Nuclear Reactors and Fuel Cycle

Nuclear Reactor IEA-R1

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Introduction

The Center for Nuclear Engineering has shown, along several years, expertise in the field of nuclear and energy systems and correlated areas. Due to the experience obtained over decades in research and technological development at Brazilian Nuclear Program, personnel has been trained and started to actively participate in the design of the main system that will compose the Brazilian Multipurpose Reactor (RMB) which will make Brazil self-sufficient in the 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 years from 2008 to 2010 numerous specialized services of engineering for the nuclear power plants Angra 1 and Angra 2 were carried out, in addition to the development of many related technologies applied to nuclear engineering, thus enabling the institute to fulfill its mission, that is to contribute in improving the quality of life of Brazilian people.

The Nuclear Fuel Center is responsible for the production of the nuclear fuel necessary for the continuous operation of the IEA-R1 research reactor. Development of new fuel technologies is also a permanent concern. A program for autonomous serial fuel element production started in the 80's, motivated by the political constraints for buying these fuels abroad in order to keep the reactor in operation. The fuel element fabricated at IPEN is a dispersion LEW (Low Enriched Uranium) Material Testing Reactor fuel type, and uses 20 wt% enriched uranium. Both the U_3O_8-Al and U_3Si_2-Al fuel are well qualified for reactor operation up to an average burn-up of 45 %. The U_3Si_2 fuel has been used in the IEA-R1 reactor since 1999, with good performance. The Institution is capable of fabricating U_3Si_2 enriched powder, allowing the domestic fabrication cycle of the dispersion fuel for research reactors, completing the fuel cell cycle from mining to the pellet/plate production. Therefore, Brazil is totally independent in materials and technology to fabricate nuclear fuels for its own research and test reactors. This achievement has placed our country among a few commercial manufacturer countries of LEU uranium for fuel elements of nuclear research reactors. In the last years, IPEN-CNEN/SP, besides its own fabrication needs, is fully accomplished to develop U-Mo and other new nuclear alloys for the next generation of fuel elements.

In the triennium considered, the IEA-R1 research reactor has been operated most of the time at a power of 3.5 MW and operation schedule of 62 hours per week. Other activities were also carried out to extend the lifetime of the reactor, to improve the conditions to comply with the user needs and to allow operation at higher powers.

Uranium-Molybdenum technology

For the last 30 years high uranium density dispersion fuels have been developed in order to accomplish the low enrichment goals of the Reduced Enrichment for Research and Test Reactors (RERTR) Program. Gamma U-Mo alloys, particularly with 7 to 10 wt% Mo, as a fuel phase dispersed in aluminum matrix, have shown good results concerning its performance under irradiation tests. That's why this fissile phase is considered to be used in the nuclear fuel of the Brazilian Multipurpose Reactor (RMB), currently being designed. For that reason efforts are under way at IPEN-CNEN/SP for developing the fabrication technology of this new fuel.

Gamma U-Mo alloys have been for long considered as fuel phase in research and test reactors using dispersion fuel in aluminum matrix. Promising results concerning performance under irradiation tests of U-Mo alloys, especially with molybdenum content ranging from 6 to 10 wt.% Mo, have encouraged IPEN-CNEN/SP to consider this fuel phase for the second stage of the RMB reactor operation, since uranium silicide compound, U_3Si_2, already produced, will be used at first.

The main challenge is the powder production from these ductile alloys. As a fuel based on dispersion concept U-Mo alloys must be used in powder form. At least three main fabrication routes for U-Mo powders could be listed: atomization (mainly centrifugal atomization by rotating disk method or even rotating electrode process, mechanical comminution, i.e., machining or grinding, and chemical comminution, i.e. hydride-dehydride process, also known by its acronym HDH. HDH of gamma U-Mo alloys can be accomplished by heating the alloy at temperatures where it decomposes in two phases, i.e. alpha and gamma' (U_2Mo). Since alpha uranium is easily hydrated, generating a very fine powder, its content must be carefully controlled (changing soaking time) in order to obtain particles within the desired size range. A variation of the last route, named HMD, combines hydriding-dehydriding with milling process. According to the authors, gamma U-Mo alloys can form a U-Mo hydride (A-15 structure) that embrittles the alloy, but not intensively, so powder could only be produced by interposing a milling operation, before dehydriding.

Each one of the previous routes are held by commercial or potential suppliers based on features like particle size range yield of the powder, costs, and the more important one, irradiation behavior. Since there are some controversial arguments, an investigation at local process conditions is necessary to find the best technological solution.

The focus of the preliminary work was to compare the characteristics of U-Mo alloy powders (10 wt% Mo) fabricated by two routes: mechanical grinding and HMDH. In this particular case, HMDH stands for hydrogenation-milling-dehydrogenation, since a hydride phase was not formed. It was reported before that gamma U-Mo alloys (particularly U-10wt% Mo) suffered a loss of ductility when submitted to a hydrogen atmosphere, by incorporating hydrogen interstitially. This fact was used to provide enough brittleness to the alloy to allow comminution.

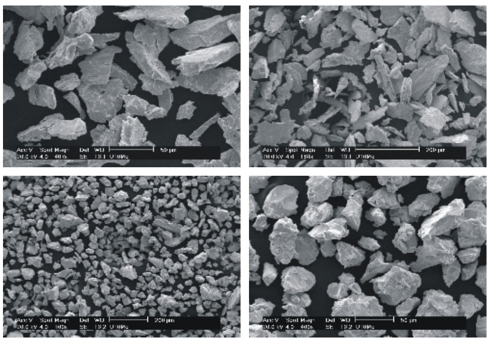
Ingots of U-Mo alloy with 10 wt% Mo were induction melted into a magnesia-stabilized zirconia crucible. Metallic uranium and metallic molybdenum were used as raw materials. Metallic uranium was home produced through magnesiothermic reduction and the metallic molybdenum was supplied with 99.95% purity, as small cylinders with 3 mm in diameter and 3 mm in height. Both materials were charged inside the zirconia crucible and heated by induction under high-purity argon atmosphere up to melting. Melting temperature was maintained for 3 minutes providing homogenization, than the furnace was turned off allowing the alloy to be solidified inside the crucible. The solid material was a cylindrical piece with near 40 mm in diameter and 50 mm in height, weighting around 1200 g with a density of 16,87 g/cm^3. The ingot was treated at 1000^oC for 72 hours under pure argon and quickly cooled for retention of the gamma phase. It was cut in pieces for studying the two routes for powder preparation, namely mechanical grinding (MG) and hydrogenation-milling-dehydrogenation (HMDH).

For the mechanical grinding route, the powder was produced by using high-speed grinding (15000 rpm) with diamond abrasive wheel. The abrasive wheel was 4 mm in diameter; having impregnated diamond particles with mean diameter of about 100 µm. Grinding was accomplished inside a glove-box under protective argon atmosphere.

For the HMDH route, small pieces were taken from the U-Mo ingot (with approximately 10x50x5 mm in thickness) and were individually heated at 400^oC for 3 hours under high purity hydrogen (99.9999%) at 3 bar. A Sievert type apparatus was used and no measurable hydrogen intake could be noticed, with the pressure gauge used (precision of 0.5 bar). At this temperature and time, alpha phase is not supposed to be formed, since about 40 hours would be necessary to start the gamma decomposition according to published TTT diagram. Next the pieces were manually crushed in a stainless steel mortar. The resulting granules were 3 mm in length. For comparison, crushing of pieces not hydrogenated treated was carried out but not succeeded due to ductile behavior of the parts. This was taken as an indication that some hydrogen intake must be occurred in former pieces. U-Mo granules were milled in a planetary ball mill at 400 rpm for 10 hours, with ball-to-powder weight ratio of 20:1. The vial and the balls were made from hardened steel. Loading and opening of the vial occurred inside a glove-box with protective argon atmosphere. After milling the powder was heat treated under vacuum at 400°C for dehydrogenization.

Figure 1 shows the morphology of the powders prepared by both routes, mechanical grinding (MG) and hydrogenation-milling-dehydrogenation (HMDH). It was observed that the powder prepared by MG route (left column in Fig. 1) presents particles with acicular and flake shapes, while the particles from HMDH route (right column in Fig. 1) are more regular and equiaxial. The powder prepared by MG route presented particles sensibly larger than the ones prepared by the HMDH route. The mean particle size (50 wt%) was about 100 μm for MG powder and 50 μm for HMDH powder. Furthermore HMDH powder particles fit very well the size requirements of dispersion fuels, with practically 100 wt% below 150 μm and about 30 wt% below 45 μm, while MG powder particles were shown to be larger, with more than 20 wt% above 150 μm. Other important difference was the aspect ratio of the particles (maximum to minimum Ferret's diameters). The aspect ratio reached 10 for the MG powder, much higher than the maximum ratio measured for HMDH powder, close to 4.

The partial results indicate the technical feasibility for producing powders from both investigated routes. The control of variables of the MG route, such as the size of the diamond granules of the abrasive wheel, the pressure of the tool under the alloy surface and the rotation of the grinding wheel machine, should promote the necessary adjustment in the particle size distribution. Powder produced by the HMDH present a particle size distribution compatible to be used as a dispersion fuel. Further work is necessary to increase the yields in order to evaluate both process routes as real technological alternatives for nuclear fuel powder production to research reactors. This project has an important cooperation of the staff of Materials Science and Technology Center of IPEN-CNEN/SP. This project was partially financed supported by FAPESP



Mechanical Grinding (MG) Hydrogenation-Milling-Dehydrogenation (HMDH)

Figure 1. Scanning electron micrographs of powder particles from both investigated routes

Advances in dispersion fuel fabrication technology

The use of radioisotopes in medicine is certainly one of the most important social uses of nuclear energy and IPEN-CNEN/SP has a special place in the history of nuclear medicine in Brazil. The CNEN institutes are the only officially allowed producers of radioisotopes and radiopharmaceuticals for use in nuclear medicine in Brazil. The production of IPEN-CNEN/SP represents nearly 98% of the total produced. Due to the serious international crisis in the supply of radioisotopes, Brazil has decided to build a new nuclear research reactor, the Brazilian Multipurpose Research Reactor - RMB, in order to ensure the delivery of radioisotopes to the Brazilian market.

Since 1988, IPEN-CNEN/SP has fabricated the fuel for the IEA-R1 research reactor. The current fuel produced at IPEN today allows the incorporation of 3gU/cm^3 by using the uranium silicide (U_3Si_2) technology. This concentration is sufficient to operate the research reactor IEA-R1 running at the power up to 5 MW. However, this level of uranium concentration is not enough for the efficient supply of reactors with higher powers and, therefore, higher neutron fluxes, as the Brazilian Multipurpose Reactor - RMB. Another difficulty inherent to low uranium concentration fuel is the generation of higher quantities of burned fuel. This is due to the low operation life of the fuel with low concentrations of uranium, requiring its frequent replacement.

Based on previous experience by IPEN-CNEN/SP in developing and manufacturing the silicide dispersion fuel, it was decided that the new reactor RMB will use the same type of fuel that is used in the IEA-R1 research reactor, with an increase in uranium concentration from 3.0 to 4.8 g/cm^3 using silicide technology. Developing activities were started to promote an adjustment of the current manufacturing procedures, allowing the incorporation of higher concentrations of uranium. The objective was the increase in the uranium concentration into the fuel from up to 4.8 gU/cm^3 using the U_3Si_2, and 3.2gU/cm^3 using the U_3O_8-Al. These concentrations are the maximum possible to reach if adopting the dispersion technology.

From the important parameters to qualify a fuel plate, it was found that meat length and width of all the produced fuel plates met the specification. However, difficulties arose with regard to the quality of homogeneity in the uranium distribution inside the fuel meat, bonding quality between the meat and the cladding and cladding and meat thicknesses. The initial fabrication tests for high uranium concentration U_3Si_2-Al plates (4.8 gU/cm^3) showed lower values for the cladding thickness as 0.33 mm, which resulted in fuel plates rejection. Another problem found refers to the uranium segregation of mixture towards the underside of pressed briquette. Especially in the case of the U_3O_8-Al dispersion, it was observed that during loading of the pressing die towards the cavity with powder, the nuclear material, which is more dense, tended to segregate at the bottom of the cavity, causing a localized increase in the uranium concentration and exceeding the maximum value of 45% by volume for the nuclear phase. In this case, the aluminum matrix ceased to act as a continuous dispersion, severely hindering the welding between the meat and cladding. Finally, for both fuel studied U_3Si_2 or U_3O_8, when the volume fraction of the nuclear compound was elevated to its maximum, there were difficulties with respect to homogeneity of the uranium distribution inside the fuel meat.

The maximum particle size of the U_3Si_2 powder was lowered from 150 to 125 µm. This change solved the problem regarding the penetration of U_3Si_2 particles into the cladding. The segregation problems observed previously were solved by changing the particle size of aluminum powder used in the manufacture of fuel meats. It was used a finer powder with smaller maximum size (45 µm) and with a size distribution more open. The aluminum powder used previously had a maximum particle size of 150 µm.

Miniplates with high uranium concentration were prepared to be irradiated at the IEA-R1 reactor of IPEN. It was decided to include in the irradiation program 3 miniplates fabricated with U_3Si_2-Al dispersion (4.8 gU/cm^3) and 3 miniplates fabricated with U_3O_8-Al dispersion (3.2 gU/cm^3). Four miniplates, fabricated according to the usual manufacturing technology adopted at IPEN for the routine fabrication of fuel for the reactor IEA-R1, were also included. This technology was extensively proven as feasibly once there were irradiated several fuel plates in IEA-R1 reactor up to 50 wt% ^{235}U burn-up. These miniplates, two with U_3Si_2-Al 3.0gU/cm^3 and two with U_3O_8-Al 2.3 gU/cm^3, will serve as a benchmark for assessing the irradiation performance of high uranium concentration miniplates. The miniplates were fabricated and qualified under the same procedures adopted to produce normal size fuel plates. All miniplates have been approved for operation in reactor. Figure 2 shows photographs of miniplates to be irradiated. The start of the irradiation is planned to be on June 2011.

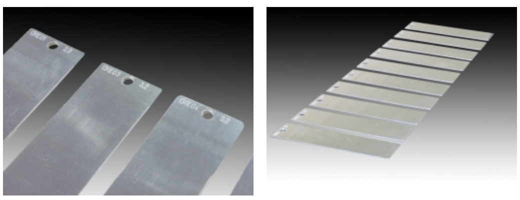


Figure 2. Photos illustrating the miniplates to be irradiated

Fabrication of targets for ^{99}Mo production by fission

As part of the Brazilian Multipurpose Reactor Project, a project aiming at developing the fabrication of targets based on UAl_x-Al dispersion technology started. The targets when irradiated produce mainly the pair 99Mo-99mTc. Brazil has a great demand for this product, which is imported. Currently, the Brazilian nuclear medicine suffers a impact due to the limited availability of these radionuclides in the foreign market. IPEN-CNEN/SP has not yet developed the technique for manufacturing this kind of targets. There are, nowadays, some techniques used worldwide. As a first step, the technique being developed at IPEN is based mainly on metallurgical production of UAl_x powders and UAl_x-Al dispersion miniplates, following the production procedures basically adopted for manufacturing of fuel elements.

UAl_2/UAl_3 powder was fabricated by melting in induction furnace. A slightly hypostoichiometric composition in relation to the nominal composition of UAl_2 was used. The XRD pattern of the powder produced confirmed the predominant presence of the UAl_2 phase, with small amount of UAl_3. However, it was detected the presence of impurities which have not yet been identified. Those peaks are indicated in the diffractogram shown in Figure 3. More studies are needed to identify this contaminant in order to eliminate it.

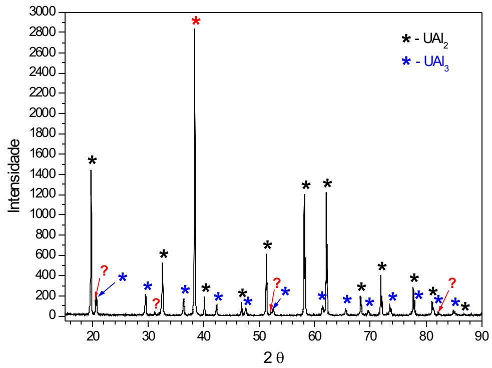


Figure 3. XRD pattern of UAl_x powder produced

The grinding of the material was done manually according the method usually adopted in manufacturing fuels by IPEN-CNEN/SP. The grinding operation was very difficult due to the ductile nature of the material, which was imposed by the presence of the ductile UAl_3. Despite the difficulty, it was possible to manually get enough powder to manufacture prototypes of the first targets, which were used for testing the dissolution and extraction of Mo99.

Eight prototype UAl_x-Al targets with uranium density of 2.8gU/cm^3 were fabricated. The x-value in this intermetallic is close to 3. This amount is the maximum density of uranium that can be incorporated by using the dispersion technology, which corresponds to 45% by volume of the nuclear phase.

At this stage, it was concluded the feasibility of manufacturing UAl_x-Al targets with uranium concentration of $2.8gU/cm^3$. Figure 4 shows the targets fabricated. These are just prototypes manufactured for testing dissolution. The development of the final specification for the target is necessary to continue the work. Additional fabrication tests will be necessary, with a view to meeting the specification that is still under development.

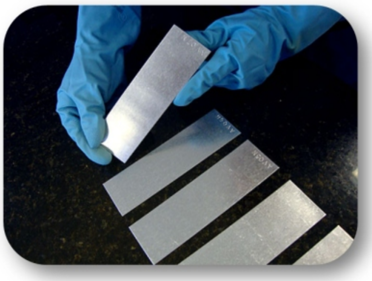


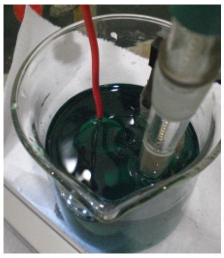
Figure 4. Photograph illustrating the targets UAl_x Al-manufactured

The after-irradiation processing of ^{99}Mo-^{99m}Tc targets could be either done by alkaline or acid route. The alkaline route is used for UAl_x-target, but it penalizes deeply the rejection procedure because it keeps great amounts of highly corrosive liquid radioactive rejection. On the other hand, the acid dissolution route is much more adequate in terms of rejection. This acid route is utilized when metallic uranium, not alloyed with aluminum, is used as target. Normally, the uranium comes involved in nickel covering, which could be removed and separated from ^{99}Mo easier.

There is an IAEA/CPR project planning to use LEU uranium foil (135µm) involved in nickel thin foil (15µm) inserted in an aluminum tube for irradiation. The target fabrication group of CCN/IPEN decided to study other possibilities to produce uranium targets involving electrochemistry. One of the routes could be the compacting of uranium powder and recovering it by electroplating of nickel. Some simulations using iron-powder compacts have been made and are shown in Figure 5. So, LEU metallic uranium powder could be produced by hydrating and then compacted to small coins and then be plated with nickel, using chemical pickling and electrochemistry techniques.

Other possibility is to use small chips of metallic uranium recovered with nickel. The cutting could be made in a cut-off into thin disks with less than half millimeter of uranium. The range of disk diameter could be between 10-30mm. This could be made by straightforward routine in IPEN-CNEN/SP. An example of the metallic uranium chip is shown in Figure 6. Then, the U-chip receives an electroplated layer of nickel and placed inside an aluminum case to be irradiated. The material is set inside the case in such a way it could be removed easily during post-processing manufacture of ^{99}Mo-^{99m}Tc targets following the acid route.

The main purpose of the electroplating is to avoid all possible contact of uranium with aluminum, in order to prevent promptly reactions resulting in U-Al intermetallic and, so, involving alkaline post-irradiation dissolution. Since all nuclear reactions produce radioactive gases, mainly Xe, it is also interesting to have this radioactive product confined inside small volume, which is met by using the nickel electroplated layer over the U-target, helping the fabrication handling inside hot cells and diminishing the hazards occurrences.



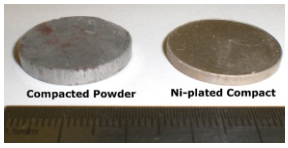


Figure 5. Nickel Electrodeposition over compacted briquette (simulation with 400µm iron powder)

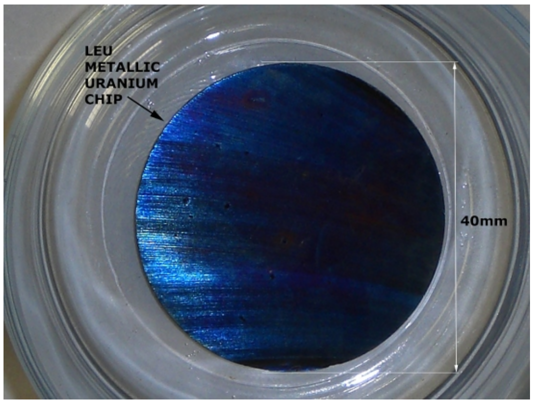


Figure 6. LEU metallic uranium chip (1 mm thick) ready to receive Ni-platting

Treatment of effluent generated by the precipitation of uranium tetrafluoride

The uranium metal used as raw material for uranium silicide production (U_3Si_2) is obtained from the reduction of uranium tetrafluoride (UF_4). There are basically two pathways for the UF_4 production: the first is called the dry route where the UO_2 is treated at 400°C in a resistive furnace under hydrogen and fluoride atmosphere (H_2 and hydrofluoric acid HF); the second is known as the wet route, where the UF_4 is precipitated in hydrofluoric acid using a reducing agent: stannous chloride II ($SnCl_2$). The uranium source for the UF_4 precipitation is uranyl fluoride (UO_2F_2), which can be obtained by the hydrolysis of uranium hexafluoride - UF_6 or from the dissolution of uranium trioxide - UO_3 with HF. In both cases, the

After the precipitation the UF_4 is filtered and washed. The washing solution is collected and used in the UF_6 hydrolysis or to dissolve of UO_3 in the next batch. The filtrate is rich in fluoride ion (F^-), which is treated with calcium oxide (CaO) for the precipitation of calcium fluoride (CaF_2). When the UF_4 is got by hydrolysis of UF_6, the average concentration of F^- ion in the filtrate is 100 g/L. But when the UF_4 is precipitated from the dissolution of UO_3, the average F^- concentration is at the level of 180 g/L. The diffractogram is shown in Figure 7.

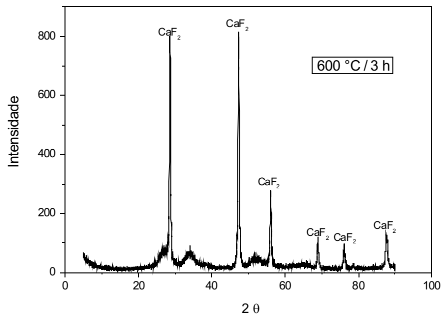


Figure 7. The diffraction diagram of obtained CaF_2

Three main parameters have been studied for the precipitation of CaF_2: initial fluoride concentration, precipitation temperature and final pH of precipitation. The initial fluoride concentration should be between 40 and 50 g/L, because the precipitate decants at this the point. With this initial concentration the filtration is better made. The best temperature was founded to be 90°C and the optimal pH was between 2.5 and 3.0. The combination of temperature and pH produced a filtrate with the fluoride concentration between 2.0 to 3.0 g/ml. The following up step is to calcinate CaF_2 at 600°C for 3 hours.

Computational simulation on reduction process of UF_4 to metallic uranium

The production of metallic uranium is essential for production of fuel elements for using in nuclear reactors manufacturing of radioisotopes and radiopharmaceuticals. In IPEN, metallic uranium is produced by magnesiothermic reduction of UF_4. This reaction is performed in a closed graphite crucible inserted in a sealed metal reactor with no contact with the outside environment. The set is gradually heated in an oven pit, until it reaches the ignition temperature of the reaction (between 600-650°C).

The modeling of the heating profile of the system can be made using simulation programs by finite element method. Through the thermal profiles in the load, we can have a notion of heating period required for the reaction to occur, allowing the identification of greater or smaller yield in metallic uranium production just by recognizing the level of residence time to promote the ignition of the system. Normally, longer time for reaction is accompanied with higher temperature isotherms curves and more homogeneous thermal conditions, but with worse metallic yield. The modeling results were compared to routine production and revealed pertinent in helping the operation routine towards achieving better metallic yield for uranium.

Design and construction of a new fuel fabrication plant

Between 1985 and 1988, IPEN-CNEN/SP worked aiming at assembling a small fuel element fabrication plant with production capacity of only 6 fuel element a year, for demonstration purpose, but produced the necessary quantity to supply its IEA-R1 research reactor that operated at this time in power level of 2 MW with a of regime of 40 hours a week. In August 1988, IPEN-CNEN/SP supplied the IEA-R1 reactor with the first national fuel element. Ever since, IPEN-CNEN/SP began a continuous fuel production, which continues up today.

During 1997, IPEN-CNEN/SP raised the fuel production capacity of the small plant from 6 to 10 fuel element a year, representing the maximum production capacity in laboratorial scale for the available facilities at that time, which was enough to keep IEA-R1 reactor operating at 3-4MW power level in 64 hours regime a week. Due to the emergent increase in radiopharmaceuticals products demand and the consequent increase in the IEA-R1 power, the reactor needed an increasing number of fuel elements to operate properly at 5MW and 120 hours a week. The number should be raised from 6 (U_3O_8-Al) to 18 elements (U_3Si_2-Al) a year. In addition, a new reactor to produce radioisotopes will be built (Multipurpose Reactor Brazilian) and aims to make the country independent in the production of radioactive isotopes for medicine, material testing reactors for energy production and possession of a neutron beam line for scientific use. The new reactor (20MW) would consume about 50 yearly U_3Si_2-Al fuel elements. The RMB is a project that will contribute decisively to the country's strategic objectives for increasing the production of radioisotopes for medical applications among others. The ^{99}Mo and ^{131}I are the main radioisotopes for application in health and will be a priority in the reactor irradiation. The ^{99}Mo and ^{131}I will be produced from targets irradiated in the reactor core consisting of miniature fuel plates. The production of ^{99}Mo and ^{131}I require weekly, with the reactor operating without interruption. It is estimated a consumption of 20 uranium targets per week, which equals 1000Ci/week.

Based on this forecast demand for fuel elements and uranium targets, a project in IPEN-CNEN/SP began in 2001 in order to adapt the production facilities seeking on improving the production capacity. The first phase of that project is now concluded (Figure 8). The fabrication step for dispersion preparation is made by: cermet core pressing, fuel plate rolling, fuel element assembling and all fuel qualification steps are properly built in the new fabrication plant. The second phase of that project is still going on. Once ready, the whole project will be constituted of an integrated production line using in industrial level facilities, which will operate according to international nuclear quality and safety standards, established by IAEA. The new facility is planned to have nominal capacity for producing 50 fuel elements yearly. That will attend entirely the fuel element demand in 4-5 years. The production capacity of the new facility could be incremented to reach 80 fuel elements in a year, to full supply the new multipurpose reactor which has been planned to be constructed. The conclusion of the physical project and equipment installation is foreseen to happen by 2012-2013.

Figure 8. Partial view- concluded parts of the new CCN plant for fuel element fabrication

Fuel element fabrication

The Nuclear and Energy Research Institute IPEN-CNEN/SP has worked in the area of Nuclear Fuel Cycle practically since its founding in 1956. In the 60s, a golden age in the area of manufacturing technology of nuclear fuel began at the Division of Nuclear Metallurgy. It started, at this period, the development of dispersion based fuel, with applications in pool type research reactors. This fuel type was assembled from fuel plates containing uranium compounds dispersed in aluminum. Between 1964 and 1965, the fuel elements for the Argonauta Reactor of Institute of Nuclear Engineering (IEN) were manufactured in this division of Nuclear and Energy Research Institute IPEN-CNEN/SP. The fuel used had a dispersion of U_3O_8-Al. The employed U_3O_8 powder was enriched to 20wt% of ^{235}U, imported from United States, through the program Atoms for Peace (UN/IEAE). Despite the relatively simple technological requirement to produce the fuel for the Argonauta reactor, it was a great start-up since this technological seed germinated strongly after 1980.

The development program for producing the fuel elements began in the mid 80s. It was motivated by "cold war" restraint purchases policy, which hindered the acquisition of fuel out of the country. The internal necessary fuel demand to promote the continuous operation of IPEN´s reactor IEA-R1 was certainly the most stimulating cause for developing skills to produce the fuel elements. In September 1988, the first fuel element produced at IPEN-CNEN/SP, were placed in our reactor. Since then, IPEN-CNEN/SP has made its own fuel elements based on plate type, initially using U_3O_8-Al dispersions in density of 2.3 gU/cm^3.

Due to the increasing demand of the radioisotopes production in Brazil, the reactor IEA-R1 increased its power from the original 2MW to 5MW. Also, the reactor routine became ready to increase up to 120 operating hours per week, requesting other fuel material types to bring more U-density to the fuel meat. To carry out this increasing in fuel meat density, a new fuel element was developed based on U_3Si_2-Al dispersion with a density of 3.0 g/cm^3, which called for a gradual change inside the IEA-R1 from U_3O_8-Al towards Al-U_3Si_2.

The intermetallic U_3Si_2 fuel was imported until 2002. Presently, it is fully produced in IPEN having the conversion chemical cycle included using enriched LEU UF_6 supplied by Navy Technology Center (CTMSP). Currently the Nuclear Fuel Center is also manufacturer of powdered U_3Si_2 passing through producing UF_4, metallic uranium and then the silicide by induction smelting. The entire fuel cycle is actually nationalized.

The fuel element, which is manufactured by IPEN, is dispersion MTR-type enriched to 20 wt% ^{235}U. The two types of dispersion are well qualified for operation up to 30% burns, on average. However, practice has shown a performance superior to that amount, leading to 50% burn-ups in many fuel elements, without any nuclear hazard.

Brazil became totally independent in terms of materials and technology to fabricate its fuel elements for their research reactors in order to produce radioisotopes. This was an important technological know-how acquired by our research institute. It places our country inside the international market, as part of a restrict group of commercial suppliers of this kind of fuel. Besides, the Al-dispersed fuel technology was established as a developing basis to further going on to reach advanced fuel material and projects for higher performance nuclear research reactors, such as our future RMB reactor.

Up to now, IPEN has manufactured 63 fuel elements of U_3O_8 (26 having meat density of 1.9 to gU/cm^3 and 37 having 2.3 gU/cm^3 and 29 fuel elements with U_3Si_2 having a concentration of 3.0 gU/cm^3. So, a total of 92 fuel elements were produced to date.

From 2010 onwards, it was installed in the reactor a new "instrumented element" system, developed by IPEN aiming at providing information and data parameters to promote better studies on thermohydraulics during operation of IEA-R1 reactor. The new device had its first operation at a power of 3.5MW in February 2010 and has produced ever since important experimental data.

This device had special difficulties to fabricate fuel plates with thermocouples attached to its surface. The IEA-R1 reactor, instrumented in this way, became a world reference for many thermohydraulic and accident analysis codes. Figure 9 shows the instrumented element (EC 208) fabricated.

Figure 9. Instrumented Element (EC 208)

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, support the demand of Nuclear Fuel Cycle Program providing chemical characterization of uranium compounds and other related materials.

Among these, one can highlight the determination of uranium content in U_3Si_2, UF_4, U_3Si_2-Al compounds and its impurities (metals and rare earths elements). 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 methodology as used at NBL (New Brunswick Laboratory) adding an improvement on lowering the mass used in the 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 of 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 are 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. Among then environmental sampling based on swipe samples for the identification of uranium and plutonium in nuclear facilities.

Whenever possible, the laboratories of Chemistry and Environmental Diagnosis Center are engaged to update or develop new methods to become the activities greener as currently expected.

Reactor physics benchmarks at the IPEN/MB-01 reactor

Since 2004, the reactor physics group of the Nuclear Engineering Center of IPEN is participating on two international programs for the elaboration of benchmarks 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, EUA) 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 the ICSBEP is to: 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 the experimenters or individuals who are familiar with the experimenters or the experimental facility, b) Evaluate the data and quantify overall uncertainties through various types of sensitivity analysis, 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 the ICSBEP group is documented as an International Handbook of Evaluated Criticality Safety Benchmark Experiments. Currently, the handbook spans over 42,000 pages and contains 464 evaluations representing 4092 critical, near-critical, or subcritical configurations, 21 criticality alarm placement/shielding configurations with multiple dose points for each, and 46 configurations that have been categorized as fundamental physics measurements that are relevant to criticality safety applications. The International Reactor Physics Benchmark Experiments (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 are being performed at the IPEN/MB-01 research reactor facility. During the last three years, several experiments have been designed, executed and analyzed at the IPEN/MB-01 Reactor. They were documented in a proper format and submitted to the working groups. More than 100 critical configurations have been approved to be included in the ICSBEP DVD handbook. From these experiments, we can mention the critical configurations with borated stainless steel used in the storage pool of ANGRA-I and II to save storage space. Another very interesting experiment was a central void simulation with a 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) we complete a series of benchmark experimental problems on the isothermal reactivity coefficient of light water reactors. This experiment was very important to give support to the nuclear data evaluation of ^{235}U in the thermal energy region of the neutron. This was a long standing problem in the reactor physics area and recently it was made very good progress in the C/E comparisons of several integral parameters of interest in the physics of reactors. Another experiment complete at the end of 2008 was the measurement of several effective delayed neutron parameters without the need of any correction factor or data from other experiments. More recently the reactor physics area of IPEN added more three contributions to the IRPhE project; the reaction rate experiments, the fission density and the power distributions of the IPEN/MB-01 core. The experiments were approved to be included in the handbook of reactor physics benchmarks issued in March 2009 and March 2010. This project is being supported by FAPESP and CNPq.

Development of a reactivity meter at the IPEN/MB-01 reactor

The development of a reactivity meter at IPEN was performed to fulfill the needs of the IPEN/MB-01 reactor researchers. Due to the good results presented by the reactivity meter, a preliminary test was done using the data of ANGRA 1 unit in the course of the start up P9 in 2001. These data (voltage signals of neutron detectors, temperature signals and bank position signals) were acquired by the Framatome staff and analyzed at IPEN by the staff of the IPEN/MB-01 reactor. The results were in good agreement with those obtained by Framatome.

Since then, the reactivity meter was tested in all start up tests of ANGRA 1 in parallel to the Framatome acquisition and analyses. In these tests, however, all the signals of interest were acquired independently and from different sensors and/or acquisition points. In all start up tests, the results were totally consistent to the Framatome ones.

As an example, Table 1 shows the comparison of results for the reactivity of the control banks and the isothermal reactivity coefficient for the ANGRA 1 P14 start up tests performed in 2007.

Parameter	Calculated Value	Framatome Measurement	IPEN Measurement
Isothermal Temperature Coefficient	-13,87 pcm/°C	-14,405 pcm/°C (heating) -14,391 pcm/°C (cooling) -14,398 pcm/°C (average) deviation = -0,528 pcm/°C	-14,208 pcm/°C (heating) -14,409 pcm/°C (cooling) -14,301 pcm/°C (average) deviation = -0,431 pcm/°C
Bank D	821 pcm	832,90 pcm (-1,43 %)	843,02 pcm (-2,61 %)
Bank C	937 pcm	974,25 pcm (-3,82 %)	957,40 pcm (-2,13 %)
Bank B	1040 pcm	1097,46 pcm (-5,24 %)	1045,78 pcm (-0,55 %)
Bank A	434 pcm	479,43 pcm (-9,50 %)	461,88 pcm (-6,04 %)
The deviation of the bank reactivities is given by: $\sigma = [(VC - VM)/VM]*100$ where VC is the calculated value and VM is the measured value			

After the good results obtained in all these start up tests, the researchers of IPEN performed the ANGRA 1-P15A , 1-P16, 1-P17 start up tests in 2008, 2009, and 2010 respectively, without the Framatome staff nor supervision of any kind. The results obtained were again totally consistent with the calculated ones and the tests were approved. It is important to mention that the reactivity meter has also been used in the IEA - R1 reactor at IPEN for almost 6 years, providing a great economy of working time, and that the reactor physics staff of ANGRA 2 unit has interest to install the reactivity meter in this power plant.

Software for medical image based PHANTOM MODELING

Latest treatment planning systems depends strongly on CT images, so the tendency is that the dosimetry procedures in nuclear medicine therapy be also based on images, such as magnetic resonance imaging (MRI) or computed tomography (CT), to extract anatomical and histological information, as well as, functional imaging or activities map as PET or SPECT. This information associated with the simulation of radiation transport software is used to estimate internal dose in patients undergoing treatment in nuclear medicine.

This work aims to re-engineer the software SCMS, which is an interface software between the Monte Carlo code MCNP, and the medical images, that carry information from the patient in treatment. In other words, the necessary information contained in the images are interpreted and presented in a specific format to the Monte Carlo MCNP code to perform the simulation of radiation transport. Therefore, the user does not need to understand complex process of inputting data on MCNP, as the SCMS is responsible for automatically constructing anatomical data from the patient, as well as the radioactive source data.

The SCMS was originally developed in Fortran-77. In this work it was rewritten in an object-oriented language (JAVA). New features and data options have also been incorporated into the software. Thus, the new software has a number of improvements, such as intuitive GUI and a menu for the selection of the energy spectra correspondent to a specific radioisotope stored in a XML data bank. The new version called AMIGO also supports new materials and the user can specify an image region of interest for the calculation of absorbed dose. The main window of the AMIGO software presented as a combo box to set the image and simulation parameters by the user. Several parameters and options are already predefined with default values, but the user can modify them by introducing new parameters or configurations according to each problem and the last modifications will be saved to the next program execution. Several tests have been done successfully using a diversity of examples of anatomic and functional virtual images. The AMIGO software is part of the development of an integrated computational dosimetric system which is being developed at IPEN for applications in nuclear medicine and radiotherapy.

Energy degrading and scattering plates for electron beam radiotherapy for skin diseases in small rooms

There are many radiosensitive epidermotropics diseases such as mycosis fungoids and the syndrome of Sézary, coetaneous neoplasics originated from type T lymphocytes. Several studies indicate the eradication of the disease when treated with linear accelerators emitting electron beams with energies between 4 to 10 MeV. In this project we developed a customized single-field electron beam for total skin therapy (TSET) and local irradiations in small treatment rooms with maximum source-surface distance (SSD) of 2.95 meters, using energy degrading and scattering plates.

The energy spectrum and geometric distribution of the 6 MeV electron beam of the VARIAN 2100C accelerator were reconstructed through Monte Carlo simulations, using the MCNP5 Monte Carlo code and based on experimental data. The simulated source was then utilized to test several materials and geometry for scatter and degrader plates, producing scattered field suitable and adequate for local irradiations and whole body TSET procedure. The simulation results were validated by experimental measurements with thermoluminescent dosimeters (TLD), radiochromic films and ionization chambers. Figure 1 shows the experimental configuration for dose distribution measurements at the Hospital das Clínicas de São Paulo, HC-FMUSP/SP.

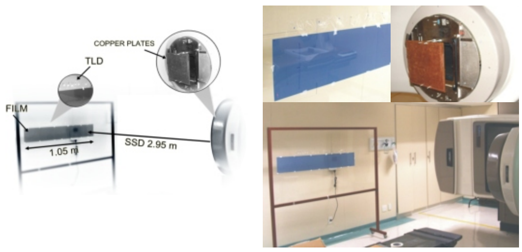


Figure 1. Experimental configuration for TSET dose distribution measurements using TLD and radiochromic films

NSECT applied to the assessment of calcium deposition due to the presence of microcalcifications associated with breast cancer

Breast cancer is the second most common cancer worldwide and the leading cause of death among women in Brazil. According to estimates for the year 2010 it is expected approximately 49,000 new diagnosed cases. One of the main signs of breast cancer at an early diagnosis is the development of microcalcifications. Because of calcium radiological properties, microcalcifications are associated with non-palpable lesions that can be visualized on mammography, which makes it the primary mode of breast cancer diagnosis. The importance of the detection of microcalcification formations in their early stages is well-known fact and according to the literature, the survival rate of patients who developed breast cancer is inversely proportional to the lesion size.

In recent years, a new technique for in vivo spectrographic imaging of stable isotopes was presented as Neutron Stimulated Emission Computed Tomography (NSECT). In this technique, using multiple projections, a fast neutron beam interacts with the stable isotopes of the irradiated tissue, through inelastic scatterings, making them jump into an excited state. When they return to their ground state, they emit photons which energies are intrinsic to the emitting nuclei. The emitted gamma energy spectra can be used for two purposes: (a) reconstruction of the target tissue image and; (b) determination of the tissue elemental composition.

BNCT research facility experiments

BNCT (Boron Neutron Capture Therapy) is a binary cancer treatment therapy which stands on loading tumor cells with boron following their irradiation with thermal neutrons. As the 10B absorbs the neutron, two high LET particles are emitted (alpha and 7Li) releasing all their kinetic energies (around 2 MeV) in a cell size volume. Its success relies therefore on the capability and specificity of the boron carrier compounds to deliver boron to the target cells and also on tailoring adequate neutron beams to impinge on these cells. Although its idealization and first treatments on brain tumors dates for more than 60 years, it has been felling an increased interest due to good results observed in the treatment of other cancer types.

A BNCT research facility was projected and constructed in IEA-R1m reactor of IPEN-CNEN-SP. It has been used in the development of studies in Radiation Physics and Radiobiology such as: - radiation field (neutrons and gammas) characterization; - neutron beam tailoring through the determination of a moderators and filters configuration which adequate the radiation field to BNCT needs; - dose estimation and development of "in vitro" and "in vivo" biological studies. Work cooperation has also been established to evaluate BNCT efficiency regarding its dependence on cell type cultures and on the use of different boron compound agents.

Multi-modular integral pressurized water reactor control and operational reconfiguration for a flow control loop

The work focused on the International Reactor Innovative and Secure (IRIS) design since this will likely be one of the designs of choice for future deployment in the U.S. and developing countries. With a net 335 MWe output IRIS novel design falls in the "medium" size category and it is a potential candidate for the so called modular reactors, which may be appropriate for base load electricity generation, especially in regions with smaller electricity grids, but especially well suited for more specialized non-electrical energy applications, such as district heating and process steam for desalination. The first objective is to evaluate and quantify the performance of a nuclear power plant comprised of two IRIS reactor modules operating simultaneously with a common steam header, which in turn is connected to a single turbine, resulting in a steam-mixing control problem with respect to "load-following" scenarios, such as varying load during the day or reduced consumption during the weekend. To solve this problem a single-module IRIS SIMULINK model previously developed by another researcher is modified to include a second module and was used to quantify the responses from both modules. This twin-module plant is shown in Figure 2, and the control schematic of a twin-unit IRIS system, with helical coil steam generators connected by the feedwater line, is shown in Figure 3.

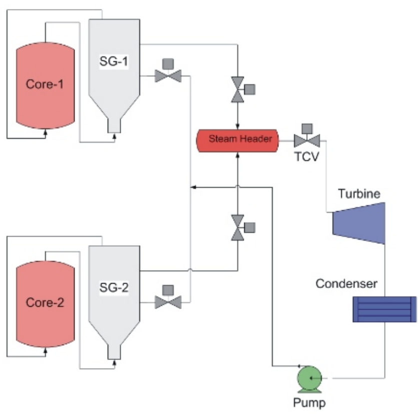


Figure 2. Schematic diagram of an IRIS-type multi-modular power block

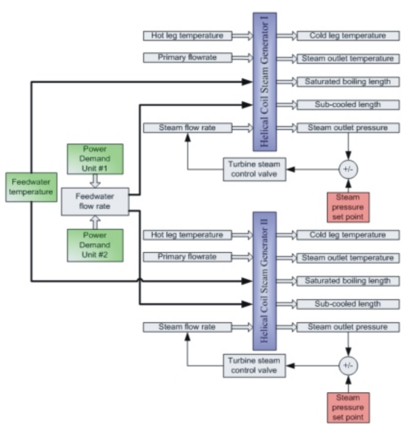


Figure 3. Schematic of a Twin-HCSG control system connected at the feedwater flow rate line

IRIS simulation results

From simulations it is possible to conclude that, for small variations in process variables, both units are somewhat independent of each other even with a common feedwater line connecting both steam generators, and are able to perform well even under such variations, with the real connection between the two units being located in the steam header. Simulations show the feasibility of having two IRIS modules in a single plant. An example of changes in steam temperature at each of the reactor modules and the temperature of the mixed steam in the common header are shown in Figure 4. At first, both steam temperatures are the same, and remained constant in the first unit while it varied in time in the second unit because of the changes in the power demand. As the power decreased, the area available for heat transfer in the steam generator increased, therefore increasing steam temperature, conversely decreasing following power increase. The control strategy of regulating the average reactor temperature and the steam pressure is robust for this load following operation.

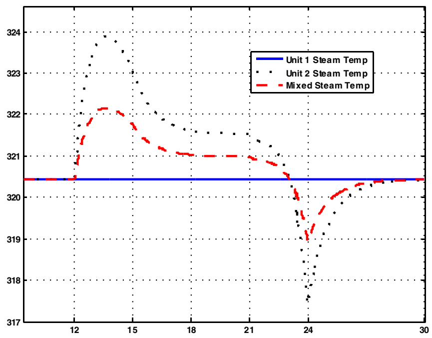


Figure 4. Units #1 & #2 and mixed steam header temperatures changes

In order to develop research related to instrumentation and control, and equipment and sensor monitoring, the second objective is to build a two-tank multivariate loop in the Nuclear Engineering Department at the University of Tennessee. This loop provides the framework necessary to investigate and test control strategies and fault detection in sensors, equipment and actuators. The third objective is to experimentally develop and demonstrate a fault-tolerant control strategy using this loop. Using six correlated variables in a single-tank configuration, five inferential models and one Auto-Associative Kernel Regression (AAKR) model were developed to detect faults in process sensors using Sequential Probability Ratio Test (SPRT). Once detected the faulty measurements were successfully substituted with prediction values, which would provide the necessary flexibility and time to find the source of discrepancy and resolve it, such as in an operating power plant. Finally, using the same empirical models, an actuator failure was simulated and once detected the control was automatically transferred and reconfigured from one tank to another, providing survivability to the system.

Flow control loop results

Actual measurements were successfully substituted with predictions and provided the system the necessary flexibility to keep on operating even under degraded conditions, thus offering survivability to the system and the time necessary to perform corrective procedures, should that be the necessary. These experiments were particularly important because they offered the opportunity to prove that a system like the multivariate loop can survive degraded circumstances, provided the empirical models used are accurate and representative of the system dynamics.

An experiment was performed by adding a drift to the level sensor measurements at a rate of +50 mmH_2O over 3 minutes. In this experiment, once the difference between measured and predicted values reached around 18 mmH_2O, the level SPRT triggered changing from normal to faulty condition causing measured values to be substituted with predicted values, therefore isolating the faulty sensor from the loop.

This work was funded by the US Department of Energy and developed in its entirety at the University of Tennessee in Knoxville, USA. It was presented in October of 2010 as a partial fulfillment of the requirements for the degree of Doctor of Philosophy, with a major in Nuclear Engineering. Graduate research and teaching assistantships were provided by the University of Tennessee Nuclear Engineering Department, under the supervision of UTNE professor Dr. Belle R. Upadhyaya.

Innovative and hybrid reactors

The renascence of Nuclear Energy is bringing new initiatives in the world such as GIF (Generation IV Initiative), and INPRO (International Project on Innovative Nuclear Reactors and Fuel Cycle). These new initiatives are looking for new reactors concepts and associated fuel cycles which take as principle the sustainability of Nuclear Energy for the next centuries. Among several concepts being considered, Fast Spectrum System is the focus of this research group, in particular subcritical reactors driven by external source of neutrons (e. g., spallation neutron source, fusion neutron source). These systems are being considered as dedicated nuclear transuranic burner reactors. In the last years, the calculation methodologies for simulation and analyses of these systems had been developed. We participate in the IAEA technical working group on fast reactors (TWG-FR) and in the Coordinated Research Project (CRP) on Analytical and Experimental benchmark analysis on Accelerator Driven Systems.

The main activities and resulted products during the year 2007-2010 in innovative hybrids reactors are:

- Participating of International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) on “Analytical and Experimental Benchmark Analyses of Accelerator Driven Systems”.The objective of this CRP is to improve the present understanding of the coupling of an external neutron source [e.g. a spallation source in the case of the accelerator driven system (ADS)] with a multiplicative sub-critical core;

- Participation in cooperative work established to study the feasibility to use Low Enriched Uranium in existing or planned facilities, under the umbrella of the main CRP and with support of DOE of the USA;

- Feasibility studies of the implementation of a compact neutron generator in the IPEN/MB-01 zero power reactor. A numerical calculation of IPEN/MB-01 driven by a compact neutron generator was performed to verify the feasibility of using Low Enriched Uranium in existing or planned facilities;

- Participation in the simulation of Reactor Physics Parameters of the sub-critical assembly YALINA-Booster at the Joint Institute for Power and Nuclear Research Sosny (JIPNR) of the National Academy of Sciences of Belarus. It was performed these calculation benchmark exercise and the results are being compared with the experimental results;

- Participation in Benchmark on Computer Simulation of radioactive nuclides production rate and Heat Generation Rate in a lead target exposed to 660 Me protons, was defined by the participants from Poland. The benchmark model is based on the earlier experiment done within the project SAD, in the Joint Institute of Nuclear Research in Dubna (Russia). In the experiment absolute activities of several long-lived radionuclide's, generated in lead target during its irradiation with 660 Me proton beam, were determined. Thus, the benchmark is oriented to compare simulation predictions, based on different available codes and physical models, with the experimental data;

- Implementation Monte Carlo burnup calculation of subcritical systems;

- Comparison between fusion hybrids subcritical reactors and accelerator driven subcritical reactors, in Figure 5 it is showed Gas Cooled Fast Subcritical Reactor (GCFR) driven by a Tokamak, and in Figure 6 it is showed a neutron flux distribution of these reactor.

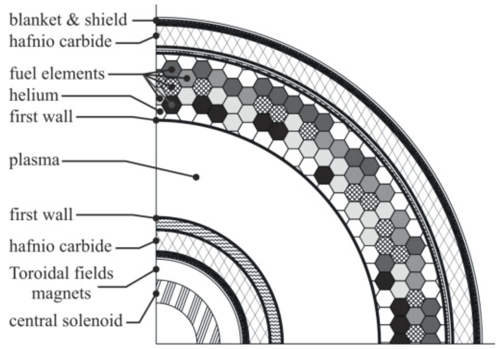


Figure 5. Gas Cooled Fast Subcritical Reactor driven by a Tokamak

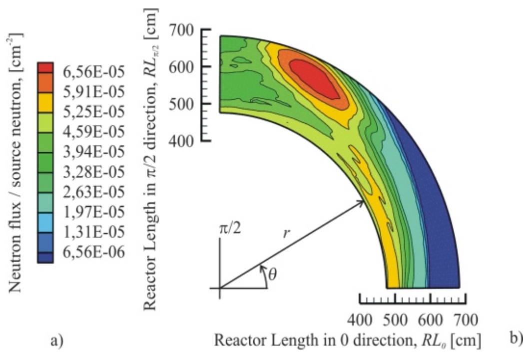


Figure 6. Neutron flux distribution of a Gas Cooled Fast Subcritical Reactor

Collaborative project INPRO/ENV “Environmental Impact Benchmarking Applicable for Nuclear Energy System under Normal Operation”

Since 2010 CEN/Ipen has participated in a IAEA study group of benchmarking of methodologies for the ranking of stressors of interest for given release and pathways scenario. This study presents a comparison of ranking methodologies and will be applied to different source terms. The specific research scopes:

- a benchmark on assessment methodology to rank radio-nuclides in terms of health related impact on human;
- a comparison of the most important radio-nuclides in terms of environmental impact for a given source term;
- the role of retained criteria to obtain this list and the relevance of sensitivity studies;
- reference scenarios for INPRO Assessment Methodology;
- feed back for review of the practical application of the INPRO methodology for environmental protection.

The participating countries are Belarus, Brazil, Czech Republic, France, India, Indonesia, Kazakhstan, Republic of Korea, Russia, Slovak Republic, Ukraine.

MTR instrumented fuel element for thermal-hydraulic analysis of research reactors

With the financial support of the IAEA and IPEN-CNEN/SP, the Nuclear Engineering Center designed an MTR Instrumented Fuel Element for the IEA-R1 research reactor. This Fuel Element was used in order to measure the temperatures on the lateral and central fuel plates, and also to compare them with the calculated temperatures. Thermal hydraulic computer codes such as PARET, COBRA, RELAP, etc were used for the calculations. The Instrumented Fuel Element, called IEA-208, was assembled by the Nuclear Fuel Center, Figures 7 and 8. Fourteen thermocouples were used to acquire the wall and fluid temperature at predefined positions in different cooling channels. IEA-R1 reactor has been operating since February 2011 with this element in different core positions.

Figure 7. Instrumented fuel element assembling

Figure 8. Instrumented fuel element

Natural circulation in nuclear reactors (theoretical and experimental)

One of the most serious problems for a nuclear reactor in accidental conditions is the residual heat removal from the fission products decay. The design of a nuclear reactor provides the establishment of natural circulation in the primary system as an alternative to cooling the reactor core. An experimental circuit was developed in order to understanding the complex phenomena involved in two phase flow natural circulation and also to obtain data for validating computer codes. These studies started at USP in the Departamento de Engenharia Química/Escola Politécnica. Experiments concerning single and two-phase flow natural circulation regimes were performed. Several papers were published. In an agreement between institutions, this circuit was moved from USP to IPEN/CNEN-SP for its better utilization. Some improvements were performed in order to updating instrumentation and data acquisition system. The experimental circuit consists of a rectangular glass loop with an electrical heat source and a coil cooler sink. Glass is used to allow the flow visualization, filming and photographing to identify the flow patterns and to observe the phenomena.

A new user interface using LabVIEW was developed for the data acquisition system. Experiments were performed for single and two-phase flow conditions. RELAP5 code was used to simulate this circuit. Figure 9 shows the comparison between measured and calculated temperatures at two positions of the circuit, T12 and T17, for single phase flow condition. Figure 10 shows the results of measured and calculated temperatures at the same positions of the circuit for two-phase flow condition.

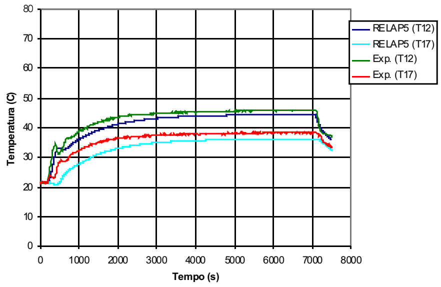


Figure 9. Theoretical and experimental temperatures at heater outlet (T12) and cooler outlet (T17) for single phase flow condition

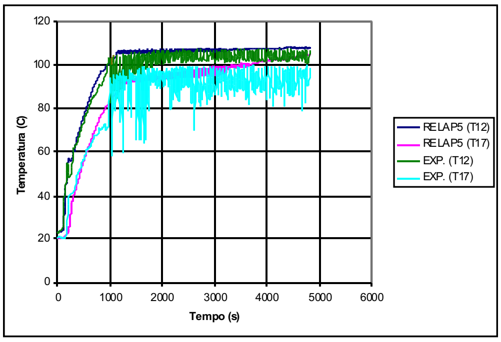


Figure 10. Theoretical and experimental temperatures at heater outlet (T12) and cooler outlet (T17) for two phase flow condition

Theoretical results from the code were compared to the experimental ones in order to validate the RELAP5 models. New instruments will be installed along the circuit for flow rate, void fraction and vibration measurements. Additional tests will be performed and the data will be compared with the code results. A high speed video/photo camera has been used for filming and photographing the two phase flow regimes and other phenomena during the transients. The acquired images are being treated in specific software for void fraction determination and boiling/condensation studies.

Safety analysis of Angra-2 nuclear power plant

The Safety Analysis group is involved with the activities related to the licensing process of nuclear power plant Angra 2. These activities have been developed through cooperation among institutes of CNEN and Pisa University in Italy. In this cooperation the safety analysis group is responsible for activities related to the elaboration of an input data set to RELAP5/MOD3 code, to be used in future independent calculations. IPEN's researchers in this cooperation were qualified to the accident analysis code in nuclear plants, which has been used in the commissioning of nuclear power plants Angra 1 and 2. The initial task consisted in the preparation of the necessary input data. Some of the Angra 2 components as well as some associated systems of the plant were considered. In a second phase, when the whole plant was modeled, researchers began the simulation of some transients and accidents, according to the description given in the Angra 2 Final Safety Analysis Report. From the results obtained, it was possible to evaluate work and proceed with alterations in order to more realistically represent the analyzed cases.

The simulated accidents will be repeated with the new detailed core nodalization as well as other accidents analyses aiming the validation of the multi-purpose nodalization, which represents a preliminary stage in the process of transient qualification of this input data. Uncertainty calculations were also in development and the results were satisfactory. The uncertainties were calculated by CIAU method. Figure 11 shows the primary loop pressure (theoretical/experimental) and the uncertainties band (CIAU). Figure 12 presents the primary loop mass (theoretical/experimental) and the uncertainties bands (CIAU).

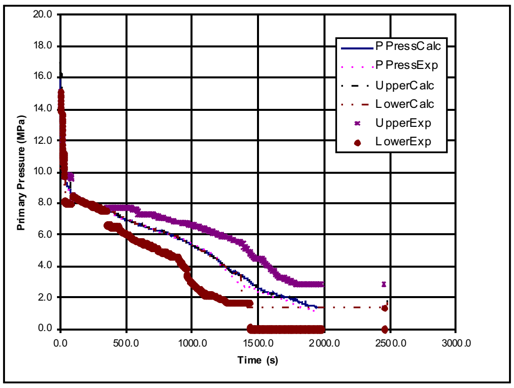


Figure 11. Primary loop pressure and the uncertainties bands

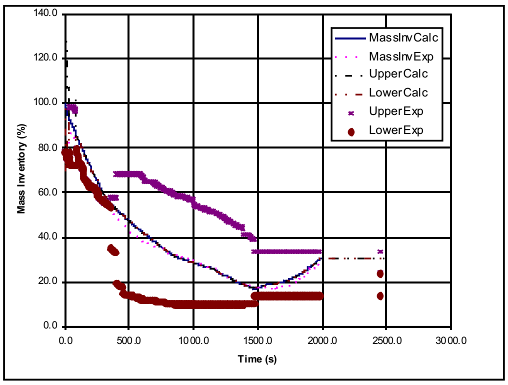


Figure 12. Primary loop mass and the uncertainties bands

Figure 11 shows the results obtained with RELAP5 code to nuclear power plant Angra 2 are into of the uncertainties bands, when compared with the experimental data (LOBI Test A1-93). This work received financial support from Coordinated Research Project (CRP) J7.20.05 of the International Agency Energy Atomic (IAEA).

Computational fluid dynamics studies

The research in this area was focused mainly in the IEA-R1 Heat Exchanger, Figure 13, and the Natural Circulation Loop.

A preliminary model was developed with ANSYS-CFX® code which was used in order to study the flow at the inlet nozzle of the heat exchanger of the primary circuit of the nuclear research reactor IEA-R1. The geometry of the inlet nozzle is basically compounded by a cylinder and two radial rings which are welded on the shell. When doing so there is an offset between the holes through the shell and the inlet nozzle. Since it is not standardized by TEMA, the inlet nozzle was chosen for a preliminary study of the flow. This research established an initial mark to the scientific cooperation between the Mackenzie Presbyterian University and IPEN.

Although with a simplified geometry, the CFD model supplied consistent results for the pressure field, velocity field, Figure 14, streamlines, Figure 15, and vectors. Mesh elements reached approximately 2.2 million, Figure 16, indicating our computational system limits.

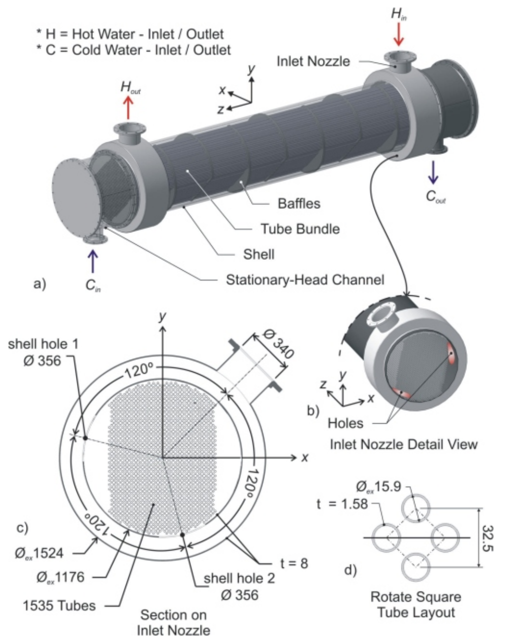


Figure 13. IEA-R1 heat exchanger

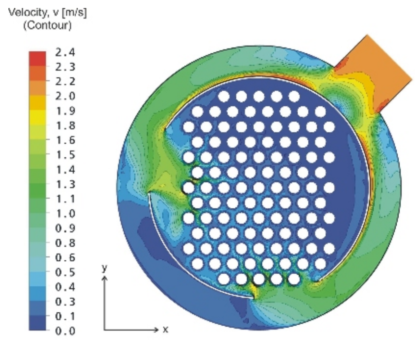


Figure 14. Velocity field

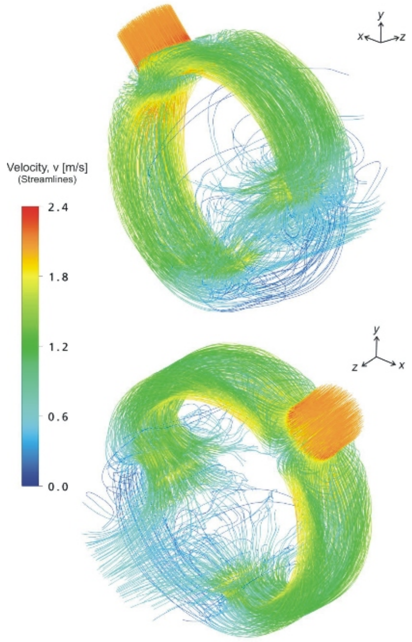


Figure 15. Streamlines

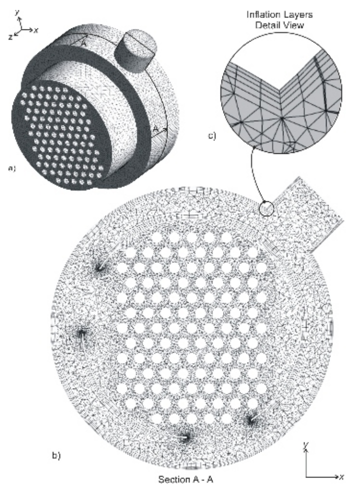


Figure 16. Mesh

Radial equilibrium design of axial flow gas turbine

In the axial flow gas turbine - reaction type - the flow gas has an angular velocity. This causes radial pressure gradient, consequently possible radial flow. The radial flow causes mixing which can result of entropy increase and less efficiency.

The early treatment of the problem is the Free Vortex method which results of constant entropy change in the radial direction and higher blade twist so it was recommended to be used in short blades. Actual measurements show higher entropy change in the tip and the root of the blades.

Later the problem was treated by Flow Through method, which is a numerical method, simplifies the flow in axial and radial directions only. This method was used by Pratt & Whitney (P&W) air craft manufacturer. The simplifying assumption, in order to solve the problem, was that the effect of losses was treated as reversible heat addition, which is not correct.

The approach suggested in our work was to divide the spacing between two successive blades, in the axial direction, in a 51 control volumes. For each control volume is written the governing physical equations, which are conservation of mass, conservation of momentum in axial direction, conservation of momentum in tangential direction, conservation of momentum in radial direction considering zero radial flow, conservation of energy, pressure loss coefficient model with exponent distribution along the axial direction and thermodynamic correlations. Observing that the pressure loss coefficient is a measurable quantity and can be estimated experimentally and there is many experimental works done about that subject.

The result was 612 nonlinear algebraic equations solved by Newton Raphson method, where the Jacobean matrix was calculated analytically forming full matrix of $(612)^2$ elements. The convergent limit can reach any value which the computer can calculate. It is used 10^{-5} as an acceptable limit for each variable.

The output calculates the exit flow angle along the radial direction which satisfies the condition of zero radial flow, also calculated the power and efficiency along radial direction. The results show that the Free Vortex and Flow Through simplifying assumptions was not true for this model.

Calculation of thermal hydraulic and vibration assessment of heat exchanger of research reactor IEA-R1 according as built information of manufacturer

The heat exchanger of IEAR-1 nuclear reactor is designed for 5 MW heat transfer load. The following additional factors were considered in the

1. The hydraulic pressure drop in the secondary side must be in a certain value which can permit flow rate of 2200 GPM in the secondary side pump, valves, filters, etc. without any change in any additional component. The value of 2200 GPM is the flow rate needed for cooling tower for 5 MW load.

2. The hydraulic pressure drop in the primary side must be in a certain value that permits 3500 GPM at least in the primary side without any change in any component of primary circuit.

3. High resistance to vibration damage.

The above three items needed mathematical modeling of hydraulic performance of secondary and primary sides. The third item needed mathematical modeling of cooling towers thermal performance.

The manufacturer IESA suggested many designs. They were checked by calculations done according to HEAT EXCHANGER DESIGN HANDBOOK correlations and TEMA vibration assessment. Finally heat exchanger with 15% cut segmental baffles without tubes in the windows satisfied both calculations done by the manufacturer IESA and IPEN-CNEN/SP. The heat exchanger was fabricated and installed in the IEA-R1 research reactor of IPEN-CNEN/SP.

Development of the reliability database for IEA-R1 Brazilian research reactor

The development of a reliability database for the research reactors located at IPEN-CNEN/SP started in 2001 when Brazil took part in an IAEA Coordinated Research Project (CRP). The IAEA CRP was entitled "CRP to Upgrade and Expand the IAEA Reliability Database for Research Reactor PSAs" and had participants from eleven Member States: Argentina, Australia, Austria, Brazil, Canada, Czech Republic, India, Indonesia, Republic of Korea, Romania and Vietnam.

In the case of Brazil, a specific reliability database for IEA-R1 reactor continued being updated and improved. The reliability database of IEA-R1 reactor consists of a set of connected Microsoft Excel spreadsheets (input data and output/final data) with necessary information to generate estimates of component failure rates/probabilities of failure on demand and accident initiating events frequencies; and to compile human error evidences related to reactor operation and maintenance. The generation of these data aims to give support to several technical areas of IPEN-CNEN/SP for the development of reliability and safety analyses of the local research reactors or other similar facilities. The information gathered in this database mainly covers:

Component Technical/Engineering Data
Technical characteristics of IEA-R1 reactor components are stored in the database (type, size, rating, fluid, manufacturer, model, location, etc.).

Component operational data
Records of continuous operating times between consecutive interruptions (either planned shutdowns or not) and the number of demands of the components per reactor operation are stored in the database. In addition, cumulative operating times and number of demands are also computed.

Component maintenance data
Every maintenance activity (preventive, corrective or predictive), concerning each reactor component, is recorded in the database.

Component failure data
All component failures are reported and verified in order to identify their causes, effects on system / subsystem, actions taken and recovery time.

Data analysis
Part of the data stored in the database can be processed in order to generate estimates of component reliability parameters. The approach implemented in this database is based on the assumption that failure times are exponentially distributed. It generates an estimate of the constant failure rate (that is the inverse of the "mean time to failure") associated to each time-related component failure mode. The analysis includes the calculation of a 90% confidence interval estimate (uncertainty limits) for each component failure rate or probability of failure on demand.

Accident initiating events and human errors data
Occurrences identified as accident initiating events precursors and/or human errors are stored in the database in order to be investigated and properly grouped. Considering the observation period from January 1999 to December 2007, 557 failures of 108 different component types were compiled. The total operating time of IEA-R1 reactor during that period was 19989,5 hours. Mean values of component failure rates / probabilities of failure on demand and respective confidence intervals are calculated using the algorithms developed during the IAEA CRP and are compiled in a specific spreadsheet. Data stored in the IEA-R1 database can also be used to estimate the frequencies of accident initiating events and to assess occurrences related to human errors during the operational and maintenance procedures. During the nine-year-observation period from 1999 to 2007, over 350 events were identified as initiating events precursors. In addition, 38 human errors were identified and grouped according to event types: failure to follow procedures or maintenance error (26); error of commission (9); and design error (3). Among these 38 events related to human errors, at least 25 could also be classified as precursors of accident initiating events. The scope of this database does not include quantitative derivation of human error data.

Neutronic analysis for production of fission Molybdenum-99 at IEA-R1 and RMB research reactors

The IEA-R1 reactor of IPEN-CNEN/SP is a pool type research reactor cooled and moderated by demineralized water and having Beryllium and Graphite as reflectors. In 1997 the reactor received the operating licensing for 5 MW. A new research reactor is being planning in Brazil to replace the IEA-R1 reactor. This new reactor, the Brazilian Multipurpose Reactor (RMB), planned for 30 MW, is now in the conception design phase. Low enriched uranium (<20% ^{235}U) targets (UAl_x dispersed in Al and metallic U foils) are being considered for the fission Molybdenum-99 (^{99}Mo) production. Neutronic calculations were performed to compare the production of ^{99}Mo for the two type of targets under similar conditions of irradiation (irradiation position, neutron flux and power density) both in the IEA-R1 reactor and RMB.

The UAl_x-Al targets of LEU type proposed and analyzed are aluminum coated miniplates. Each miniplate measures 52 x 170 mm, 1.52 mm thick, corresponding to a total volume of 13.437 mm^3. The UAl_x-Al core fuel is 40 x 118 mm, 0.76 mm thick, leading to a total volume of 3.587 mm^3. Considering this volume and a ^{235}U mass in the target equals to 2.01 g, the ^{235}U density (ρ_{U-235}) in the target core is 0.58 g $^{235}U/cm^3$. For a 19.9% ^{235}U enrichment, the uranium density in the target is ρ_U = 2.91 gU/cm^3.

A special Miniplate Irradiation Device (MID) was designed for the UAl_x-Al miniplates irradiation in the IEA-R1 reactor. Figure 17 shows the MID which has the external dimensions of the IEA-R1 fuel element. The miniplates will be allocated in a box with indented bars placed inside the external part of the MID.

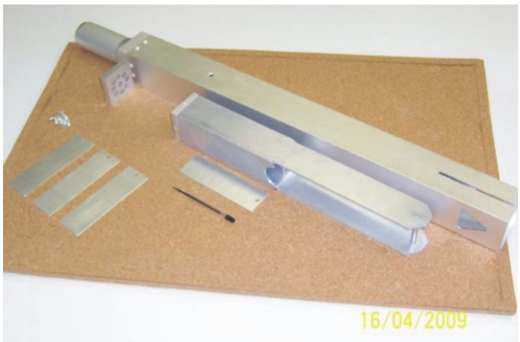


Figure 17. Miniplate Irradiation Device - MID

The targets of metallic U were mounted in cylindrical geometry, in a tubular arrangement. The metallic U foil was covered with a Ni sheet before being placed concentrically inside the aluminum tubes. The dimensions of the target are:

- One foil of uranium (LEU) of 44 cm x 76 mm x 135 µm;
- Coating nickel foil of 15 µm thickness;
- Two aluminum cylinder having 44 cm length, outside diameters of 27.99 and 30.00 mm, and inside diameters of 26.21 and 28.22 mm, respectively;
- ^{235}U mass of 20.1 g, distributed in 10 miniplates, with 19.9% enrichment of ^{235}U.

For the performed calculations, the U-Ni target was located in the same irradiation device, whose external dimensions are 76.2 mm x 76.2 mm x 88.74 cm (with nozzle). A mass equals to 20.1 g of ^{235}U in the metallic U foils was considered for the neutronic calculations.

From the neutronic calculations it was conclude that for the same amount of uranium in the analyzed targets (20.1 g) and the same irradiation conditions, a higher total ^{99}Mo activity was obtained for the U-Ni targets. In the IEA-R1 case, the total ^{99}Mo activity calculated at the end of the irradiation period for U-Ni targets was 1,275.8 Ci, while for the Al-UAl_x targets it was 581.89 Ci. For the RMB, the total ^{99}Mo activity obtained at the end of the irradiation time was 4,409.83 Ci for the Ni-U targets and 2,070.69 Ci for the UAl_x-Al ones. Initially, ^{99m}TC generators will be distributed five (5) days after the end of the irradiation. Consequently, the total ^{99}Mo activity is expected to reach values of 361.35 Ci and 164.7 Ci for the U-Ni and the UAl_x-Al targets irradiated in the IEA-R1, respectively. For the U-Ni and UAl_x-Al targets irradiated in the RMB, the total ^{99}Mo activity at the distribution time is expected to be 1,249.06 Ci and 583.51 Ci, respectively. From these values, it is noted that the Brazilian current demand of 450 Ci of ^{99}Mo per week and the future projected demand of 1,000 Ci may only be addressed by the RMB reactor under conception. This research is part of the IAEA's Coordinated Project (CRP) T1 2018 - Developing Techniques for Small Scale Indigenous Molybdenum 99 Production using Low Enriched Uranium (LEU) Fission or Neutron Activation.

Application of non-destructive methods for qualification of high density fuels in the IEA-R1 reactor

The IEA-R1 reactor of IPEN/CNEN-SP in Brazil is a pool type research reactor cooled and moderated by demineralised water and having Beryllium and Graphite as reflectors. Since 1990, IPEN/CNEN-SP has been fabricating and qualifying its own U_3O_8-Al and U_3Si_2-Al dispersion fuels. The U_3O_8-Al dispersion fuel is qualified to a uranium density of 2.3 gU/cm^3 and the U_3Si_2-Al dispersion fuel up to 3.0 gU/cm^3. The IEA-R1 reactor core is constituted of the fuels above, with low enrichment in U-235 (19.9% of U-235).

Nowadays, IPEN/CNEN-SP is interested in qualifying the above dispersion fuels at higher densities. Fuel miniplates of U_3O_8-Al and U_3Si_2-Al fuels, with densities of, 3.0 gU/cm^3 and 4.8 gU/cm^3, respectively, which are the maximal uranium densities qualified worldwide for these dispersion fuels, were fabricated at IPEN/CNEN-SP. The miniplates were put in an irradiation device, with similar external dimensions of IEA-R1 fuel assemblies (FA), which was placed in a peripheral position of the IEA-R1 reactor core.

IPEN/CNEN-SP has no hot cells to provide destructive analysis of the irradiated fuel. As a consequence, non destructive methods are been used to evaluate irradiation performance of the fuel miniplates:

- monitoring the fuel miniplate performance during the IEA-R1 operation for the following parameters: reactor power, time of operation, neutron flux at the position of each fuel assembly, burnup, inlet and outlet water, and radiochemistry analysis of reactor water;

- periodic underwater visual inspection of fuel miniplates and eventual sipping test for the fuel miniplate suspected of leakage.

The miniplates have been periodically visually inspected by an underwater radiation-resistant camera inside the IEA-R1 reactor pool, to verify its integrity and its general plate surface conditions. A new special system was designed for the fuel miniplate swelling evaluation. The fuel swelling evaluation is being performed by means of the fuel miniplate thickness measurement during the shutdown periods between successive irradiation cycles at the IEA-R1 reactor. During the measuring period, the fuel miniplates are transferred from the reactor core to the measurement system positioned at the pool border. This measurement system was sponsored by FAPESP.

Neutronic, thermal-hydraulics and accident analysis calculations for an irradiation device to be used in the qualification process of dispersion fuels in the IEA-R1 research reactor

Neutronic, thermal-hydraulics and accident analysis calculations were developed to estimate the safety of an irradiation device placed in the IEA-R1 reactor core. The irradiation device will be used to receive miniplates of U_3O_8-Al e U_3Si_2-Al dispersion fuels, LEU type (19.9% of ^{235}U), with uranium densities of, respectively, 3.0 gU/cm^3 and 4.8gU/cm^3. The fuel miniplates will be irradiated to nominal ^{235}U burnup levels of 50% and 80%, in order to qualify the above high-density dispersion fuels to be used in the Brazilian Multipurpose Reactor, now in the conception phase. For the neutronic calculation, the computer code CITATION was utilized. The computer code FLOW was used to calculate the coolant flow rate in the irradiation device, allowing the determination of the fuel miniplate temperatures with the computer model MTRCR-IEA-R1. A postulated Loss of Coolant Accident (LOCA) was analyzed with the computer codes LOSS and TEMPLOCA, allowing the calculation of the fuel miniplate temperatures after the reactor pool draining.

The MID has the external dimensions of the IEA-R1 fuel element. The miniplates will be allocated in a box with indented bars placed inside the external part of the MID. Up to ten miniplates can be placed in the box inside of the MID. The qualification of the U_3O_8-Al and U_3Si_2-Al dispersion fuels with higher ^{235}U density will be made in use, which means that it is based on the irradiation of the dispersion fuel miniplates in the IEA-R1 reactor followed by the use of non-destructive analysis techniques, mainly fuel miniplate visual inspections performed regularly with a radiation-resistant underwater camera. A new special system was designed for the fuel miniplate swelling determination. The swelling determination will be by means of the fuel miniplate thickness measurement during the irradiation time in the IEA-R1 reactor. The so called “Fuel Miniplate Thickness Measurement System” will be located at the fuel storage area of the IEA-R1 reactor pool. It will be operated from the reactor pool border, allowing the measurement of the fuel miniplate thickness along its surface by electronic probes (LVDT). The results will be collected by instrumentation connected to the probes.

The result of neutron calculations showed that the best position for the irradiation in the reactor core is the 36, for having the highest power density, and the inclusion of MID in the reactor core does not affect the operation of the same, since the change in reactivity is irrelevant. Through the thermal-hydraulics calculations it was determined a minimum flow for cooling the fuel miniplates diverting a small fraction of the flow of the reactor core without affecting the cooling of the core. From the analysis of accidents concluded that there is any damage to miniplates in the case of a postulated Primary Coolant Boundary Rupture. This project was partially sponsored by FAPESP.

Development of computer codes for loss of coolant accident analysis of IEA-R1 reactor

Two computer programs, LOSS and TEMPLOCA, were developed to analyze postulated Loss of Coolant Accidents (LOCA) in the IEA-R1 reactor. The LOSS program determines the time to drain the reactor pool down to the level of the bottom of the core. The TEMPLOCA program calculates the maximum temperature reached in the fuel, due to the decay heat of fission products and when there is complete loss of coolant in the core. These programs were used to assess the safety of a

Miniplate Irradiation Device (MID), placed in the IEA-R1 reactor core, during the occurrence of a postulated loss of coolant accident. The MID are being used to receive miniplates of U_3O_8-Al and U_3Si_2-Al dispersion fuels, LEU type (19,9% of ^{235}U) with uranium densities of, respectively, 3.0 gU/cm^3 and 4.8 gU/cm^3. The fuel miniplates will be irradiated to nominal ^{235}U burnup levels of 50% and 80%, in order to qualify the above high-density dispersion fuels to be used in the Brazilian Multipurpose Reactor (RMB), now in the conception phase.

Studies were performed to evaluate possible postulated loss of coolant events, which could lead to the reactor pool emptying. Five of them were considered to be mostly critical: a) Tube rupture of the Irradiation Pneumatic System (IPS); b) Pool drainage failure rupture of the access tubes for the Water Retreatment System (WRS); c) Primary system boundary rupture; d) Undue opening of the WRS drains; e) Failure in the collimator tubes of Beam Hole-3 (BH-3).

Out of the five accidents analyzed with the LOSS program, the primary system boundary rupture was found to be the most critical. The calculations showed that about 7.5 minutes are necessary to drain the reactor pool during a postulated primary system boundary rupture. After the pool draining, the maximal fuel miniplate temperatures calculated with the TEMPLOCA was 125 ºC, below the blistering temperature, which is the fuel temperature design limit.

Fuel miniplate thickness measurement system for dispersion fuel swelling evaluation

A special system was designed and constructed at IPEN/CNEN-SP for fuel swelling evaluation. The system will be used in the qualification process of U_3O_8-Al and U_3Si_2-Al dispersion fuels with 3.0 gU/cm^3 and 4.8 gU/cm^3, respectively. The determination of the fuel swelling will be performed by means of fuel miniplate thickness measurements along the irradiation time, during the shutdown period between the operational cycles of the IEA-R1 reactor. The system will be located at the reactor pool fuel storage area and it will be operated from the reactor pool border, allowing the measurement of the fuel miniplate thickness along its surface by electronic probes (LVDT). The results will be obtained by the instrumentation connected to the probes.

The Figure 18 shows the fuel miniplate thickness measurement system. It consists of a mobile metallic column, held by an X-Y coordinate table system for miniplate thickness measurement. The table is supported by another metallic structure fixed at the border of the reactor pool. The thickness measurement is performed by electronic probes (LVTD). The results are obtained by the instrumentation connected to the probes.

(a) (b)

Figure 18. Fuel miniplate thickness measurement apparatus at IEA-R1: (a) lateral view; (b) profile view (thickness)

For commissioning of the measurement system, two sets of measurements were made. The first set of measurements has consisted of thickness measurements of an aluminum dummy miniplate, outside the reactor pool. These measurements have evaluated the system performance in dry condition. The second set of measurements was performed for another dummy miniplate in the IEA-R1 reactor pool. The fuel miniplate thickness measurement system showed good response. Dimensional results were in good accordance. The results show that the equipment is efficient and accurate, with measurement precision of 1 µm.

Improvement of computer codes used for fuel behaviour simulation - FUMEX III

The Brazilian nuclear organizations are conducting joint efforts toward the coordination of existing personnel and infrastructure in order to create competence to study, design and develop nuclear fuel elements for high performance and extended burnup. As part of this program, IPEN-CNEN/SP joined the FRAPCON/FRAPTRAN users group in order to obtain the permission to utilize the computer codes FRAPCON-3 and FRAPTRAN. Since them, IPEN has made a joint effort between personnel and students to understand fuel rod modeling of these computer codes in both normal and off-normal conditions. With project “IAEA Improvement of Computer Codes Used for Fuel Behaviour Simulation FUMEX III”, IPEN will have the opportunity to test fuel modeling of FRAPCON-3 and FRAPTRAN codes against data and cases provided by IAEA and OECD/NEA. It is IPEN intention to promote with this project interaction, collaboration and discussions amongst Brazilian fuel modelers which will result in a better understanding of physical processes and phenomena, and thus allows improvements to be made in both Brazilian organizations codes and their models. The following Brazilian organization are participating of the FUMMEX III program: IPEN-CNEN/SP, CDTN-CNEN/BH, Eletronuclear (Eletrobrás Termonuclear S.A.), INB (Indústrias Nucleares Brasileiras) and CTMSP (Centro Tecnológico da Marinha em São Paulo). The project is being sponsored by IAEA.

Design, analyses and tests of a nuclear research reactor spent fuel storage and transportation packages

The applied qualification requirements for the packages used in the transportation of nuclear spent fuel elements are very severe due the nature of the radioactive content. They include the so-called normal conditions of transport and the hypothetical accident conditions. The 9 m drop tests are the most critical hypothetical accident conditions. The package qualification under these conditions shall be conducted using full scale models (prototypes), small scale models, numerical simulations and a combination of physical tests and numerical simulations. The choice of the qualification approach depends on economical and safety aspects. To comply with the nuclear safety functions, as the containment of the internal products and biological shielding, the package itself has several components connected to each other in different ways (impact absorbers, welded parts, flanged connections, surface contacts, etc.).

This research involves other groups in Brazil and abroad (mainly, Argentina) and it is sponsored by IAEA. It uses a combination approach with tests and numerical simulations for the structural assessment of a half scale model of a package for transportation of nuclear research reactor spent fuel elements under 9 m drop tests. The numerical simulations of the 9 m free drops over a rigid surface of half scale model of the transportation package under different orientations were conducted using a finite element explicit code considering several nonlinear aspects as the nonlinear materials models and properties, with emphasis on the impact absorbers behavior, the different package materials stiffness, and the different types of the contacts between the package components and between the package and the rigid surface, including the friction in the contacts. Also, several 9 m drop tests were conducted in a half scale model in different drop orientations.

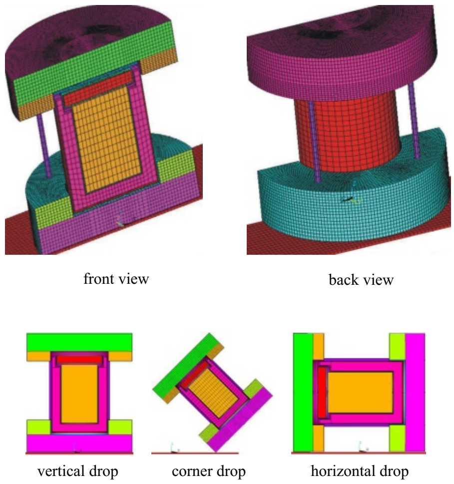


Figure 19. 180° finite element model and drop orientations

Monitoring steam generator tubes using wavelet transforms in eddy current test signals

Eddy current inspection is the standard method for routine steam generator tubes inspection. To improve the rate of correct diagnosis, a time - frequency domain transform, namely the wavelet transform is used to process the signals which are transient by nature, avoiding subjective inspectors decisions. The use of Wavelet Transform allows for automatically denoising the signal from sensor wobble, geometrical and material properties fluctuations and electronic noises. The method has been tested successfully in actual tubes from steam generators as well as laboratory made tubes with known defects implanted. Improved precision in the localization of the fault is also achieved by the right choice of the frequency band.

Integrated analysis of the USEXA postulated accidents

Under request of Centro Tecnológico da Marinha in São Paulo (CTMSP) was developed an integrated analysis of postulated accidental scenarios presented in the Preliminary Safety Analysis Report (PSAR) of the Uranium Hexafluoride Production Facility (USEXA), installed at Iperó, São Paulo, Brazil. The main purpose was to identify the propagation conditions of these accidents, both, on and offsite of process units, as well as to nearby plants (domino effect). Internal and external hazards chemical releases, fire and explosions at the facility were evaluated.

The PHAST Professional Software - version 6.51 (DNV Risk Management Software licensed to CEN) was used to perform consequence assessment. The domino effect analysis was carried out with methodology proposed in MSc. thesis “A Study on Domino Effect in Nuclear Fuel Cycle Facilities”, developed at IPEN/CNEN-SP.

Analysis of the USEXA hydrogen fluoride postulated accidents

Under request of Centro Tecnológico da Marinha in São Paulo (CTMSP) was developed an analysis of accidental scenarios involving Hydrogen Fluoride in the Uranium Hexafluoride Production Facility (USEXA), installed at Iperó, São Paulo, Brazil. The main purpose was to identify the toxic consequences of these accidents, both, on and offsite of process units, as well as to nearby areas. Internal and external hazards releases at the facility were evaluated.

The SAFETI Professional Software - version 6.51 (DNV Risk Management Software licensed to IPEN/CNEN-SP) was used to perform consequence assessment.

Decision support system for major accident prevention in the chemical process industry

The chemical industry today is processing a lot of hazardous substances within densely populated areas. The risks emerging from the processing, storage, handling, and transport of these hazardous materials are becoming more and more complex. Consequently chemical plants worldwide are faced with the growing importance of safety issues. Comprehensive and detailed hazard mapping and an understanding of possible consequences are necessary.

In last years it was developed decision support tools for investigating (internal and external) major hazards in the chemical industry to assist risk management decision makers in implementing organizational decisions on plant safety. Consequently an inherent approach to domino effect prevention in chemical process industries can be applied in early plant design or during modification in existing plant units.

Operation, maintenance and experimental utilization of the IPEN-MB/01 reactor

The first criticalization of the IPEN/MB-01 reactor (Figure 20) was obtained on November 9th, 1988. Since this date more than 2600 operations were made to measure very important in core parameters of the Reactor Physics such as kinetics parameters, spatial and energetic neutron flux distribution, power density, nuclear reaction rate, spectral indices, buckling, criticality rods positions to several and different reactor core (Figure 21) configurations. These experimental parameters are very important to correlate reactor calculation and its nuclear data libraries with experimental results. Thus is possible to improve the accuracy and precision of calculation methodology used to design the nuclear reactors core.

Figure 20. IPEN/MB-01 Reactor building

The IPEN/MB-01 has a maintenance program (each 2 weeks the reactor is shutdown for this purpose) that consists from a simple visual inspection of the reactor systems to a preventive and corrective maintenance made by the operators. For complex problems is contracted external maintenance. Normally are reserved until 15 weeks by year to maintenance of the IPEN/MB-01 Reactor.

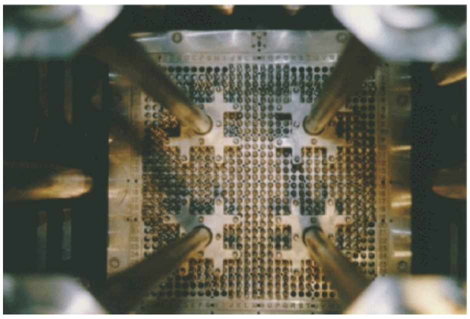


Figure 21. : IPEN/MB-01 Reactor core

Basically the reactor has three kinds of activities: Operational, Experimental and Educational. The operational is to testing the reactor systems and calibration of control rods, nuclear channels, power level, IAEA inspections, and training operators. The experimental activities are to measure reactor physics parameters to several purposes such as academic, benchmarks (OECD-NEA), educational (São Paulo University - Graduated and Post-Graduated) and to internal (IPEN-CNEN/SP) and external (Reactor Operator Training: CTMSP-Brazilian Navy and ELETRONUCLEAR).

The Table shows the number of operations, experiments and criticality time in 2008, 2009 and 2010 years. The total operational time to starting and shutdown of the reactor is about 1 hour and this time in not considered in the table below but may be estimated as being the same that the number of operations by year in hours.

Year	Number of Operations	Time Criticalization (H:Min)	Experimental and Operational Activities
2008	173	236:34	Neutronic characterization at Neutron Flux trap, Spectral Indices, Subcritical experiments, Instrumentation test (electrometers and filters) to noise analysis, Measurements of Spectral Cross Density (CPSD), Spectral ratios using fission chambers, Criticality using gadolinium in fuel rods, Solid detector testing Graduate Experimental Reactor Physics Course, reactivity measurements, training of reactor operators, Radiological emergency simulation-exercise.
2009	137	181:05	Operational testing, Reaction rate along of radial fuel pellet, Instrumentation tests, Neutronic characterization at Neutron Flux trap, Subcritical Reactivity Experiments, Measurements of Spectral Cross Density (CPSD), Buckling, Measurements, Effective neutron temperature, Neutrons box reflector experiments, Out-of-Core experiments, Critical configuration using stainless steel slabs (Benchmarks), Measurements of Neutron Spectrum Energy inside the fuel rods, Post-Graduate Experimental Reactor Physics Course, training of reactor operators (Eletronuclear).
2010	134	133:29	Neutronic characterization at Neutron Flux trap, Reaction rate along of radial fuel pellet, Cross power spectral density (CPSD), Measurements of Neutron Spectrum Energy inside the fuel rods, Reaction Rates measurements, Fission reaction rates measurements, Reactivity temperature coefficient measurements, Experiments using boron acid diluted at water moderator, training of reactor operators, Graduate experimental Reactor Physics Course.

Modernization of the IPEN/MB-01 reactor

The main goal of this modernization project (FINEP Support) is the replacement of the current control board IPEN/MB-01 reactor by a system with Programmable Logic Controllers (PLC). This new system will implement all the existing functions in the control board due to their flexibility and may implement others, so as to facilitate reactor operation and control.

The reactor protection system, consisting primarily through the channels and the nuclear safety interlocks, will not be modified. Only his connections with the current control board will be transferred to the system with PLCs. Being a development model for future reactors to be designed in Brazil, the reactor IPEN/MB-01 must accompany the state of the art control using digital systems. Thus, these digital control systems will be tested and evaluated in the safe operation of a zero nuclear reactor, lends itself to this kind of experience because of their safety, as evidenced in their 23 years of operation

Items controlled by the system with Programmable Logic Controllers (PLC)

The system with Programmable Logic Controllers (PLCs) will be able to act in 237 input variables and output of the scorer's table, with forecast growth of up to 315 variables.

Architecture of the system with PLC

The system with PLCs for data acquisition, display and control of the reactor IPEN/MB-01 will comprise three units interface with the field: Acquisition Unit Signal Command and General, for the acquisition of the variables the scorer's control board and to generate control signals for operating the reactor IPEN/MB-01; Unit Command Bars (UCB) to the command manual/automatic neutron absorbing rods (two control and two safety rods) and Acquisition Unit Position rods, which is to make acquisition of data from 08 position indicators of the reactor rods (04 Absolute Position Indicators and 04 Indicators of Relative Position).

The man-machine interface (HMI) will comprise 03 units, each composed mainly of micro-computer, video terminal, keyboard and mouse. The main unit for operation and control of the reactor called “SCADA/HMI” and two others only for monitoring the reactor called “clients”. The connection of these units with units to interface with the field should be done through Ethernet and the communication with the signs of the reactor will be held by the input and output interfaces (I / O) of the PLC.

Researches in energy, environment and development

In 2004 was formed a multidisciplinary group to study several relationships between Energy, Environment and Development with their social reflexes. Researchers from IPEN and others institutions (IEE, UNICAMP and UFRJ) are part of the group qualified by CNPq. The sub-lines studied are: basic concepts, consumption of energy and development, energy scenarios, energy and life quality, environment and development, sustainable development indicators, environmental solutions, alternative energy and energy conservation and consumerism. There are also partnerships to other institutions that already have tradition in this area.

During the last years, several studies have been executed and analyzed by the group, for example: the study of the consumerism as a relevance component in global environmental degradation, the relation between human activities and climate and environment changes since the primordium of the man up to the Industrial Age, study of the Emergy (Embodied-energy) methodology and its scientific fundamentals that takes into account the embodied energy during a process transformation, issues of Climate Change and Global Sustainability based in the Game Theory, analysis and actions aimed to human values as a form of contribution to the planet sustainability, multivariate analysis applied to the study of energy-environment-development relationship, etc.

The sustainable development is no doubt, the most divulged concept in the present time, taking into account the economic effectiveness, simultaneously with the requirements of ecological, social, cultural, technological and politics order. The concept of sustainable development started to be widely used after the United Nations Conference on Environment and Development that occurred in Rio de Janeiro in June of 1992. After that, some countries started presenting the sustainable development as a component of its political strategy conjugating environment, economy and social aspects. One of the challenges of the construction of the sustainable development is to create measurement instruments, such as the sustainable development indicators. Sustainable development Indicators are essential instruments to guide the action and to subsidize the accompaniment and the evaluation of the reached progress toward the sustainable development. One of the critical aspects is the methodology to be adopted in the determination of the indicator, its reading and interpretation. Independent of the choice of the methodology, it must be clear and transparent, about the principles that are in the base of the process.

Services on nuclear engineering and specialized services on nuclear and conventional power plants

Services were performed by the Nuclear Engineering Center in the nuclear engineering field, mainly among activities directly related to power and researches reactors, considering the following specialties: nuclear fuel, waste management, reactor physics, thermal and hydraulics, electricity, structural mechanics, piping, instrumentation, monitoring and diagnosis of reactor systems and components and safety analysis. The expertise of our professionals in the above fields is directed to the identification and analysis of technical problems and proposition of solutions to improve safe operation, better maintenance and efficient upgrade of systems.

Complementing the services described in the nuclear engineering field, other specialized technological services were also performed in risk analysis of fuel cycle installations and benchmarking of conventional fossil fueled power plants. The services were directed to solve technical problems related to safe design, reliable and efficient maintenance operation. The background acquired is also applied to others industrial chemical process and conventional industrial power plants.

All the services and specialized services described above were conducted by a team of 20 (twenty) professionals. A total of approximately 12.000-man-hour per year was spent in all these activities related to experiments, laboratory and engineering services. The outcome of these technical activities was demonstrated through technical reports, engineering documents, calibration sheets, inspections reports and training notes.

The customers were some other departments of the Brazilian Nuclear Energy Commission (CNEN) and other companies such as the Brazilian oil company (named PETROBRAS), Electrical and Energy Institute from São Paulo University, the Brazilian Nuclear Utility (named Eletronuclear) and the Brazilian Navy Technological Center (named CTMSP). All the engineering services are certified with the ISO 9001:2008.

In the triennial 2008-2010 the IEA-R1 Research Reactor has been operated most of the time at a power of 3.5 MW and operation schedule of 62 hours per week, achieving the following results:

- Reactor power: 3,5 MW
- Time of reactor operation: 8090 h
- Energy dissipation: 24697 MWh
- Number of samples irradiated in the grid plate: 3107
- Number of samples irradiated in the pneumatic system: 1950
- Production of ^{131}I: 1958 Ci
- Production of ^{153}Sm: 57 Ci

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.

Pneumatic system to transfer irradiated sample from the IEA-R1 reactor to radiopharmaceutical facility and Activation Analysis Laboratory

This system are been installed with the aim to facilitate the transfer of irradiated material to these facilities. The line are installed and a new facility is been built in the reactor to place the terminal station and to link this to the reactor pool.

Pneumatic system to short irradiation in the IEA-R1 reactor

This system needed to be rebuilt due to timing deterioration. The project and construction of the new one was done by IPEN staff. Another station located in a different place and with a different neutron flux is been constructed by the same staff in order to comply with the user needs.

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:

- Data acquisition system (SAD) concluded;
- Neutron flux monitoring system final phase of implementation;
- Online system to monitoring aerosol emission in implementation;
- Core temperature monitoring by using a instrumented fuel element concluded;
- Reactivation of a meteorological station at IPEN site in implementation;
- Pump vibration monitoring system a wireless module is being included.

Many of these equipments were obtained with IAEA financial support.

Ageing program

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

- Cooling tower: Tower B was rebuilt and Tower A will be tested in order to assure it performance if the reactor power is increased to 5 MW;
- Beam hole draining system: totally revised, including the placement of a flange in the BH#4 and BH#12, in order to minimize leak risks and worker exposing when the reactor power is increased to 5 MW;
- Electronic system: control room racks are been replaced and a new table control is been projected.

Physical protection system

New equipments like cameras, 42” televisions, tape records and electronic locks were installed to improve the access control and surveillance system.

Training program

Besides of the internal operator training program, part of the staff participated of scientific visits and training in foreign facilities to learn about different reactor systems. These visits and training were supported by IAEA resources.

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.

Nuclear Reactors and Fuel Cycle

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