicative of secondary scattering events. Prototypes of the system in one, two, and three dimensions have shown promising initial results when coupled with a script that algorithmically identifies candidate neutron double scatter events and back projects probability cones in the direction of possible sources. Imaging resolution/quality, double scatter efficiency, and cost for the system are quantified. Paths forward for further improvement of a future system based on this camera' operating principles are discussed.

04 Research Reactors and Particle Accelerators / 167

#04-167 First in-core gamma spectroscopy experiments in the zero power reactor CROCUS

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Gamma rays in nuclear reactors, arising either from nuclear reactions or decay processes, significantly contribute to the heating and dose of the reactor components. Zero power research reactors offer the possibility to measure gamma rays in a purely neutronic environment, allowing for validation experiments of computed spectra, dose estimates, reactor noise and prompt to delayed gamma ratios. This data can contribute to models, code validation and photo atomic/nuclear data evaluation. To date, most experiments have relied on flux measurements using ion chambers or spectrometers set into low flux areas. The CROCUS reactor allows for flexible detector placement in the core, and has recently been outfitted with gamma detection capabilities to fulfill the need for in-core gamma spectroscopy, as opposed to flux. In this paper we report on the experiments and accompanying simulations of gamma spectrum measurements in a zero power reactor core. The CROCUS reactor is a two-zone, uranium-fueled light water moderated facility operated by the Laboratory for Reactor Physics and Systems Behaviour (LRS) at the Swiss Federal Institute of Technology Lausanne (EPFL). With a maximum power of 100 W, it is a zero power reactor used for teaching and research. Herein we also introduce, in detail, the new LEAF system: A **L**arge **E**nergy-resolving detection **A**rray for **F**ission gammas. It consists of an array of four detectors – two large 127x254 mm Bismuth Germanate (BGO) and two smaller 12x50 mm Cerium Bromide ($CeBr_3$) scintillators. We describe the calibration and characterization of LEAF followed by first in-core measurements of gamma ray spectra in a zero power reactor at different sub-critical and critical states and different locations. The spectra are then compared to code results, namely MCNP6.2 pulse height tallies. We were able to distinguish prompt processes using photon production tracking, and delayed peaks from decay databases. The results indicate the possibility of on line isotope tracking and burn-up validation. We provide the data as validation means for codes that attempt to model these processes for energies up to 10 MeV. We finally draw conclusions and discuss the future uses of LEAF.

08 Decommissioning, Dismantling and Remote Handling / 169

#08-169 Radiation Detection Instruments Ideally Suited for Robotic Autonomy and a Novel Approach to Reduce Collimator Field-of-View

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There are many applications in nuclear environments that benefit from the use of robotics, such as the response to accidents, decommissioning and routine monitoring. Robots have some advantages over humans in the aforementioned contexts, for example: they are often better at repetitive tasks or working in confined spaces. Such situations in nuclear environments can also present a radiological or chemical risk, which can be mitigated by deploying robots in the place of people. As the needs of the situation can be very diverse, the robots available match this with a multitude of shapes and methods of locomotion, wheeled, tracked, aerial, submersible etc. To navigate their environment, robots are equipped with sensors, which can be optical and infrared cameras or Light Detection And Ranging (LiDAR), for example. Radiation detection instruments are almost essential when operating in a nuclear environment, as they can provide valuable information when decommissioning or responding to an accident. When these are combined with a tether-less robot with a degree of autonomy, more complex tasks can be undertaken. Robotics in nuclear environments have already demonstrated significant use: e.g., submersible use at Sellafield, aerial mapping over the Chernobyl Red Forest and tracked robot deployment at the CEA etc. Unfortunately, radiation instruments are seldom designed to fit within the strict limits for robot use, such as power, weight, size and operation by onboard computer. Along with improvements in hardware, there is a wealth of opportunity for better onboard data analysis going beyond mapping count-rate data.

This paper will report on the testing and deployment of two γ-ray detectors adapted for robotics and a novel data analysis method for reducing the Field-of-View (FOV) of collimation. One of the detectors is a 10-mm cubic $CeBr_3$ scintillator with photomultiplier tube (PMT), paired with a Red Pitaya STEMlab 125-14 which is repurposed as a multichannel analyser (MCA). The second detector is a combination of NaI:Tl with a PMT and a Brightspec bMCA MCA. These were selected as they are capable of spectroscopy, compact and well suited to a variety of robot platforms and applications. The associated software for them has been adapted and written to fit a number of criteria, these are: to use a widely adopted robot middleware, minimal performance loss on the robot and negligible additional work for the operator. The robot middleware used is the Robot Operating System as it is used by the robots in the collaboration. Both detector setups have been tested in a laboratory and demonstrated excellent performance, for example, achieving 5.4% and 8% resolution for $CeBr_3$ and NaI:Tl respectively when counting ^{137}Cs.

The detectors underwent preliminary field-testing at a low-level waste (LLW) drum store using a Clearpath™ Husky robot in a routine monitoring application. During field-testing, the $CeBr_3$ detector was placed inside a thin-walled collimator and mounted onto the Pan-Tilt Unit on the robot to be able to point around the environment. The NaI:Tl detector was mounted in a fixed location without collimation. Field-testing was successful in satisfying the above criteria, and the preliminary data from the $CeBr_3$ detector was combined with LiDAR. This combination was used to overlay the radiation data onto the 3D LiDAR map to highlight the activity of the adjacent drum.

For γ-ray detectors to be able to localise a source of radiation it is usually necessary for them to be surrounded by a dense material such as lead or tungsten. Producing a narrow FOV collimator means adding additional dense material to reduce the angular magnification, but this increases the mass of the system significantly. This increase in mass is a particular problem if the collimator is mounted to a Pan-Tilt Unit, which often have limited payload restrictions. An alternative technique has been developed here which processes the output from the barrel collimated detector by fitting it to a transform function, to test the hypothesis that this approach can aid source location. The arrangement is a 2-mm lead barrel collimator, the detector being panned over a single γ-ray point source. The detector is panned incrementally using only a single dimension as a proof-of-concept for the technique. This method relies on the response of the transform function of a step input, with the accumulated counts at each increment fitted to this response. The transform is then applied in reverse, to recover the response (now being a step), revealing the location of the point-source in the data. Preliminary results demonstrate that the transform technique is successful. This is currently being tested using a collimator with ~90° FOV and a point-source at 0.5 m.

09 Environmental and Medical Sciences / 170

#09-170 A whole gamma imaging prototype with a two-layer depth-of-interaction GSO scatterer detector

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Whole gamma imaging (WGI) is our new imaging concept which combines PET and Compton imaging. By inserting a scatterer detector ring into a PET ring, two different modalities can be evaluated on the same platform. We developed a WGI prototype and demonstrated Compton imaging of 909 keV photons emitted from a ⁸⁹Zr-injected mouse. While Compton imaging can avoid theoretical limitation in PET resolution due to positron range and angular deviation, the quality of obtained Compton images was worse than that of PET images. Therefore, in this work, toward our final goal of achieving Compton imaging to outperform PET, sensitivity and energy resolution of the scatterer detector were improved.
Depth of interaction (DOI) detection was introduced for the scatterer to improve the sensitivity while suppressing the parallax error. GSO crystals sized at 2.85 × 2.85 × 7.5 mm³ each were used. The GSO crystals were arranged into a 7 × 7 array for the 1st layer and an 8 × 8 array for the 2nd layer, and these layers were stacked with the staggered arrangement. Each 2-layer GSO array was coupled to a multi-pixel photon counter (MPPC) array module (3 × 3 mm², 8 × 8 array, sub-pixel size of 50 × 50 µm², Hamamatsu, S13360-3050PE). The redesigned scatterer detector ring had four rings, and each ring had ten GSO detectors. The total axial length was 10.4 cm, which was twice as long as the previous WGI prototype. The newly developed scatterer detector ring was inserted into the PET ring, which was the same absorber ring used in the previous WGI prototype.
Energy resolution at several energies, angular resolution measure (ARM), and Compton imaging sensitivity at 662 keV gamma ray were investigated. In addition, imaging tests using a ⁸⁹Zr Derenzo phantom and a ⁸⁹Zr-injected mouse were conducted. The PET and Compton image were compared.
The new WGI prototype showed a better energy resolution (e.g., at 202 keV, 12.7% vs. 15.9%) than the previous WGI prototype. However, the improvement in the angular resolution was not so remarkable (6.4 deg. vs. 6.7 deg.). We think that the increased crystal size may have impacted on compromised improvement in the angular resolution. On the other hand, doubling the axial length of the scatterer in addition to the use of the thick crystals resulted in 1.5 times higher Compton imaging sensitivity. For the ⁸⁹Zr Derenzo phantom images, the PET image resolved the 2.4-mm rods while the Compton image resolved the 3.2-mm rods roughly. For the ⁸⁹Zr-injected mouse images, the Compton imaging result which was close to the PET image was obtained.
In conclusion, the new WGI prototype showed a promising basic performance. Further improvement of the WGI system will be performed toward the goal of achieving Compton imaging that outperforms PET.

11 Current Trends in Development of Radiation Detectors / 171

#11-171 Novel modular X-ray detection system based on Silicon Drift Detector

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A new X-ray detection system composed of an arrays of multichannel Silicon Drift Detectors (SDDs) is presented. This custom-designed multilayer can be adapted to the required geometry and caracteristics. The presented layout is composed by a 8 monolithic array of SDD each with 8 cells with a total area of 570 mm^2, ultra-low noise front-end electronics, integrated dedicated acquisition system and software, digital filtering, control and stabilization of the temperature. This 64-channel integrated detection system is optimized to work in an energy range of 3-30 keV. It allows a count-rate of 15.5 Mcount/s and an energy resolution below 170 eV FWHM at the Mn 5.9 keV Kα line at room temperature.

06 Severe Accident Monitoring / 172

#06-172 Long-term transmission characteristics of CYTOP fiber exposed by gamma radiation

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Polymer optical fibers (POFs) have been attracting substantial attention from research and industry thanks to particular mechanical properties, simple handling, biomedical compatibility and safety. Among POFs, cyclic transparent amorphous fluoropolymer (CYTOP) fiber demonstrates radically low attenuation in the telecom transparency windows of 850 and 1310 nm, thus becoming an efficient solution for short distance communication links. Besides fiber communications, CYTOP fiber has been also intensively studied for various sensing applications: fiber Bragg gratings, Brillouin scattering and intermodal interference are good examples of investigated sensing principles.

A particular topic of research is the use of POFs in radiation environments, including possible solutions for gamma radiation dosimetry. A basic step in this direction is the study of gamma radiation influence on fibers' properties aiming for evaluation of their transmission characteristics degradation. Recently, the gamma-radiation induced attenuation (RIA) in CYTOP fibers was extensively investigated. The RIA was measured in the visible and near-infrared spectral bands, the influence of temperature and relative humidity on irradiated fibers was studied, and the distributed gamma-radiation sensor was proposed. However, evolution of the RIA properties versus time was not considered.

In this work, we focused on the long-term (residual) RIA of the CYTOP fiber. We irradiated 2-m samples of the graded-index CYTOP fiber with core diameter of 50 μm (Chromis Fiberoptics) to 1, 5, 20 and 50 kGy doses, and measured the RIA five months after irradiation. The experimental setup was based on broadband supercontinuum light source Super K Compact, 450-2400 nm (NKT Photonics) and optical spectrum analyzer Yokogawa AQ6374, 350-1750 nm. We utilized cut-back technique and the attenuation spectrum of unirradiated CYTOP fiber, measured in advance. The results demonstrated significant residual RIA in the visible range as well as at wavelengths above 1300 nm. For example, values of 40 dB/m at λ = 600 nm and 68 dB/m at the RIA spike in the vicinity of λ = 1450 nm were measured for the sample irradiated to 50kGy.

To investigate the evolution of the RIA with time, we irradiated two additional fiber samples to 5 and 17.5 kGy doses and continuously measured the RIA up to 80 hours with the measurement started shortly after irradiation. The results demonstrated contrasting RIA time evolutions for visible and near-infrared spectral ranges: partial recovery of transmission properties in the visible range, while in the near-infrared spectral range the RIA increased with time.

Thus, we have found that the post-irradiation time-evolution of the RIA in gamma-irradiated CYTOP fibers strongly depends on the wavelength range. The transmission partly recovers in the visible range, while at wavelengths above 1300 nm the RIA grows and saturates. We conclude that the RIA induced by gamma irradiation in CYTOP fibers persists for a long time, and can be considered as permanent in the NIR.

04 Research Reactors and Particle Accelerators / 173

#04-173 Progress and prospects on nuclear instrumentation development achieved through JSI – CEA collaboration

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The collaboration on nuclear instrumentation started more than ten years ago between the CEA Experimental Physics, Safety experiment and Instrumentation Section and the Reactor Physics Division of the Jožef Stefan Institute in the frame of bilateral agreement between CEA and the Slovenian Ministry of Higher Education, Science and Technology started in 2008.

More than ten development projects have then been successfully achieved on several subjects: miniature fission chambers, Self-Power Neutron Detector (SPND) and gamma-ray measurement techniques (using ionization chambers and Thermo-Luminescent Detectors (TLDs)), kinetic parameter measurement techniques (□eff), reactor dosimetry unfolding techniques and nuclear data improvements.

Since 2018, three common projects have been started: the first is dedicated to the calorimetry measurement techniques testing different materials sensitive to nuclear heating, the second project concerns improvement of neutron and gamma-ray measurement during power transient pulses and the last aims at establishing an experimental benchmark for a modelling scheme of neutron and gamma-ray sensors. After a short presentation of the two organizations' expertise in nuclear instrumentation and the aims and status of these three projects, the paper presents several themes identified as possible topics for future joint projects. The following topics will be addressed to cover the needs of experimental measurements for the future Jules Horowitz reactor reactor and fusion facilities: improvement of off-line neutron field characterization with the search for new inelastic scattering nuclear reactions for epithermal neutron dosimetry, and on-line, with the study of solid-state silicon carbide (SiC) based neutron sensors and the use of optical fibers as neutron or gamma-ray sensors. The development of liquid neutron filters, containing low concentration aqueous solution of boron, is considered as promising for experimentally simulating energy shifts in the thermal neutron peak in the neutron spectrum and thus allowing to obtain experimental data to support nuclear data evaluation and validation for various actinides.

11 Current Trends in Development of Radiation Detectors / 175

#11-175 Perovskite Semiconductor X-ray and Gamma Detectors

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The application of perovskite materials for radiation sensors is a rapidly emerging field, with strong cross-over from perovskite research on photovoltaic devices. Perovskite materials offer new technologies for digital X-ray and gamma ray sensors with potential application areas in medical imaging systems, industrial X-ray inspection, and airport security systems. Perovskite materials, principally metal halide perovskites, offer several advantages over traditional silicon radiation sensors, primarily their high quantum efficiency due to the presence of high atomic number ("high Z") atoms, and the use of solution growth techniques to realise low cost, large area, sensors.

We present our work on the direct detection of X-rays and gamma rays using semiconductor lead halide perovskites. Single crystal lead halide perovskites are conveniently grown using simple solution-based methods, and we have studied the radiation sensitivity and detector performance of single crystal detectors based on FAPbBr3 (FA = the organic cation Formamidinium). These crystals show strong room temperature photoluminescence (PL) with a peak emission wavelength of ~550 nm, and absorption spectroscopy give a band gap energy of 2.16eV. Temperature dependent PL shows an unusual red shift in the emission wavelength as the temperature reduces, which is the opposite to that seen in conventional semiconductor materials. Time resolved PL shows long radiant lifetimes up to ~400 ns, which is consistent with high quality crystalline material. Charge transport studies show that the drift mobility in these materials is relatively low compared to other semiconductors, however the carrier lifetimes are long and the resulting mobility-lifetime products are reasonably high. Optimal contact preparation is important to limit dark currents in these materials, where the bulk resistivities are typically in the range 1E8– 1E9 ohm-cm.

We have also studied the X-ray sensitivity of polycrystalline lead halide perovskites, which are particularly suitable for X-ray imaging and dosimetry applications. Fabrication of large area devices up to 20mm in diameter has been achieved using millimetre thick films of polycrystalline perovskite, and the dark current and X-ray sensitivity have been measured using a 50kV X-ray tube. Due to the high effective Z number of these materials containing lead atoms, a high detection efficiency can be achieved for relatively thin sensor layers. We will compare the X-ray sensitivity of polycrystalline perovskite films with other traditional imaging materials such as a-Si and selenium.

04 Research Reactors and Particle Accelerators / 176

#04-176 Design of a 150-miniature detectors 3D core-mapping system for the CROCUS reactor

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The interest of the scientific community in studying the three-dimensional propagation of induced perturbations inside nuclear reactor cores and their prediction with deterministic codes triggered the need for experimental full-core mapping systems. In this optic, the Laboratory for Reactor Physics and Systems Behaviour (LRS) at the École polytechnique fédérale de Lausanne (EPFL), Switzerland, has been designing a 3D core-mapping system for its zero-power research reactor CROCUS. The system is composed of 150 miniature neutron detectors distributed within the core double lattice at three different axial levels. The detector technology is based on the well-proven coupling of a miniature $ZnS:Li^6(Ag)$ scintillator to a state-of-the-art silicon photomultiplier (SiPM) via jacketed optical fibers. The challenges in the design of such an extensive mapping system are numerous and the synergy between detector physics, core criticality, mechanical design and electronics acquisition is fundamental.

In CROCUS, the limited space between fuel rods and the need for positioning tools required a downsizing of the detector dimensions compared to the prototype version. The scintillator screen volume was reduced from 0.2 mm^3 to a volume <0.1 mm^3 and the fiber diameter was decreased from 3 mm to 1 mm. This detector arrangement was chosen as a cut-off between the need for size reduction and the preservation of the minimum level of neutron sensitivity required. In addition, smaller scintillator screens reduce the neutron absorption and consequently the reactivity impact on the core. Preliminary results showed that a single of these <0.1mm^3 detector positioned at the core periphery contributes to a reactivity reduction of around 0.7 ± 0.2 pcm. Based on this experimental result, a modelling of the full-system via the MCNP6.2 Monte Carlo code allows to assess the total reactivity impact given by the 150 detectors spread in the CROCUS core and the eventual need for reactivity compensations.

The 150 detectors, split among three identical axial layers, will be arranged to have the best mapping of the induced perturbations in the CROCUS core, e.g. fuel rod oscillations within the COLIBRI program, absorbers insertions or rotations, etc. The mechanical system to held in positions the 150 detectors was designed to be as practical as possible in the manipulation, of easy manufacturing and respectful of the safety standards for nuclear reactors. The detectors will be held in place through 3-mm thick plastic guides running between fuel rods and never entering in contact with the fuel cladding. The optical fibers will be glued to the plastics guides with epoxy and their front-end, consisting in the detector active zone, will be glued and embedded within the plastic itself in order to avoid any unwanted movement of the detectors or loss of neutron absorbers in the core.

An essential role is played by the acquisition system, which was considerably upgraded with respect to the analog read-out used in the prototype. The need for the processing and acquisition of data from 150 channels induced the development of in-house electronics tailored for the application to the 3D mapping system. Stand-alone electronics modules, each able to process the light signal coming from 32 detectors, were designed and built at LRS. The output signals of these 32-channels modules, consisting in the photon counting in every channel, is sent to an FPGA board equipped with a custom firmware that performs a software-based processing of the photon counting into neutron counting.

The installation of the full-core mapping system for the CROCUS reactor is planned for spring 2021 and it will mark a milestone in proving the feasibility of such first-of-a-kind system, and in providing valuable localized experimental data for the validation of high-fidelity neutronics codes.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 178

#07-178 URANIA-V: An innovative solution for neutron detection in homeland security applications

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Detection of neutrons is becoming of the utmost importance, especially in the studies of radioactive waste and in homeland security applications. The crisis of He^3 availability has required the development of innovative techniques. One solution is to develop light gas detectors for neutron counting to be used as portals for ports and airports. The neutron is converted on the Boron-coated cathode, releasing a charged particle, which passage can be identified by the gas detector. While several technologies have been deployed in the past, the project μRANIA-V (μRwell Advanced Neutron Identification Apparatus) aims to detect thermal neutron by means of the μRwell technology, an innovative gas detector. The goal is to produce tiles to operate as portals in homeland security or for radioactive waste management. The technological transfer towards the industry has started, thus the production can be cost-effective also owing to a construction process relatively easier compared to similar apparatus. By reading directly the signals from the amplification stage, the neutrons can be counted with simplified electronics further reducing the total cost. The first part of the project, more dedicated to the optimization of the neutron detection technique, has been developed within the ATTRACT-EU call. In this presentation, the project will be described, with details on the μRwell technology and on the neutron counting, on the test beam performed, and on the future plans.

08 Decommissioning, Dismantling and Remote Handling / 179

#08-179 Updates on a real-time in-line monitoring of tritium in water for Fukushima

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Tokyo Electric Power Company (TEPCO), in order to enhance the monitoring capability for site contamination and surrounding environment is looking to continuously monitor Tritium (3H) concentration in seawater. We hereby present the latest developments of a system by CEA List, in partnership with TEPCO, to achieve the challenging real-time/in-line monitoring of Tritium (3H) in water.
During Phase 1 of the project, a prototype has been developed, based on the SAFEWATER real-time in-line beta/gamma contamination monitoring system. Initially based, for SAFEWATER, on coated scintillating fibers, the TEPCO-dedicated design is based on ZnS-coated PMMA plates to make it sensitive to 3H. Digital pulse processing is performed to improve the 3H detection. An experimental campaign undertaken at Laboratoire National Henri Becquerel (LNE-LNHB), France's National Metrology Laboratory in the field of ionizing radiation, showed promising results. This experimental campaign run during the first Phase of the project, while conclusive with 3H measurable at a concentration of 1.5 kBq/L has highlighted some possible paths for improvement.
The focus of the current Phase 2 work, is to take into account the feedback and lessons learned from the promising Phase 1 results, in order to improve the design, for a better measuring efficiency and robustness of the system.
The first part of the present study is dedicated to the design of the new measurement apparatus, using computer models and simulations carried out using the MCNP6.2 code and Solidworks CAD software; in order to design and eventually confirm assumptions made during Phase 1. The second part is focused on two measurements campaigns, and the adjustments that were made in-between, both undertaken at LNHB, in order to achieve metrology-grade assessment of the 3H concentration.

11 Current Trends in Development of Radiation Detectors / 180

#11-180 Lead Halide Perovskite Nanocomposite Scintillators

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Lead halide perovskites (LHPs) have many characteristics that are of interest when considering their use as scintillators: due to the high atomic number of lead, LHPs have high X-ray stopping power; their luminescence wavelength may be tuned throughout the visible region through adjustments to their composition (this wavelength could be chosen to fit the application of the detector, e.g. to match the maximum quantum efficiency of a given photomultiplier tube); LHP nanocrystals are solution-processable at room temperature, which allows for low-cost manufacture; these materials have also been shown to have a fast response time as well as a high light yield. During the last three years, articles relating to the development of LHP nanocrystal scintillators have been published by several groups. In each of these cases, all-inorganic nanocrystals were set into films of inert plastic that were thinner than 2 mm. The development of thicker composites with high nanocrystal loadings is typically limited by transmission losses caused by scattering of light from the nanocrystals. The negative effect of this scattering increases with particle size, and is therefore exacerbated by the formation of nanocrystal aggregates within the composite scintillators.

Here, nanocomposites up to 1 cm thick have been produced by loading nanocrystals of either formamidinium lead bromide ($FAPbBr_3$) or caesium lead halide ($CsPbX_3$) into plastic scintillator. The resultant samples combine the scintillation properties of both components, while protecting the nanocrystals from environmental degradation. A chemical ligand has been used to suppress aggregation of the nanocrystals, paving the way for production of nanocomposites with high loadings of LHP nanocrystals. Ultimately, the aim of this work is to produce a plastic scintillator in which the inclusion of a large proportion of high-Z LHP nanocrystals allows for the observation of photoelectric absorption peaks.

Hybrid organic-inorganic $FAPbBr_3$ nanocrystals have been synthesised using a room-temperature ligand-assisted reprecipitation method. A solution of perovskite precursors is injected into a second solution which contains chemicals that serve to stabilise the perovskite and to prevent excessive crystal growth, as well as an antisolvent which causes the yellow nanocrystals to precipitate. This method produces particles of a range of sizes, which may then be separated into two groups using a centrifuge. The smaller set of particles are typically less than 30 nm in size and are small enough to demonstrate quantum confinement effects, while the larger particles are typically 300 - 500 nm in size and are stable in a toluene dispersion for a period of more than a year.

Photoluminescence spectra were measured at room temperature using a 20 mW, 405 nm laser. For $FAPbBr_3$ nanocrystals in a toluene dispersion, luminescence maxima were recorded at 525.2 nm for the smaller set of nanocrystals, and 541.5 nm for the larger set of particles. The shorter emission wavelength measured for smaller nanocrystals is an effect caused by quantum confinement, which also leads to brighter emission for the small nanocrystals. The Stokes shift (the difference between emission and absorption maxima, which provides a quantitative measure of self-absorption) for LHP materials is small, but also has a size-dependence: consistent with the literature, we measured Stokes shifts of 70 meV for our smaller nanocrystals, and 10 meV for our larger nanocrystals. This means that there is reduced self-absorption in the smaller nanocrystals, which will reduce losses in the finished nanocomposite. The photoluminescence decay time of the FAPbBr3 nanocrystal dispersion was measured to be 36.1 ns, consisting of two components with time constants of 8.2 ns and 60 ns respectively.

A PVT-based plastic scintillator was used a basis for the nanocomposites, produced from the Eljen Technology EJ-290 casting kit. Prior to casting the scintillator, a concentrated dispersion of LHP nanocrystals was mixed into the partially-polymerised resin. The mixture was poured into moulds and then cured in a water bath.

In order to prevent aggregation of nanocrystals within the nanocomposites, a chemical ligand was attached to the nanocrystals after synthesis. The ligand allows bonds to form between the LHP nanocrystals and the PVT polymer chain during the casting of the scintillator, hence preventing the nanocrystals from aggregating. This ligand-assisted strategy had previously been used by other groups to achieve aggregate-free loading of up to 60% in composites of inorganic quantum dots in PVT. However, the ligand had never before been applied to perovskites.

Initially, nanocomposite scintillators were produced that contained a low (1%) loading of larger nanocrystals either with or without the chemical ligand. The ligand was successful in preventing aggregation of nanocrystals, and the increased uniformity of the nanocrystal loading could be seen visually in the samples. Elemental mapping of the scintillators was also carried out, using energy dispersive x-ray spectroscopy. This confirmed the location of nanocrystals within the surface layer of the composites, providing an image of the differences in nanocrystal aggregation between the two types of sample.

Photoluminescence and radioluminescence spectra have been recorded for these nanocomposites. In these, a luminescence peak resulting from the nanocrystals can be clearly seen, replacing the five luminescence peaks that would normally result from the un-loaded plastic scintillator. The LHP emission peak recorded in radioluminescence was weaker than that for photoluminescence, which is a consequence of the opacity of the sample: while in photoluminescence measurements the luminescence is generated at the surface of the sample and easily escapes to the detector, in radioluminescence measurements luminescence is generated deeper into the sample and suffers losses from scattering and re-absorption while travelling out of the scintillator.

In future work, nanocomposites will be produced which contain smaller nanocrystals (<30 nm size) as well as the aggregate-reducing ligand. Our calculation of the effect of Mie scattering with different particle sizes leads us to believe that this improvement in the nanocrystal size will improve the efficacy of the ligand and will have a significant effect on scattering. This development will therefore increase the transparency of the composites, improving the light output and allowing for higher loadings of nanocrystals.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 181

#07-181 Timepix3 detector network for nuclear waste monitoring

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Distributed networks of hybrid semiconductor pixelated detectors have proven their usefulness in characterization of mixed radiation fields, e.g. the ATLAS TPX network, which has provided data on particle flux within the ATLAS cavity at LHC over a span of several years by now; or the similar case at the MoEDAL experiment in the LHCb facility. In the framework of the Horizon 2020 funded MICADO project (Measurement and Instrumentation for Cleaning and Decommissioning Operations), we propose a long-term nuclear waste monitoring system based on Timepix3 technology as an alternative/complementary solution to the conventionally used detectors. Particle detectors based on Timepix3 are capable of particle discrimination and provide spectrometric information in common with precise timing information. The particle flux encountered by these detectors, depending on nuclear waste composition, volume, geometry and the distance between the waste packages and detectors, is expected to cover a challenging range and to be composed of mixed radiation field of ionizing and non-ionizing radiation. Moreover, in some cases, part of the detectors will be attached directly to the waste drums, while others are fixed to the storage walls. To maximize the versatility of the setup, the Timepix3 detectors employed are equipped with a range of different sensors, namely Si sensors with thicknesses of 100, 300, and 500 µm, as well as CdTe sensors with a thickness of 1 mm. To enable measurement of neutrons, two quadrants of the detectors are covered by neutron converters. Thermal and epithermal neutrons are detected below a ^{6}LiF foil through products of a ^{6}Li(n,α)^{3}H reaction and fast neutrons are visible through recoil protons under a polyethylene layer. Energy calibration of all detectors was performed in the Institute of Experimental and Applied Physics of Czech Technical University in Prague, using an X-ray tube with several fluorescence foils and gamma rays from an ^{241}Am source. The neutron detection efficiency calibration was performed at the Czech Metrology Institute by thermal neutrons from ^{238}PuBe source moderated in a graphite prism and by neutrons from ^{252}Cf and ^{241}AmBe sources. A specific software for the Timepix3 network control and data acquisition was developed in-house. This software allows to reproduce and visualize particle fluxes and dose on-line in all detectors separately, as well as to combine quantities from some of them on demand. An early warning system in case of a specified increase of radiation is incorporated as well. First results of long-term laboratory test where various radiation sources were used to simulate radwaste will be presented. However, comprehensive measurements on radwaste sites are currently being prepared.

08 Decommissioning, Dismantling and Remote Handling / 182

#08-182 Follow-up on characterizing low-activity waste containers: a case study for Compton Suppression Systems under challenging signal-to-noise ratio

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Gamma-ray spectrometry is a reference technique for measurements carried out during the decommissioning process. Compton continuum is a major contribution to the signal-to-noise ratio when considering low-energy peaks measurement. Lead shielding is oftenly used to reduce background contribution but its weight is a major drawback.
This work investigates a compact Compton Suppression System (CSS) based on a High-Purity Germanium (HPGe) diode as a primary spectrometer. Among a wide range of guard detectors, including BGO, NaI(Tl), and LaBr3, a plastic scintillator is selected on account of its lightweightness.
For this purpose, CEA investigates a compact Compton Suppressor System "CSS" consisting of an HPGe "High Purity Germanium" spectrometer coupled with a plastic "EJ 200" scintillator. Hence, when an event temporally coincides in both detectors, corresponding to at least on Compton scattering event, it is rejected by an anti-coincidence filter. The signal-to-noise ratio improvement of total-absorption peaks over Compton continuum hence betters the Minimum Detectable Activity "MDA".
The present article is focused on the experimental study of said Compton Suppression System. It consists in using HPGe detectors of various relative efficiencies (10 %, 20 % and 40 % at 1.3 MeV), set inside a plastic annulus scintillator from Scionix Holland. The acquisitions were carried out using "Hexagon" and "DSPEC 502A" digital MCAs, from CAEN SpA and AMETEK Ortec respectively. The sources and peaks of interests shall be 59 keV (for 241Am, 238U's 63 keV and 133Xe's 81 keV), 122 keV (for the 100 – 200 keV region of Plutonium), 137Cs's 662 keV, 60Co's 1.17+1.33 MeV.
The figures of merit of interest quantized with and without anti-Compton filtering are the overall Compton continuum "C(E)" reduction at different energies "E", and the P(E)/C(E) ratio between total absorption peak "P" and the Compton continuum; the latter corresponding to the overall improvement of the measurement.

11 Current Trends in Development of Radiation Detectors / 183

#11-183 Large Area SiPM Pixels for SPECT: from high energy astrophysics to medical imaging

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The gamma camera is still employed in most Single Photon Emission Computed Tomography (SPECT) clinical scanners. In particular for large cameras it provides a balance between cost, reliability and performance that is hard to obtain for instance with modern CZT cameras. A standard gamma camera for full-body SPECT features a large area 50×40~cm^2) scintillator coupled to an array of 50-100 photomultiplier tubes (PMTs) of 4-8 cm diameter. The camera is shielded by a thick layer of lead, turning it into a heavy and bulky system that can weight a few hundred kilograms. The volume, weight and cost of a camera could be significantly reduced if the PMTs are replaced by silicon photomultipliers (SiPMs). The main obstacle to use SiPMs in full-body SPECT is their limited physical size. A few thousand channels would be needed to fill a camera if using the largest commercially-available SiPMs of of 6×6~mm^2. We propose to reduce the number of readout channels by using Large-Area SiPM Pixels (LASiPs), built by summing individual currents of several SiPMs into a single output. Our LASiPs employ the MUSIC, an ASIC designed for high-energy astrophysics, to perform the sum of the SiPM individual currents. We built a LASiP prototype with a sensitive area 8 times larger than a 6×6~mm^2 SiPM and evaluated its performance in a proof-of-concept detector consisting of a $40\times40\times8$~mm^3 NaI(Tl) crystal coupled to 4 LASiPs. We were able to reconstruct simple images with an intrinsic spatial resolution of $\sim$ 2~mm, achieving an energy resolution of $\sim$ 11.6\% at 140 keV. We also performed a detailed study on the SiPM noise and its impact on the performance of a SPECT camera, focusing on the additional noise introduced by the summing stage. We simulated the proof-of-concept detector with Geant4 and validated it with experimental data. Once validated, we simulated a larger camera with more and larger pixels, which we used to study how to optimize the pixel size, geometry and trigger settings of a full-body SPECT camera equipped with LASiPs. We present the results of such optimization, which sets the basis for a first compact full-body SPECT camera based on LASiPs.

04 Research Reactors and Particle Accelerators / 184

#04-184 Local and high distance neutron and gamma measurements of fuel rods oscillation experiments

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Within the COLIBRI program, reactor noise related to fuel vibrations is investigated in the CROCUS zero power reactor. It consists in experiments on rod lateral displacement (static) and oscillation (dynamic) with different rods' numbers at various relevant amplitudes and frequencies. Its main motivation is the increased amplitudes in the neutron noise distributions recorded in ex- and in-core detectors that have been observed in recent years in Siemens pre-Konvoi type of PWR reactors. In particular, with this program, the Laboratory for Reactor Physics and System behaviour (LRS) is contributing to the Horizon 2020 European project CORTEX, which is dedicated to the understanding and simulation of reactor perturbations for the development of novel core monitoring techniques.

During the first phase of COLIBRI, the observation of a spatial dependence of the induced noise, also called neutron modulation, was demonstrated. In the second phase of COLIBRI starting 2021, it is planned to use a core mapping array of neutron detectors to record its propagation. It consists in about 150 miniature scintillators coupled to optical fibers and SiPM readouts, to be distributed in the reactor core. As a feasibility test, experiments were performed using a miniature scintillator prototype placed on one moving fuel rod, or the one directly adjacent to it. In addition, it is theoretically possible to measure branching or induced reactor noise using gamma radiation. Following recent developments on gamma measurements in CROCUS, the fuel oscillation was simultaneously recorded with a gamma detection array, LEAF. Its two large and high efficiency BGO detectors were used by placing them at the maximum distance to the core, i.e. seven meters away with a clear line of sight using an experimental channel through the reactor cavity.

We report here on the successful observation of the lateral oscillation of one fuel rod ±2.5 mm around nominal and 0.1 Hz frequency, using the miniature neutron scintillator at the rod level, and the BGO gamma detectors seven meters away from the reactor core.

04 Research Reactors and Particle Accelerators / 185

#04-185 Calibration of CFUL01 fission chambers in the standard neutron fields of BR1 reactor at SCK CEN

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Recent subcritical VENUS-F experiments showed that fission chambers with a threshold deposit like U-238 can essentially improve the on-line sub-criticality monitoring with the beam interruption method, which is currently supposed to be the main method for the ADS MYRRHA. To suppress the uncertainty caused by fissions in the U-235 impurities, the fraction of U-235 in the U deposit should be accurately known. Three PHOTONIS CFUL01 type fission chambers with a high purity of U-238 were purchased for sub-critical experiments in the VENUS-F reactor. To verify the nominal purity (U-235 content) of the deposits ofthese fission chambers, the effective U-235masses were measured in 2017 in the empty cavity of the BR1 reactor with a well-known thermal neutron spectrum. The measured effective mass was determined as the mass of the isotope corresponding to a certain discrimination level and is obviously expected to be smaller than the total mass of the isotope, especially for heavy, thick deposits, which is the case for the fission chambers under investigation. It turned out that the measured effective U-235 mass in two fission chambers is lower than the nominal (as it should be), but this is not the case for the U-235 content in the third fission chamber. To obtain the correct of U-238 content in these three FCs, the effective U-238 mass was measured in the well-known fast spectrum of the MARK-III convertor in the BR1 reactor. Finally, the isotopic composition was obtained by dividing the U-238 effective mass (measured in MARK-III) by the U-235 effective mass (measured in the empty cavity) at the same discrimination levels. It was found that the purity of two CFUL01 FCs is in agreement with the values declared in the certificates but it is not the case for the third fission chamber. The statistical and systematic uncertainties and corrections applied in the data analysis are discussed. The developed procedure using the BR1 standard irradiation fields can be of interest for calibration and impurity determination of large fission chambers.

11 Current Trends in Development of Radiation Detectors / 188

#11-188 Scanning of a Germanium Double Sided Strip Detector

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The response of a position-sensitive planar High Purity Germanium (HPGe) detector has been studied using pulse shape comparison and positron annihilation method. Such detectors will be useful in the DESPEC (DEcay SPECtroscopy) experiments at the FAIR facility to study the exotic nuclei. The characterization of the detector has been performed using a novel scanning system available at GSI Helmholtz Centre for heavy-ion research, Germany, which consists of a LYSO (Cerium doped Lutetium Ytturium Orthosilicate) scintillation crystal-based position-sensitive scintillator detector (PSD) [1]. The crystal is coupled to a position-sensitive photomultiplier tube, a mesh of 16X and 16Y anodes. The electrically segmented HPGe detector has dimensions 6cmx6cmx2cm consisting of 10 segments each along the horizontal and vertical directions [2]. The pulses have been stored for scanning along the front and side view of the segmented HPGe detector. The 2D image from the PSD has been used to characterize the depth of interaction information in the planar strip detector using pulse shape analysis. The analysis and results from the scanning of the planar HPGe detector will be presented at the conference.

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09 Environmental and Medical Sciences / 189

#09-189 Characterization of lead tungstate crystals for the Clear-Mind Project

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ClearMind project aims to develop a fast detection module for TOF-PET.
We propose a position-sensitive detector consisting of a monolithic $PbWO_4$ scintillating crystal on which is directly deposited a bialkali photoelectric layer. This detector optimizes the transmission of light photons to the photoelectric layer. Photons are generated by two processus : the Cherenkov effect and the scintillation.
To fully exploit the potential of this detector, we need to caracterize crystal properties such as the scintillation yield and the different time constants. According to the literature, the time dependence of the scintillating processus can be modelized by a multi-exponential decay function ; each of exponential component are related to a different luminescent center. The scintillation of $PbWO_4$ presents two main scintillation time constants, a fast one about 2 ns and a slow one about 6-10 ns.
We studied the scintillation light yield and time constants as a function of the crystal temperature for four different $PbWO_4$ crystals, provided by three different producer : CRYTUR (CMS doping), EPIC (undoped) and SICCAS (CMS doping, Y-doping).
All crystals show 4-fold increase in the scintillation light yield when cooled to -25℃. EPIC undoped crystal shows the highest yield. The time measured constants are similar to the different technologies and we observe that scintillation is slowing down when crystal is cooled. This trend is more marked on the slow time constant.

04 Research Reactors and Particle Accelerators / 190

#04-190 High resolution measurements with miniature neutron scintillators in the SUR-100 zero power reactor

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Three 0.95-mm^2 miniature fiber-coupled scintillators have been used to perform cm-wise resolution measurements of the thermal neutron flux within experimental channels of the SUR-100 facility, a zero power thermal reactor operated by the Institute of Nuclear Technology and Energy Systems at the University of Stuttgart. The detection system is developed at the École polytechnique fédérale de Lausanne in collaboration with the Paul Scherrer Institut.

Reaction rate distributions are measured along the experimental channels I and II of SUR-100, which cross the reactor at the center and tangentially to the core, respectively. The results for the experimental channel I are compared to neutron activation measurements. In addition, reaction rate gradients across the 2.6 cm and 5.4 cm diameters of the channels are measured. The reactor was modelled with the neutron transport code Serpent-2.1.31, and the collected experimental data are compared with Monte Carlo simulations.

The comparison of experimental and computed reaction rate distributions showed a good agreement within the core region, with discrepancies within 2σ. An unexpected discrepancy, probably caused by a geometric inconsistency in the computational model of the reactor, was observed in the reflector region of the experimental channel I, where a 20% difference (i.e. 8σ) was found between experimental and simulated results. Significant discrepancies, respectively worth 10σ and 15σ, were noticed at distance, in the lead shielding region, for both experimental channels I and II.

An horizontal reaction rate gradient of $(9.09 \pm 0.20)\%$ was measured within 2.4 cm across the diameter of the experimental channel II, with a difference from computed results of 2%. The absence of a vertical reaction rate gradient inside the experimental channel I was confirmed by measurements.

The performed measurements highlighted the suitability of the miniature fiber-coupled scintillators for high resolution reaction rate distribution measurements and for the characterization of highly localized gradients in reaction rate. The negligible flux perturbation induced by the miniature fiber-coupled detectors has been found to be a great advantage compared to the previously used manganese activation techniques. In the case of the activation technique, the presence of the samples holder caused a maximum 35% increase of thermal neutron flux in the core region of experimental channel I.

01 Fundamental Physics / 191

#01-191 Preliminary results of the CGEM-IT of the BESIII experiment

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The BESIII (Beijing Spectrometer III) experiment is hosted at the leptonic collider BEPCII (Beijing Electron Positron Collider II) at IHEP in Beijing and it is successfully running since 2009. Recently, an extension of the data taking for the next 10 years has been approved, so an upgrade program has started. A central element of the upgrade is the replacement of the present inner tracker with a new detector based on the cylindrical GEM (Gas Electron Multiplier) technology. The CGEM-IT detector is composed of three layers of cylindrical triple-GEMs with time and charge readout to guarantee excellent performance in a wide range of incident angles in 1 Tesla magnetic field. Two of the final layers are already at IHEP and are undergoing commissioning with cosmic rays events. The third layer is being completed by end of 2020 and will be taking data with the other two by the summer of 2021. The installation is foreseen in Summer 2022. In this presentation, the general status of the project will be presented with a particular focus on the recent preliminary results from the cosmic data taking and future plans.

04 Research Reactors and Particle Accelerators / 192

#04-192 Measuring gamma doses over the mGy-MGy range with a single type of TLD detector

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Thermo-luminescent detectors are currently used to measure gamma doses in the medical and environmental surveillance fields. During the past few years, the CEA Reactor Studies Division tested and validated the use of these detectors for gamma flux characterization and nuclear heating measurements in mixed neutron/gamma fields of low power reactors. Doses were comprised between a few mGy and a few Gy for dose rates up to a few Gy/h. However, in MTR or TRIGA reactors, the gamma flux level is much higher (> 10^{12} n/cm²/s) and the TLD currently in use (TLD400 (CaF_2:Mn) and TLD700 (^{7}LiF:Mg,Ti)) and their readout protocol are no longer suitable for the resulting doses. In order to extend the applicable dose range up to ~1MGy (dose rate of a few kGy/h), several options were explored. On one side, some adjustments were made to the readout protocol of TLD400 and TLD700, notably by testing the use of filters to reduce the amount of light received by the reader PMT to avoid saturation. On the other side, a new type of TLD (LiF:Mg,Cu,P) with different Li enrichments (natural – MCPN or enriched in ^{7}Li – MCP7) was tested.

This paper presents the calibration measurements results performed in pure gamma fields, at the irradiation platform of the CEA Cadarache Radioprotection Division, between 250 mGy and 3 Gy for all detector types. In addition to the calibration, these measurements also studied the MCPN and MCP7 response: reproducibility, dose rate dependence, incoming photon energy dependence, high temperature effect when reading TLD, etc. In parallel, a numerical model of the thermoluminescence process was developed. This numerical model was validated using the calibration measurements and used to optimize the heating laws of all TLD types.
Using these optimized heating laws, TLD detectors will be used to characterize the gamma field and nuclear heating in the TRIGA Mark II reactor at the Jožef Stefan Institute (JSI) in Slovenia. Measurements are expected to take place in the second half of 2021, in the frame of a CEA-JSI collaboration dedicated to the benchmark of the TRIGA reactor neutron and photon 3D transport model.

04 Research Reactors and Particle Accelerators / 194

#04-194 Diamond detectors for neutron flux measurement during transient neutron flux changes

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Precise measurement of neutron flux is crucial for reactor operation, especially during fast reactor transients. Measurement is valuable not only in reactor operation itself, but also for the prediction of reactor behavior via calculation or experiments. Especially fast transients with significant change in neutron flux are interesting and in order to analyse them, an advanced detection system is needed.

Since 2017, diamond-based detectors are being developed in collaboration with CEA Saclay. Detectors are now in the testing phase at the VR-1 reactor. Diamond detectors are rarely used in reactor core measurements but are promising for this particular application.

To cover changes in neutron flux during a very short time, a large number of interactions must be recorded, especially if significant changes in neutron flux are measured. Due to this prerequisite, the detection system and data processing must be advanced.

Several approaches are available to process the signal from diamond detectors. In a nuclear reactor, detector signal contains plenty of different pulses. Some are created by noise or by other electronic influences should be separated. Others that have an origin in particle interactions must by sorted by the source particle, as not only neutrons interact in the detector volume. For example, gamma radiation creates a significant amount of impulses that needs to be identified and separated, so only the count rate from neutron detection can be obtained. In order to solve the separation problem, several approaches are used in current detection systems. For diamond detectors, we tested pulse shape analysis, where pulses are separated by their shape. This approach is very accurate, however, it requires a robust detection system, including a fast AD convertor that allows continuous data recording, if the experiment is performed in reactor core. Alternatively, the subtractive gamma compensation may be a better solution for online measurement signal analysis.

In order to examine the subtractive gamma compensation, we are testing a set of diamond detectors that allows to collect data that can be analysed by subtractive gamma compensation. Experimental work is performed at the VR- 1 reactor and adjacent laboratories, where various radiation sources are used. Preliminary data should be available at the beginning of the year 2021.

This detection approach should be used in detection part of a new experimental device that focuses on the rod SCRAM transient. This device is currently under development at the Department of Nuclear Reactors, Faculty of Nuclear Sciences and Physical Engineering, Czech Technical University in Prague and should be installed at the VR-1 reactor.

01 Fundamental Physics / 195

#01-195 Measuring the delayed neutrons multiplicity and kinetic parameters for the thermal induced fission of ^{235}U, ^{233}U and ^{239}Pu

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Depending on the used library, nuclear data regarding the emission of delayed neutrons present significant discrepancies. Therefore, in the framework of the NEEDS/NACRE collaboration, the ALDEN (Average Lifetime of DElayed Neutrons) program was launched in 2018 by CEA (DES and DRF), CNRS (LPSC, CENBG, LPC Caen), ENSICAEN and Caen University. This program aims at measuring the (α_i,λ_i) parameters describing the emission kinetics of the delayed neutrons as well as their average multiplicity ν_d.
To this end, a new experimental device, conceived using TRIPOLI4® Monte-Carlo simulation, was designed to measure the thermal induced fission of actinides on the PF1b cold neutron beam-line of the Institut Laue-Langevin (ILL, Grenoble, France). This device consists in a cylindrical matrix of polyethylene, shielded using a thin B4C layer, and at the centre of which a miniature fission chamber containing the actinide to study is placed. This chamber is irradiated for a period of time t_{irr}, then the beam is shut using a motion-controlled thermal neutrons screen. Delayed neutrons, measured over a period of time t, are thermalized in the polyethylene matrix and detected using 16 ^{3}He proportional counters. They are positioned into three rings to ensure a detection efficiency as constant as possible between 0.1 and 1 MeV (energy range of the delayed neutrons). Irradiation cycles (t_{irr}+t) are repeated until a good counting statistics is obtained. All events are recorded using the NOMAD digital acquisition system developed by the ILL and the resulting curve is fitted using the CEA-developed CONRAD regression software.
After detailing the experimental setup, this paper presents the adaptations that were made to it in order to allow studying ^{239}Pu. Results obtained during the experimental campaign of June 2019, dedicated to the thermal induced fission of ^{235}U, and first results of the experimental campaign of the beginning of 2021, dedicated to ^{239}Pu and ^{233}U, are presented.

04 Research Reactors and Particle Accelerators / 196

#04-196 Summary of Thermocouple Performance in the Advanced Gas Reactor Experiment AGR-5/6/7 During Irradiation in the Advanced Test Reactor

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The Advanced Gas Reactor -5/6/7 (AGR-5/6/7) experiment was the fourth and final experiment in the AGR experiment series and completed irradiation in July 2020. It serves as the formal fuel qualification test for the TRISO fuels under development by the US Department of Energy. This experiment was designed to irradiate fuel particles at temperatures ranging from 800℃ – 1500℃. The high end of the range created unique challenges for thermocouple-based temperature measurements. Commercially available high-temperature platinum-rhodium thermocouples (Types S, R, and B) and tungsten-rhenium thermocouples (Type C) suffer rapid decalibration due to transmutation of the thermoelements from neutron absorption. A special low neutron cross-section thermocouple system based on molybdenum/niobium thermoelements called High Temperature Irradiation Resistant thermocouple (HTIR-TC) has been under development at INL since circa 2004. Several examples of this thermocouple type were incorporated into regions of the test operating above 1200℃. For regions in the test operating in the 1000℃ – 1200℃ range, high-performance versions of Type N thermocouples recently developed at Cambridge University were installed. Standard Type N thermocouples were used in regions of the test operating below 1000℃. A total of 54 thermocouples were incorporated into the test. A report of the performance of this large heterogenous set of thermocouples over the first third of the irradiation was previously provided (ANIMMA 2019). This follow-on paper discusses results from the balance of the experiment (Feb 2019 – July 2020). Special attention is focused on the estimated drift of thermocouples operating in the higher temperature regions. This is particularly important as an input to the final irradiation report for AGR-5/6/7, in which estimates are presented of the temperature distribution of the 570,000 particles irradiated.

05 Nuclear Power Reactors Monitoring and Control / 197

#05-197 Position evaluation of ex-core neutron flux measurement in new type graphite reactors

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Several concepts of new reactors use graphite. Some of them use graphite as a moderator, some of them as a reflector. There are at least two concepts of these graphite-type reactors under development in Czech Republic – the Energy Well and the Teplator. Both reactors use graphite as the reflector. An in-core measurement might be impossible to use due to various reasons, for instance high temperature or aggressive environment. Therefore, this article focuses on ex-core neutron flux measurement system placed in the graphite reflector and the optimization of ex-core detector position. A set of experiments were performed at LR-0 reactor. The LR-0 is a light water reactor with well-defined neutron field, which can be used for different material insertion testing and testing of its influence on criticality. The three modifications of LR-0 cores were modelled in Monte Carlo codes Serpent and KENO. A set of calculations were performed for verification of the criticality and neutron flux course in reactor core and graphite reflector. Further investigation was focused on the influence of graphite reflector presence on neutron distribution in the reactor core. The LR-0 graphite experiments were also used for models' calculations verification. Based on the results of this article, the optimal position of ex-core detectors in the mentioned new reactor types were proposed.

11 Current Trends in Development of Radiation Detectors / 198

#11-198 A multi-scale clinical translation strategy for photon-counting CT

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Currently first clinical CTs equipped with photon-counting technology are becoming available. Whereas such scanners will remain to be very rare in the time to come, laboratory based photon-counting μCT setups have become much more common in previous years. For this reason, method development in photon-counting CT is likely to be driven by translational research in the coming years. With Freiburg being among the first recipients of a clinical photon-counting CT, we aim to develop and provide a scaling and translational toolbox for this system, allowing researchers in the photon-counting imaging community to transfer their methods, developed on laboratory scale systems, to clinical scale systems.
Our approach is based on multi-scale phantom based system characterization in combination with robust physics and detector simulation. By extracting the characteristics of the respective source and target system via known ground truth, and by analyzing systematic scaling effects, translation of methods between such systems can be facilitated greatly.
In collaboration with the Institute of Experimental and Applied Physics of CTU Prague, we plan to perform a proof of principle investigation demonstrating this concept.
Within this investigation multi-contrast agent methods for pre-clinical / clinical purposes will be developed and optimized on laboratory scale systems and our clinical system in parallel. These data will be used to calibrate and verify our translational toolbox.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 199

#07-199 Optimising Detecting Geometry for Improved Pulse Shape Discrimination Performance in Plastic Scintillation

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Pulse Shape Discrimination is a technique used extensively for the detection and analysis of neutron emissions in mixed fields of ionising radiation. Pulse Shape Discrimination utilises a single detector and digital processing setup that can discriminate between incident γ-rays and neutrons based on the decay rates in the scintillation light caused by the different radiations interacting in the detector media. Extensive work has been carried out on optimising Pulse Shape Discrimination through improvements in digital signal processing, however few works focus on the physical geometry of the detection medium and how this influences discrimination.

This work uses Monte Carlo codes (OpenMC, FLUKA, GEANT4) to model the ratio of energy deposition from incident gamma and neutron isotropic radiation fields (1 MeV – 9 MeV range) in a series of EJ299 plastic scintillator geometries. The general shape types simulated included, spherical, cuboid, cylindrical, and conical geometries in a range of volumes. The simulated shapes were compared with regard to their volume and their surface-area-to-volume (SAV) ratio. Where SAV offers a normalised parameter to assess the cross-section of the detector covering the solid angle of the emitted radiations.

Simulated results indicate that at lower particle energies (< 3 MeV) conical detectors demonstrate a lower gamma/neutron energy deposition ratio, i.e. a higher proportion of neutron energy is deposited compared to the gamma energy deposition, when compared to other detector shapes with similar volumes and SAV ratios. A conical detector with an approx. 40 cm3 volume (height 6 cm, 2.5 cm base radius) has a gamma/neutron energy deposition ratio of 8.31%, compared to 10.21% for a cylindrical detector (height 2 cm, 2.5 cm base radius). Similar differences are found throughout a range of comparable volumes. Though modest, this difference would stimulate greater production of optical photons with a slower decay times which enhance the ability of digital signal processors to discriminate between the signals produced by the γ -rays and neutrons. The difference in energy deposition ratios between shape geometries diminishes as radiation energy increases.

Further work is being conducted to model additional geometric refinements, different detector media, and simulations of optical photon production occurring as a result of the energy deposition.

10 Education, Training and Outreach / 200

#10-200 Remotely controlled laboratory gamma-ray spectrometry with CdZnTe-detectors

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Now, when the entire educational process had to be transferred to distance learning, we faced difficulties in performing laboratory and practical exercises, which require real laboratory equipment.
One option is virtual labs using a variety of distance learning platforms. In the 2020-2021 academic year at the Odessa National Polytechnic University to study the courses "Experimental Methods of Nuclear Physics" and "Spectrometry of Ionizing Radiation", a project of a virtual educational laboratory of gamma spectrometry with remote laboratory experiment is being implemented. A remote experiment is a real experiment with real laboratory instruments and equipment that can be controlled by a teacher or a student from their computer through the Internet.
The special laboratory kit is based on μSPEC microspectrometers (ZRF Ritec SIA). Also, we use the spectrometers based on SDP500 (Ritec) CdZnTe-detectors connected to the multichannel analyzer MCA-166 (GBS-Elektronik GmbH). A LattePanda single-board computer is used to control the operation of spectrometers, collect, and analyze data. LattePanda - A Windows 10 Computer with integrated Arduino. This explains the choice of LattePanda. Windows 10 application allows you to use the WinSPEC software to control the multichannel analyzer operation supplied with the spectrometer. The built-in Arduino allows you to remote control the movement of the radiation source during laboratory experiments. A VNC server is used to implement remote control and access to a Windows GUI running on LattePanda. The vendor recommends TightVNC, a free and easy way to set up this service.
Laboratory exercises for students include both the traditional tasks of calibration of the spectrometer (energy calibration and efficiency curves), including those for various source geometries, processing the measured spectra using standard programs, calculating the activity of sources, and creating a spectra processing program and a spectrometer MCA-166 control program. Samples containing natural radionuclides and sources of low activity, which do not require a special permit, are used as sources.

10 Education, Training and Outreach / 201

#10-201 The European Nuclear Experimental Educational Platform – ENEEP: Progress, Prospects and Remote Education Capabilities

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The European Nuclear Experimental Educational Platform – ENEEP is currently being established by five European educational / research organizations in the framework of a Horizon 2020 project, initiated in 2019. The ENEEP partner institutions are the Jožef Stefan Institute (JSI) in Ljubljana, Slovenia, the Slovak Technical University (STU) in Bratislava, Slovak Republic, the Czech Technical University (CTU) in Prague, Czech Republic, Technische Universitaet Wien (TU Wien) in Vienna, Austria and the Budapest University of Technology and Economics (BME) in Budapest, Hungary. ENEEP is intended as an open educational platform, offering experimental hands-on educational activities at the ENEEP partner institution facilities: four research reactors and one Radiation Physics Laboratory.

ENEEP educational activities will be offered in different formats: group educational activities ("package" and "custom" courses) and individual activities, and are targeted at university students at the bachelor, master and Ph.D. educational levels and young professionals in the nuclear field wishing to deepen their knowledge and gain valuable practical experience in nuclear facilities.

This paper gives an overview of the ENEEP project activities and the progress achieved so far, highlighting the experimental capabilities which will be offered. In the first implementation phase, ENEEP will be based on a comprehensive set of experiments which constitute the basics in Reactor Physics and Nuclear Engineering curricula, e.g. approach to criticality, reactor response to changes in reactivity, neutron flux mapping, as well as more specific experiments focusing on particular aspects – investigated phenomena, types and working principles of detectors, etc., e.g. neutron emission rate measurements with the manganese sulphate bath technique, radiation measurements with semiconductor detectors. Subsequently, novel education activities will be introduced and implemented in ENEEP, following scientific development in nuclear science and technology and nuclear instrumentation detectors, stemming from research activities. Attention will be devoted to the development and optimization of remote education capabilities at the ENEEP partner institutions, of particular relevance during the current Covid-19 pandemic which is responsible for major changes in education activities worldwide.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 202

#07-202 A comparison of bounding approach with isotopic correction factors and Monte Carlo sampling in burnup credit method

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In nuclear criticality safety analysis, burnup credit is an approach that credits the reduction in reactivity due to fuel burnup. The calculation using burnup credit method consists of two main steps: a burnup calculation, which estimates nuclide concentrations in the spent fuel, and criticality calculation, which uses the nuclide concentrations determined in the first step. However, computational prediction of fuel composition introduces an additional source of bias or uncertainty. There are several methods that deal with these uncertainties and in general, they can be divided into two main categories: bounding methods and best-estimate methods.

The bounding approach aims to get the most conservative result by adjusting the concentrations of nuclides in a system. This can be done using isotope correction factors (CF), which are determined based on the difference between the experiment and the calculation results (E/C) and are part of validation of each calculation code. The concentration of selected nuclides in the model is then multiplied by these factors so that concentrations of fissile nuclides are increased and concentrations of absorbing nuclides are decreased. This leads to a more reactive system with the conservative estimate of multiplication factor keff.

The best-estimate methods aim to determine the uncertainty in spent fuel in a more accurate and realistic way. This can be done using Monte Carlo sampling of the different parameters in a model based on their uncertainty and selected distribution. Other options, such as direct-difference method or sensitivity calculations, are also possible.

In this work, the comparison of bounding approach with correction factors and Monte Carlo sampling was made on the model of spent fuel pool. The isotopic composition of the fuel was calculated for different values of burnup using TRITON code from SCALE code system. In a bounding approach calculation, we have considered 48 nuclides to which the correction factors were applied. The criticality calculation was then made by KENO-VI code from SCALE. In a Monte Carlo sampling calculation, the concentrations of nuclides were multiplied by a factor, which was sampled with normal distribution based on average E/C value and its uncertainty. A total of 5,000 sampled input files were then calculated using KENO-VI code. The average multiplication factor keff of the results and its standard deviation σ were subsequently determined.

The comparison of results of both methods shows how much conservative the bounding approach is. In this case, the multiplication factor is higher by 0.02935, which is about 5.4× more than the value of the standard deviation of the sampled Monte Carlo tasks. The using of Monte Carlo sampling method can reduce the maximum multiplication factor for models where the coefficient is slightly above the legislative limit based on the current conservative burnup credit methodology. This could lead to an alleviation of possible restrictions resulting from the exceed of the limit. However, the Monte Carlo sampling approach requires a calculation of large number of tasks and is therefore more computationally demanding.

05 Nuclear Power Reactors Monitoring and Control / 203

#05-203 Characterization of high harmonics frequencies in reactor noise experiments within the CORTEX project

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The Laboratory for Reactor Physics and Systems Behavior (LRS) of the école Polytechnique Fédérale de Lausanne (EPFL) has in recent years been involved with the European CORTEX project, which aims to characterize reactor perturbations such as local bubbling, mechanical vibrations, etc. by neutron flux measurements. The goal of the LRS is the preparation and execution of experiments and experimental data analysis, to provide quantities of interest for newly developed neutron transport code validation. In this paper, we present a method of discriminating the origins of spectral power peaks at different harmonic frequencies: either due to higher modes of the reactor perturbation inducing device or due to effects of different harmonics.

The European CORTEX project aims to develop an instrumentation system framework with the capability of localization and identification of different reactor neutron noise sources, leading to improved reactor diagnostics and ultimately safety. The project involves development stages from ground up: from experiment design, execution and analysis for code development and validation to instrumentation development supported by machine learning.

As part of the experimental data analysis is to provide the data for code validation. Initial developments enabled simulations of frequency effects of the same order, which changed during the CORTEX project. In terms of the actuall experiments, a more rigorous approach was employed by including the position signal of the induced perturbation device in the form of oscillating absorber or fuel. This enabled for an estimate of two different contributions: due to higher harmonics of the oscillator itself, or higher order harmonics of lower order oscillations, which have different origins. An estimate on the two contributions can be made by comparison of spectral power peaks of the detector signals and the oscillator position signal. This can help distinguish between the two effects and provide higher quality data for code validation.

This is possible by utilizing a high frequency resolution analysis technique based on bootstrapping with replacement. The oscillation and the detector data are synchronized in time. The data is then sectioned by well defined oscillator position data features (e.g. crossing neutral position), and used for spectral density calculations using Fourier transform and the bootstrapping re-sampling process. The developed analysis scripts also use modern graphics processors (GPU's) to speed up the re-sampling process by a factor of ~10.

The methodology has already been applied to the data from the 1st experimental campaign of the CROCUS reactor, yielding higher harmonics up to the 14th order. The data is already being utilized for the code validation and further developments.

The improved capabilities of reactor noise propagation codes identified the need for higher harmonics experimental oscillator and detector data. The newly developed analysis techniques enable the distinction between higher order effect of the oscillator movement itself and higher order effects due to lower harmonics oscillations. The analyzed experimental data is available to the CORTEX consortium members for code validation and comparison.

10 Education, Training and Outreach / 204

#10-204 Radiation Sensor Based on PIN Photodiode

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The main idea of our project is to build a cheap gamma photon detector that can be assembled without buying expensive parts. This idea will be an Open Hardware Project so that everyone can try it out and make it better.
The basis of this project is the ability of the photodiodes to react to the gamma particles that pass through them. Since photodiodes are inexpensive, this makes it possible to make the device cheaper than other radiation detectors (like the Geiger counter).
What components should be used?
- PIN photodiodes, - the usual Texas Instruments microcontroller (the cheapest one), - Wi-Fi and Bluetooth module (if necessary), - battery (if necessary), - MicroSD module (if necessary).
By connecting the photodiode to the microcontroller, we can begin downloading the software to the microcontroller, which will be in open access as well as other information in our project. The software was created by using libraries that are free and available on the Internet. After completing these steps, the device consider to be ready. (In case of using elements identical to ours, it may not even necessarily make a calibration). The next step might be to connect the battery. Due to the low voltage of the microcontroller, it can work from the battery for several months. Then by connecting the MicroSD module, appears the ability to store data on the memory card. Connecting the Wifi and Bluetooth module can view and manage the data wirelessly. Also, with good project prevalence, it will be possible to collect data from such detectors around the world and perform radiation analysis from all over the world and collect this information on the project server.
This solution allows us to make a cheap gamma photon detector by ourselves, using affordable parts and open source libraries.
Thanks for this project we can:
1) Construct an interesting and useful device;
2) Improve our technical skills;
3) Make a community for improving this project and creating new ones;
4) Give a start impulse for developers and engineers to make interesting and useful devices and projects for Open Source.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 206

#07-206 Safeguards investigations based on gamma spectrometry to determine the activity ratio of fission products in spent fuel assemblies of VVER-440 Nuclear Reactor

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Safeguards investigations deal with the inspection of fission materials and their goal requires a reliable experimental verification method to determine the power history of spent fuels. In nuclear power plants it is a key task to continuously monitor the fuel burnup, for this purpose different types of verified deterministic and stochastic codes are applied. For both tasks, it is absolutely necessary to validate experimentally the theoretical models and calculations, for which one of the most appropriate methods is in-situ gamma spectrometry.

High resolution gamma spectrometry measurements have been carried out on the VVER-440 type spent fuel assemblies of Paks Nuclear Power Plant, Hungary. The purpose of the measurements was to verify the final burnup values calculated by the method of the power plant's staff. Until now, gamma spectra of more than 100 fuel assemblies were collected and analysed by using coaxial high-purity germanium semiconductor detectors. The majority of the measured spent fuel are so called work assemblies, but few of the absorber follower assemblies were also measured. The measured assemblies have varying initial enrichments (1.6 − 4.7%), cooling times (1 − 5 y) and operational histories, the latter resulting in differing levels of burnup. All of the hexagonal assemblies were measured from either all six or three non-neighbouring sides, which means that some information regarding the whole assembly can be derived. Additionally, the data collected throughout the years can be used for safeguards method investigation and development as well.

Due to the measurement conditions and the relatively short cooling times, only prominent gamma emitting fission products are able to be measured. However, since the production of these isotopes depends solely on the fuel's initial composition, geometry and operational parameters, their final activity is connected to the relevant safeguards quantities, namely the burnup, initial enrichment and cooling time. By using high resolution gamma spectrometry, we were able to detect the 137Cs, 134Cs, 154Eu, 106Ru, 144Ce, 125Sb, 110mAg, 95Zr, 103Ru and 91Y fission products. The last three isotopes could only be measured in assemblies with cooling times shorter than two years, due to their short half-lives. The difficulties raised by an absolute calibration procedure limit us, at the present, to use a relative efficiency calibration method. With such a "self-calibration" technique, only activities relative to some other isotope can be calculated.

In the present study, we examine our dataset to reveal connections between the measured activity ratios and the three relevant quantities by utilizing model functions, dependent on the burnup, initial enrichment and cooling time, and statistical methods such as cluster analysis. The clustering of certain isotopic activity ratios shows at a glance which assemblies have similar or dissimilar initial compositions and operating histories. We also investigate the axial distribution of fission product activity ratios from assemblies measured in several different axial positions. From this, we can determine the burnup level seen by the detector at a given axial position of the fuel and compare that to the known average assembly burnup.

06 Severe Accident Monitoring / 208

#06-208 Optimized High Temperature Irradiation Resistant Thermocouple for Fast Response Measurements

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The High Temperature Irradiation Resistant thermocouple (HTIR-TC) is the only temperature probe proven to withstand both the high temperatures (e.g. >1290 ℃) and high radiation (e.g. up to a fluence of 1E21 n/cm2) of nuclear reactor fuel design tests and/or over-temperature accident conditions. The HTIR-TC heat treatment, calibration, and in-pile performance during the Advanced Gas Reactor 5/6/7 fuels tests inside the Idaho National Laboratory's Advanced Test Reactor have been previously shared. The current work describes the improved performance of the molybdenum versus niobium thermocouple by utilizing a coaxial design—i.e. single-wire grounded to the outer sheath. The thermocouple junction is formed at the end of the coaxial cabling by swaging the sheath down and welding to the inner wire. This optimized HTIR-TC is more concise by simplifying the design while allowing for more robust individual components. The niobium and molybdenum thermoelements can be interchanged as sheath or wire depending on the application. Using a plunge test in flowing water, the coaxial build of the HTIR-TC was found to be 30x faster in response time than ungrounded Type K TCs, and 10x faster than both grounded Type K TCs and traditional ungrounded HTIR-TCs (i.e. two-wire configuration). Further, by capitalizing on the coaxial design, a multi-core HTIR probe has been proven with multiple 'single-pole' wires down the length of the sheath. Each wire is then terminated and formed into a junction with the inside of the outer sheath at the location where temperature measurement is desired. This multi-core thermocouple design has been dubbed "demicouple." The primary application of the HTIR demicouple is during fuel experiments to achieve multiple fuel pin centerline temperature measurements in one compact sensor. However, the shared 'common' leg between demicouple junctions reduces error propagation in secondary measurements such as temperature differentials.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 209

#07-209 New development from Orano Mining in the field of nuclear instrumentation to improve Uranium extraction and recovery.

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Facing more complex mining extraction issues, Orano Mining has implemented a development plan on implementing new instrumentation techniques. These new techniques are mainly in support of the geological control of our mine at SOMAIR in Niger and to respond to the problem of radioactive imbalance in the roll-fronts for our ISR (In Situ Recovery) mines of KATCO in Kazakhstan and in support of our projects in Mongolia and Uzbekistan.
A. In support of our KATCO site and our ISR mining projects, we can cite the following ongoing projects:
- The development of a LaBr borehole probe for measurement by gamma spectrometry using an algorithm patented by Orano with CEA to separate Uranium more quickly from its radioactive decay products
- The same type of detector is also used to characterize cores or ore samples

B. In support of our SOMAIR mine, we have the following ongoing projects:
- a new connected stick coupling a very precise differential GPS (10cm uncertainty in altitude) and a gamma measurement, make it possible to ensure the selectivity and traceability of the ore, while avoiding having to go through a gantry measured.
- a gamma measurement on a belt conveyor after crushing of the ore, allowing to have a precise mine balance and to perform a good grade selectivity between the ore leached in heaps and that of higher grade entering our processing plant
- an X-ray fluorescence measurement to characterize the uranium, the penalizing agents (carbonate and clay) in order to optimize the quantity of reagent necessary to leach the ore in the front end workshop of the ore processing plant.

All these developments allow Orano mining to better characterize these ores with an objective of industrial performance aimed at:
- a reduction in local uncertainties on Uranium reserves of our ISR deposits, for a better positioning of our production cells
- an improvement in mining selectivity
- an increase in leaching yields

09 Environmental and Medical Sciences / 210

#09-210 Radionuclides contamination in soil; effect, source and spatial distribution

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Radionuclide concentrations in the soil depend on the geological and anthropogenic activities of an area. They influence level of gamma radiation in the environment, which can cause significant health risk in humans. Due to the non-uniform distribution of radionuclides in the soil, various measurement methods have been adopted to monitor our environment. The challenges involved in collecting environmental samples, duration, technicality, and cost of measurement have led to various models for predicting unmeasured locations. This article presents geostatistical method using kriging techniques, which adopt the theory of regionalized variables, to characterize the spatial distribution of radionuclide in unsampled locations using data obtained from the sampled location. Knowledge of spatial distribution of radionuclides provides important information needed by regulatory authorities in identifying the contaminated area that are in need of remediation.

01 Fundamental Physics / 211

#01-211 Production and Monitoring of Neutron Flux by Activation Detectors

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At the microtron M-10 of Uzhgorod National University there are carried out experimental investigations of photonuclear reactions as well as applied studies of irradiation influence on the properties of new technological materials and electronical components. It is possible to use electron, bremsstrahlung and neutron fluxes.
The bremsstrahlung is generated by output electron beam at few millimeter thick tungsten or titan converter. Neutrons can be generated on 9Be target in (γ,n) reaction (with a reaction threshold 1.67 MeV). To measure neutron flux we use 59Co as activation detectors. Cobalt detector is activated by neutrons to 60Co ground state and to isomeric level 60mCo with half-life 10.47 minutes and following isomeric transition Eγ = 58.6 keV to the ground state. After β-decay (half-time 5.27 years) the ground state gives well known two lines Eγ = 1173 keV and Eγ = 1332 keV.
The neutron flax was tested at the microtron for 8.6 MeV output electron beam, which was converted to bremsstrahlung by 2 mm thick tungsten plate. The cobalt detector (boxed powder CoCO3*Co(OH)2*nH2O, mass 31 g) was irradiated during 10 min by neutrons generated by bremsstrahlung at beryllium block of size Ø10x14 cm and weight 2 kg.
Measurements of γ-radiation from the activated cobalt were performed with a spectrometer, which includes a NaI(Tl) scintillation detector with an SBS-40 amplifier-to-digital converter board, which is connected to a computer (the spectrum is built using AkWin). The energy resolution of the spectrometer was 7% (FWHM) for line Eγ = 1173 keV. The flux density of thermal neutrons was 2·107 n/(cm2·s).
Due to the presence of isomer state 60mCo cobalt as neutron detector is very convenient.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 212

#07-212 Evaluation of the characteristics of CdZnTe-detectors for the quality of non-destructive assays of nuclear fuel using passive tomography methods

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Non-destructive assays (NDA), carried out to account for nuclear materials, generally include the measurement of the initial enrichment of fresh fuel, the burnup, the colling time and the initial enrichment of the irradiated nuclear fuel, and distribution of nuclear materials and fission products over the volume of fuel assemblies in order to identify the fact of extraction of nuclear materials. In addition, the distribution of fission products makes it possible to detect a damaged fuel rod during refueling of the reactor core. Now the amount of research aimed at developing such systems has increased significantly. One of the universal technologies for such control is passive reconstructive tomography. We have previously presented systems designed to solve the listed tasks in real-time based on spectrometric measurements with CdZnTe detectors. All implementations require a multi-detector system that is installed on the mast of the refueling r machine. It has been shown that the implementation of the SVD decomposition algorithm for passive reconstructive tomography using several energies allows for two to three orders of magnitude to improve the signal-to-noise ratio to improve the quality of measurements in comparison with traditional algorithms. Now there are publications that the use of machine learning methods can provide a similar result. However, the implementation of spectrometric measurements in real-time places high demands on hardware and software.

We have developed a method for evaluating and optimizing the parameters of the measuring cycle based on the analysis of the amount of information. At the first step, a measurement model is created that formalizes the relationship between physical quantities describing the measured object and the results of observations. As a rule, this is a system of equations in which the instrumental spectrum of one detector is represented by the sum of the products of the sensitivity and the radiation flux density in a given energy range, taking into account the background component. In this case, the measurement channel is understood as the digital radiation spectrum for one detector for a given spatial configuration of radiation sources. It is obvious that the complex dependences of per-channel sensitivities (here it is a complex indicator depending) on the parameters of the detectors and the measured fields limit the possibilities of obtaining them analytically. Therefore, the experimental results on the dependence of per-channel sensitivities were supplemented by simulations in GEANT. Then, for each experiment, we build an information matrix and use it to determine the amount of information in the experiment. The analysis of the previously obtained results showed that this matrix can be used to analyze the quality of NDA by passive tomography methods. In spectrometric measurements, the information matrix explicitly contains the registration efficiency, energy resolution, peak/Compton ratio, and the range of measured energies. In the future, we solve the optimization problem for one of the parameters. To check the obtained calculated dependences, laboratory experiments were carried out. The experimental setup consists of a fuel assembly simulator and a multi-detector measurement system. The fuel assembly simulator consists of aluminum tubes located at the vertices of a regular triangle and thorium sand, the number of tubes may vary. The measuring system is based on μSPEC micro spectrometers (ZRF Ritec SIA) and SDP310, SDP500 CdZnTe-detectors of several models with different values of the studied parameters. A detailed analysis of the obtained dependencies will be presented in the presentation.

11 Current Trends in Development of Radiation Detectors / 213

#11-213 Neutron Beam Monitoring Using Nitrogen-doped Optical Fiber

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For the last three decades, the number of active irradiation facilities is continuously increasing, for various needs, from fundamental research to radiation testing or medical applications. To develop and guarantee the facility's technologies, requests for its control and reliability are becoming more and more precise and multiply the number of beam diagnostics from the source to the final beam. Among the different kinds of beam diagnostics, the control of the beam dimension and its particles flux characterization are the most crucial needs for irradiation facility operators. Moreover, information about the available integrated dose or dose rate also helps following the evolution of the irradiation from the user's standpoint. For these three characterizations (beam dimension, flux and the dose rate), beam monitoring sensors based on the use of optical materials are showing very high interest for their response under radiation.
Fiber-based sensors mainly exploit the optical and structural properties of the pure or doped amorphous silica structure. The emerging and fast development of various manufacturing processes (chemical vapor deposition, sol-gel, 3D printing, polymers) allow defining and manufacturing materials with very diverse properties useful for sensing. Optical fibers combining extremely low losses and sub-millimetric volumes are now available and are progressively used in harsh environments for both data transfer and sensing
Under irradiation, the optical fiber properties are modified by different radiation effects, the amplitudes of these changes are depending on both the dopant and the beam parameters such as dose, dose rate, flux, fluence, beam particle type and energy. The radiation induced behaviours of the optical fibers are known in three categories [1]. The first one, which will not be treated in this paper, is the compaction. This appears for neutron fluences [2] largely exceeding those encountered in our experiments. The second effect is described as an opacification of the optical fiber material under radiation and is called radiation-induced attenuation (RIA) [1]. This darkening is cumulative with the dose absorbed by the optical fiber and can be used for dose monitoring. Finally, the last induced effect is a light emission induced by the energy transferred from the radiation or particle strike with the matter. When a particle or a photon interacts the doped SiO2 structure, the energy loss is high enough to ionize the atoms of the structure. The presence of selected dopants adds new defect levels into the SiO2 bandgap which may trap the recombining electrons. These trapped electrons recombine by radiative decay which can emits a photon guided by the optical fiber. This process is named radiation induced luminescence (RIL). The RIL intensity is a function of the particle flux and can be used after calibration to monitor the beam flux, and then the beam profile. This last phenomenon is the one employed in this work, based on a small-size (50µm doped core, 250µm coating) optical fiber that already demonstrated very efficient radiation detection performances under both X-ray and high energy protons [3].
For this study, we used this RIL phenomenon to monitor the flux of a 14 MeV neutron beam at the GENEPI2 (Intense Pulsed Neutron GEnerator) facility (Grenoble, France) [4]. We characterize the evolution of the fiber response in terms of RIL vs the neutron flux between 1x106 n.cm-2.s-1 and 3x107 n.cm-2.s-1 and fluence, defining eg. the threshold limit for the detected flux. During this experimental campaign using a specific doped optical fiber, we demonstrate the potential of this nitrogen-doped optical fiber to monitor in situ the flux evolutions and the adaptability of this kind of sensors to extend the neutron flux range.

01 Fundamental Physics / 215

#01-215 Experiments with fast neutrons at nELBE

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The neutron time-of-flight facility nELBE at Helmholtz-Zentrum Dresden-Rossendorf features the first photo-neutron source at a superconducting electron accelerator. The electrons are focused onto a liquid lead target to produce bremsstrahlung which in turn produces neutrons via photo-nuclear reactions. The emitted neutron spectrum ranges from about 10 keV up to 15 MeV with a source strength of above 10^{11} neutrons per second. The very precise time structure of the accelerator with a bunch width of a few ps enables time-of-flight measurements at very short flight path and experiments to investigate the time response of novel detector concepts.
The high repetition rate of 100 to 400 kHz in combination with the low instantaneous flux and the absence of any moderating materials provide favorable background conditions.
The very flexible beam properties at nELBE enable a broad range of nuclear physics experiments. Examples for the versatility of nELBE will be presented: From transmission measurements and inelastic neutron scattering and fission experiments to determine nuclear reaction cross sections with relevance for fundamental nuclear physics, reactor safety calculations, nuclear transmutation or particle therapy to experiments to investigate the response of novel particle detectors e.g. for dark matter search experiments, nuclear instrumentation or the range verification in cancer treatment.

11 Current Trends in Development of Radiation Detectors / 216

#11-216 Experimental and Simulation Investigation of Micro- and Nano-Structure Neutron Detectors

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In this project, we are investigating different micro- and nano-structure approaches to neutron detection based on inorganic scintillators. Specifically, we have been assessing various micro- and nano-geometries to maximize the fast-neutron detection efficiency. Our approach is based on extensive Geant4 simulations that are supported through tailored experiments aimed at thorough validation of our simulation models.
Some inorganic-scintillator-based sensors that are used for neutron detection utilize two types of materials: One of them serves as a neutron converter (to convert neutrons into another, shorter-range radiation type), while the other one is typically an inorganic scintillator. The neutron converter captures incoming neutrons and emits heavy charged particles (HCPs), while the scintillator converts the HCP energy into visible light. One such detector is EJ-426 sheet from Eljen Technologies. EJ-426 neutron detector sheets are manufactured by mixing 6LiF and ZnS(Ag) powders in a transparent binder material, and this scintillator has served as the reference material for our project in terms of neutron detection efficiency.
Accurate modeling of such micro- and nano-structure materials has proven challenging for a variety of reasons. One of those reasons is the highly irregular nature of the detector's constituents with varying grain size of powders and the volumetric grain nonuniformity within the detector medium. Other reasons include unknown or poorly known optical properties of the constituent materials. The method that we have investigated to accurately model the existing and new-generation neutron detectors necessitates mutiple experiments in which grain size distribution, positional distribution of powders, and optical densities and refractive indices of constituent materials are measured. Moreover, in order to improve our Geant4 simulations, we are in the process of measuring the single photon response (SPR) of the photosensors that are being utilized for the neutron detection experiments. Experiments that involve extensive material characterization have been carried out in Material Characterization Laboratory (MCL) at the Pennsylvania State University. Specifically, Cary 7000 UV-VIS-NIR spectrophotometer from Agilent has been used to measure the total reflectance and transmittance for different EJ-426 sheets. Absorption length, which is one of the most crucial parameters for scintillator simulations, has been calculated for these EJ-426 sheets and used for the simulations. In addition, the Scanning Electron Microscopy (SEM) facility of MCL is being used to investigate the morphology of the different types of EJ-426 sheets. These investigations will involve the characterization of grains size variation and grain distribution in the sheets to improve the detector simulation models. In addition to measuring the optical properties of bulk EJ-426 sheets, optical properties of individual LiF and ZnS powders are investigated after depositing them in thin layers.
Resulting data from the aforementioned experiments will substantially strengthen our simulation models. Finally, single photon counting (SPC) experiments are being carried out for various Silicon Photomultipliers (SIPMs) to investigate the single photon response (SPR) for these sensors. ZFL-1000LN+ low noise amplifiers from Mini-Circuits are used to amplify the single photon signals. In addition, an air-cooler from TE technology will be employed to cross-check the SPR as a function of temperature. The SPR information will be beneficial for comparing the experiments to simulations more accurately since the ultimate simulation result in Geant4 is the number of light photons detected by photosensors. The experimental results, which will include optical absorption depth, grain size morphology, and SPR will be used as experimentally acquired inputs to accurately simulate EJ-426 sheets by using realistic geometry models and optical parameters. These improved simulations are paramount for our ongoing investigation of designing and optimizing various micro- and nano-structure neutron detectors.

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#09-218 Particle identification and tracking by the use of a pixel-based semiconductor radiation detector coupled with voltage controlled oscillators

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In this paper, an approach based on a semiconductor radiation detector exhibiting a matrix organization coupled with the use of rings oscillators for current detection is developed. This approach is based on the reading of the information related to indirect output parameters of the detection chain instead of directly measuring the current from the sensor. This solution is interesting because it avoids most of the design problems related to the use of a charge sensitive amplifier and keeps the current shape. Indeed, in applications such as gas prospection or medical therapy, the knowledge of the electrical signature of the particles is required. Moreover, knowing the actual shape of the generated current at the output of the detector allows easier post processing of the signal. Then, in our system, the information is extracted by correlating the initial oscillation of the system with that of the system after the particle has passed the detector (Fig. 1). To be able to make a diagnosis, the requirement is then to link the output information (i.e. the mean voltage of the oscillator) to the input information (current stimuli). The initial version of the chain was implemented in bulk technology. As the full integrated circuit (detector + readout electronic) is supposed to work in a radiative environment, the chain has been implemented on a 130nm partially depleted Silicon On Insulator (SOI) process, in order to make it more tolerant to radiation.
The detection chain has been simulated at circuit level using "Spectre" simulator (SPICE-based) under Cadence DFII ©. The readout circuits is composed of a ring voltage controlled oscillator (VCO) working at high frequency. Each VCO is composed of three specific delay cell in CMOS-SOI technology making their implementation very simple. This simplicity allows several configurations of use. The current source at the input of the VCO comes from the realistic simulation of the 3x5 matrix using TCAD device simulation tools (Synopsis ©). The effect of the ion strike is simulated using the Heavy Ion module of Synopsis [1], considering an electron-hole pair column centered on the ion track axis. The Linear Energy Transfer is defined as the energy lost by the particle, by unit of length [2]. An actual variation of the LET was integrated in our simulations, based on the value given by SRIM tables [3]. The VCO based chain presented here is optimized for the detection and identification of particle fluxes lower than 109 particles per detection area and per second. This chain could be particularly suitable for the detection of low energy particle such as alpha particle crossing the device with an initial energy of 0.9MeV, corresponding to a range of 2.3 m in the silicon matrix .
The aim is to study the end of the path corresponding to an alpha particle of 1.47MeV generated by the initial interaction of thermal neutron with boron-10. The case of a more energetic particle is also considered. This is a 50MeV Aluminum which could be produced by the interaction of fast protons with silicon, for instance [4].
In a previous work, the various currents generated in the matrix of detection were studied and several ways of improvement of the detection ability were explored [5]. We propose here a complementary solution to extract the total collected charge from the detector what should facilitate particle identification. Other particle species will be presented in the final paper.
After being generated by the sensor, the signal has to be conditioned in order to allow particles counting and/or recognition. The key point is how the output parameters of the VCO chain can give information related to the input current. This could be done through the analysis of various characteristics extracted from the VCO output voltage (Fig. 1) that is the variation of the average output signal (ΔVmax) versus the maximum of the input current (Imax) for different particle charges. In [6], a linearity curve linking input to output parameters was determined for a 2GHz VCO. A new linearity curve has been obtained for this new 4.3GHz VCO [5]. Through calibration curves, the output parameters can be linked to the input currents, which could allow the incoming particle tracking.
The matrix studied here contains fifteen contacts (Fig. 2). All the currents of the matrix, generated using Synopsis have been injected in the fifteen VCOs contacts. The novelty in this work is that one more VCO is added in order to collect the the total collected charge. Indeed, the linearity of the VCO response may be affected if too high magnitude level of current is produced by the pixel detector, leading to a saturation of the VCO response. In order to overcome this limitation, we increased the size of the bias transistor of the delay cell composing the oscillator, leading to a high sensitive readout circuit but working a lower frequency. This second topology has been used for the total charge calibration. Then oscillator used at the bottom contact oscillates at 2.98GHz.

Fig.3 presents first results obtained with this VCO on four cases: an Al particle an alpha particle following two configuration corresponding to an horizontal crossing and a diagonal crossing of the cell. On the axis, the total collected charge at the bottom contact of the matrix is plotted. Its corresponds to the amount of charge collected in the whole structure. This gives a direct information about the deposited charge. Then the charge deposited by the Al particle is higher than the one of the alpha, and the charge collected for an horizontal track crossing 5.05µm is higher than the diagonal case crossing 2.82µm. Theses analysis will be details in the final paper. This trend is respected by the integral of the mean voltage value of the VCO. Then the linearity between the output and the input response is preserved (Fig.4). This will be checked for other particles species in the final paper. The tracking ability will be also detailed through the analysis of the currents from the fifteen other VCO.

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#11-220 Neutron Measurement Using Pulse Shape Discrimination Method with 3D-Printed Plastic Scintillator

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Neutron measurement technologies are being studied due to the demand in nuclear fusion, accelerator facilities, nuclear non-proliferation treaty, etc. Since neutron fields are typically present with gamma-ray, the separation technique is necessary such as pulse-shape discrimination (PSD) for neutron detection. Organic scintillators, such as plastic scintillators, are capable PSD using different decay times depending on particles. Plastic scintillators can be manufactured by 3D-printing technique based on digital lighting processing (DLP).

The PSD performance of plastic scintillators composed of polymer, which is doped with the 2,5-diphenyloxazole (PPO), is affected by concentration of PPO. In this study, 3D printing resin contain PPO concentration of up to 30wt%, and the scintillator based on these resins can measure fast neutron. The PPO and wave length shifter concentration were optimized, and 6LiF was doped for thermal neutron detectable scintillator. The scintillators, which have diameter of 2.54 cm and thickness of 1.27 cm, were connected to PMT (Hamamatsu-H6410) and FADC (Notice-NGT400). Data acquisition (DAQ) system based on ROOT framework was operated in PC through Ethernet. For neutron irradiations of this work, ^{252}Cf source (88.3 μCi), which shielded with 5 cm of lead and moderated with high density polyethylene of 11 cm, was used. The energy calibration was performed with Gaussian fitted integral through Compton edge peaks of 137Cs source (10.4 μCi) and 22Na source (5.0 μCi), as an organic scintillator calibration method. The PSD capability were evaluated by Figure of Merit (FOM) value. The reasonable definition of the capability of PSD for well separated is FOM ≥ 1.27 similar to that of single crystals. The three samples, which were 3D printed with high concentration PPO and 6LiF, met the target value in measurable energy region. It is expected that the 3D-printed scintillators which are comparable to the existing commercial detectors, can be developed through continuous research and optimization.

04 Research Reactors and Particle Accelerators / 224

#04-224 Optimization of Next-Generation Fast-Spectrum Self-Powered Neutron Detectors using Geant4

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Self-powered neutron detectors (SPND) have been a common diagnostic tool for intra-core neutron flux mapping in thermal nuclear reactors for more than 45 years. They are attractive flux monitors as they are compact, simple, and produce a signal proportional to local neutron flux without the need for an external source of power. Signal in these detectors is driven by electrons generated from nuclear reactions within the emitter in the detector. As the next-generation reactors are on the horizon and some of them are designed to be operating with harder neutron spectra, it is imperative to develop diagnostic tools tuned to their neutron spectrum, peaking around 0.5 MeV. However, the current state-of-the-art for SPNDs is optimized for thermal neutron interactions. We will be discussing our efforts to develop fast-spectrum SPNDs (FS-SPND) that are sensitive to neutrons with energies ranging from thermal up to 1 MeV. We have performed an in-depth analysis of ENDF VII.1 neutron-capture cross sections and have identified 4 novel materials that are suitable emitter candidate materials to measure fast neutrons, all are stable mid-shell nuclei in the region between doubly-magic 132Sn and 208Pb. We will be discussing the results of Geant4 simulations for each emitter candidate with detector parameters optimized to maximize overall detector efficiency as well as an exploration of complex detector geometries.ocess.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 226

#07-226 Study of natural uranium fuel for new reactor design TEPLATOR

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The TEPLATOR is a new type of nuclear reactor which the main purpose is producing heat for district heating. It is designed as a special thermal reactor with 55 fuel channels for fuel assemblies, which is moderated and cooled by heavy water and operated around atmospheric pressure. The TEPLATOR DEMO is designed for using irradiated fuel from PWR or BWR reactors. Using heavy water as the moderator and coolant in this reactor concept allows to use natural uranium as an alternative fuel in case that the irradiated fuel is not available for some reasons. This solution is suitable because of the price of natural uranium and the absence of costly fuel enrichment. This article is focused on deeper analyses of alternative suitable fuel for TEPLATOR based on natural uranium and new fuel geometries. This work builds on previous research on alternative fuel material and geometry for the TEPLATOR. It is mainly concerned with the neutronic development of fuel assemblies, the possibility of manufacturing of developed fuel types and optimization of fuel management and uranium consumption. This article contains predetermined candidates of suitable fuel geometries and new untested types of fuel geometry with some new advantages. Finally, the optimization of the whole reactor core and number of fuel channels in terms of higher safety and higher fuel burnup were made. Presented calculations were performed by Monte Carlo code Seprent.

08 Decommissioning, Dismantling and Remote Handling / 227

#08-227 Handheld Spectrometers: From CZT To NaI, an Embedded Template Analysis for Isotope Identification

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This paper presents a solution for embedding an isotope identification algorithm on portable gamma spectrometry systems based on template approach. There exist many algorithms able to identify isotopes in a spectrum. These algorithms can be classified into two large families. The first family is based on peak search or Region Of Interest (ROI) in the spectrum, the second on a global template analysis or deconvolution [1]. All of these algorithms can be evaluated using off-line solutions as described in a large amount of papers [2-4]. In the case of algorithm embedded on handheld spectrometers, the first family is commonly used due to their low requirements in term of computing resources and memory. However, the new generation of microcontrollers or small board computers allows template analysis algorithm that need more computing resources to be embedded.
In this paper, we present a template analysis, based on MLEM algorithm [5], and its implementation on two different targets, a STM32 microprocessor and a Raspberry PI. Time performances show that this kind of approaches can be used in handheld system such as spectrometers and portable isotope identifier. Experimental results are based on two use cases, the first applies the requirement of ANSI42.34 [6], the algorithm can be performed at the end of the measurement, the second needs to provide evolution of the isotope identification as the progress measure.
Some promising results have already shown that this kind of algorithms can be uses for handheld spectrometers. Both implementation will be described and results of their implementation will be presented for a set of different detectors from CZT to NaI. The study will be focused on CZT, LaBr3/CeBr3 and NaI detectors. For each detectors, a spectra database is performed based on isotopes described in ANSI 42.34 norm. We have developped pocket spectrometer providing spectral information on an ultra-portable and wireless CZT prototype capable of providing identification information on the radioelement being detected. The system includes a battery, a screen allowing the visualization and an on-board PC (raspberry PI) that perform online identification of the detected. Base on this system and others developments for others detectors, the impact of energy resolution and algorithm implementation will be discussed in the full paper.
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01 Fundamental Physics / 228

#01-228 A high speed data link optimization for digitalized data transfer to processing FPGAs

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State-of-the-art arrays of detectors, that require digital processing, may have a sizeable number of digitalized signal links. This is the case in several experimental nuclear physics instruments. Moreover, the sampled signals data rate, defined primary by the signal bandwidth of the individual detector, may not exhaust the capabilities of a single FPGA transceiver input.
This is a critical issue in position sensitive HPGe segmented detectors as AGATA, where each crystal detector has 36 segment signals plus a core signal. The 37 signals for each crystal detector are digitalized and processed with an independent electronics. The sampling is performed with 14 bit (about 12 ENOB) at 100MHz due to the bandwidth limitation to about 30 MHz of the detector and preamplifier response. Each ADC has a link to the preprocessing system at 2Gbps in a JESD204 protocol.
The preprocessing is carried out in a modern FPGA with transceiver data rate capabilities over 10Gbps. Moreover, cost effective FPGA have a limited number of transceivers for a given FPGA processing capabilities. The investigation of a cost-effective and efficient solution to the mismatch between both data rates, optimizing the use of the FPGA resources, is the topic of the present work.

We have developed a solution based on the Time Domain Multiplexing link aggregation, in the form of a Mezzanine board. This mezzanine combines four channels from an optical or copper input up to 2.5 Gbps to one up to 10Gbps, and serves them to the FPGA via the mezzanine connector. The board itself is controlled by a small FPGA by Two Wire Interface (TWI) protocol as a standalone intelligent device, with minimum slow control needed. An associated firmware has been developed to de-aggregate the data in the FPGA and recover the original digitalized data, with the JESD204 cores, inside the FPGA. The method has been validated and applications, beyond the development of the AGATA electronics, may be envisioned.

02 Space Sciences and Technology / 229

#02-229 Introduction of hybrid silicon/scintillator detector for space experiment BION-M2

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Space experiment BION-M2 focuses on research of effects of ionizing radiation to biological samples such as geckos and mice. The return module of BION-M2 carries the life support systems for all living organisms. The launch of the satellite is planned on the year of 2023, the average altitude will be 800 km.
The newly developed hybrid silicon/scintillator detector will support this experiment with measurement of radiation doses in combination with other detectors such as thermoluminescence detectors, track-etched detectors CR-39 and silicon-based detectors.
The detector is based on silicon strips which are deployed in 4 layers. The layers are placed in such pattern that when a charged particle crosses all four layers the incident angle of the particle can be calculated. Each layer contains 64 strips. Altogether there are 256 strips in all layers. The block of plastic scintillator EJ-276 is placed between second and third layer. Plastic scintillator is coupled with silicon photomultiplier (SiPM) which converts scintillation photons to electrical signal. EJ-276 supports technology of pulse shape discrimination (PSD) which can be used to estimate the linear energy transfer (LET) of the particle. The considerable effort has been made to optimize the PSD performance with the SiPM component. The detector is designed to manage large fluences of radiation which can be encounter for example when satellite passes through South Atlantic Anomaly (SAA) which is located at the coast of Brazil. Such arrangement of the detector provides detection capabilities to calculate radiation quantities related to biological response to ionizing radiation. The goal is to obtain time resolved LET spectrum, absorbed dose and dose equivalent rates.
We would like to introduce newly developed hybrid silicon/scintillator detector and present the preliminary results from experiments performed at high energy accelerators. The type of the particles and their energy were chosen so it was close to the ionizing radiation at Earth orbit.

10 Education, Training and Outreach / 230

#10-230 Development of Active-Learning Units in Nuclear Engineering

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Active learning engages students in activities that could enhances their ability to analyze, synthesize, and evaluate the material being learned. The students participate in doing things instead of just listening. Evidence-based studies have shown that active learning increases student performance in Science, Technology, Engineering, and Mathematics (STEM) courses. The goal of this project is to develop active-learning units to enhance students learning and technical skills to improve their preparation for success in pursuing STEM graduate programs and careers in nuclear engineering. Three modes of active learning that are of interest include: problem-solving, lab-based hands-on activities, and simulation. This paper focuses on the development of problem-solving interactive units aimed at mastering fundamental principles and concepts, and better understanding of how equations translate and apply to real-life engineering situations. It also enhances the understanding of how different parameters in an equation interact with each other (such as dependency relationships). The practicality is in understanding how different components of an engineering systems function together to accomplish the goals for which they are designed. The design approach to each problem-solving unit starts with a brief review of the fundamental concepts, and identification of key parameters. Students participates in determining independent and dependent variables, constant parameters, and equations/formulas associated with the problem. The steps in each unit include understanding the problem, identification of the equations/formulas needed to solve the problem, listing the knowns and the unknowns, solving the problem, testing/checking the solutions/answers, and write down any difficulties encountered. Part of the post-activity evaluation include identifying any difficulties encountered, how were they resolved, and any lessons learned. The problem units presented in this paper are in the areas of neutron interactions, cross-sections, and attenuation.

11 Current Trends in Development of Radiation Detectors / 231

#11-231 Regional variation in neutron/gamma pulse-shape discrimination in an organic scintillator

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This paper describes the use of a Hamamatsu H13700 16×16 multi-anode photomultiplier tube (MAPMT) to quantify regional variations in the confidence of neutron/gamma discrimination across the volume of a continuous, organic scintillator. The MAPMT outputs are multiplexed to a single analogue input channel on a mixed-field analyser performing pulse-shape discrimination (PSD) by pulse-gradient analysis (PGA). Accuracy of the PSD response is compared for events occurring within different regions of the scintillator volume by varying the centre of interaction using an aperture collimator. Width and uniformity of the light pulse dispersion is inferred by varying the number of readout anodes used and comparing the change in analogue output to supplied crosstalk data. Although this study is ongoing, these findings could inform future PSD developments to increase certainty in particle identification and position-sensitive neutron counting methods for nuclear safeguarding and materials assay.

05 Nuclear Power Reactors Monitoring and Control / 234

#05-234 Advanced sensor technologies for real-time temperature measurement in nuclear reactors

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In 2012 the US Department of Energy Office of Nuclear Energy (DOE-NE) initiated the Advanced Sensors and Instrumentation (ASI) program as part of the Nuclear Energy Enabling Technologies (NEET) research as part of crosscutting RD&D activities to advance the state of nuclear technology, improve its competitiveness, and promote continued contribution to meet the nation's energy and environmental challenges. The objective of the ASI program is to provide reliable, cost-effective, real-time, accurate, and high-resolution measurement of the performance of existing and advanced reactor core and plant systems. Instruments are designed, fabricated, and tested in relevant and operational conditions to advance their technological readiness to a point in which they can be integrated in I&C systems without the significant cost and risk associated with development activities. This technology maturation is made possible by the deployment of developmental instrumentation in Material Test Reactors (MTRs) irradiation experiments aimed at the characterization of advanced reactor components, such as advanced fuel forms. In the near term therefore the requirements on sensor technologies are driven by the irradiation test conditions and their research objectives.

This paper reports on the effort to develop technological solutions to the measurement of temperature in irradiation experiments aimed at the development of core components of nuclear reactors, including its fuel assemblies. The scope of the work extends from the current fleet (primarily Pressurized Water Reactors) to advanced reactor concepts, including Small Modular Reactors and microreactors. Considering experiments that address both operational conditions and design basis accident cases, the requirements on temperature measurements extend up to 2000℃ under a wide range of neutron flux and integrated fluence depending on the facility under consideration – from a peak thermal flux of 1x1015 n/cm2-s and fast flux of 5x1014 n/cm2-s in the Advanced Test Reactor to the sub-second burst of neutron and gamma radiation in the Transient Reactor Test Facility (TREAT), both located at the Idaho National Laboratory. In addition to operating temperature and radiation resistance, continuous or discrete distributed sensing capability is another important requirement to consider for technologies development.

Progress in three areas of research are reported: High Temperature Irradiation Resistant (HTIR) thermocouples, Ultrasound Thermometers and optical fiber sensors. HTIRs are thermocouples based on Molybdenum and Niobium alloys and they have been investigated at INL for more than 10 years and have demonstrated reliable performance up to 1600℃. The recent deployment in irradiation test aimed at the characterization of TRISO fuel, in parallel with commercialization efforts, enabled the collection of sufficient data to satisfy the Preliminary Design Review towards their qualification process. Ultrasound Thermometers are acoustic sensors that utilize a radiation resistant transducer and a passive waveguide to derive temperature from the measured sound speed in the waveguide materials. Discrete features in a solid waveguide or a waveguide design based on a bundle of thin wires with stacked ends allows discrete multi-point sensing. The selection of refractory alloys for the waveguide (Molybdenum or Tungsten) allows temperature measurement beyond 2000℃, providing the transducer is maintained at or near reactor coolant temperature. Optical fiber sensing is now considered for many irradiation tests in TREAT, and through accurate selection of fiber materials (radiation hardened fibers) and interrogation techniques their performance characterization has been extended to neutron fluences that are compatible with extended testing in high flux facilities such as ATR. Temperature and radiation limits depend strongly on the sensing method and the fiber material, limiting their current applicability to less than 800℃. Several examples of the use of optical fibers for temperature measurement in irradiation experiment are presented and their results discussed.

09 Environmental and Medical Sciences / 235

#09-235 The Wearable PET project for a compact clinical exam for an early diagnosis

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The Wearable PET (WPET) project was successfully supported by EU-ATTRACT (https://attract-eu.com). Its objective is to demonstrate the feasibility of a light PET systems for cancer preventive screening which can be hosted by wearable vest.
Positron Emission Tomography or PET is a very precise imaging modality commonly used in hospital oncology units to test patients with cancer to establish possible metastasis or other complex diseases difficult to diagnose with traditional methodologies. In PET a radioactive positron emitter is injected into the patient's body. Sugar is attracted by infected human cells and cancer cells, which therefore accumulate the positron emitter. The emitted positrons annihilate producing two high energy gamma rays back to back that can be detected outside the human body.
Traditional PET systems detect the two emitted gammas and reconstruct the emission point. These devices produce a 3D image of the full body reconstructing the cancer location with spatial resolution of the millimetre. The disadvantages of these systems are that are expensive, massive and can only be used for a limited number of tests daily.
On the contrary, the WPET, that performs a similar, but less detailed, measurement for a longer time, would allow one to use this modality for routine screening for a large fraction of the population at risk, improving early stage detection.
The WPET project aims at building a wearable PET scanner with a modular, flexible design, able to image different body parts according to specific needs, as well as to track detector motion and correct its effects in real-time. This is made possible not only by lighter and compact LYSO-SiPM detector modules, but also by accompanying developments in miniaturized readout electronics, battery technology, wearable sensors, wireless data transmission and fast PET image reconstruction algorithms. WPET requires a novel system design to combine wearable electronics with novel materials, controls, batteries and data transmission components to be hosted in a comfortable wearable support. Storage, handling and visualization are also part of the package on the medical side for an easy diagnosis of the collected data.
Within the project Geant4 MonteCarlo simulations were performed to optimize the jacket design and evaluate the feasibility. Reconstruction algorithms were used to demonstrate the proof of concept performances. With this paper we would like to present the WPET concept and the early results obtained. Using simulation and tests with single modules we can conclude that the fabrication and use of a WPET system is feasible. The key to feasibility is scalability, miniaturization, data storage and handling and precise position monitoring. With current technology, WPET can detect tumours as small as 2 mm with 10 kg crystal weight in 6 hours, using the same tracer dose as conventional clinical PET.

11 Current Trends in Development of Radiation Detectors / 236

#11-236 Large area scanning of painted arts with photon counting detectors

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The restoration and preservation processes of the old painted arts are often combined with more or less extensive inspection of work. Any manipulation with the investigated object needs to be non-invasive and non-destructive as the historical price of the artefacts can be incalculable. Information about the sample surface can be obtained by visual methods varying from digital photography to advanced methods like infrared reflectography or ultraviolet fluorescence. Nevertheless, visual techniques are unable to provide information about the internal structure of the investigated sample. Such information can be obtained only by transmission methods utilizing more penetrating radiation such as X-ray radiography or various adapted techniques of computed tomography (limited-angle tomography, laminography or tomosynthesis).

Radiography systems used in this work are equipped with the large area detector based on Timepix photon counting technology. Such detectors are operated in a dark-current-free data collection mode with energy-resolving capabilities. Besides conventional X-ray radiography or CT, these detectors are, therefore, convenient for advanced approaches like energy-sensitive radiography or X-ray fluorescence imaging.

The historical painted arts are typically flat and vary in dimensions from several decimetres to square metres. This range is significantly larger than the standard size of any available imaging detector. This arises a need for advanced scanning devices and techniques allowing imaging of objects significantly larger than the detector field of view. The Institute of Experimental and Applied Physics, Czech Technical University in Prague in cooperation with ALMA laboratory of the Academy of Fine Arts designed and constructed a device dedicated for radiographic inspection of large-area painted artwork. The device allows automatic scanning of paintings in step-and-shoot manner in which the sample is acquired in a set of partially overlapping tiles. The tiles are then co-registered and merged into a final radiographic image.

The obtained radiographies typically consist of hundreds of megapixels and bring very detailed information about the condition of the inspected artwork that can be used for evaluation of any structural damage, later restoration interventions or even complete overpainting of the whole artwork. We would like to summarize the recent results in this field achieved at the Institute of Experimental and Applied Physics in close collaboration with our partners. High-resolution X-ray images of historical paintings of different scales are shown. Different scanning strategies are discussed. Potential upgrades of the mentioned set-up to further exploit its functionality like topographic mapping of elemental composition using X-ray fluorescence are mentioned.

09 Environmental and Medical Sciences / 237

#09-237 Applicability of large-area single-photon counting detectors Timepix for high-resolution and high-contrast X-ray imaging of biology samples

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High-resolution X-ray imaging techniques, usually known as micro-radiography and micro-CT, have become highly required and frequently used tools for biology, biomedical and pre-clinical research. State-of-the-art micro-CT scanners are capable of achieving spatial resolution of few micrometers or even less thanks to the constant development of compact micro-focus X-ray sources with simultaneous progress in detector technologies.
X-ray micro-CT is frequently used for phenotyping, study of zoomorphology, drug development and others. Moreover, it is nowadays used for non-destructive inspection of ex vivo soft biology tissue. Such technique has become known as virtual histology. High-resolution X-ray imaging currently becomes a competitor to conventional research techniques used in biology research like optical microscopy and histology.
The current standard in X-ray detection is a digital read-out chip coupled with a scintillation sensor. Such detectors are available in variety of different sizes, they are easy to use and relatively affordable. Nevertheless, the mentioned technology suffers from inherent technology limitations, like for example undesirable generation of dark-current, that compromise the quality of the provided data.
This work demonstrates the advantages of large-area hybrid-pixel photon-counting detectors Timepix for high-resolution X-ray imaging in biology research. Photon-counting detection technology provides dark-current-free quantum-counting operation. Therefore, enhanced contrast-to noise ratio is of the acquired data is achieved. Furthermore, the biased semiconductor sensor achieves almost ideal point-spread-function resulting in high spatial-resolution of images. And finally, the detectors are operated with user-adjustable detection threshold opening possibilities for energy-sensitive X-ray imaging. Abovementioned features make photon-counting detectors to be excellent tools for high-resolution X-ray imaging of samples with low intrinsic absorption contrast like for example soft biological tissue.
We evaluate the imaging performance of large-area Timepix detectors compared to widely used scintillation-based X-ray imaging detectors dedicated for high-resolution X-ray imaging. Further, we summarize and demonstrate the applied results in the field of biology and pre-clinical research achieved at Institute of Experimental And Applied Physics, Czech Technical University in Prague. The institute, as a member of Medipix Collaboration, has actively participated in the development of Timepix technology from its introduction. The presented data obtained in cooperation with Charles University demonstrate the versatility of the used detectors as it covers a wide range of samples from laboratory animals to single-cell marine organisms. Finally, practical experience from long-term usage is discussed and the limitations of Timepix technology for X-ray imaging are mentioned.

11 Current Trends in Development of Radiation Detectors / 238

#11-238 Design and first tests of the S^3 detector of reactor antineutrinos

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The new experiment S^3 devoted to the study of reactor antineutrinos was designed and constructed as a common activity of IEAP CTU in Prague and JINR (Dubna). The S^3 detector (40 x 40 x 40 cm^3) is a highly segmented polystyrene-based scintillating detector composed of 80 detector elements (40 x 20 x 1 cm^3) with a gadolinium neutron converter between elements layers. A positron and a neutron are produced in an inverse beta decay initiated with an electron antineutrino in the detector. The high segmentation of the detector enables the identification of the antineutrino interaction which has a specific time, energy and spatial pattern. The signature of the signal event is an occurrence of a prompt signal from a positron and a delayed signal from a neutron interaction. Light produced by each detector element is collected via 19 wave-length shifting fibers to Silicon Photomultipliers (SiPM), which ensures transformation of light into an electric signal. A modular multi-channel fast ADC was developed for the data acquisition for the whole 80-channel S^3 detector and the 4-channel cosmic veto system. For the real-time visualization of signals and DAQ from the S^3 detector software has been developed and tested.

The detector meets very strict safety rules of nuclear power plants and can be installed in a chamber located immediately under the reactor. The close vicinity from the reactor enables to study neutrino properties with a higher efficiency, to investigate neutrino oscillations at short baselines and try to verify the hypothesis of a sterile neutrino. Since antineutrinos produced in the nuclear processes in the reactor fuel penetrate the reactor vessels and other reactor materials almost without an interaction, they can be also used as a reliable monitor of the reactor processes. Therefore, the S^3 detector can be used for the real-time measurement of the reactor power, determination of fuel burnout and control of the illegal extraction of ^{239}Pu.

The details of the design and construction of the S^3 detector, as well as properties of the modular multi-channel fast ADC will be presented. The whole detector setup was tested in an on-surface laboratory, in an underground bomb shelter (cosmic muons suppression ~ 5x), and with installed gamma and neutron shielding in order to measure signature of cosmic muon signals as well as background events. The properties of the S^3 detector will be demonstrated on the analyzed data.

06 Severe Accident Monitoring / 239

#06-239 X-Ray Imaging Calibration for Fuel-Coolant Interaction Experimental Facilities

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During a severe accident in sodium-cooled fast reactors, jets of molten nuclear fuel may penetrate into the coolant resulting in fuel-coolant interactions (FCI). Experimental programs are being conducted to study this phenomenology and to support the models development for evaluating consequences of severe accident. Due to the optical opacity of the test section walls and sodium coolant, high-speed X-ray imaging is the preferred technique for FCI visualization. The configuration of these X-ray imaging systems, whereby the test section is installed between a fan-beam X-ray source and a scintillator-image intensifier projecting an image in the visual spectrum onto a digital camera, entails certain imaging artefacts and uncertainties, not limited to vignetting, geometric distortion and noise in the detected photon flux. Although the X-ray imaging configuration can observe FCI process qualitatively, it ideally requires precise calibration to enable detailed quantitative characterization of the FCI. Calibration tests have been conducted for a new, enlarged, sodium test section at the MELT facility. To this end, 'phantom' models have been fabricated using polyethylene, either steel or hafnia powder, and empty cavities to represent the sodium, molten fuel and sodium vapor phases, observed during FCIs, respectively. The checkerboard configuration of the phantom enables calibration and correction for distortion artefacts which magnify features towards the edge of the field of view with fan-beam X-ray imaging. Polydisperse steel ball configurations enable precise determination of the minimum object size detectable by the camera, and the estimation of parallax errors which introduce uncertainty in an object's silhouette dimensions due to the uncertainty in its position within the depth field. Analysis of these calibration tests is presented with the objective of establishing a universal procedure for the optimization of FCI visualization experiments.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 241

#07-241 The Mini Labyrinth – A Simple Benchmark For Radiation Protection And Shielding Analysis

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Since World War II there has been a significant development of methods and approaches used in the calculation of radiation shielding. This development was directly supported by the needs of industry (military technology, nuclear power plants, food processing, medical applications, accelerators, etc.). Over time, modelling and simulation of relevant effects shifted from an analytical modelling to methods based on the so-called primary principles and their stochastic nature. Even nowadays it is necessary to know the accuracy of available computation codes, used nuclear data and it is desirable to evaluate the influence of the user on the final calculated parameter. One of the most effective ways of gaining user experience and minimizing user effects on the results of calculation is international collaboration comprising the designing and constructing of relevant benchmark experiments, following simulation with tools available at the workplace (engineering tools vs. high-fidelity methods), comparison of work group results and subsequent identification of the source of observed deviations from the experiment. The proposed paper comprises a definition of the simple neutron and gamma shielding benchmark, inspired by the ALARM-CF-AIR-LAB-001 ICSBEP experiment. The experimental setup consists of the PuBe neutron source, several NEUTRONSTOP C5 shielding blocks (polyethylene with 5 % boron), H2O filled PLA tank, plastic source holder and the active and passive detectors. The measured quantities are compared to values calculated by MONACO (as a part of SCALE 6.2.4 system) and MCNP 6 stochastic codes. The influence of different cross section libraries and propagation of cross section uncertainties is studied through the shielding analysis. The achieved results are included and finally, some discussions on further needed development are also included.

03 Fusion Diagnostics and Technology / 242

#03-242 In-situ gamma irradiation testing of radiation hardened chips till 1MGy

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Most of today's commercial off-the-shelf (COTS) electronics are not specified to meet the demanding requirements of advanced nuclear applications requiring MGy-level TID tolerance. Examples are maintenance and diagnostics tasks in future burning plasma fusion reactors, for example the ITER. Applications such as interventions during nuclear accidents, dismantling of old nuclear power plants and disposal of radioactive waste also call for rad-hard electronics. Therefore, first, the development and, secondly, the testing of custom tailored MGy hardened integrated solutions become necessary for use in these environments. It will not only reduce the shielding but will make it possible to place electronics closer to front-end sensor transducers and actuators in a radiation environment.
To address this need, Magics Instruments has developed a set of 5 chips which are highly resistant to ionizing radiation of over 1 MGy. The pioneering technology of Magics Instruments offers unprecedented opportunities for the use of electronics. Furthermore, Magics Instruments developed a modular test system to assess the radiation hardness of these chips even during irradiation (in-situ). Due to the modularity, this test setup can be customized for any other chip.
The set of these 5 chips developed by Magics Instruments on behalf of F4E deploys an RS485 communication network for reading out sensors and driving actuators. The performances of 3 of these chips were tested on a set of samples during an irradiation assessment. In total 30 samples at once were measured, from which 24 in-situ. The first chip is a BiSS (RS485) interface communication Application Specific Integrated Circuit (ASIC) used for building up a sensor/actuator network. Another chip is a resistive bridge sensor signal conditioner ASIC capable of reading out resistive based sensors like temperature sensor, strain gauges, potentiometers and so on. The last chip tested is a resolver or Linear Variable Differential Transformer (LVDT)-to-digital converter that can be used to read out these sensors typically used in closed-loop motor control applications. The remaining 2 chips out of the set of 5 were tested during a previous irradiation assessment. One chip is a 10-channel relay driver chip and the second one a 10-channel limit switch sensing ASIC.
The irradiation assessment was performed in the Co60 gamma underwater test facility at SCK CEN (called RITA) in Mol. The 30 chips were loaded in a closed container that was lowered down in a pool of water with the Co60 radioactive sources at the bottom. Prior to the actual irradiation assessment of these chips a reference dosimetry was performed to determine the dose rate distribution inside the irradiation volume. Axial variations of the dose rate along the chip-support were ~20%. On average, the exposure dose rate was 484 kGy/h. The chips themselves were connected to a Data-Acquisition system, using cables pulled through 4 tubes of 10 meters length what made it possible to perform a challenging task of the in-situ measurements.
To achieve in-situ measurements, Magics developed a modular test rack computer making it possible to measure the chips fully autonomous in a remotely controlled setup with only limited lab equipment. Data was locally stored in a database on a hard drive and replicated continuously to a database in a cloud storage server for reasons of redundancy. A test setup failure warning mechanism was built in the test setup to lower the risk of data loss. When the test software would crash or power to the setup was interrupted, a warning email and text message was sent to undertake fully remotely necessary actions. An uninterruptable power supply was included to overcome short power interruptions. Even a self-restoring functionality, restarting measurements after failure, was built in.
Multiple tests were continuously performed in a loop on all samples for 97 days to reach at a total ionizing dose of at least 1 MGy for each sample in the test. The higher located samples received even 1.2 MGy. The irradiation itself was interrupted twice to investigate the dynamic balance between defect generation due to irradiation and recovery. These interruptions were performed for 1 hour when average TID of 10 kGy and 100 kGy were reached. After the end of the irradiation, all samples were measured in-situ for 2 more weeks to investigate the recovery during the annealing phase. First, 1 week of annealing at room temperature was performed followed by another week of

high temperature annealing at 100 °C. These tests were performed in a climate chamber. The full assessment was performed in-line with the ESCC 22900 standard: Total dose irradiation test standard . Furthermore, the ITER EEE Nuclear Radiation Compatibility Handbook was used as the reference throughout the full project, from design in the very beginning till the final post-processing of the data.
The collected data was processed into readable plots showing the performance of the chips along the TID they received and their recovery. Almost 9000 plots were generated. Some of these data plots show a drift in the measured data, but all of these were acceptable. Therefore, it can be concluded that performance was proven over the full TID range. The data obtained during this irradiation assessment complete the data yet obtained for the 2 other chips, out of the set of 5, tested in a previous irradiation assessment. The performance of these chips was also proven over the full TID range. This means that the radiation hardness of the RS485 communication network to read out sensor and drive actuators, build up by the chips of Magics Instruments is proven.
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08 Decommissioning, Dismantling and Remote Handling / 243

#08-243 An advanced blind-tube monitoring instrument to improve the characterization of subsurface radioactive plumes

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Nuclear legacy waste storage facilities, such as ponds, tanks and silos, at nuclear waste management facilities or waste disposal sites, often contain radioactive waste under water. Some of these structures in the UK were not lined and, over time, cracks and slits have occurred providing paths for egress of radioactivity to the environment. Consequently, a mixture of radionuclides and water, defined as liquor, can leak into the subsurface dispersing downward and outward in the unsaturated ground from the waste source, contaminating the soil and sediment during its path.

The scenario on which this research is focussed concerns the potential for nuclear material to have leaked into the ground beneath the Magnox Swarf Storage Silo (MSSS), which was built in the 1960s at Sellafield (UK). In the main, solid waste from the reprocessing of nuclear fuel is stored in the MSSS underwater but it also contains residual material held within the water that is highly mobile and thus has the propensity to migrate into the ground. Sellafield Ltd. is in the process of retrieving the waste and decommissioning the MSSS facility, which requires radiological surveillance of the ground. The current techniques of measurement of the subsurface radioactivity generally involves periodic campaigns to collect soil and groundwater samples from several boreholes, and subsequent laboratory analysis. In addition, ground surveys employing geophysical logging probes have also been deployed in specific boreholes/blind-tubes, in preference to the laboratory analysis. The available radiometric sensors operate at only relatively low dose rates, have low radiation resistance and have low spectroscopy potential. Thus, they are not satisfactory for deployment in a high dose rate environment and constitute only a short-term solution. Therefore, an innovative technological solution is necessary to achieve increased resolution and better understanding of soil and groundwater contamination around the MSSS building. It will also allow to identify and predict spatial migration pathways and guide site remediation programs.

A design for a resilient radiometric logging probe developed in this research will be presented that fulfils these requirements for down-hole rapid monitoring of long-lived contaminants, such as caesium-137 and strontium-90 (and hence yttrium-90). The device (Figure 1) holds a CeBr3 scintillator detector and a digital data transmitter (MCA), in a resistant and waterproof housing. The probe is then connected to a cable and lowered down in the blind-tube by means of a winch system, and the position in the hole calculated by a depth encoder. The logging cable has usually three functionalities: transmit electrical signals to recording instruments at surface, provide power, and support the lower and raise of the probe. The assessment of caesium-137 and strontium-90 in the surrounding soil formation (but closed to the borehole), is done by detecting the high penetrating gamma-ray photons and Bremsstrahlung photons, emitted by the deceleration of the energetic beta particles in the steel blind tubes, respectively. Moreover, it is anticipated that the extension of a single detector to a blind-tube string network will provide 3D spatial arrangement in order to characterize radioactivity in the ground system, at interim storage sites for high-level radioactive waste. Modelling and laboratory experiments are underway to validate the concept and calibrate the system in a soil-filled phantom arrangement that replicates the in-ground blind-tubes set on site which will be reported on in this work. The ultimate goal is to achieve a validation of the complete system in the existing blind-tube network at the Sellafield site.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 245

#07-245 Development of a miniaturized furnace for Scanning Electron Microscopy: Thermal modelling, manufacturing and tests

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In the field of samples observations by scanning electron microscopy (SEM), imaging material samples at elevated temperatures has got an increase of interest during the last decades. Performing in-situ observations at high temperature facilitates the understanding of materials behaviour when they are used under severe environments [1]. It is for instance the case of superalloys used for turbine blade in jet engine which must withstand high mechanical loads and extremely high temperatures. Making high temperature SEM imaging possible requires specific instruments to be developed in order to overcome several constrains (thermoionic emission, detectors thermal sensitivity…). Heating samples at elevated temperature (1000°C and more) can be achieved by the means of miniaturized furnaces directly implemented in the SEM chamber. Most of them are built with a thin electrical resistance that uses Joule effect to warm up the sample. The accurate control and monitoring of the sample temperature needs a precise understanding of the micro-furnace thermal behaviour that is conditioned by the material components properties. In order to predict the micro-furnace capabilities, thermal modelling can be done by using finite element analysis (FEA). This tool is very helpful to design and to make an efficient micro-furnace but it requires a validation by testing a real furnace prototype. It is in this aim that a new micro-furnace prototype called FurnaSEM is manufactured and tested in controlled environments by the means of a specific bench test. The comparison of experimental data and numerical solutions is used to validate the prototype operability at high temperature.

Numerical model of FurnaSEM
Thermal modelling of the micro-furnace is ensured by a FEA industrial software: SolidWorks Flow Simulation which allows to take into account and to calculate all the heat transfer mechanisms operating around a solid component (conduction, convection and radiation) in a large variety of situations. Relevant numerical solution requires to consider the material properties such as thermal conductivity, specific heat capacity and radiative emissivity in a large temperature domain. The use of thermophysical properties database is crucial to adjust the parameters for the thermal modelling of solids. These properties are not available in litterature for the temperature range of interest. Consequently we have performed sensibility studies as a function of several parameters (emissivity and thermal contact resistance) to better estimate the micro-furnace thermal behaviour.
Results obtained by numerical modelling are compared with experimental measurement achieved within a housemade bench test. This bench test is designed to monitor the furnace under vacuum while measuring the temperature at different points of interest using both thermocouples and thermal cameras. The experimental data are used to identify the best fitted parameter values. Finally the more accurate thermal model (i.e. set of parameters adjusted between the numerical model and experimental datasets) is defined to predict the 3D temperature fields. The full thermal map of the micro-furnace gives tips about its weaknesses. This will help us to further optimise the design and efficiency of the micro-furnace and to develop a new generation of micro-furnaces that works at higher temperature (1300-1450°C).

Manufacturing
The manufacturing of the miniaturized furnace was essentially 3D computer-aided. The prototype design is cylindrical and contains few elements that simplify its use. The sample is directly put on a sample holder that is a metallic cylinder piece. This cylinder is directly sustained by the heating system. The heating system contains a heating element that is embedded in a cylindrical hot body. Both pieces are assembled together mechanically. They are associated to the body of the furnace by a thin metallic wire which limits the heat conduction leaks. Thermal shield pieces enclose hot

components and are cooled by conduction with a cold casing. The casing temperature is maintained below 50°C by a cooled fluid circulation. The chose of the material compositions were motivated by all these constrains. The heating element is made of a nickel superalloy that withstand high temperature constrains and presents a good resistance to high temperature oxidation. The shields and the sample carrier are made with stainless steel and the casing is made with copper. The micro-furnace temperature is monitored using two K-type thermocouples which are located near the heating element and the sample holder. The diameter of the furnace as well as its height are lower than 30mm. It can be easily placed in the SEM chamber. The compacity of the whole system allows to perform in-situ observations at high temperature using a relatively low working distance between the sample surface and the objective lens of the microscope (10 to 15 mm).

Tests and SEM applications
The bench test was specially designed to assess thermal capacities before operating inside the SEM. It contains a vacuum chamber instrumented with a thermal infrared camera which provides surface thermography measurements. The system can operate under a minimum pressure of 3.10-3 Pa and was mainly used for ex-situ thermal characterisation of the micro-furnace. The main advantage of this system is the temperature monitoring by independent measurement devices (i.e. thermocouple sensors, infrared camera and pyrometer) which help to enhance robustness of experiment data. Moreover, continuous thermal cycling of the furnace in repeatable conditions is a good way to monitor the system durability and to identify the degradation mechanisms.
When it is implemented in the SEM chamber, the miniaturized furnace enables new imaging possibilities, mainly due to the flatness of the sample holder.
• The furnace can be tilted in the -5° / +5° range while recording images in the SEM chamber. Then, the tilted image series are used to calculate 3D images of the sample surface and thus observe directly the surface rugosity variations as a function of temperature [2].
• A high temperature backscattered electron (BSE) detector (provided Crytur company) can be used in combination with FurnaSEM. It allows to record BSE images at high temperature, i.e. to characterize continuously the sample chemical modifications during a heat treatment [3].
• Working with a shorter working distance allows to record images at high temperature with a lower high voltage and with a high resolution.
This was used in the field of nuclear materials: in-situ observations of uranium oxide microparticles sintering were achieved in order to understand better the physical and chemical mechanisms that operates during the fabrication of the nuclear fuel [4].

11 Current Trends in Development of Radiation Detectors / 246

#11-246 A High-Granularity Timing Detector for the Phase-II upgrade of the ATLAS Calorimeter system: detector concept, description and R&D and beam test results

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The increase of the particle flux (pile-up) at the HL-LHC with luminosities of L ⊠ 7.5×10^{34} $\text{cm}^{-2}\text{s}^{-1}$ will have a severe impact on the ATLAS detector reconstruction and trigger performance. The end-cap and forward region where the liquid Argon calorimeter has coarser granularity and the inner tracker has poorer longitudinal vertex position resolution will be particularly affected. A High Granularity Timing Detector (HGTD) is proposed in front of the LAr end-cap calorimeters for pile-up mitigation and for luminosity measurement.
It will cover the pseudo-rapidity range from 2.4 to 4.0. Two Silicon sensors double sided layers will provide precision timing information for MIPs with a resolution better than 30 ps per track in order to assign each particle to the correct vertex. Readout cells have a size of 1.3 mm × 1.3 mm, leading to a highly granular detector with 3 millions of channels. Low Gain Avalanche Detectors (LGAD) technology has been chosen as it provides enough gain to reach the large signal over noise ratio needed. A dedicated ASIC is being developed and some prototypes have been already submitted and measured
The requirements and overall specifications of the HGTD will be discussed. LGAD R&D campaigns are carried out to study the sensors, the related ASICs, and the radiation hardness. Laboratory and test beam results will be presented.

01 Fundamental Physics / 247

#01-247 ATLAS LAr Calorimeter Commissioning for LHC Run-3

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Liquid argon (LAr) sampling calorimeters are employed by ATLAS for all electromagnetic calorimetry in the pseudo-rapidity region $|\eta| < 3.2$, and for hadronic and forward calorimetry in the region from $|\eta| = 1.5$ to $|\eta| = 4.9$. In the first LHC run a total luminosity of 27 fb^{-1} has been collected at center-of-mass energies of 7-8 TeV. After detector consolidation during a long shutdown, Run-2 started in 2015 and about 150fb^{-1} of data at a center-of-mass energy of 13 TeV was recorded. With the end of Run-2 in 2018 a multi-year shutdown for the Phase-I detector upgrades was begun.
As part of the Phase-I upgrade, new trigger readout electronics of the ATLAS Liquid-Argon Calorimeter have been developed. Installation began at the start of the LHC shut down in 2019 and is expected to be completed in 2020. A commissioning campaign is underway in order to realize the capabilities of the new, higher granularity and higher precision level-1 trigger hardware in Run-3 data taking. This contribution will give an overview of the new trigger readout commissioning, as well as the preparations for Run-3 detector operation and changes in the monitoring and data quality procedures to cope with the increased pileup.

01 Fundamental Physics / 248

#01-248 Development of the ATLAS Liquid Argon Calorimeter Readout Electronics for the HL-LHC

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To meet new TDAQ buffering requirements and withstand the high expected radiation doses at the high-luminosity LHC, the ATLAS Liquid Argon Calorimeter readout electronics will be upgraded. The triangular calorimeter signals are amplified and shaped by analogue electronics over a dynamic range of 16 bits, with low noise and excellent linearity. Developments of low-power preamplifiers and shapers to meet these requirements are ongoing in 130nm CMOS technology. In order to digitize the analogue signals on two gains after shaping, a radiation-hard, low-power 40 MHz 14-bit ADCs is developed using a pipeline+SAR architecture in 65 nm CMOS. Characterization of the prototypes of the frontend components show good promise to fulfill all the requirements. The signals will be sent at 40 MHz to the off-detector electronics, where FPGAs connected through high-speed links will perform energy and time reconstruction through the application of corrections and digital filtering. Reduced data are sent with low latency to the first level trigger, while the full data are buffered until the reception of trigger accept signals. The data-processing, control and timing functions will be realized by dedicated boards connected through ATCA crates. Results of tests of prototypes of front-end components will be presented, along with design studies on the performance of the off-detector readout system.

11 Current Trends in Development of Radiation Detectors / 249

#11-249 Towards the experimental validation of a small Time-Projection-Chamber for the quasi-absolute measurement of the fission cross section

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Improvement in neutronics codes jointly with the advent of high performance computing systems made a deeper knowledge on nuclear data more sensitive. The latter is used both for solving the neutron transport equation together with nuclear instrumentation validation and operation. Hence, it becomes relevant to improve the knowledge of the fission cross section of fertile secondary actinides as the 242Pu one, which is fissile in a fast neutron flux. This isotope is, in particular, chosen as a deposit for the fission chamber for the online monitoring of the fast flux in the experimental irradiation reactor Jules Horowitz (RJH) at CEA Cadarache. The latter, as any nuclear thermal or fast neutron reactor, has a high neutron flux around 1 MeV. This motivates the improvement of the fission cross section for the fertile 242Pu isotope, whose the various observations show a dispersion of 10 to 15% around 1 MeV.
The standard measuring technique of a fission cross section is based on simultaneous comparison between the one measured and another one so-called reference nucleus and jointly located in the experimental fission chamber. With a fission cross section known within a few percent, this nucleus enters in the class of the [secondary standard], themselves calibrated to a primary standard of almost absolute precision. Our approach aims to bypass the secondary standard by performing the measurement directly with the primary standard. This is the reaction cross section H (n, n) p which was chosen to achieve our goal since the latter is known from 0.2 to 0.5% over the energy range [0-20 MeV]. Quantifying the neutron flux requires a precise count of the number of recoil protons emitted by a hydrogenated sample of chosen thickness irradiated by a neutron flux. It is therefore essential to use a recoil proton detector having a perfectly known intrinsic efficiency in all operating regimes and a linear response with respect to the input signal. Above 1 MeV, the use of one or two silicon junctions is fully adequate. However, this device is unsuitable at lower energies when a large number of gamma and electrons generate a crippling background noise. This presentation will therefore focus on the recent development and validation of a Recoil Proton Gas Detector (DGPR), insensitive to gamma/electrons noise. This one especially contains a double small time-projection chamber and will be used for the 242Pu fission cross section measurement from 200 keV.

03 Fusion Diagnostics and Technology / 251

#03-251 Long Term Neutron Activation in JET DD Operation

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The Joint European Torus (JET) is the largest operating fusion tokamak in the world. This allows performance of unique experiments that cannot be performed anywhere else. One such experiment is long term irradiation of different fusion relevant materials with neutrons produced in deuterium-deuterium (DD) and deuterium-tritium (DT) plasmas. During the last DD plasma campaign at JET in 2019 and 2020 several different ITER relevant materials and activation foils were irradiated in a specially design long-term irradiation station located inside the vacuum vessel with the purpose of testing activation of ITER materials by fusion neutrons. The samples were exposed to neutron fluence of 1.9×10^{14} n/cm^2 during JET discharges performed in the experimental campaign over a period of 5 months. After the irradiation with DD neutrons, activation of materials was determined with measurements of long-lived isotopes in the samples such as Co-58, Co-60, Fe-59, etc.
In order to support the long-term irradiation experiments at JET a series of Monte Carlo neutron transport calculations is required to calculate neutron fluxes, neutron flux spectrum, reaction rates, etc. A detailed computational model of the complex JET tokamak structure and the experimental position was developed in MCNP code and validated on experiments previously performed at JET. Another important parameter for computational analysis is the detailed description of the materials in the JET structure and activation samples. The detailed material composition together with the nuclear data libraries is used to properly describe physics of neutron interaction with materials during Monte Carlo simulations. The last important parameter for the Monte Carlo simulation is the source of neutrons. In DD plasma some DT reaction take place due to production of tritium during the DD reaction. Due to this the fraction of neutrons produced by DT reaction is of significant importance in computational analysis.

07 Nuclear Fuel Cycle, Safeguards and Homeland Security / 252

#07-252 Passive neutron multiplicity counting with PVT plastic scintillators for plutonium characterization in radioactive waste drums

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In the framework of plutonium characterization in radioactive waste drums by passive neutron coincidence counting, the Nuclear Measurement Laboratory of CEA Cadarache is studying plastic scintillators as a cheaper alternative to ^{3}He gas proportional counters. Plastic scintillators offer a three order of magnitude faster time response than ^{3}He detectors, larger volumes and a similar neutron detection efficiency. However, the high sensitivity to gamma rays and crosstalk make this technology difficult to use in neutron-gamma mixed field and for large detectors without PSD capabilities. A new patent-pending data processing, based on a time discrimination of triple coincidences, permits to isolate useful fission coincidences from parasitic ones, such as those due to (α,n) reactions and gamma-ray cascades. The performances are studied with an experimental setup designed for 100L to 200L waste drum measurements, and composed of sixteen 10x10x100 cm^{3} plastic scintillators positioned in circle around the package. MCNPX-PoliMi simulations are also performed and compared to experimental data, first to validate the numerical model, then to optimize data processing, and finally to study main causes of uncertainties. For instance, we investigate the linearity of the method with the plutonium quantity, in presence of increasing background noises and matrix effects. To this purpose, different neutron and gamma calibration sources (^{252}Cf for the useful signal, AmBe, ^{60}Co and ^{137}Cs for the background) are placed in mockup drums filled with metallic (iron) or organic (wood, polyethylene, PVC…) materials mimicking common technological waste. Without unfolding real and accidental coincidences at this step, the new data processing increases the signal-to-noise ratio (SNR) recorded with bare calibration sources (no drum) by up to 50% compared to classical triple coincidence calculation with the shift register method. On the other hand, MCNPX-PoliMi simulations are in good agreement with experiment, with a relative difference between the measured and calculated number of coincidences lower than 20%, whatever the coincidence order from total counting to triples. Then a linear response is observed by simulation with the equivalent mass of ^{240}Pu. In addition, this linearity is preserved up to an "alpha ratio" of 100 between (α,n) and fission neutrons. We are currently studying the deconvolution of real and accidental coincidences at high count rate, especially due to high gamma-ray fields, as it is frequently the case with radioactive waste (^{60}Co, ^{137}Cs). We will also discuss the benefit of crosstalk rejection algorithms, as well as the ^{240}Pu detection limit with different waste matrices, plutonium localization and background noise level. The ultimate goal of this work is to reduce the cost of passive neutron coincidence collars for radioactive waste drums by approximately a factor five, compared to ^{3}He-based systems.

11 Current Trends in Development of Radiation Detectors / 253

#11-253 Liquid Argon Instrumentation and Monitoring in LEGEND-200

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The LEGEND Collaboration aims to develop a phased, ^{76}Ge-based double-beta decay experimental program with discovery potential at a half-life beyond 10^{28} years. The first 200-kg phase, LEGEND-200, targets a discovery potential of 10^{27} years by a background index of $< 2{\cdot}10^{-4}$ cts/(keV·kg·yr).
Based on the success of GERDA a liquid argon (LAr) detector system will be deployed. It will offer secondary event information which will allow the identification of background events. The system utilizes the property of LAr to scintillate upon the interaction of ionizing radiation. The primary light, emitted at 128 nm, is shifted to the optical spectrum and read out by silicon photomultipliers mounted at the end of optical fibers.
Crucial parameters for the modeling the LAr detector's response are the light yield, the triplet lifetime and the attenuation length valid at the vacuum ultraviolet emission wavelength. These values are dependent on the actual impurity concentrations in the liquid. To this end, the dedicated LEGEND liquid argon monitoring apparatus (LLAMA) was designed and installed in the LEGEND-200 cryostat for permanent in-situ measurements of the mentioned parameters.
The design of the LEGEND-200 LAr instrumentation will be presented and discussed. Furthermore, an overview of LLAMA will be shown.

10 Education, Training and Outreach / 254

#10-254 EASY: Educationa Alibava System

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EASY, a plug-and-play educational system, is portable, compact and a complete system for microstrip sensor characterization. Ideal for making basic or complex experiments. It is based on the Alibava System, largely used within the CERN community to test micro-strip detectors for particle experiments. The system can be configured to work with laser light or radioactive sources.
The aim of this system is to illustrate students in the operation of a silicon strip detectors, in particular:

• To observe the noise of a silicon strip detector as a function of bias voltage-
• To observe the signal spectra due to a minimum ionising particle in a silicon detector and demonstrate the Landau distribution shape of collected charge.
• To determinate the Charge Collection efficiency of a silicon detector and depletion voltage.
• To observe charge sharing between strip and relate this to the position resolution of the detector.
• To illustrate the structure of a typical micro-strip detector.
• To compare the charge deposition of minimum ionization particles and gamma particles.
• And, to calculate the laser penetration in silicon.

The components of the EASY systems are the Control Unit and the Sensor Unit. The Control Unit is the heart of the system communicating with the Sensor Unit and the Computer software. It contains the Data Acquisition Control and it is also in charge of processing of the sensor data and trigger inputs. In addition, it contains an adjustable Hight Voltage unit for microstrip sensor bias, with voltage and current display and Includes the laser source. The Control Unit communicates with computer software via USB. The Sensor Unit accommodate a p on n silicon micro-strip sensor segmented in 128 strips.
EASY comes with an activity book where the students, through 10 exercises, are introduced in the main concepts and functionalities of microstrip silicon detectors, used in the actual particle physic experiments. The book also provides a full description of the EASY device and the data Acquisition system.

10 Education, Training and Outreach / 255

#10-255 The LUCID experiment - releasing the potential of school students in space research

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The LUCID experiment involved groups of students in a school in Kent, UK, working with a huge number of colleagues to put Medipix technology in open space on TechDemoSat 1. Students worked on this outside their formal studies over a period of ten years and benefitted from real engagement with research. LUCID [1, 2] studied the radiation environment in Low Earth Orbit for three years, collecting over 2.1 million frames of radiation data from its five Timepix detectors. Students published their results in Advances in Space Research in 2019. The students who were involved in LUCID are now becoming leaders in science research themselves, working across many areas of physics and engineering.
Becky Parker [3] the teacher supporting students in this, will outline how this genuine involvement of young people in research can not only inspire students and teachers but also contribute to scientific advance.

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10 Education, Training and Outreach / 256

#10-256 Advanced neutron detection hands-on exercises with MX-10 particle camera educational kit

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MX-10 particle camera [1] as a modified variant of the read-out interface for Timepix pixel detector is dedicated for educational purposes to demonstrate real-time detection and visualization of basic kinds of radiation as alpha, beta, gamma rays and cosmic muons. However, its use is not strictly limited to perform the fundamental exercises that are formulated in the complementary manual book Experiments Using Pixel Detector in Teaching Nuclear and Particle Physics [2], which contains detailed instructions for a set of 50 experiments practicable with Timepix detector. After necessary adaptation, MX-10 can also serve well in more advanced practices as the neutron detection experiments undoubtedly are.

The contribution presents a set of exercises, methods involved and arrangements of a compact tailor made set-up based on MX-10 edu kit to allow effective performance of neutron experiments in a quite undemanding way easily comprehensible to students. The newly developed experiments are focused on demonstration of basic interactions of neutrons with the matter to provide clear evidence on them. Specifically, the individual exercises cover the following objectives:

i) The measurement of radiation emitted by a compact a laboratory ^{241}AmBe radionuclide neutron source and the radiation induced by neutrons interacting in the surrounding materials based on observation of particle tracks resulting from interactions of neutrons and gamma rays in a bare silicon sensor of Timepix detector. The particle track analysis will be applied to recognize the type and energy of every individually interacting radiation quantum. ii) The demonstration of a highly effective way of neutron detection with an adapted sensor is shown applying suitable neutron converters of ^{6}LiF (permitting the registration of slow neutron by means of ^{6}Li(n,α) reaction) and of polyethylene (the hydrogen rich material for detection of fast neutrons by means of recoil protons). Detection of moderated and fast neutrons (in range 1 - 10 MeV) and their recognition is presented. A possibility of the spectral analysis of a mixed neutron field is also discussed.

iii) The experiments dedicated to key aspects of neutron shielding and moderation performed with the Timepix detector adapted for detection of slow and fast neutrons realized in various experimental set-ups of different types of materials (high density polyethylene, parafine, Pb, Cd, B, Li and their combination) used for shielding the detector against the radiation from the neutron source. Shielding efficiency of the particular materials is examined considering also the undesirable secondary radiation generated by neutrons interacting with shielding materials.

The lecture arises from many years of experience gained in using the MX-10 camera to teach the interaction of ionizing radiation at Czech high schools and at a number of international schools/workshops organized in 2018-2020 by the IEEE NPSS focused on representative students of high school and universities. The feedback on the hands-on exercises organized recently as a part of XXI Jorge Andre Swieca Summer School on Experimental Nuclear Physics [3] organized in Sao Paulo 2020 and the virtual IEEE NPSS Workshops on Application of Radiation Instrumentation in Jakarta [4] and Dakar [5] 2020, has proven well how the newly developed exercises help students to understand the physics of neutron interactions with matter. It is worth mentioning here that the practical exercises described in this contribution will be a part of this conference program.

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#10-257 Visualization of ionizing radiation with MiniPIX-EDU turns boring theory into wonderful exploration

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The MiniPIX EDU is miniaturized imaging detector visualizing traces of individual particles of ionizing radiation in the normal environment such as classroom. It is simplified but still very powerful version of the MiniPIX detectors operated by NASA in space for radiation monitoring on the orbit. This CERN based technology gives very illustrative and easy to understand insight into complicated processes of particle physics.
This miniature particle tracking detector can visualize even very small amounts of ionizing radiation which is present everywhere. Students can "see" the different types of radiation emitted by common materials such as piece of granite or ash, radioactivity accumulated in paper bag from vacuum cleaner or deposited on the face mask used for protection against infections. They can explore the variation of the air radioactivity during the day, hunt for cosmic muons and check their directions, see how altitude affects presence of radiation types. They can try to prepare their own (safe) radioactive source and try to construct the shielding against the radiation it emits. Students can directly observe how different radiation types interact with matter and what happens then. They can watch the laws of radioactive decay.
The MiniPIX EDU device and its utilization for educational extents experiences collected by earlier projects such as "CERN technology in schools" of Mrs. Becky Parker. About 450 schools with more than 15.000 students participated since 2007. Similar project of the Institute of Experimental and Applied Physics (IEAP) of Czech Technical University in Prague and Jablotron company resulted in the prize winning MX-10 detector device. All these projects were done in close collaboration with CERN Medipix group. The new MiniPIX EDU device is based on the same Timepix chip as MX-10 but it is much smaller, more stable and less expensive.

09 Environmental and Medical Sciences / 258

#09-258 A prototype of pCT scanner: first tests

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Proton therapy is a cancer treatment technique that allows for a more selective application of dose to tumors in comparison with conventional radiotherapy with X- or γ-rays. This is due to the fact that higher dose is concentrated in the region where the protons stop, whereas far less dose is deposited in the traversed tissue. In this context, a system of imaging and dose verification to define effective and accurate treatment plans and to guarantee the correct location of the applied dose is mandatory. However, the treatment planning in proton therapy facilities is guided via X-ray computed tomography (X-ray CT) images. This requires the conversion "a posteriori" of the map of Hounsfield Units (HU) obtained from X-ray CT to Relative Stopping Powers (RSP) useful for proton therapy treatment plans [1]. This conversion induces a large uncertainty in the range of the protons (up to 5% in the abdomen and up to 11% in the head) [2-4]. Meanwhile, treatment plans made via proton-CT (pCT) images will offer more accurate estimations of proton ranges with an uncertainty below 1% and a better control of the treatment [1].
With this purpose, we are building a prototype for pCT scanner using particle detectors extensively used in experimental nuclear physics. Those are the Double-Sided-Silicon-Strip-Detectors (DSSDs) and the $LaBr_3$(Ce) scintillation detectors. The former, being segmented horizontal and vertically, are to be used as tracking detectors and the latter, an array of 2x2 modern scintillators of $LaBr_3$(Ce) offering fast response and good energy resolution, is to be used as residual energy detector. With these detectors, an image of the sample is taken by mapping the energy losses with respect to horizontal and vertical positions. In this way, we obtain a 3D distribution of Relative Stopping Powers (RSP) needed to design proton therapy treatment plans.
A first test using low-energy protons (10 MeV) was carried out at the CMAM tandem (Madrid, Spain) in June 2019 to test the proton tracker. Monte Carlo simulations were used to optimise the setup and obtain predicted images of the scanned 2D objects. A second experiment at proton energies relevant in proton therapy (100-200 MeV) will be performed in Cyclotron Centre Bronowice (CCB) in Krakow (Poland) in May 2021. This second measurement will allow to test and optimise the full setup, including the residual energy detector by scanning 3D objects.
In this contribution we report on the results of the first test at CMAM (Madrid) which demonstrated the viability of the pCT scanner prototype. Additionally, preliminary results from the study at CCB Krakow will be shown.

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11 Current Trends in Development of Radiation Detectors / 259

#11-259 Josephson Travelling Wave Parametric Amplifiers for a neutrino mass measurement

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In parametric amplification, a signal is amplified by mixing it with a pump wave in a nonlinear medium. During the last five years, Josephson Travelling Wave Parametric Amplifiers (JTWPAs) have proven to deliver near-quantum limited amplification for signals in the 5 to 10 GHz range with a large relative bandwidth of about 10%. This performance made them the workhorse for multi-qubit system readout.

The unmatched low-noise characteristics of JTWPAs open up new opportunities in radio wave detection for fundamental physics. The work presented demonstrates the ability to design and operate a JTWPA at higher frequencies around 25 GHz, combined with an antenna.

This frequency was chosen as it is of particular interest for the Project 8 collaboration. Project 8 employs a technique termed electron Cyclotron Radiation Emission Spectroscopy (CRES) to lower the bound on the electron neutrino mass. The ability to perform CRES in large volumes crucially depends on these low-noise cryogenic amplifiers to collect radio waves emitted by magnetically trapped electrons from tritium decay.

05 Nuclear Power Reactors Monitoring and Control / 260

#05-260 EPR In-core Instrumentation Fabrication and Operation

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The core instrumentation of the EPR reactors consist of 12 exchangeable instrumentations, each with 6 fixed cobalt-self-powered neutron detectors and 3 type k thermocouples for core exit temperature, 40 measuring fingers of the aeroball measuring system using moveable steel ball stacks with vanadium as probes, as well as 4 level and dome temperature measurement probes. The exchangeable instrumentation and the aeroball fingers are installed in 12 instrumentation lances guided through the reactor pressure vessel head.

The poster explains the purpose, concept and operation of the instrumentation:

The aeroball measuring system, as reference system providing 3D core flux maps, allows calibration of the fixed in-core cobalt self-powered neutron detectors, which provide continuous input to the reactor protection system. The core exit temperature system is used for accidental monitoring and also for level measurement, which is based on a heat transfer principle. Material selection and manufacturing of the instrumentation with its welding and brazing steps are shown as examples. The components of the instrumentation are shown with dimensions and weights. As results, examples of radial and axial profiles of neutron flux measurements in the EPR reactor with this instrumentation are shown.

10 Education, Training and Outreach / 268

#10-268 A complete training program dedicated to nuclear instrumentation at Aix-Marseille University

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Since 2004, the "Filière Instrumentation" unit of the Physics Department of the Faculty of Sciences at Aix-Marseille University has been collaborating with the CEA to develop different education activities on the topic of nuclear instrumentation. These activities involve also several partners such as EDF, IEEE NPSS, international nuclear centers (SCK-CEN in Belgium, NCBJ in Poland, CNESTEN in Morocco) and other key international partners such as the Nuclear Reactor Laboratory of the MIT in USA and the Jozef Stefan Institute in Slovenia. These activities benefit from a strong recognized research activity carried out in the framework of the IM2NP laboratory and the joint laboratory LIMMEX with the CEA.
The presentation will deal with these different activities and detail the three most recent in particular:
- A speciality dedicated to test facilities instrumentation. It was developed in collaboration with the CEA in 2004. It is now a diploma co-delivered by AMU and the national nuclear science and technology institute of CEA INSTN. It is part of the master's degree in Instrumentation, Measurement and Metrology (IMM) and more recently is integrated into the graduate school of a new institute created by AMU in July 2019 called ISFIN (Institute for fusion sciences and Instrumentation in Nuclear environments);
- A summer school EFMMIN "Ecole Franco-Marocaine de la Mesure et de l'Instrumentation Nucléaires" since 2010. It is a two-year summer school co-organized with the CEA, the CNESTEN, the Sciences Faculty of the Mohammed V University and the AMSSNUR. After editions devoted to research reactors, the fuel cycle, decommissioning, medicine and the environment, and safety, the next edition will focus on innovation in nuclear sensors and detectors;
- A mobility program called MOBIL-APP since 2018. It is labelled and funded by the Excellence Academy of the A*MIDEX foundation promoting pedagogical innovation, improvement of existing training and international attractiveness. This program concerns the outgoing international mobility of block-release apprentices preparing the IMM master's degree. It consists in selecting, preparing and sending a group of block-release apprentices for 2 weeks per year to a partner. The two groups realize visits to reactors, laboratories and companies, participate in seminars, short courses, experimental projects and workshops together with local students, visit cultural places and prepare an activity report in English;
- A new international specialty. It is currently in the process of being set up administratively and pedagogically. This new course will belong to the IMM master's degree and to the new ISFIN institute. It will dedicated to nuclear instrumentation and measurement for major nuclear research facilities in the fusion and fission fields such as MTRs and tokamaks (for instance the Jules Horowitz Reactor and ITER located in the South of France, 70 km from Aix-Marseille University). This speciality will involve several international partners. For instance the Nuclear Reactor Laboratory of the MIT will carry out training courses, seminar and practice exercises. The students will have to do internships in international facilities after a project realized at Aix-Marseille university labs;
- A student branch and a chapter dedicated to nuclear instrumentation and associated to IEEE NPSS. It is in the process of creation with about 30 students.

04 Research Reactors and Particle Accelerators / 276

#04-276 Characterization of the external pulsed beam at CMAM

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The Centro de Micro-Análisis de Materiales (CMAM) is one of the two research centres with an ion accelerator in Spain. It belongs to the Universidad Autónoma de Madrid (UAM) and the building that hosts the laboratory is at the university campus.

The equipment of the facility consists on an electrostatic ion accelerator with a maximum terminal voltage of 5 MV and six beam-lines dedicated to various application areas such as the analysis and modification of materials, the study of the nuclear reactions or archaeometry studies. The accelerator, built by High Voltage Engineering Europe (HVEE), is of the tandem type with a Cockroft-Walton acceleration system. It is provided with two sources: a plasma source for gaseous substances and a sputtering source for obtaining any element from H to U from a solid target.

The accelerator feeds up six beam-lines after the bending magnet at different angles: the standard multi-purpose line which ends in the internal micro-beam line, the time of flight line, the implantation line, the external micro-beam line and the nuclear physics line. For a more detailed description of each line, see Ref. [1]. There are two lines where the beam can be extracted from vacuum to air: the implantation line (IMP) and the external micro-beam line (EuB). Until now, the first has been devoted to the implantation or irradiation in large areas for material modification. The second has been focused onto archaeometry studies. Now, a new project has started to use this two lines for proton-therapy preclinical studies.

What will be presented at the conference is the characterization of the pulsed beam in air obtained in the IMP line. The main aim of this beam-line is to perform homogeneous implantations or irradiations in large areas (up to several cm^2). For this purpose, the line has a HVEE electrostatic scanner consisting of four plates for vertical and horizontal beam deflection and the corresponding power supplies. This system is capable of scanning Hydrogen of 10 MeV at a size of 100×100 mm^2. Until recently, this system, called RASTER, has been used merely for scanning and irradiating large areas. However, it can open the door to obtaining external pulsed beams for pre-clinical studies on proton-therapy in the FLASH regime.

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11 Current Trends in Development of Radiation Detectors / 296

#11-296 Characterisation of Germanium Detectors in LEGEND-200

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The LEGEND Collaboration is searching for neutrinoless double beta ($0\nu\beta\beta$) decay in Germanium-76 (^{76}Ge) via a phased approach. The first of two phases, LEGEND-200 (L200), is currently under construction at the Laboratori Nazionali del Gran Sasso (LNGS) in Italy and will use 200 kg of enriched High Purity Germanium (HPGe) detectors submerged in a liquid argon active shield in order to target a half life sensitivity of 10^{27} years and a background index of < $2 \cdot 10^{-4}$ cts/(keV·kg·yr). Brand new Inverted Coaxial Point Contact (ICPC) HPGe detectors will be utilised in L200 and their unique geometry provide the advantage of superior energy resolution and strong pulse shape discrimination power whilst having large detector mass. These detectors must be comprehensively characterised before their deployment at LNGS in order to verify their performance and behaviour, including the important task of energy resolution evaluation at the Q-value for $0\nu\beta\beta$ decay in ^{76}Ge of 2039 keV. This characterisation process is currently being performed in the underground laboratories of HADES in Belgium and SURF in the US. This talk will focus on the characterisation activities of LEGEND at HADES, with a particular emphasis on the active volume determination of ICPC detectors. Active volume determination is a crucial characterisation task required for $0\nu\beta\beta$ analyses directly affecting the measured half-life sensitivity.