Nuclear Reactors and Fuel Cycle

introduction

The Nuclear Fuel Center (CCN) of IPEN produces nuclear fuel for the continuous operation of the IEA-R1 research reactor of IPEN. The serial production started in 1988, when the first nuclear fuel element was delivered for IEA-R1. In 2011, CCN proudly presents the 100th nuclear fuel element produced. Besides routine production, development of new technologies is also a permanent concern at CCN. In 2005, U_3O_8 were replaced by U_3Si_2-based fuels, and the research of UMo is currently under investigation. Additionally, the Brazilian Multipurpose Research Reactor (RMB), whose project will rely on the CCN for supplying fuel and uranium targets. Evolving from an annual production from 10 to 70 nuclear fuel elements, plus a thousand uranium targets, is a huge and challenging task. To accomplish it, a new and modern Nuclear Fuel Factory is being concluded, and it will provide not only structure for scaling up, but also a safer and greener production.

The Nuclear Engineering Center has shown, along several years, expertise in the field of nuclear, energy systems and correlated areas. Due to the experience obtained during decades in research and technological development at Brazilian Nuclear Program, personnel has been trained and started to actively participate in design of the main system that will compose the Brazilian Multipurpose Reactor (RMB) which will make Brazil self-sufficient in production of radiopharmaceuticals. The institution has participated in the monitoring and technical support concerning the safety, licensing and modernization of the research reactors IPEN/MB-01 and IEA-R1. Along the last two decades, numerous specialized services of engineering for the Brazilian nuclear power plants Angra 1 and Angra 2 have been carried out. The contribution in service, research, training, and teaching in addition to the development of many related technologies applied to nuclear engineering and correlated areas enable the institution to fulfill its mission that is to contribute in improving the quality of life of the Brazilian people.

The nuclear fuel cycle is a series of steps involved in the production and use of fuel for nuclear reactors. The Laboratories of Chemistry and Environmental Diagnosis Center, CQMA, support the demand of Nuclear Fuel Cycle Program providing chemical characterization of uranium compounds and other related materials.

In this period the Research Reactor Center (CRPq) concentrated efforts on improving equipments and systems to enable the IEA-R1 research reactor to operate at higher power, increasing the capacity of radioisotopes production, samples irradiation, tests and experiments.

Nuclear fuel production

Historical background
IPEN has worked in the area of Nuclear Fuel Cycle essentially since its founding, in 1956. Between 1964 and 1965, Nuclear Fuel Elements (NFE's) based on U_3O_8-Al were manufactured for the Argonauta Reactor of Institute of Nuclear Engineering (IEN). It was a great start-up, since this technological seed germinated after 1980, when, to maintain the operation of the IPEN IEA-R1 reactor, national production of nuclear fuel was demanded. Thus, in September 1988, the first Nuclear Fuel Element (NFE) produced for IEA-R1 was delivered by CCN. Since then, IPEN has made its own fuel elements, based on plate type, initially using U_3O_8-Al dispersions with density of 2.3 gU/cm^3. As the IEA-R1 reactor operating power raised, accomplished with longer operation time regimes, more U-densities in the fuels were needed. To carry out this increase, a new fuel element was developed, based on U_3Si_2-Al dispersion, with a density of 3.0 gU/cm^3, and since 2005, it is fully produced at IPEN from UF_6 delivered by Brazilian Navy - CTMSP. This uranium concentration is sufficient to operate IEA-R1 running at the power up to 5MW. The fuel element manufactured is dispersion MTR-type enriched to 20wt% ^{235}U (Low Enriched Uranium - LEU).

High concentrations of uranium represent longer operation life of each NFE, requires fewer replacements and generates lower quantities of nuclear waste. In this sense, for supplying the new Brazilian Multipurpose Reactor - RMB, a different fuel should be used, with even higher concentrations of uranium. The nominal power of RMB should be 30MW. As specified by the Center for Nuclear Engineering of IPEN (CEN) and Argentinean INVAP, RMB will use an analog type of fuel of IEA-R1, based on uranium silicide technology. It will have uranium concentrations around 3.5 gU/cm^3, thinner plates and a cadmium wire embedded as burnable poison. The development of this new fuel are to be started at CCN. Recently, CCN has successfully produced miniplates based on U_3Si_2 containing up to 4.8gU/cm^3, and it endorses CCN expertise and capability for producing the fuel for the RMB.

100th nuclear fuel element
In September of 2011, as IPEN completed 55 years of operation, CCN proudly presented the 100th nuclear fuel element produced for IEA-R1. Since the first fuel delivered, in 1988, CCN supplies IEA-R1 research reactor with its fuel elements and promotes development of processes and products in nuclear fuel area and in correlated fields, aiming excellence in quality and pursuing technological advances.

Evolving from the first NFE, based on U_3O_8-Al dispersion, to the state-of-art UMo prototypes required a lot of effort. Not only to develop technology, but also to apply new technologies at the production line, aiming the raise of productivity and use of safer, more effective and greener procedures. CCN has fully qualified crew formed by Technicians, Engineers, Masters and PhD, specialized in several areas such as material science, chemical processes and nuclear technology. Undergraduate, Master, PhD and Postdoctoral students, with different educational backgrounds, developing nuclear fuel-related researches, completes CCN staff. This group is able to perform highly qualified technical and analytical activities, which allow the production of NFE's and help to improve the knowledge with new methods and technologies.

Up to now, IPEN has manufactured, for IEA-R1, 63 fuel elements of U_3O_8 (26 having meat density of 1.9 gU/cm^3 and 37 having 2.3 gU/cm^3) and 48 fuel elements of U_3Si_2 having a concentration of 3.0 gU/cm^3. Therefore, 111 fuel elements were produced to date, for our reactor.

Figure 1. CCN staff and the 100th nuclear fuel element produced.

The new nuclear fuel element fabrication plant

The Brazilian Multipurpose Reactor (RMB) project consigned the Nuclear Fuel Center of IPEN to be the future fuel plant to supply the fuel elements to RMB and irradiation targets to produce ^{99}Mo for nuclear medicine. The future production, as account in the end of 2013, would be around 1000 uranium targets and 60 fuel elements a year. This production request is projected for RMB full power operation at 30 MW. There are two basic steps to reach this position:

First Step - The development of planned fuel, with uranium density around 3.5 g/cm^3, with an alike profile of the fuel produced for IEA-R1 Reactor. Nevertheless, it represents a challenge for not only for development, as the new fuel plates are thinner, and the NFE's will include an embedded cadmium wire as burnable poison, but also for technological industrialization and for scale up, from 10 to around 70 NFE's (60 for RMB and 10 for IEA-R1) per year.

Second Step - The development of irradiation targets to produce ^{99}Mo, which is already been carried out. Considering the available technologies, the first choice for producing uranium targets is based on UAlx dispersed in aluminum, which is rolled to mini-plate. However, there are alternatives as monolithic uranium film or by electrochemical deposition of uranyl, whose are being studied as well. The UAlx process route should be concluded in the next years, but its routine production will depend on the integration with the new fuel factory. To achieve these goals, the new plant will be launched in 2015. There will be a fully revamp in all chemical and metallurgical equipment. FINEP resources from Brazilian Government funded many buildings and equipment. Laboratories for routine inspection and fuel characterization will receive, in 2014, top of line equipment as X-ray diffraction, X-ray fluoroscopy, chemical analysis and SEM/EDS equipment. A FINEP project, around US$ 10 million, for building of a critical core for RMB simulation at MBR-01 Reactor was approved in 2013. This project allows the renewal of metallurgic and mechanical fuel plant, improving of environmental control of the plant and routine inspection at laboratory facilities.

Figure 2. The new Center of Nuclear Fuel fabrication plant.

Advances in characterization methods

The expansion of the production scale is a big challenge for the next years. To achieve it completely, it is vital that the methods for characterization of the materials involved in the several steps of the production are the most modern possible, with faster, more precise and less residue generation analysis. In accordance, some of these methods were recently updated, and others are subject of studies and certification.

Crystallinity of uranium silicide
Uranium silicide (U_3Si_2) is an intermetalic used as nuclear fuel in most modern research reactors. For this use, the material is dispersed in aluminum, and allows high densities of uranium in the fuel core, up to 4.8 gU/cm^3, comparing to UAlx-Al at 2.3gUcm^{-3} and U_3O_8-Al at 3.2gUcm^{-3}. Considering its use in a nuclear reactor, U_3Si_2 must be submitted to a strict quality control. An important requirement of the uranium silicide powder is the amount of U_3Si_2 present in crystalline

form. The specification is imposed by the Nuclear Engineering Center of IPEN (CEN), with minimum concentration of crystalline U_3Si_2 as 80wt.%. This is because the U and Si can combine in many different forms, and each presents a different behavior under irradiation. Due to the adopted manufacture route at IPEN, the product usually carries two crystalline phases, U_3Si_2 and Usi.

To perform this analysis, the Rietveld refinement method seems to be the most suitable route, since X-ray diffraction (XRD) patterns of crystalline substances are virtually unique, and independent of each other. However, the use of this method implies in several precautions, as adequate granulometry of sample, enough counting time during analysis and carefulness with the refined parameters, as a good fit does not always represent physical likelihood. Additionally, the method itself does not offer a reliable error estimative, and the results can vary depending on the operator. In this sense, aiming the development of a routine methodology for quantification of phases via analysis of XRD data using the Rietved method, uranium silicide samples produced at CCN were submitted to X-ray diffraction. For Rietveld refinement, the data were analyzed using non-commercial software GSAS.

We observed that the intensity of X-ray diffracted peaks could not be well fit in Rietveld refinement. Probable causes include association of preferential orientation, inadequate granulometry, and differences in thermal factors and fractional occupation of U_3Si_2 atoms. Despite that, the refinements were acceptable and the results for phase quantification seem reliable and in accordance with the specification limit. The step from where U_3Si_2 was sampled from the process plays an important role, considering granulometry and crystallite size for XRD and Rietveld refinement, bearing in mind phase quantification. More studies including additional milling should be done to investigate its influence on crystallite size and intensity of diffracted peaks.

Analysis of effluents
The polarography of wastewater at the laboratory of electrochemical analysis of CCN has achieved great performance since the installation of a Metrohm VA797 analyzer, which is capable to measure tap water uranium content around 2ppb. This established a statistical routine control of wastewater at IPEN nuclear fuel factory. FAPESP financed this development by Project 2013/08514-3.

Automation in X-ray radiography, measuring and tracing
After the U_3Si_2 powder is prepared, classified, mixed with aluminum powder and pressed at CCN, it is ready to compose the nuclear fuel core. Next, the U_3Si_2-Al briquette is mounted inside the aluminum frame; the set is welded, heated and submitted to several hot rolling passes.

Figure 3. Mechanical-metallurgical steps of fabrication: Picture in frame assembly of fuel, welding, heating and rolling.

After the rolling passes, the plate must be cut to fit properly inside the fuel element. To locate the U_3Si_2 precisely inside the plate, an X-ray radiography is performed, and then the plate is traced to be correctly cut afterwards. Until recently, the X-ray image was obtained using films, and tracing was done manually. It is a very accurate method, however, it takes considerable time, and films are expensive and difficult to buy, not to mention that its development generates residues. To improve this step, it was installed a digital fluoroscope at CCN.

This new equipment is able to obtain digital radiographs of the plates, eliminating the use of films and the chemicals used for its development. Additionally, it uses computer interface and automated system of tracing, reducing the time of image acquirement, plate measuring and tracing from two hours to 30 minutes.

Figure 4. X-ray fluoroscope image system.

Figure 6. X-ray source, fluoroscope and automated tracing system (the aluminum fuel plate is located in the middle of table).

Automation of thickness measuring of plates and fuel core
During irradiation, fission gases are formed inside the fuel plates, and the cladding is subject of pressure. Still, considerable amount of heat is formed within the fissile material. Therefore, the cladding must be thin enough to maintain a good thermal exchange with reactor water, and sufficient thickness to preserve mechanical integrity under pressure, keeping fission gases and avoiding the contact of the fuel with the water of reactor. In this sense, for a safe operation of the reactor, the specifications of thickness of NFE plates are strict and rigorously followed.

To measure the internal thickness of fuel meat and cladding, from every 24-fuel plates produced at CCN, one is cut and submitted for measurements, in destructive tests. The measurements are performed at the center and terminal parts of the fuel meat (dog-boning area). If a plate is reproved, a second one is tested. If the result is the same, all the 24 plates of the three corresponding lots are destroyed and sent to uranium recover.

For measurement of fuel meat thickness, metallographic techniques for preparing the sample are used, including cutting, embedding and polishing. After these steps, a digital image of the cross-section of NFE is acquired with optical microscopy, analyzed and measured, in a semi-manual process.

The new characterization process consists in acquiring the image with a Scanning Electron Microscope (SEM) and analyze the image by computational method. It measures thickness of core and cladding of fuel plates. With the developed method, such measurements can now be performed in less time (from 288 to 72 hours) and with better statistical data, when compared with the old method of measurement.

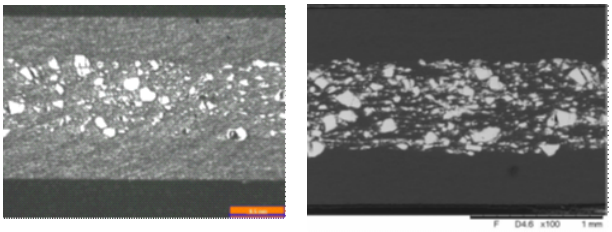


Figure 7. Images of the core of nuclear fuel plates, obtained by optical microscopy (left) and scanning electron microscopy (right).

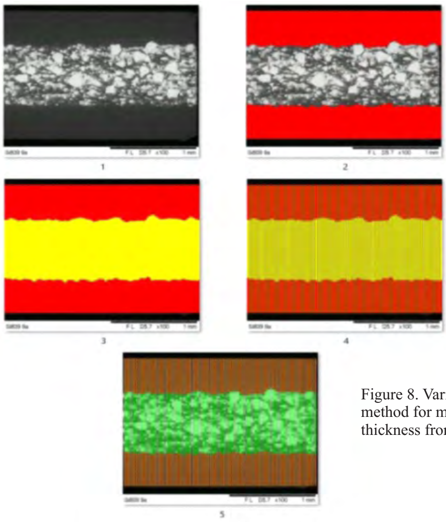


Figure 8. Various steps of the developed method for measurement of nuclear fuel plates thickness from the SEM image.

Studies on chemical treatment and surface contamination of nuclear fuel plates
Over the last 10 years, significant increases were observed in the radioactivity in the water of IEA-R1 reactor pool, even reaching the reactor hall detectors. It was noticed that the increased activity in the reactor IEA-R1 environment was related to newly fabricated fuel elements coming into operation. Despite the current surface treatment process to be perfectly stable and reproducible, a possible cause for the activity rise of water in IEA-R1 reactor pool could be the presence of residual uranium contamination on the surface not removed by surface treatment. For years, this problem was not observed due to the lower power reactor operation at 2 MW. However, with the increase in power to above 3.5 MW, this issue began to be spotted.

In this sense, this work adopted the hypothesis of residual uranium contamination on the surface of the fuel plates, which may be responsible for increased activity in the IEA-R1 environment. Fuel plates used in NFE fabrication were submitted to a chemical treatment to clean the surface, aiming the complete removal of any impurity present on their surfaces, including residual uranium. The process was developed based on information gathered from the literature and information from other manufacturers. It was shown statistically that the uranium content carried to the reactor pool is approximately 5µg of uranium per fuel element. It is considered a low level to cause any problems of increasing activity in the IEA-R1 reactor pool.

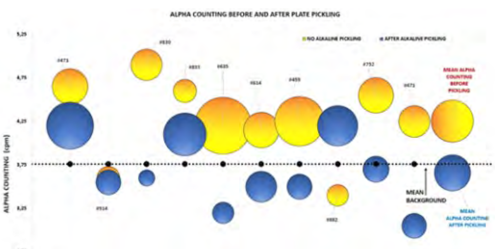


Figure 9. Effect of chemical treatment on fuel plates decontamination. Y-axis shows the level of contamination. Yellow circles represent the plates without treatment, blue circles with treatment, and white is the mean.

Uranium targets

Every year the world demands more than 30 million medical imaging procedures that use technetium-99m radioisotope (^{99m}Tc), which correspond to approximately 80% of all nuclear medicine diagnoses, by the means of a number of radiopharmaceuticals to assist the diagnosis of problems in different parts of the human body including heart, brain, liver, lungs, kidneys, bone, thyroid and mammary glands. This radiopharmaceutical product derives from the radioactive decay of molybdenum-99 (^{99}Mo), which is commercially produced in research reactors by irradiation targets that contain uranium-235, and it is delivered weekly to hospitals and clinics as ^{99}Mo generator from where ^{99m}Tc is extracted.

There are currently two established technologies available to produce uranium targets. One is based on a uranium-aluminum alloy dispersed in an aluminum matrix, and the second, based on metallic uranium thin foils. The first is already subject of good results at CCN, despite the fact that some development in the process still should be done, and the second has just started to be the subject of our research. Additionally, a third route, based on electrodeposition, is being studied, with some initial results.

Uranium-aluminum miniplates
CCN has been working on the development of uranium-miniplates manufacturing process. Because of its experience acquired over the years in the manufacturing technology based on dispersion fuels, UAl_x-Al dispersion targets seems to be the most suitable route for IPEN to the future production of ^{99}Mo in Brazil. For this purpose, uranium and aluminum are melted together, and this powder is mixed with aluminum, pressed submitted to hot rolling.

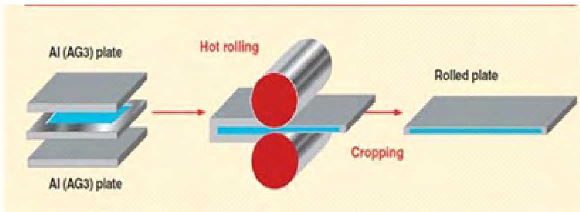


Figure 10. UAl_x miniplates rolling process.

The fissionable uranium-235 is limited to 20wt.% (LEU) in the total uranium contained in the intermetallic UAl_x powder. The index x identifies the phase composition of the compound, usually a mixture of UAl_2, UAl_3 and UAl_4. Characterizing the phase composition in UAl_x powder used as raw material for target fabrication is important because the maximum uranium concentration depends on the phase composition presented in the starting powder. Furthermore, it is important to mention that the UAl_3 and UAl_4 are more easily dissolved in alkaline solutions than the UAl_2, which defines, ultimately, the radiochemical processing throughput after the irradiation.

Pursuing the best variables for the process, CCN has manufactured several miniplates. The phase composition was quantified in the UAl_x powder and UAl_x-Al miniplates by means of image analysis and X-ray diffraction, applying the Rietveld method. The results from the two methods differed from each other with respect to the concentration determination. As a rule, the method based on X-ray diffraction showed higher potential to be applied to the RMB Project for phase quantification in UAl_x-Al dispersion targets, which is required by specification.

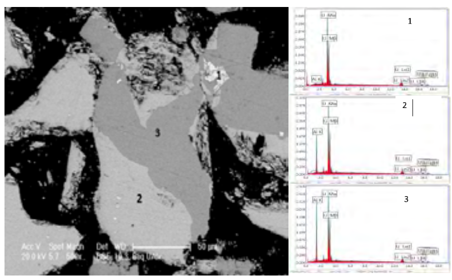


Figure 11. SEM and EDS analysis of the regions designated by 1, 2 and 3. The compositions of regions 2 and 3 indicate the presence of UAl_2 and UAl_3, respectively.

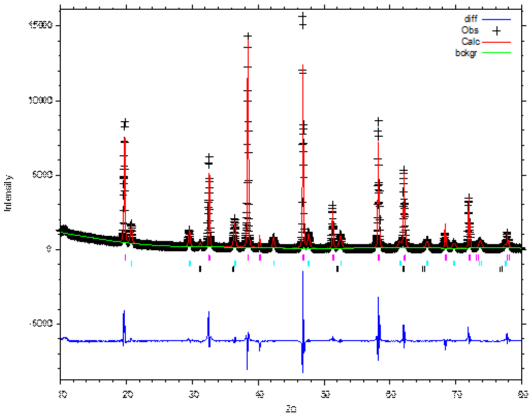


Figure 12. Experimental (black) and calculated (red) diffractograms for the UAl_x powder. The peaks marked with purple bars are from UAl_2, cyan bars from UAl_3 and black bars from UO.

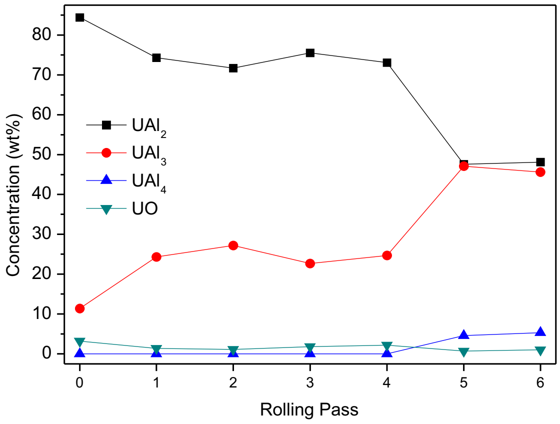


Figure 13. Evolution of phase composition (measured by XRD) in light of number of rolling passes.

During 2013, CCN was also in charge of fabricating ten Low Enriched Uranium targets (LEU UAl_x-Al) for irradiation purposes. These miniplates contained, on average, 2.75 gU/cm^3. It is worth to point out that the targets were successfully irradiated in the Critical Unit (UCRI), at the Center for Nuclear Engineering of IPEN (CEN), providing data for neutronic model validation for Mo-99 production.

Figure 14. Image of an individual uranium target (left), and the uranium targets packed and ready to be transported (right).

Uranium targets produced by electrodeposition

The electrochemistry of uranium at low temperature is important as a development route to produce ^{99}Mo irradiation LEU targets by electrodeposition. Usually, electrodeposition of uranium uses ionic or aqueous solutions to produce uranium deposits in acidic medium electrolytes (pH>2.5), and the performance of uranium electrodeposition is relatively erratic, since there is a high competition with H_2 evolution inside the reduction potential window. In the present studies, it was used an ionic electrolyte based on uranyl nitrate concentrate dissolved in 2-propanol, diluted in a proportion 1:20, containing 0.05 $mol.L^{-1}$ of natural uranium at a relative high acidity (pH<1). Working in a non-equilibrium state, when enough H_2 evolution happens, we carried out experiments using direct cathodic polarization -3.0 VAg/AgCl with electrodeposition time of 1800 seconds over nickel pre-plated substrate. With these settings, it was achieved an optimum mass deposit of 3.2 µg of uranium/cm^2, emitting alpha in the range of 40 Bq/cm^2.

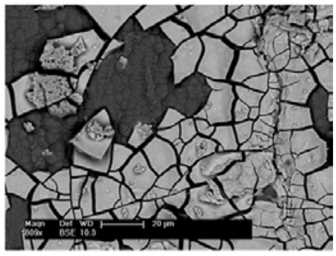
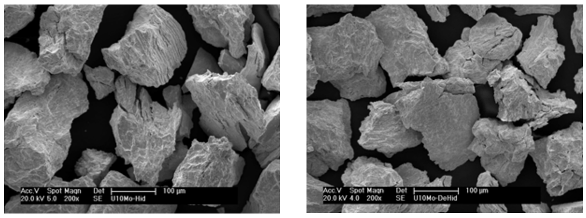


Figure 15. Uranyl electrodeposit over nickel substrate, after drying. Direct polarization at -3V during 1800s, using an electrolyte of 50mM of $UO_2(NO_3)_2$ in 2-propanol (1:20); pH<1. Drying process: 24h under room temperature atmosphere.

UMo-Al fuel manufacturing

UMo powder fabrication

We investigated the feasibility of powdering ductile U-10wt%Mo alloy by hydriding-milling-dehydriding of the gamma phase (HMD). The study was performed together with the Intermetallics Laboratory of CCTM. Hydriding was conducted at room temperature in a Sievert apparatus following heat treatment activation. Hydrided pieces were fragile enough to be hand milled to the desired particle size range. Hydrogen was removed by heating the samples under high vacuum.



A B

Figure 16. SEM micrographs (secondary electrons images) of hydrided (A) and dehydrided (B) U10wt%Mo powder particles.

X-ray diffraction analysis of the hydrided material showed an amorphous-like pattern that is completely reversed at the following dehydriding process. The hydrogen content of the hydrided samples corresponds to a trihydride, i.e. $(U,Mo)H_3$. SEM analysis of HMD powder particles revealed equiaxial powder particles together with some plate-like particles. A hypothesis for the amorphization (or nanocrystallization) of the hydrided alloy was formulated based on the high strains developed during hydriding, but without the severe or bulk rupture of the hydrided material observed when hydriding pure uranium.

UMo miniplates fabrication

Four miniplates of UMo-Al dispersion fuel containing U-10wt%Mo with high densities of natural uranium densities were produced. Table 1 shows the main characteristics of the fabricated miniplates.

Table 1 - Main Characteristics of the Fabricated Miniplates

Identification		*Thickness*	*Uranium Mass*	*Meat Dimensions* (mm)		*Uranium Density*
		(mm)	(g)	*Length*	*Width*	(gU/cm^3)
UMo-01		1.54	22.19	117	42	6.65
UMo-02		1.54	22.18	116	42	6.72
UMo-03		1.54	22.18	117	42	6.68
UMo-04		1.54	22.18	116	42	6.70

The UMo-Al meat integrity presented no cracks and good uranium homogeneity. The miniplates presented good metallurgical bond. No blisters or decladding was detected in blister and bending tests.

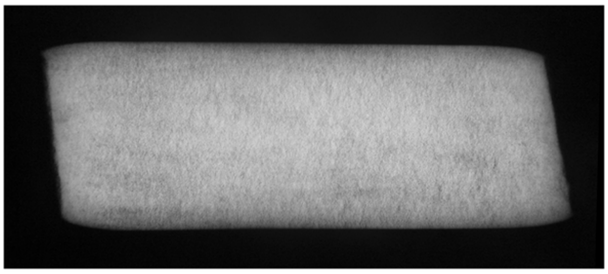


Figure 17. X-ray radiography of UMo-Al miniplate with uranium density around 6.7gU/cm^3 (sample UMo-04).

However, the macroscopic appearance of the dispersion after rolling was unsatisfactory. This was probably due to the particle size of the UMo powder used, between 150 and 44 µm. Since no fragmentation of UMo particles occurs during rolling due to its ductility, the use of finer powder is needed, with a maximum size of around 90 µm. The activities seeking to improve the quality of the microstructure of Al-UMo meats are in progress.

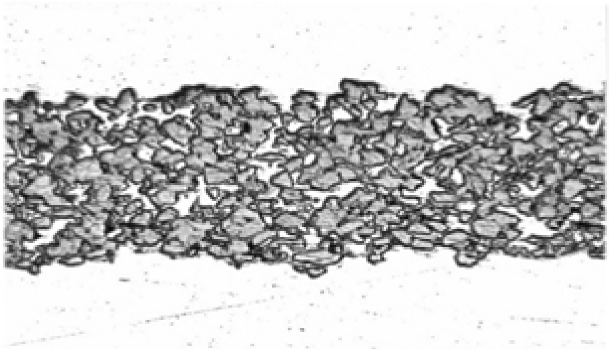


Figure 18. Typical micrograph of the meat of UMo-Al dispersion miniplate.

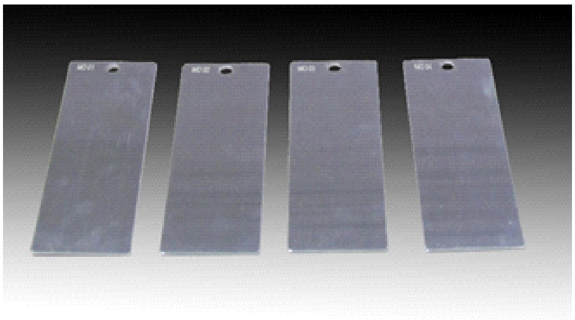


Figure 19. UMo fabricated miniplates.

Assuming the theoretical density of 17 g/cm^3 for the U-10wt%Mo powder, the maximum concentration of uranium in the Al-UMo is 6.9 gU/cm^3. The slightly lower densities achieved in this work, around 6.7 gU/cm^3, could be explained by the density of the powder used, 16.93 g/cm^3.

The value predicted for porosity was based in U_3Si_2 cores, which is around 8vol%. However, examining the microstructure of the U-Mo meat obtained, the residual porosity was much lower. Therefore, as the final porosity in Al-UMo meats was overestimated, the volume fraction of UMo particles exceeded the projected amount of 45vol%. This mismatch generated a greater variation in the thickness of the meat and a discontinuity in the aluminum matrix. Additionally, due to the very low particle fragmentation of ductile UMo, it was concluded that the size of these particles should be reduced to 90 µm. New manufacturing tests should be conducted to obtain a more suitable microstructure.

After the overcome of the problems presented, the next step is to fabricate full-sized fuel plates with natural uranium for evaluation of end defects. After that, will be possible to manufacture UMo-Al miniplates with enriched uranium and start irradiation testing in the IEA-R1 reactor. The development will be completed with the production of a partial UMo fuel element. It would be tested under irradiation for qualification in the RMB, eventually replacing U_3Si_2-based fuels in the future.

Chemical characterization of nuclear fuel

The nuclear fuel cycle is a series of steps involved in the production and use of fuel for nuclear reactors. The Laboratories of Chemistry and Environmental Diagnosis Center, CQMA, support the demand of Nuclear Fuel Cycle Program providing chemical characterization of uranium compounds and other related materials.

Among these, the determination of uranium content in U_3Si_2, UF_4, U_3Si_2-Al compounds and its impurities (metals and rare earths elements) can be highlighted. The last ones are determined using extraction chromatography and ICP-OES measurement. The method development to quantify the uranium content was based on Davies & Gray (Figure 20) methodology as used at NBL (New Brunswick Laboratory) adding an improvement on lowering the mass used in previous methodology. To the analyst, less radiation exposition is a safer condition to work. Also, lower quantities of chemical and radioactive wastes are produced. Another improvement achieved was the reducing of time consumed on determination of silicon and uranium in the respective alloy.

The XRF laboratory runs routinely analyses related to nuclear materials (U_3Si_2, Al powder and its alloys, AgInCd alloys, U and Th compounds). The XRF technique allows developing methods with a minimum or no chemical treatment. The laboratory participated in an Interlaboratory Comparison for the verification of international target value of uranium content in several uranium compounds using Davies and Gray methodology supported by ABACC (Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials) and NBL (New Brunswick Laboratories).

The Group of Chemical and Isotopic Characterization supports IPEN's nuclear fuel production program to IEA-R1 research reactor providing isotopic analysis for uranium compounds. Isotopic analysis is performed by using a high resolution inductively coupled plasma mass spectrometer. By means of a technical cooperation with the Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials (ABACC) and with the United States Department of Energy an intensive program, based on laboratorial inter-comparison exercises, for improvement of results from analysis of nuclear samples, has been in progress. New protocols for safeguards purpose were established and among them, environmental sampling based on swipe samples for the identification of uranium and plutonium in nuclear facilities. Whenever possible, CQMA laboratories are engaged to update or develop new methods to become the activities greener as currently expected.

Figure 20. Davies & Gray potenciometric titration for uranium content determination.

Reactor physics benchmarks at the IPEN/MB-01

During last decades, the reactor physics group of the Nuclear Engineering Center of IPEN is participating on two international programs for elaboration of benchmarks of experiments on critical facilities. The programs are the working groups ICSBEP (International Criticality Safety Benchmark Evaluation Project) and IRPhE (International Reactor Physics Evaluation Program) both sponsored by INL (Idaho National Laboratory, USA) and NEA (Nuclear Energy Agency).

ICSBEP is devoted to criticality safety benchmarks and IRPhE is more related to reactor physics experiments in general. The purpose of ICSBEP is: a) identify a comprehensive set of critical benchmark data and, to the extent possible, verify the data by reviewing original and subsequently revised documentation, and by talking with experimenters; b) evaluate the data and quantify overall uncertainties through various types of sensitivity analyses; c) compile the data into a standardized format; d) perform calculations of each experiment with standard criticality safety codes; e) formally document the work into a single source of verified benchmark critical data.

The work of ICSBEP group is documented as an International Handbook of Evaluated Criticality Safety Benchmark Experiments. The IRPhE project aims to provide the nuclear community with qualified benchmark data sets by collecting reactor physics experimental data from nuclear facilities worldwide. More specifically, the objectives of the expert group are as follows: a) maintaining an inventory of the experiments that have been carried out and documented; b) archiving the primary documents and data released in computer-readable form; c) promoting the use of the format and methods developed and seek to have them adopted as a standard.

The experiments have being performed at the IPEN/MB-01 research reactor facility. Several experiments have been designed, executed and analyzed. More than 100 critical configurations have been approved to be included in the ICSBEP handbook. From these experiments, it is possible to mention the critical configurations with borated stainless steel used in the storage pool of Angra-I and Angra-II power plants to save storage space. Another experiment was a central void simulation with an aluminum block. More recently, the reactor physics group completes a series of experiments with a heavy reflector made of SS-304 to give support to the EPR development in Europe. In the reactor physics area (IRPhE) a series of benchmark experimental problems on the isothermal reactivity coefficient of light water reactors were carried out. Those experiments contributed to give support on data evaluation of ^{235}U in the thermal energy region of neutron.

Establishing methods and facilities for seismic qualification of nuclear components

Many components of nuclear power and research reactors and of other nuclear facilities shall be seismic qualified by tests. In Brazil, the main needs for seismic qualification by tests are related to: (i) maintenance, repair and dedication programs for Angra I and Angra II NPPs; (ii) implementation of Angra III NPP; (iii) implementation of 4 new planned NPPs; (iv) implementation of a new Brazilian multipurpose research reactor (RMB); (v) implementation of the Brazilian Navy research reactor (LABGENE).

These tests must be conducted in adequate facilities with triaxial actuators to simulate earthquake loadings. In Brazil, there is not any experimental facility to perform triaxial earthquake simulations for masses greater than 100 kg. Also, there is a lack of experience in perform seismic qualification of nuclear components by tests. To start the assessment of this problem IAEA supported during 2012/2013 a project, BRA2018 conducted in CENM to help in the establishment of the type and size of seismic experimental facility to perform nuclear components qualification by tests and the experimental methods to be used according to Brazilian needs.

The project is the starting point considering that its main targets are: (i) specify a seismic testing facility to comply with Brazilian nuclear program needs; (ii) develop approaches and methods to perform nuclear components seismic qualification by tests. The future laboratory will give a good opportunity to Brazilian suppliers to develop qualified nuclear components for the national nuclear development plan.

The IAEA project BRA2018 allows people from the participating organizations (IPEN, ELETRONUCLEAR and CTMSP) to perform 3 scientific visits and 3 fellowships in laboratories in EUA, France, UK and Macedonia. Also, a short course and an expert mission on “Establishing methods and facilities for seismic qualifications of nuclear components for Brazil” were held in Rio de Janeiro attended by more than 20 people from the project participating organizations. The project main results were the draft technical specification for the laboratory shake table and the draft procedures to conduct seismic qualification tests. The next steps depend on the financial support to the laboratory implementation and some alternatives are being discussed with ELETRONUCLEAR and CTMSP.

STAR test section for of loss of coolant experiments in IEA-R1 research reactor

STAR Test Section was designed to simulate loss of coolant experiments using the Instrumented Fuel Assembly EC-208 of the IEA-R1 Research Reactor. STAR has received financial support of the Nuclear Engineering Center of the IPEN-CNEN/SP. The STAR was designed to conduct experiments of partial and complete uncovering. The proposed experiments aim are the reproduction of heat transfer conditions similar to those expected in loss of coolant accident in researches reactors, with safe and controlled way. Experimental data can be used to validation or development of computational tools for LOCA analysis. STAR Test Section comprises a base, a cylindrical stainless steel vessel, and the instrumentation. It uses the compressed air system of the IEA-R1 reactor for uncovering process. In addition to the fourteen thermocouples of the EC-208, STAR has more four thermocouples and one differential pressure transducer to the water level measurement inside the vessel. Figure 21 shows the base, with a “dummy” fuel assembly, representing the EC-208, and the Fig. 22 shows the STAR with the vessel mounted on the base.

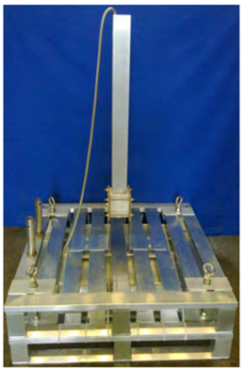


Figure 21. Base of the STAR.

Figure 22. STAR test section.

Revisiting stainless steel as PWR fuel rod cladding after Fukushima Daiichi accident

Austenitic stainless steel was the chosen material for fuel rod cladding in the first PWRs. Since 1960 stainless steel claddings have been replaced in commercial reactor cores by zirconium-based alloys, due to its lower absorption cross section for thermal neutrons and its higher melting temperature. This lower absorption for thermal neutrons allows cores using zirconium-based alloys as fuel rod cladding to operate with lower enrichment cost than cores using stainless steel as cladding material. Despite the above factors, there are some benefits in using stainless steel as cladding material in PWRs. During steady-state and controlled transients, austenitic stainless steel is more resistant than Zircaloy and is therefore less susceptible to damage due to pellet cladding mechanical interaction (PCMI). Stainless steel is also less susceptible than Zircaloy to stress corrosion cracking generated by fission products in the fuel. As a result, the formation of cracks on the cladding inner wall is less likely and higher concentrations of fission products can be tolerated in stainless steel fuel rods. During loss of coolant accidents (LOCA), in which the cladding temperatures remain below 1200°C, austenitic stainless steel exhibits a metal-vapor reaction rate, a quantity of liberated hydrogen and a reaction heat lower than Zircaloy.

The potential for oxygen embrittlement is almost inexistent for stainless steel and its mechanical strength and ductility are better than those of Zircaloy. This results in a smaller cladding deformation and reduced cooling channel blockage. The use of stainless steel cladding has the advantage of not presenting the violent oxidation reaction that occurs with zirconium-based alloys at high temperatures. During the Fukushima Daiichi accident this reaction caused the release of large amounts of hydrogen and major structural damage to the reactor core structure. The austenitic stainless steels used as fuel rod cladding in the first PWRs were of types AISI 304, 347 and 348. With the exception of small isolated failures, its performance was excellent. Nevertheless, only limited efforts have been made to model the thermal-mechanical behavior of PWR fuel rod using stainless steel as cladding.

In recent years, detailed investigations of the interrelated effects of the fuel rod thermal-mechanical behavior under irradiation have been carried out by fuel performance codes. Due to the proprietary nature of this codes, a group of IPEN-CNEN/SP researchers concluded that efforts should be applied in the construction of a fuel performance code named IPEN-CNEN/SS to model the thermal-mechanical behavior of PWR fuel rod using stainless steel as cladding and a particular emphasis was given for type AISI 348. The basis for this new code was the FRAPCON-3.4 code sponsored by the United States Nuclear Regulatory Commission (U.S.NRC) for the licensing of nuclear power plants. Since this code is restricted to evaluate zirconium-based alloys cladding, it was modified to evaluate stainless steel as cladding.

Using IPEN-CNEN/SS to compare the performance of type 348 stainless steel and Zircaloy-4 claddings (Tables 2 and 3) under a common power history reveals that type 348 stainless steel rods display higher fuel temperatures and wider gaps than Zircaloy-4 rods. No gap closure is observed for type 348 stainless steel rod. On the other hand, the Zircaloy-4 cladding spends a high fraction of life in a state of tensile stress at the ridge. Nevertheless, the steady state thermal performance of the two fuel rod types is very similar, confirming that stainless steel could be a good option to replace zirconium-based alloys as cladding material in PWRs in order to improve the safety conditions under accident conditions. Neutronic assessments have shown that the higher stainless steel neutron absorption cross section could be compensated, for the evaluated PWR rod, by increasing about 20% in the ^{235}U enrichment. Also, there is an alternative of combining enrichment increase with pitch size changes in order to compensate the neutron absorption penalty due to the use of stainless steel.

Table 2: Data obtained from simulation for Zircaloy-4 fuel rod

	Time (h)	Burnup (MWd/ kgU)	Power (kW/m)	Average cladding temp. (K)	Gap (cm)	Fuel center line temp. (K)	Cont. (MPa)	Hoop stress (MPa)	Fuel outs. diam. (cm)	Internal Pressure (MPa)
1	235	0.25	12.17	584	0.00531	922	0	-78.77	0.8335	5.98
2	5035	6.87	15.91	591	0.00376	1033	0	-77.85	0.8342	6.14
3	7373	10.05	24.70	606	0.00216	1304	0	-72.81	0.8367	6.90
4	9353	14.20	24.18	606	0.00145	1276	0	-71.58	0.8372	7.09
5	12629	20.93	23.62	605	0.00018	1237	0	-69.09	0.8380	7.49
6	14069	23.63	20.51	600	0.00013	1146	0	-69.17	0.8382	7.50
7	16711	28.32	20.47	600	0.00013	1168	0	-67.09	0.8374	7.83
8	20237	34.45	19.95	600	0.00013	1175	20.22	16.23	0.8371	8.04
9	22404	33.75	15.42	608	0.00013	1041	16.73	-7.29	0.8361	7.93
10	25368	37.67	15.39	609	0.00013	1052	26.03	55.74	0.8367	7.94
11	28704	42.10	15.39	609	0.00013	1065	27.57	66.42	0.8373	8.06
12	30480	46.43	9.74	591	0.00013	876	9.27	-58.05	0.8371	7.95
13	34320	49.91	11.09	596	0.00013	929	26.31	58.12	0.8378	8.09
14	37512	53.14	11.94	599	0.00013	972	32.80	110.79	0.8385	8.23
15	40080	55.96	13.19	603	0.00013	1015	33.76	109.49	0.8391	8.36

Table 3: Data obtained from simulation for AISI 348 fuel rod

	Time (h)	Burnup (MWd/ kgU)	Power (kW/m)	Average cladding temp. (K)	Gap (cm)	Fuel center line temp. (K)	Cont. (MPa)	Hoop stress (MPa)	Fuel outside diam. (cm)	Internal Pressure (MPa)
1	235	0.25	12.17	584	0.00650	944	0	-81.88	0.8336	5.52
2	5035	6.87	15.91	590	0.00582	1087	0	-81.56	0.8346	5.56
3	7373	10.05	24.70	604	0.00447	1408	0	-77.25	0.8375	6.22
4	9353	14.20	24.18	604	0.00409	1394	0	-76.51	0.8381	6.33
5	12629	20.93	23.62	603	0.00340	1380	0	-75.04	0.8391	6.55
6	14069	23.63	20.51	598	0.00338	1266	0	-75.61	0.8388	6.46
7	16711	28.32	20.47	598	0.00290	1268	0	-74.38	0.8396	6.65
8	20237	34.45	19.95	596	0.00231	1251	0	-72.85	0.8405	6.88
9	22404	33.75	15.42	602	0.00282	1107	0	-73.82	0.8391	6.73
10	25368	37.67	15.39	601	0.00241	1107	0	-72.61	0.8398	6.92
11	28704	42.10	15.39	601	0.00196	1108	0	-70.46	0.8405	7.24
12	30480	46.43	9.74	587	0.00188	903	0	-72.24	0.8402	6.97
13	34320	49.91	11.09	590	0.00142	951	0	-69.63	0.8410	7.36
14	37512	53.14	11.94	593	0.00102	988	0	-67.07	0.8418	7.75
15	40080	55.96	13.19	595	0.00066	1021	0	-64.54	0.8424	8.13

Neutronic comparison of the nuclear fuels U_3Si_2/Al and U-Mo/Al

Nuclear fuels composed by uranium metal alloys in monolithic and dispersed forms have been considered for research and power reactors due to their density properties and fast heat transfer. Among several candidates, U-Mo alloys are one of the most promising systems for plate type fuel elements owing to its broad gamma-phase stable field.

This fact allows extensive fabrication capability since cubic gamma-phase shows good plasticity, higher strength and elongation. Because of the high uranium density and good irradiation stability of U-Mo alloys, this fuel in the form of dispersion in an Al matrix is the choice for the conversion of research and material test reactors currently using highly enriched uranium (HEU) to low-enriched uranium (LEU). The formation of an interaction layer between U-Mo particles and the Al matrix as a result of interdiffusion has become a major issue for the performance of this fuel. The formation of an interaction product in this dispersion fuel is unfavourable because of its low thermal conductivity and volume expansion as it consumes the Al matrix.

Depending on the irradiation conditions (high burnup or high heat flux), large pores are formed at the interface of the interactions products and the Al matrix, which could eventually lead to a fuel plate failure. Many post irradiation tests have been conducted for uranium alloys with molybdenum content between 6 to 10% by weight allowing the characterization of U-Mo/Al interaction, and this fuel qualification is an on-going process.

An U_3Si_2/Al dispersion fuel with a uranium density of 4 gU/cm^3 was once considered as the fuel for the first core of the new Brazilian Multipurpose Reactor (RMB). This research compares the calculated infinite multiplication factor (K_∞), obtained through neutronic calculation with the code Scale 6.0, for fuel plates reflected in all directions using U_3Si_2/Al and U-Mo/Al dispersion fuels. The U_3Si_2/Al dispersion fuel used in the calculation has a uranium density of 4 gU/cm^3 and the U-Mo-Al dispersion fuels have densities ranging from 4 to 7,52 gU/cm^3 and 7 to 10% Mo addition. The percentage by weight of molybdenum (Mo) in the dispersion changes the neutronic behavior of the fuel since the neutron absorption by Mo is considerable higher than that by Si. The calculated infinite multiplication factor (K_∞) obtained from the simulations with the code scale 6.0 are shown in Tables 4 and 5. Table 4 shows the values obtained for U-7wt%Mo/Al fuel and Table 5 for U-10wt%Mo/Al fuel. For U_3Si-2/Al the values 1.65782 and 0.00012 were obtained for K_∞ and *σ K_∞, respectively. Figure 23 presents the infinite multiplication factors plotted against uranium density for all studied fuels in this work. The top line represents the K_∞ value obtained for U_3Si_2/Al.

It can be seen from Figure 23 that the K_∞ versus values obtained for different uranium densities with U-10wt%Mo/Al fuels are below those obtained with U-7wt%Mo/Al fuels. This behavior was expected due to the different absorption cross section of the two materials. The potential benefits of the high density fuel will depend on the research reactor to be upgraded and, a priory, it is difficult for potential users to clearly understand what kind of economic or improvement benefits can be expected. Further works are being conducted in order to identify improvements in core performance (higher neutron fluxes) and on the impact of fuel density on the cost of the research reactor fuel cycles (to reduce the number of fuel assemblies needed for operation).

Table 4: Infinite multiplication factors for U-7wt%Mo/Al fuel ranging from 4 to 7,52 gU/cm^3

Density Uranium (gU/cm^3)	K_∞	*σ K_∞
4.01	1.64642	0.00012
4.55	1.65203	0.00012
5.02	1.65634	0.00012
5.55	1.65596	0.00012
6.02	1.65561	0.00012
6.55	1.65439	0.00012
7.02	1.65270	0.00013
7.52	1.65059	0.00012

* Uncertainty

Table 5: Infinite multiplication factors for U-10wt%Mo/Al fuel ranging from 4 to 7,52 gU/cm^3

Density Uranium (gU/cm^3)	K_∞	σ K_∞
4.01	1.64040	0.00012
4.52	1.64581	0.00012
5.02	1.64827	0.00012
5.56	1.64905	0.00013
6.00	1.64865	0.00012
6.54	1.64727	0.00012
7.01	1.64517	0.00012
7.11	1.64478	0.00012

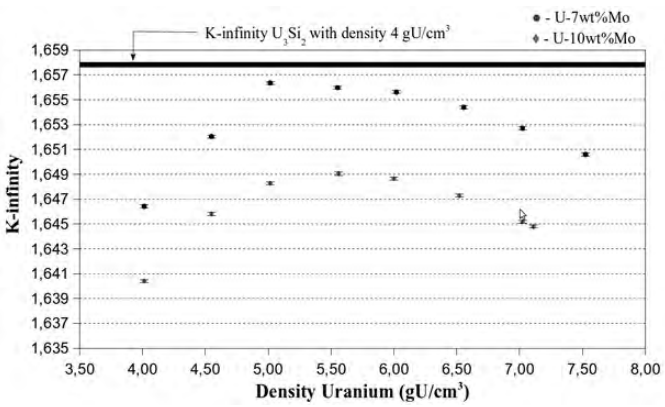


Figure 23. K_∞ for U_3Si_2/Al fuel (4 gU/cm^3) and for U-7wt%Mo/Al and U-10wt%Mo/Al with uranium densities ranging from 4.01 and 7.52 gU/cm^3.

Neutronic calculations using different methodologies (transport and Monte Carlo) to characterization and technical specification generation of targets for ^{99}Mo production by fission

The objective of this work was to develop studies on the characterization and technical specification of targets for production of ^{99}Mo. A detailed bibliographical study selected three types of targets: UAlx-Al plate type, and U-Ni cylindrical and plate types. Neutronic calculations were performed to analyze whether the targets would produce the minimum required amount of ^{99}Mo, 450 Ci per week, that meets demand from Brazil, and to verify the target impacts in the reactor operation. The cross sections of all the reactor core materials were generated with HAMMERTECHNION. The CITATION was used to make the 3D modeling of the reactor core and to determine parameters such as k-effective, neutron flux and power density. The program SCALE 5.1 was used to find the targets burnup, and the inventory of nuclides generated. Neutronic calculations showed that the current Brazilian demand of ^{99}Mo, 450 Ci per week, and the projected future demand of 1000 Ci, can be met by using targets of UAl_x-Al and U-Ni. The analyzes were realized for the same amount of uranium present in the targets (20,1 g) and the same irradiation conditions. From equation $\{[N * (\sigma 1\Phi 1 + \sigma 2\Phi 2 + \sigma 3\Phi 3 + \sigma 4\Phi 4) * V * y * (1\text{-}e\text{-}\lambda t)] / 3.7 * 1010\}$ and from the microscopic fission cross sections produced and collapsed into 4 groups by HAMMERTECHNION it was possible to calculate the expected results and compares them to the results generated with the SCALE 5.1.For UAl_x-Al and U-Ni plate type targets the expected calculations converge to values very close to those performed by the SCALE, but for the U-Ni cylindrical target the results were inconsistent.

The inconsistency was due to the fact that the HAMMERTECHNION does not have a module for calculating self-shielding, which makes it unsuitable for this analysis. To solve the problem was used the software package AMPX from the Oak Ridge National Laboratory to generate the cross sections of the homogenized cell (aluminum radiator + U-Ni cylindrical target + cooling channels) of the U-Ni cylindrical target. This program package contains the module Rolaids which executes a integral transport calculation to handle the effect of self-shielding in multi- regions. The programs SCALE 5.1 and MCNP 5 were utilized for calculation the cross sections of the reactor core materials, the 3D modeling of the core and to determine parameters such as k-effective, the neutron flux and power density. The calculations were compared with each other to ensure consistency of methodology.

Neutronic and thermal-hydraulic analysis of a device for irradiation of LEU UAl_x-Al targets for ^{99}Mo production in the IEA-R1 reactor

Technetium-99m (^{99m}Tc), the product of radioactive decay of molybdenum-99 (^{99}Mo), is one of the most widely used radioisotope in nuclear medicine, covering approximately 80% of all radiodiagnosis procedures in the world. Nowadays, Brazil requires an amount of about 450 Ci of ^{99}Mo per week.

Due to the crisis and the shortage of ^{99}Mo supply chain that has been observed on the world since 2008, IPEN decided to develop a project to produce ^{99}Mo through fission of uranium-235. The objective of this work was the development of neutronic and thermal-hydraulic calculations to evaluate the operational safety of a device for ^{99}Mo production to be irradiated in the reactor core IEA-R1 at 5 MW. In this device was placed ten targets of UAl_x-Al dispersion fuel with low enriched uranium (LEU) and density of 2.889 gU/cm³. For the neutronic calculations were utilized the computer codes HAMMER-TECHNION and CITATION, the maximum temperatures reached in the targets were calculated with the code MTRCR-IEAR1. The analysis demonstrated that the device irradiation will occur without adverse consequences to the operation of the reactor. The total amount of ^{99}Mo was calculated with the program SCALE, obtaining an activity of 620 Ci for 3 days irradiation, 831.96 Ci for 5 days and after 7 days irradiation the activity was 958.3 Ci.

Study of necessary equipment for cogeneration viability in commercial installations at concession area of the São Paulo Gas Company

This study aims to identify the characteristics of the equipment, to generate the energy balance, and to assess the potential for energy conservation that different equipment configurations may offer, to make them economically attractive in commercial facilities of São Paulo.

To achieve the above goals, energy balance studies were carried out in detail accounting for thermal energy flow rejected for different temperature ranges of engines and turbines used for CHP. The results were applied to heat recovery systems assembling different retrieval settings, such as domestic hot water, steam, chilled water production (lithium bromide-water absorption chiller) cold production at low temperature (ammonia-water absorption chiller). Figure 24 shows the schedule for cogeneration system using natural gas engines for hot and cold water production from flue gases. Surveys were also conducted in the field of commercial premises where it would be possible to apply cogeneration, such as laundries, data centers, supply centers (central food supply), among others. Important research was conducted at the Port of Santos in order to evaluate the energy matrix of the harbor and the possibility of implementing cogeneration thermoelectric generation from natural gas. The purpose would be the supply of power to both terminals on land and to make the cold ironing, namely the supply of electricity to ships at berth. This supply electricity mainly aims at reducing emissions for ships that currently use oil to generate the energy required for this step. Figure 25 shows a photo of a large container ship, the Cap San Marco docked at Santos.

This type of vessel carries containers, including refrigerated / frozen called reefers that require electricity so while remaining on ships as when they are downloaded to the terminals on land, when they should be connected as fast as possible, while providing power terminal. The application of cogeneration in data centers is a system that has more possibilities for deployment in Brazil. The energy consumption per square meter will be higher because of the concentration of processing volume. The challenge of this new generation of data centers will be the pursuit of energy efficiency. The project involved a survey carried out in ECC (Electronic Computing Center) of University of São Paulo. The Electronic Computer Center (ECC) of USP involves coordinating body of the main functions of computer and data communication at the University of São Paulo, also providing computer services to the university community of the USPA.

Surveys and records electrical equipment currently used in air conditioning and ventilation system were made from the center. The part of the field survey was conducted by IEE - USP. Several simulations were also performed to evaluate the installation of cogeneration systems in CCE-USP. This research is part of a research project for Distributed Generation and Cogeneration with Natural Gas: Technological and Institutional Barriers and was conducted in conjunction with the IEE-USP.

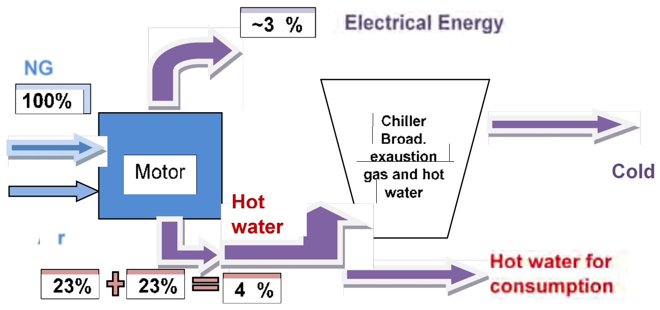


Figure 24. Schedule for cogeneration system using natural gas engines for hot and cold water production from flue gases.

Figure 25. Cap San Marco docked at Santos.

Flow regime identification and heat transfer coefficient in the core of ANGRA-II nuclear power plant during small break LOCA simulated with RELAP5 code

The present project aims to use RELAP5/MOD3.2 gamma code to simulate the behavior of Angra-II nuclear reactor core for a postulate loss of coolant accident in the primary circuit, Small Break Loss of Coolant Accident (SBLOCA). This accident and boundary conditions are described in detail in Chapter 15 of the Final Safety Analysis Report of Angra-II - FSAR [2]. The accident consists basically of the total break of a pipe of the hot leg Emergency Core Cooling System (ECCS) of Angra-II, which is a PWR reactor with four primary loops, and power of 1,400MW(e). The rupture area is 380 cm^2, which represents 100% of the ECCS pipe flow area. In this simulation, failure and repair criteria are adopted for the ECCS components, in order to verify the system operation, in carrying out its function as expected by the project to preserve the integrity of the reactor core and to guarantee its cooling. SBLOCA accidents are characterized by a slow blowdown in the primary circuit to values that the high pressure injection system is activated. The thermal-hydraulic processes inherent to the accident phenomenon, such as hot leg of ECCS vaporization and consequently core vaporization causing an inappropriate flow distribution in the reactor core, can lead to a reduction in the core liquid level, until the ECCS is capable to refill it. Results were obtained with RELAP5 to the core of ANGRA2, for the considered SBLOCA. Figures 26 to 28 summarizes the obtained results of SBLOCA of Angra-II analysis using RELAP5 code. Pressures, flow rates and primary system mass results, were compared obtained with FSAR, and showed to be in a reasonable agreement.

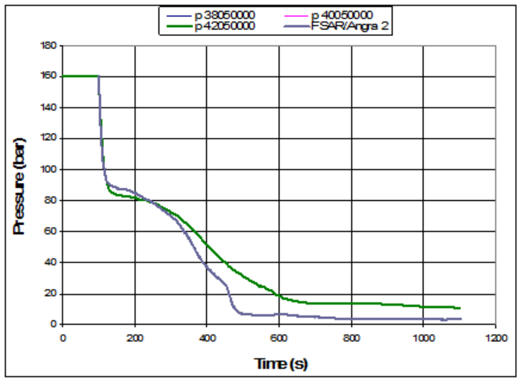


Figure 26. Pressure in the Angra-II NPP core (RELAP5 and FSAR).

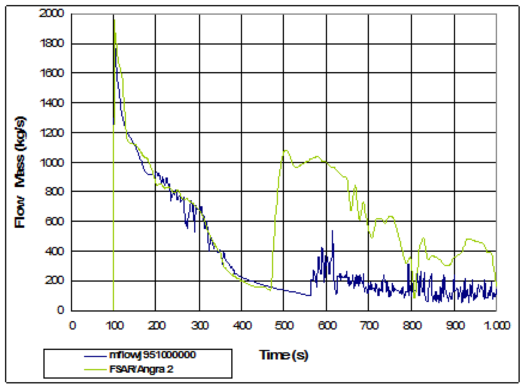


Figure 27. Flow mass in the break (RELAP5 and FSAR).

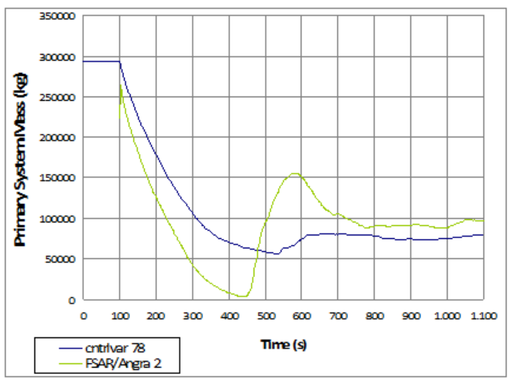


Figure 28. Primary system mass (RELAP5 and FSAR).

Integrated environmental management: a case study at the federal office of Brazilian research

The globalization process establishment has been a major impulse for profound transformations as to environmental issues in the social, political and economic scenario of both industrialized and developing countries. Within this scope, the concern with climate changes, global warming, biodiversity, population growth and public health have promoted the dissemination of environmental values and the induction to a community participative culture. Notwithstanding, a growing demand by the society related to the environment and social issues has been evidenced, converging the environmental theme to a holistic approach and, also, to the life quality concern. Therefore, private and public organizations have given more attention to issues involving their internal and external clients and/or users, in view of their products or services and social aspects, including those covering their workers and collaborators health and safety: with this overall purpose, an Integrated Management System (IMS) for Quality, Environment, Health and Safety was created. This management policy has been, commonly, employed in the private sector, even though a small, but yet expressive part of it refers to the public area. In face of this scenario, it may be foreseen that the motivations for adopting such management tool and the methods used for this goal may differ, according to the economic context. Under this point of view, this work had the target of analyzing, qualitatively, the process of setting the IMS in a federal. Eventually, a targeted result was to identify advantages and disadvantages for a public institution.

Risk communication importance

Risk Communication has shown its importance in the elaboration of emergency plans in the Chemical industry. In the 90's, the UNEP developed the APELL (Awareness and Preparedness for Emergency at Local Level) plan, a risk management methodology used by dangerous chemical facilities. The methodology comprises the commitment of both Government and the community located in the risk area in the development of the emergency plan. In the nuclear sector, there is no similar methodology developed so far. However, establishing a communication channel between the nuclear segment and the community is essential. In Brazil, the construction of Angra-III NPP and the RMB (Multipurpose Research Reactor) project stand as nuclear initiatives that improve the importance of a good communication to the public. Security issues of these projects are natural sources of concernment to the public, which is aggravated by events such as the Fukushima disaster. Without an effective communication about what means the presence of nuclear plants and reactors in a specific area, the interested public will only have an alarmist vision of the subject, given by those against these facilities.

Experimental and numerical thermal-hydraulic studies on plate type fuels for research nuclear reactors

The objective of this research project is to carry out experimental and numerical thermal-hydraulic studies on plate type fuels for research nuclear reactors. One of the research in development is the test of an instrumented fuel element pattern, plate type, for thermal-hydraulic evaluation of the cooling of the fuel plates, especially the side plates, which form channels between adjacent elements, and thus provide information, not available in the literature, to improvements in design, construction and operation conditions of cores of research reactors. This was motivated by ensuring the safety of the IEA-R1 reactor increasing power up to 5 MW. Another research project aims to design and build a test section for the study of induced flow in fuel elements of nuclear research reactors parallel plate type vibrations. The study of the dynamic of the parallel plate fuel elements behavior is of great importance to the safety of nuclear research reactors where the flow of cooling fluid may reach high operational conditions (critical speeds) causing vibrations and may, in the latter case lead to the collapse of fuel plates, which can result in serious accident proportions.

Knowledge of the operational limits of the fuel elements designed to be used in the research reactor under development in Brazil requires the testing that can be compared with the criteria required for your license of operation. In this research project an experimental test bench for fuel elements of nuclear research reactors was designed and reconstructed at the Center for Nuclear Engineering (CEN-IPEN-CNEN/SP). The tests consists of analysis of vibration induced by the flow in the channels formed between the plates and deformation of fuel plates. Numerical simulations on plate type fuel for research reactors were carried out as can be seen in Fig. 29.

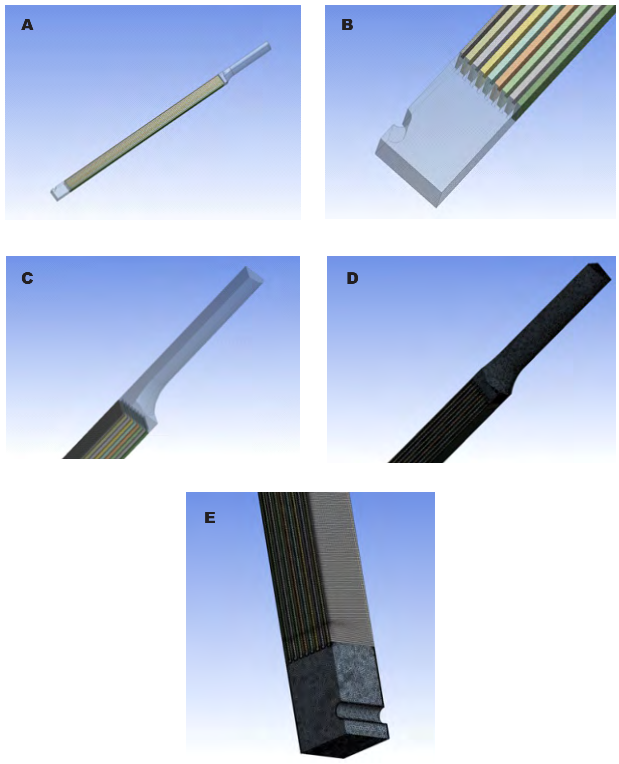


Figure 29. Numerical modelling of the plate type fuel for research reactor: A) geometry general view; B) geometry detail of fuel top section; C) geometry detail of fuel bottom section (nozzle); D) numerical mesh detail of fuel bottom section (nozzle); E) numerical mesh detail of top fuel section.

Mechanical analysis of nuclear research reactor IEA-R1 pipelines

An engineering project was developed to identification failure of the decay pipeline, brackets, flanges and screws in channels of the nuclear research reactor IEA-R1 at IPEN-CNEN/SP. The project analysis recommend the work of a consulting firm specialized in corrosion testing with gammagraphy pipe channels of IEA-R1. It was performed a stress analysis simulation of primary circuit pipeline of IEA-R1 coupled pipe, props and equipment indicating a probable failure scenarios pipe of channels. The after failure scenario of the pipeline indicated: a new project supported channels, and changes in the design of some supporters of the primary circuit. The stress analysis, coupled pipe, props and equipment of primary circuit pipeline of IEA-R1 indicated design modifications of the media. Technical reports were prepared as part of the announcement to reform of pipeline with the following specifications: specification of the pipe; specification supports; pipe stress analysis; stress analysis of media. It was detected in advance the problems that occurred with the pipe IEA-R1, new project of supported channels, and specifications for piping and brackets were developed.

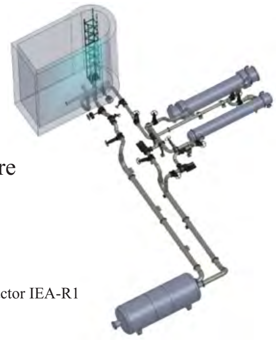


Figure 30. Schematics of the nuclear research reactor IEA-R1 primary circuit pipeline and pool.

Software for 3D images and dose distribution calculation

The “AMIGOBrachy” is planning software for treating cancer with brachytherapy sources developed in MATLAB at the Nuclear Engineering Center. The main resources and tools offered by software AMIGOBrachy are: Acquisition of tomographic images of clinical diagnoses of patients and engagement with the Monte Carlo code MCNP6 to calculate 3D dose distribution in the patient; Modeling of radioactive sources used in cancer treatment; Compatibility for exchanging information with commercial planning systems of large companies; Incorporation of innovative resources for planning and increased effectiveness in treating.

Several studies have reported methodologies to calculate and correct the transit dose component of the moving radiation source for high dose rate (HDR) brachytherapy planning systems. However, most of these works employ the average source speed, which varies significantly with the measurement technique used, and does not represent a realistic speed profile, therefore, providing an inaccurate dose determination. In this work, the authors quantified the transit dose component of a HDR unit based on the measurement of the instantaneous source speed to produce more accurate dose values. The present work demonstrated that the transit dose correction based on average source speed fails to accurately correct the dose, indicating that the correct speed profile should be considered. The impact on total dose due to the transit dose correction near the dwell positions is significant and should be considered more carefully in treatments with high dose rate, several catheters, multiple dwell positions, small dwell times, and several fractions.

In the triennial 2011-2013 the IEA-R1 Research Reactor has been operated most of the time at a power of 4.5 MW and operation schedule of 62 hours per week, achieving the following results:

- Reactor power: 4.5 MW
- Time of reactor operation: 6473 h
- Energy dissipation: 27057 MWh
- Number of samples irradiated (grid plate and pneumatic system): 4473
- Production of ^{131}I: 1963 Ci
- Production of ^{153}Sm: 79 Ci

The results related to 2013 were negatively impacted by an unscheduled stop of the reactor due to problems detected in the primary cooling system. Besides of the routine operational schedule, other activities were carried out to extend the operational lifetime of the reactor, improve the conditions to comply with user needs and allow the operation in higher powers:

Monitoring systems

Monitoring systems are being improved in order to increase the confidence level in the safety of IEA-R1 reactor operating conditions. These include:

- Online system to monitoring aerosol emission - in implementation;
- Reactivation of a meteorological station at IPEN site - conclude.

Ageing program

In order to extend the reactor operation lifetime some equipments and systems are being replaced or rebuilt. These include:

- Conclusion of the replacement of the electronic racks of the control and the emergency rooms (Fig. 31);
- Replacement of the steel cable of the chimney;
- Adequacy of the floor of the engine room (basement floor) and of the reactor hall (third floor) (Fig. 32);
- Improvement in the electric system building (Fig. 33);
- Adequacy of the diesel tank cover (Fig. 34);
- Project and hiring supplier for cooling tower substitution.

before after

Figure 31. Racks of control and emergency rooms (before and after substitution).

before

after

before after

Figure 32. Floor of the engine room (basement floor) and of the reactor hall (third floor) (before and after adequacy).

before after

Figure 33. Electrical system building (before and after improvement).

before after

Figure 34. Adequacy of the diesel tank cover.

Physical protection system

The main gate was renovated and access control has been improved by the installation of a tourniquet released by an electronic system. (Fig. 35).

before

after

after

Training program

Besides of the internal operator retraining program, it was started the training of eight new operators.

Maintenance program

An extensive program of testing, preventive maintenance and calibration was carried out to assure performance and reliable measures by equipments and systems related to reactor operation and control.

Monthly activity report

A report containing the activities and monitored and controlled operational parameters of the reactor are issued monthly. This report encloses the number of reactor operation, dissipated energy, reactor core data, chemical and physical characteristics of the pool water, radioisotopes concentration in the pool water, number of reactor shutdown, number of irradiated samples in the reactor core and pneumatic system and, finally, the radioprotection data.

Management system

Since 2002 the Quality Management System that support the scope “Operation and Maintenance of the IEA-R1 Reactor and Irradiation Services” was considered certified by Fundação Carlos Alberto Vanzolini in compliance with NBR ISO 9001, being submitted to annual internal and external reevaluation.

Honor Mention and Awards

The study developed in the LEER: “Determination of Inorganic Elements in Blood of Golden Retriever Muscular Dystrophy Dogs using Neutron Activation Analysis” was awarded with financial support from Women in Physics Travel Grant Award, IUPAP (International Union of Pure and Applied Physics) to be presented in the 5 International Congress of the FESTEM of the European Societies for Trace Elements and Minerals, Avignon, France, 2013.

Tatiane da Silva Nascimento - melhor pôster pelo trabalho: “Standardization of Y-90 in Liquid Scintillation Counting System by Means of the Ciemat/Nist Method” no XXXVI Reunião de Trabalho sobre Física Nuclear no Brasil, 2013.

The study “Concentrations of Ions in Blood of Amateur and Elite Runners using NAA” was awarded with the Young Scientis, Bangkok, Thailand, 2012.

Two studies developed in the LEER: “Concentration of Ca in Blood of Amateur Runners using NAA” and “The Half Life of Te131g,m” were awarded in the XXXV Reunião de Trabalho sobre Física Nuclear no Brasil, 2012.