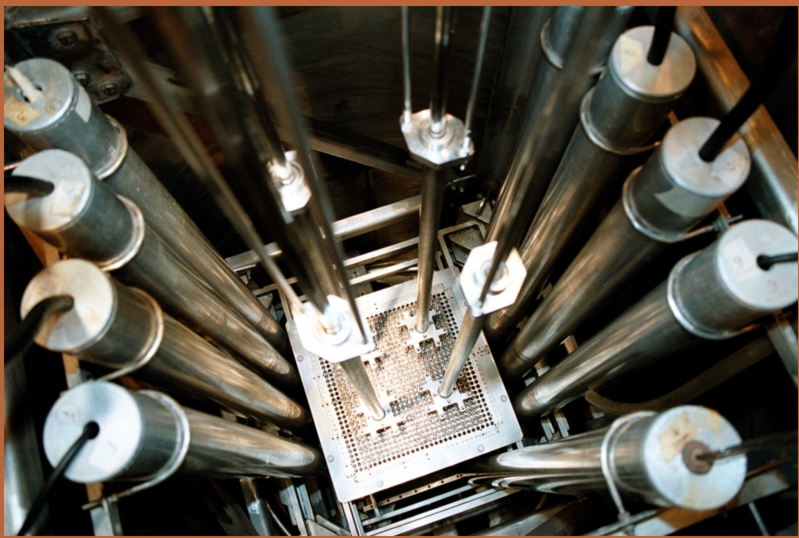
Nuclear Reactors and Fuel Cycle



Introduction

The years between 2005 and 2007 will be known as historical ones due to the countless discussions and publications on the new start of the use of nuclear energy worldwide. In Brazil, itself, the nuclear community was granted with governmental decisions that indicate the retaking of the necessary investments and the adoption of a long term nuclear program.

As to the nuclear reactors, the construction of Angra 3 and at least four more nuclear power plants until 2030 are predicted. There is also a high chance that a new research reactor will be built. Concerning the fuel cycle the goal is to become self-sufficient in the production of nuclear fuel. The nuclear reactors and fuel cycle areas at IPEN-CNEN-SP are fully prepared to attend the challenges that will come with the retaking of the Brazilian nuclear program.

The Nuclear Engineering Center (CEN) has shown during all these years high competence in the field of nuclear reactors providing specialized engineering services, research and development activities and educational competence for education activities. The specialized engineering services are mainly provided for the nuclear power plants Angra 1 and Angra 2. The research activities are mainly focused on the power and research reactors in operation nowadays. However, research is also performed on advanced and innovative nuclear reactors in order to prepare a competent staff for the challenges of the Brazilian nuclear program to be launched.

The Research Reactor Center (CRPq) installed in 2007 a new heat exchanger that will enable the IEA-R1 research reactor to operate at higher power, increasing the production capacity of radioisotopes, sample irradiation, tests and new experiments that could be needed for the operations and will represent an important tool for the measurement of parameters in the field of reactor physics. Both reactors will be playing a decisive role at the formation and training of the human resources needed in the nuclear area.

At the Nuclear fuel Center (CCN), the design and construction of a new factory of MTR fuel elements are being developed. This new plant will allow the production increase of research reactor fuels and benefit any governmental decision relative to building a new research reactor. The development of CCN concerning the production procedures of UO2-Gd2O3 nuclear fuel shows that the Center competency could also be expanded to respond to new challenges from the Brazilian Nuclear program.

Uranium recovery from slags of metallic uranium

The Nuclear Fuel Center of IPEN-CNEN/SP has already concluded its development program to fabricate uranium silicide fuel type (U_3Si_2-Al) for nuclear research reactors as IEA-R1. The process of fuel fabrication starts at LEU UF_6. The process to produce U_3Si_2 involves metallic uranium as an intermediate product, through magnesiothermic reduction which produces slags containing uranium. The recovery process consists on slag lixivium of calcined by-products from metallic uranium reduction. The results from researching this process confirmed that this method could be integrated in treatment and recovery routines of uranium. The chemical routes avoid to deal with metallic uranium since this material is instable, pyroforic and extremely reactive. On the other hand, U_3O_8 is a stable oxide with low chemical reactivity, and it justifies the slags calcination of metallic uranium reduction by-products. This calcination occurs in oxidizing atmosphere and transforms the metallic uranium into U_3O_8. Some experiments have been carried out using different nitric molar concentrations, acid excess contents and temperature control of the lixivium process. The nitric lixivium main chemical reaction for calcined metallic uranium slags is represented by the equation:

U_3O_8 (s) + 8 $HN0_3$ (l) $\rightarrow$ 3 $U0_2(N0_3)_2$ (l) + 2 $N0_2$ (g) + 4 H_20 (l).

It was adopted the following process parameters:

- temperature and time: calcination of slag of metallic uranium at 600°C during 3h;
- granulometric control: sieving and segmentation of calcined slag between at 100-200 mesh;
- concentration: lixivium adjustment of HNO_3 at 1 molar; HNO_3 excess (120%);
- lixivium temperature: 40 - 50°C;
- agitation: 300 rpm, turbine stem type (45 degrees inclination).

As results, we have: lixivium occurrence minimum time: 9 hours; fluoride concentration in lixivium: 0,002g/L. Lixivium made at low temperatures and low nitric concentrations reduce both the solubility of magnesium and calcium fluoride and the corrosion effect caused by fluoride ions, ensuring a stable and secure lixivium from operational point of view. The nitric dissolution of metallic uranium sludges produces uranyl nitrate solution, which is the utilized compound to feed the uranium purification system by solvent extraction method, with diluted n-tributhylphospate. The purified uranium product is precipitated as ammonium diuranate (ADU) at 60°C, by injecting gaseous ammonium diluted with air. Aiming at returning the recovered product to the fuel fabrication cycle with nuclear quality level, the purified ADU has to be converted to uranium tetrafluoride (UF_4) by U_3O_8 route. The yield of 94%, which is obtained by this process, proves the viability of this slag recovery of uranium magnesiothermic reduction. Figure 1 displays the sequence of recovery operation presently used in CCN/Chemical Laboratories for this recovery.

Figure 1. Development of uranium recovery

Uranium-Molybdenum technology

Since the start of the RERTR (Reduced Enrichment of Research and Test Reactors) program, IPEN-CNEN/SP/CNEN-São Paulo is working to fulfill its requirements. The conversion of high to low enriched fuels requested the development of the U_3Si_2 plate fabrication technology, but it is known that silicides, even being the current dispersion fuel of IPEN-CNEN/SP IEA-R1 reactor, are beyond the expected density goal of 9 gU/cm^3, necessary to convert HEU high production reactors in the world to LEU. U_3Si_2 is inadequate for HEU reactors for its technological and compatibility concerned to rolling and under irradiation behavior. To avoid these problems, alloys of UMo, in the range of additions from 5 to 10% weight Mo, are the most studied and employed in the fabrication of the plates. Its gamma allotropic form is the most suitable for the use as fuel for research reactors. With the technical cooperation project Brazil / AIEA BRA 4053, “Development of Alternative High Density Fuel Based on LEU UMo Alloy” and the Ph.D. thesis “Development of a High Density Fuel Based on the UMo Alloy with High Compatibility in High Temperatures”, IPEN-CNEN/SP is in the way to produce its first UMo fuel plates for seeking of its own fabrication technology also in this kind of nuclear fuel material. In addition, there are at our fuel center two projects with large budgets approved to upgrade our facility for fuel fabrication, mainly to produce U_3Si_2 but also to produce and research on UMo fabrication line. Two new MSc projects are underway to help the comprehension on UMo-Al chemical interaction and compatibility studies. The first step studied in the development of the technology of UMo alloys and plates fabrication was their synthesis and characterization. Here, factors like the condition of melting (arc or induction), hardness, grain size, gamma and alpha content, were determined by the usual techniques of characterization. The right thermal treatment conditions were also studied, enabling us to state and to reach the correct grain dimensions. A second and important step was the compatibility studies, where the interaction between oxygen, aluminum and UMo alloys were analyzed by means of the thermogravimetric and differential thermal analysis techniques. Both studies enabled us to state conditions of stability in high temperatures, as a function of the Mo addition. The third step was the production of UMo powders, and the route considered here was hydration-dehydration (HDH), instead of the widely utilized atomization and mechanization, for experimental reasons. The behavior of the alloys was studied under several conditions of thermal treatments, mainly times, temperatures and hydrogen pressure, by means of the thermogravimetric technique. To produce higher amounts of powder, in the order of 30 to 60g of UMo, the hydration-dehydration facility in the Magnetic Material Laboratory of IPEN-CNEN/SP was utilized, with good results mainly in the percentages between 5 to 7% Mo. The fourth step will be developed soon, since we have approved a project which will enable us to study and prepare several dispersion fuels, including UMo, to the determination of its behavior under rolling. Also, productions of large amounts of UMo powder of about 500g, are comprised by this project, which will be necessary to our production facility. In the next figures, there are some of the results of the first three steps. Figure 2 shows a typical curve of UMo behavior under hydrogen atmosphere, its powder correlated micrography.

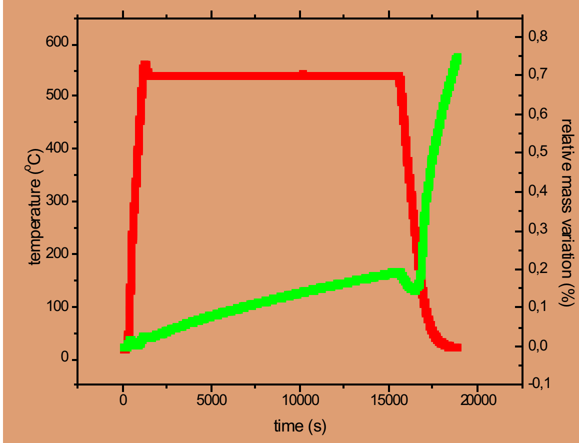


Figure 2. Typical mass variation curve showing the H2 acquisition

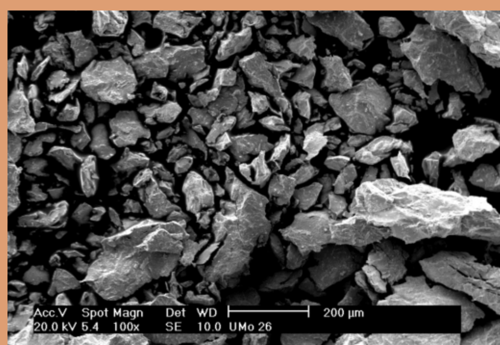


Figure 2.1. Produced powder structure

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 (Comissão Nacional de Energia Nuclear - National Commission of Nuclear Energy, which is a federal institution of Science and Technology Ministry in Brazil) 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. There has been a significant increase in the demand of radioisotopes over the past years. Between 2002 and 2004 the increase was about 30%. The distributed production supplied Brazilian hospitals and clinics. In 2006, it was attended more than 3 million patients, with an increase of about 10% relative to the previous year. To face this scenario, IPEN-CNEN/SP has been increasing continuously its production of radiopharmaceutical medicines to come along with the expanding demand imposed by Brazilian welfare. One of the most important project aims at own production of Mo-99 in order to provide cheaper Tc-99 generators than the ones produced from imported raw materials. So, this medicine will be accessible to a bigger amount of patients. The IPEN-CNEN/SP's research reactor IEA-R1 is the facility responsible to produce the radioisotopes for use in medicine. IPEN-CNEN/SP reactor's crew, for many years, has been working on upgrading mainly to increase its power from 2 to 5 MW and its operational time from 64 to 120 hours by week. The reactor control board and ventilation system were upgraded and additional safety items were incorporated. Besides, CNEN is planning to build a new facility in order to expand and spread the use of radiopharmaceutical medicines all over the country. Due to uncertainness in the international nuclear fuel market and in order to assure a continuous provision of fuel for its reactor, IPEN-CNEN/SP decided to fabricate its own fuel. The history of the fuel element fabrication technology in IPEN-CNEN/SP is very old. The work started in 1960 aiming at fabrication of the fuel for the ARGONAUTA research reactor. Between 1964 and 1965 the fuel elements were manufactured with 20% enriched U_3O_8 powder provided by IAEA in the program Atoms for Peace. In spite of the low demand of the ARGONAUTA fuel (very low power), a seed was planted and germinated 20 years later, in the decade of 80, and matured definitively in the decade of 90, when IPEN-CNEN/SP dominated the fabrication technology and began the production of the IEA-R1 fuel. The relative high power (2 MW) demanded a significant technological progress in the fabrication techniques. Starting from 1980, IPEN-CNEN/SP intensified its efforts to develop the fabrication technology of dispersion fuel element, aiming at improving the manufacturing technology for more advanced fuels. At that time, IPEN-CNEN/SP could not buy fuel elements from the international market to supply its research reactor, since there was a political growing difficulty to get fuel elements in international market. This upheaval acted as an initial impelling force for IPEN-CNEN/SP to push ahead its program for fuel element fabrication. The technology previously developed in the 60's was updated since 1985, based on the recent technological advances in this area. Between 1985 and 1988, IPEN-CNEN/SP worked in assembling a small fuel fabrication facility as a laboratory level, with capacity to produce 6 fuel elements by year. This was enough to supply the IEA-R1 reactor operating at 2 MW and 40 hours a week. On August, 31st, 1988, as part of the commemorations of their 32o anniversary, IPEN-CNEN/SP provided the IEA-R1 reactor with its first fuel element fabricated in Brazil, only fifteen days before the full exhaustion of the reactor fuel feeding. The fissile material at this period was the same U_3O_8 powder previously used for the production of the ARGONAUTA fuel. There was a reserve of about 30kg of this material. From 1988, after the production of the first fuel element, IPEN-CNEN/SP began a continuous production of fuel element, which continues until nowadays. After the production of 26 fuel elements, the enriched U_3O_8 powder finished in 1996. So, in 1994 IPEN-CNEN/SP started developing the processes for UF_6 conversion to U_3O_8 and for recovering the uranium scraps generated in fuel plate fabrication. In 1996, IPEN-CNEN/SP did the conversion of about 20kg of imported enriched UF_6. IPEN-CNEN/SP was then prepared for the routine production of fuel elements starting from UF_6 as raw material. In 1997, IPEN-CNEN/SP raised the fuel production capacity from 6 annual fuel elements to 10, which was the maximum considering the available facilities. In order to increase the radioisotope production of IPEN-CNEN/SP, the IEA-R1 reactor power capacity was increased from 2 MW to 5 MW. In 1997, the development of a new higher uranium density fuel started in order to attend the reactivity needs for reactor continuous operation. It has a compact core for better irradiation flux and also to dispose lower number of irradiated fuel elements to be stored at the spent fuel pool. The new fuel was based on the U_3Si_2-Al dispersion with uranium loading of 3.0 gU/cm^3. In 1998, the fuel plate fabrication technology of the new silicide fuel was implanted. At that time, U_3Si_2 powder was imported from France. Between 1999 and 2000 sixteen silicide fuel elements were manufactured. Starting on 1998, the efforts to develop the U_3Si_2 powder production technology began, aiming at the nationalization of all the production process: LEU enriched UF_6 reduction to produce UF_4; its

reduction to metallic uranium; from uranium producing the intermetallic U_3Si_2, with powder fabrication and, finally, arriving at the fuel plate fabrication and fuel element assembly. At this time, IPEN-CNEN/SP participated in an international cooperation through IAEA under the Technical Cooperation Project TC BRA/4/047.
In 1999 IPEN-CNEN/SP got the technology of UF_4 production using $SnCl_2$ as reducing agent. In the area of metallic uranium, IPEN-CNEN/SP had a valuable previous experience in producing up to 150kg of natural uranium ingots in 90's decade. Based on this experience, IPEN-CNEN/SP initiated efforts to scale down the size of the metallic uranium ingot, trying to produce pieces less than 3kg using 20% enriched material (LEU) as raw material for the U_3Si_2 production. In 2002, the process for producing metallic uranium was dominated, which allowed the development of the U_3Si_2 intermetallic. In 2004, IPEN-CNEN/SP produced the first lot of natural U_3Si_2 powder, manufactured with national technology, dominating then the "uranium silicide cycle". In 2006, IPEN-CNEN/SP consolidated the fabrication technology of the silicide fuel by manufacturing the first fuel element (figure 3), with fully national technology. This fuel element was put in the IEA-R1 reactor core on June 26, 2007. The nationalization of whole the production cycle of dispersion fuels for research reactors have been accomplished. After the mining and enrichment steps, IPEN-CNEN/SP became capable to execute all the other fabrication steps of dispersion fuel. The first U_3Si_2 fuel element manufactured with materials and technology totally national began to operate in the IEA-R1 nuclear reactor of IPEN-CNEN/SP. So, Brazil became totally independent in terms of materials and technology to supply fuel elements for their research reactors for producing radioisotopes. This was an important technological conquest, because it puts the country inside the international market, as part of a restricted group of commercial suppliers of this kind of fuel. Besides, the Al-dispersed fuel technology was developed also as a base for developing the fabrication technology of advanced dispersion based fuel for compact necessary for high performance nuclear power reactors for nuclear propulsion.

Figure 3. First U_3Si_2 fuel element fabricated at IPEN

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 elements 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 elements a year, representing the maximum production capacity in laboratorial scale for the available facilities at that time. That production capacity, reached its limit, which was enough to keep IEA-R1 reactor operating at 3-4 MW 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 for radioisotope production was cogitated to be constructed in the Northeast region of Brazil. This decision would be very significant in near future, since IEA-R1 reactor is already quite old (more than 50 years) and this reactor is practically the only one producer of radioisotope in the country. The new reactor (probably 20 MW) would consume about 30 yearly U_3Si_2-Al fuel elements. Therefore, a planned demand of about 50 fuel elements a year was quite realistic at that time. Based on this forecast demand for fuel elements, IPEN-CNEN/SP began a project in 2001 in order to adapt the production facilities seeking on improving the producing capacity. The first phase of that project is now concluded (figure 4). The fabrication steps regarding to the dispersion preparation, 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 under way. Once ready the whole project, all the production will be made through an integrated line in facilities with industrial character, which will operate according to international nuclear quality and safety standards, established by IAEA. Nowadays, this project is being completed. The new facility is planned to have nominal capacity for producing 30 fuel elements yearly. That will attend entirely the fuel element demand in a medium period. The producing capacity of the new facility could incremented to reach 80 fuel elements in a year, supplying so a new research reactor which has been planned to be constructed. The conclusion of this project is foreseen for 2008-2009.

Figure 4. Partial view - concluded part of the new CCN plant for fuel elements fabrication

Technological aspects concerning the production procedures of UO_2-Gd_2O_3 nuclear fuel

The activities regarding the UO_2-Gd_2O_3 fuel were continued. The mechanism that explains the sintering behavior of UO_2-Gd_2O_3 fuel prepared by the dry mechanical blending method was studied on early years and it was found that the bad sintering behavior is based on the occurrence of the Kirkendall effect. A significant difference in the interdiffusion coefficients of the gadolinium into UO_2 and of the uranium into Gd_2O_3 causes a misbalance in material transport during formation of the solid solution. As a consequence, the densification during sintering occurs simultaneously with the formation of pores in locations where $Gd2O_3$ agglomerates were originally present. The diameters of these pores are proportional to the initial diameter of these agglomerates. The pores formed are stable, since they are formed at high temperature in an essentially closed pore structure. Given this situation, the elimination of these pores after their formation is not possible in the subsequent sintering process. The pores remain in the sintered pellet and are the cause of the low densities observed. The work was continued in order to study possible adaptations in fabrication procedures for overlapping the sintering problem. The actions that could minimize or eliminate the problem would be based on changing certain fabrication procedures, as follows:

- Activity optimization of the UO_2 powder used in the preparation of UO_2-Gd_2O_3 mixed powder, which can be achieved by adjusting the conditions for AUC reduction;
- Sintering cycle optimization by adjusting the heating rate, temperature and duration of the isothermal sintering step;
- Adjustment in the homogenization procedure of the UO_2 and Gd_2O_3 powders aimed at obtaining a mixture presenting a high level of homogeneity.

Activity optimization of the UO_2 powder

The formation of pores due to solid solution formation began to occur at temperatures above 1000°C, during the second stage of sintering, and ended around 1350°C. One possibility for acting on this process would be to delay the sintering process. In other words, to act on the sintering kinetics in order to determine that it occurs at higher temperatures, after the formation of the solid solution and, consequently, after pore formation. The delay in the sintering process could be compensated in the isothermal stage of the sintering cycle. One way to achieve this objective is to reduce the activity of the UO_2 powder used to prepare the mixed powder. This possibility was tested using UO_2 powder derived from AUC with different areas of specific surface when preparing the UO_2-Gd_2O_3 mixed powder. Controlling the activity of UO_2 powder was possible through controlling the parameters for AUC reduction. There is an optimal specific surface which provides the UO_2 powder a reserve of activity in order to guarantee that sufficient sinterability exists for densification at high temperatures, after solid solution formation. This reserve of activity for sintering at high temperatures would promote the efficient elimination of the pores formed due to the Kirkendall effect and optimize densification. The best result was obtained using UO_2 powder with a specific surface of 4.5 m2/g.

Sintering cycle optimization

Other possibility of acting on the sintering kinetics by delaying the sintering process in order to determine that it occurs after pore formation by the Kirkendall effect, would be to increase the heating rate of the sintering cycle. If the kinetics for solid solution formation, or for pore formation due to the Kirkendall effect, occurs faster than sintering kinetics, the effect of increasing the heating rate would be beneficial in terms of residual porosity, since a larger fraction of pores formed could be eliminated in the sintering process subsequent to pore formation. A delay in the sintering process could be compensated in the isothermal stage of the sintering cycle. Figure 5 illustrates the sintering process under different heating rates The positive effect of increasing the heating rate can be verified in the final sintered density. Elevation of the heating rate from 5°C/min to 30°C/min led to an increase in the final sintered density of almost 1.5 vol%. When the heating rate was very low (1°C/min), pore formation during the formation of the solid solution was very clearly observed and densification in the isothermal period was very low. In contrast, when the heating rate was superior to 10°C/min, the decrease in the shrinkage rate due to pore formation was smaller and densification at the onset of isothermal treatment was pronounced, resulting in a beneficial effect in eliminating porosity, which led to higher densities in sintered fuel pellets.

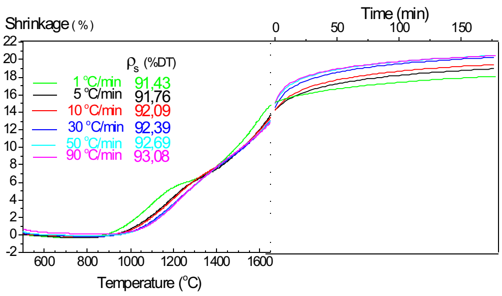


Figure 4. Effect of the heating rate on the sintering process of the UO_2-Gd_2O_3 fuel

Adjustment in the homogenization procedure

Previous experimental results demonstrated that homogeneity in the Gd_2O_3 distribution within the mixed powder exerted a decisive influence on the sintering process of UO_2-Gd_2O_3 fuel. In other words, the better the homogeneity in the mixed UO_2-Gd_2O_3 powder, the higher the final sintered density. This behavior is explained based on the Kirkendall effect mechanism. The development of an alternative method for UO_2 and Gd_2O_3 powder homogenization, which permits good homogeneity at a microscopic level, is another possible action that could improve the sintering behavior of UO_2-Gd_2O_3 fuel. The homogenization method should preserve the original morphology of the particles of the AUC powder, which gives good flowability to the UO_2 produced and the desirable direct pressing procedure. The use of comilling, in spite of resulting in good sintered densities (microscopic level of homogeneity), is not technologically interesting, because this procedure destroys the original morphology of the UO_2 powder, which implies the incorporation of a granulation procedure. The coprecipitation process via AUC permitted the acquisition of good homogeneity, at a microscopic level, in the distribution of Gd_2O_3 in the UO_2-Gd_2O_3 mixed powder, which resulted in an adequate densification level during sintering. In fact, coprecipitation does not occur in the AUC case. Probably, simultaneous precipitation occurs and homogenization is achieved in the liquid phase. Although this method for Gd_2O_3 incorporation during the precipitation stage resulted in a homogeneity level that was adequate for obtaining sufficiently good sintered densities, this method has the disadvantage of contaminating the precipitation reactor, which would demand exclusive equipment. To avoid duplication of the precipitation installation, an alternative would be the incorporation of the Gd_2O_3 powder into the AUC suspension prior to filtration. Thus, homogenization would be achieved in a liquid media, which is much more efficient at allowing the desegregation of Gd_2O_3 powder and would permit its dispersion in individual particles among the AUC crystals. In this case, after AUC precipitation in the traditional reactor, the suspension would be pumped towards a homogenizing tank and then towards a second filter, different from that used in the process for producing the standard UO_2 fuel. In this case, the duplication of the filtration system and the installation of an additional tank for homogenization would only be required. The development of alternative techniques for homogenizing the powders, together with the use of Gd_2O_3 powder with special characteristics (low tendency to agglomerate), would probably allow the obtaining of UO_2-Gd_2O_3 fuel pellets with the minimum specified density. However, a specific work in that area should still be conducted.

Reactor physics benchmarks at the IPEN/MB-01 reactor

Since 2004, the reactor physics group of the Nuclear Engineering Center of IPEN-CNEN/SP 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:

- 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;

- Evaluate the data and quantify overall uncertainties through various types of sensitivity analysis;

- Compile the data into a standardized format;

- Perform calculations of each experiment with standard criticality safety codes;

- 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 located in the area of the Nuclear Engineering Center of IPEN-CNEN/SP. 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 70 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 ANGRA-II to save storage space. Another very interesting experiment was a central void simulation with an aluminum block. More recently the reactor physics group completes a series of experiments with a heavy reflector made of SS-304 to give support to the EPR development in Europe. In the reactor physics area (IRPhE) 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 2007 was the measurement of several effective delayed neutron parameters without the need of any correction factor or data from other experiments. The experiment as approved to be included in the handbook of reactor physics benchmarks to be issued in March 2008. Also, in the same subject, IPEN-CNEN/SP and the Nuclear Center of Cadarache are working jointly in an international collaboration inside of the BRAZIL-FRANCE cooperation program. IPEN-CNEN/SP received a series of miniature fission chambers from Cadarache which will allow the measurement of several spectral indices in the IPEN/MB-01 Reactor. A mission from France visited IPEN-CNEN/SP in September 2007 to support the measurements at IPEN-CNEN/SP and to train the reactor physics staff on the utilization of the equipments.

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

The development of a reactivity meter at IPEN-CNEN/SP 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-CNEN/SP 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

A pos-doctoral student worked during 2006 and 2007 and developed a novel theoretical model to describe collisions among charged particles in an electric field. This model presents a novel form to treat coulomb interactions between charged particles. In this model the coulomb interaction is divided into two parts: interactions caused by the mean potential field generated by the electric field (electric field caused by the electrodes) and a short range interaction between two ions that takes into account the local effective potential. The short range interaction is used to correct the mean potential due to local changes when two charged particles are closed. The short range interaction is defined by a collision operator which is introduced in the Boltzmann equation. This modified Boltzmann equation allows a more realistic solution for the particle distribution in an ion field and is simple to be solved numerically. This model is now being adapted to study the space and energy distribution of the ions inside an IECF device. A master student is working since the beginning of 2007 to design an IECF device based on spherical electric grids. To provide subsides for the design a parametric study is under way to verify the influence of the electric potential and the distances between the grids on the nuclear fusion rates. In this study a simple model that does not considers ion-ion collision is being used to calculate the ion space and energy distribution.

Numerical simulation and experimental validation for application in medical physics

The goal of this multidisciplinary research group is the development of new numerical dosimetry methodologies using Monte Carlo techniques for the application in medical physics area comprehending: radiotherapy, radioprotection and nuclear medicine. During the last three years the group has accomplished some important tasks to help medical physicists in clinical and instrumentation issues. In the period of 2005-2007 it was developed a project to completely characterize the electron beam of various energies produced in a specific linear accelerator used for patient treatment in radiotherapy. The construction of the energy spectra for each clinical energy beam enabled the simulation of a series of experiments in phantoms used for the system calibration. Monte Carlo simulation using voxel phantoms also allowed the estimation of dose distribution in patients under treatment. The results provided insights for the medical physicists about the effectiveness of the treatment proposed. Figure 5 shows an example of an output of the methodology developed.

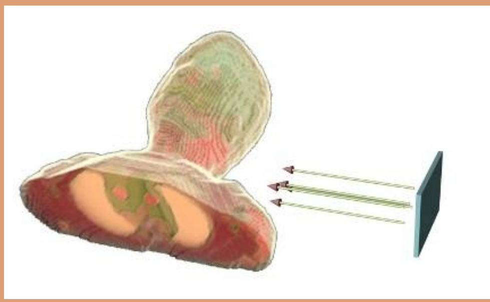


Figure 5. Head and neck irradiation with a typical clinical electron beam

Later, many efforts have been done to adapt the present methodology to be able to provide several dosimetric data for the implantation of new clinical modality: the total skin irradiation. The ultimate goal is to capacitate the radiotherapy service to offer an alternative treatment for patients with skin cancer.

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 waste burner reactors. The activities under way are mainly related in development and qualification of calculation methodologies and fuel cycle simulations. We are participating 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 2005-2007 in innovative hybrids reactors are:

- Participation in the 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.

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

With the financial support of the IAEA the Nuclear Engineering Center has designed an instrumented MTR fuel element, with the objective to measure the temperatures in lateral and central plates of the fuel element and compare them with the temperatures calculated with thermo hydraulic computer codes available in this Center. In this design was considered the installation of 8 thermocouples, 4 in lateral channel and 4 in the central channel. In the central channel there are 3 thermocouples to measure the clad temperature and one thermocouple to measure the fluid temperature near to the outlet channel. In the lateral channel there are also 3 thermocouples to measure the clad temperature and one thermocouple to measure the fluid temperature near to the outlet channel. A dummy fuel element was constructed by the Center of Nuclear Fuel (CCN) to solve some construction problems, and the next step will be the construction of the instrumented fuel element to be used in the research reactor IEA-R1 of IPEN-CNEN/SP.

Natural circulation in nuclear power reactors (theoretical and experimental)

One of the most serious problems for a nuclear reactor in accidental conditions is the residual heat removal from the decay of the fission products. The design of a nuclear reactor providing the establishment of natural circulation in the primary system may be an alternative to refrigerate the reactor core. In order to understand the complex phenomena involved in two phase natural circulation and to generate data for validating computer codes, in 1994 an experimental rectangular glass loop was designed by IPEN-CNEN/SP and the Chemical Engineering Department of USP, with financial resources of Chemical Engineering Department. During the period 1994 to 1999, the operation of this circuit has generated several studies, articles and a master dissertation. In 2003 researchers of IPEN took the initiative to put into operation this natural circulation circuit and in the recent years, 5 students of basic scientific research, guided by researchers of the Centre of Nuclear Engineering, have worked to put this circuit in operation again with the replacement of broken thermocouples; leaks fix, replace of the heater; installation of flow meters for the secondary circuit; design, development and implementation of a new data acquisition system based on new tools. All these modifications and repairs were carried out with resources of IPEN-CNEN/SP.

Safety analysis of Angra-2 nuclear power plant

The Safety Analysis group is involved with the activities related to the licensing process of 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 Angra 1 and 2. The main 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 are also in development.

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 IEA-R1 nuclear reactor is designed for 5 MW heat transfer load. The following additional factors were considered in the design:

- 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;

- 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;

- 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 IEASA 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 IEASA 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 and IPEN/MB-01 Brazilian research reactors

In January 2000, the IAEA initiated the Coordinated Research Project (CRP) “To Update and Expand the IAEA Reliability Database for Research Reactor PSAs”. The main objective of this CRP was to update and expand the existing IAEA reliability database, published as the IAEA-TECDOC-930, compiling, processing and including adequately all the data collected by the participant institutions in their research reactors. The second objective of this CRP was to promote the development and use of Probabilistic Safety Assessment (PSA). As these objectives presented interests in common with the ones of IPEN/CNEN-SP, this Institute signed a contract with the IAEA in March 2001, aiming to carry out a research project with the following purposes:

- Development of a component reliability database for IEA-R1 and IPEN/MB-01 Brazilian research reactors;
- Development of a computerized system to store and to process reactor operational, maintenance and failure data;
- Collection of component failure data to compose the IAEA Reliability Database;
- Documentation of findings on the research areas of Accident Initiating Events, Human Errors and Common Cause Failures;
- Contribution to the new IAEA technical document that will supersede IAEA-TECDOC-930.

The database software was developed in order to store and to process operational, maintenance and failure data of IEA-R1 and IPEN/MB-01 components. It also includes the storage of components descriptions and provides the user with several types of reports. Microsoft Access has been used to develop the database software. PSA analysts and the personnel in charge of operation and maintenance of the facilities led data collection activities at IEA-R1 and IPEN/MB-01 reactors. The data collected and reported in this study cover major components of selected systems of the two Brazilian research reactors. Concerning IEA-R1 reactor, failures of 49 component types were compiled within an observed period from January 1999 to December 2003. Concerning IPEN/MB-01 reactor, failures of 59 component types were investigated within an observed period from October 1997 to March 2004. The component failure rates and associated 90% confidence limits derived from data collected at the reactors were tabulated and sent, via Internet, to the IAEA Project Officer. These results were then incorporated in the IAEA Reliability Database together with data collected by other participant Institutes in their research reactors. The descriptions of the component types (component boundaries, equipment ratings, size, application, etc.) of both IEA-R1 and IPEN/MB-01 reactors were also provided. As a completion of the objectives of the research project, IPEN/CNEN-SP elaborated the topic "Statistical Data Analysis" to be included in an Appendix of the new IAEA TECDOC. Finally, the research project carried out at IPEN/CNEN-SP has shown that it is essential to promote an integration of data collection activities with the existing administrative routines of the facilities in order to minimize the efforts and to provide good quality data.

Safety culture enhancement at the nuclear research reactors of IPEN-CNEN/SP

For the IEA-R1 research reactor, the following activities were carried out:

- Survey of the state of the art in safety culture. This task was accomplished through seminars given by invited speakers and panel sessions where the participants made presentations based on literature studies in this area;

- Safety culture self assessment at IEA-R1 research reactor. The aim of this self assessment was to evaluate the safety culture at the reactor objectively. For this purpose, it was used the self assessment questionnaire proposed in the document Safety Series No. 75-INSAG-4 published by the IAEA. Important improvement opportunities in the safety of the reactor were identified in these appraised aspects. It is worth mentioning two examples: 1. the safety policy of the organization must be rewritten in a clearer and complete format; 2. training program of the reactor performance must be focused on safety issues. Presently, theses aspects are being addressed by the reactor management.

For the IPEN-MB/01 research reactor, an assessment of its safety culture was performed, based on Three Level model of safety proposed by Edgar Schein, with the participation of the whole staff of the reactor. This assessment was developed during a workshop in which the main concepts of the book "Organizational Culture and Leadership" and the guidelines IAEA-TECDOC-1329, "Safety culture in nuclear installations", were also presented.

Qualification program of research reactor U_3O_8-Al and U_3Si_2-Al dispersion fuels manufactured at IPEN-CNEN/SP

The strategy adopted by IPEN for qualification the own fuel elements is to conduct the fuel tests and fuel performance evaluations in pile, during IEA-R1reactor operation. The program for fuel element evaluation include monitoring the reactor power; time of operation; neutron flux calculation at the position of each fuel assembly; burn up calculation; inlet and outlet water temperature in core; water pH; water conductivity; chloride content in water; radiochemistry analysis of reactor water. Periodic underwater visual inspections and eventual sipping tests for fuel assembly have been carried out. A special equipment to perform underwater thickness measuring was projected by the Fuel Engineering Group of IPEN-CNEN/SP to evaluate the fuel miniplate swelling during the irradiation time. A Brazilian supplier constructed the mechanical parts and the electronic measurement instruments were acquired by IAEA. Since the 80's IPEN has been producing and qualifying its own LEU MTR fuels (19.75% ^{235}U). Fuel element assemblies have been constructed with U_3O_8-Al dispersion fuel plates with densities of 1.9 gU/cm^3 (from 1988 to 1996) and 2.3 gU/cm^3 (from 1996 to 1999). Since 1999, IPEN has been fabricating 3.0 gU/cm^3 U_3Si_2-Al dispersion fuels. Regarding the qualification in-use of MTR fuel assemblies, until 2007, the 3.0 gU/cm^3 U_3Si_2-Al dispersion fuels achieved an average burnup of 42.49%. The in-pile irradiation of 2.3 gU/cm^3 U_3O_8-Al dispersion fuel achieved an average burnup of 40.50%.

Studies on irradiation performance of U-Mo dispersion fuels

A multidisciplinary team of the Nuclear Engineering Center of IPEN-CNEN/SP has been studying on irradiation performance of the U-Mo alloy dispersion fuels aiming for generating technical specifications for fabrication of U-10Mo-Al dispersion fuel to be irradiated in the IEA-R1 Research Reactor. The irradiation performance aspects were associated to the neutronic and thermal-hydraulics aspects in order to propose a new core configuration to the IEA-R1 Reactor using U-Mo dispersion fuels. Core configurations evaluations with the computational programs CITATION and MTRCR-IEAR1 showed that U-10Mo-Al dispersion fuel with uranium densities ranging from 3 to 5 gU/cm^3 would be adequate to be used in the IEA-R1 Reactor and would present a stable in-reactor performance even at high burnup. It was proposed the meat uranium density of 5gU/cm^3 for the miniplates of U-Mo dispersion fuel to be fabricated at IPEN-CNEN/SP and tested in the IEA-R1 Reactor. With this uranium density in the fuel meat, the number of fuel elements used in the IEA-R1 reactor would be reduced, bringing economic advantages and reducing the number of spent fuel elements to be stored in the reactor pool.

Options for the interim storage of the IEA-R1 research reactor spent fuels

A multidisciplinary team of the Nuclear Engineering Center and from CDTN-CNEN/SP has carried out an analysis of different options for the interim storage of research reactor spent fuels. The analysis results showed a dual-purpose metallic cask as one of the most feasible options for the interim storage of the spent fuels of the IEA-R1 Research Reactor. In case this reactor keeps its continuous operation at 5 MW in the next years, it will be necessary to transfer the spent fuels, today located in the reactor storage pool, to a spent fuel interim storage, until a definition of the Brazilian national policy about the final disposal of these spent fuels be established. Those conclusions were shared with other Latin-American institutions in the IAEA Latin American Regional Project RLA/4/018 (IAEA-TECDOC-1508, 2006). The conceptual design of a dual transport cask for the interim storage of IEA-R1 spent fuels was finalized and a scale prototype was manufactured in Argentina for qualification tests to be held at CDTN-CNEN/SP in 2008. The dual-purpose cask Safety Analysis Report (SAR) is being prepared for submission to the licensing authority in Brazil. IPEN is now developing the general requirements and the design aspects of the conceptual project of a spent fuel storage facility that will accommodate the dual-purpose casks.

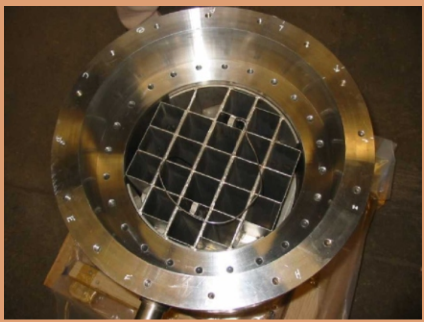


Figure 6. Dual purpose metallic cask - main body and basket

Modeling for release of fission products from a breached fuel plate under wet storage

MTR fuel elements burned-up inside the core of nuclear research reactors are stored worldwide mainly under the water of storage pools. When cladding breach is present in one or more fuel plates of such elements, radioactive fission products are released into the storage pool water. This work proposes a model to describe the release mechanism considering the diffusion of nuclides of a radioactive fission product either through a postulated small cylindrical breach or directly from a large circular hole in the cladding. In each case, an analytical expression is obtained for the activity released into the water as a function of the total storage time of a breached fuel plate. Regarding sipping tests already performed at the IEA-R1 research reactor on breached MTR fuel elements, the proposed model correlates successfully the specific activity of ^{137}Cs, measured as a function of time, with the evaluated size of the cladding breach.

Characterization of spent fuel elements stored at IEA-R1 research reactor based on visual inspection and sipping tests

Aluminum spent nuclear fuels are susceptible to corrosion attack, or mechanical damage from improper handling, while in pool reactor storage. Storage practices have been modified to reduce the potential for damage, based on recommendations presented at 2nd WS on Spent Fuel Characterization, promoted by IAEA. The Fuel Engineering Group of the Nuclear Engineering Center carry out an inspection program proposed to the IEA-R1 stored spent fuel elements, in order to provide information on the physical condition during the interim storage time under wet condition at the reactor pool. The inspection program is based on non-destructive tests results (visual inspection and sipping tests) already periodically performed to exam the IEA-R1 stored spent fuel and fuel elements from the core reactor. To record the information and examination results it was elaborated a catalogue containing the proposed inspection program for the IEA-R1 stored spent fuel, the description of the visual inspection and sipping tests systems, a compilation of information and images result from the tests performed for all stored standard spent fuel element. That document constitutes an important step of program's effective implementation, and it has been used to address regulatory and operational needs to guarantee a safe storage throughout the pool storage period.

Evaluation of fuel failure in nuclear power plants

Efforts are being made by Fuel Engineering Group of the Nuclear engineering Center to understand and evaluate the mechanisms involved in the fuel degradation subsequent to postulated primary failure in Brazilian nuclear power plants. Controlling the degradation process is extremely important to avoid large radioactivity levels in the plant and unscheduled shutdowns for removing the leaking assembly. On the majority of the cases, the inquiries on primary and secondary failures in PWR fuel rods are based on results of analysis were made use of the non-destructive examination results (coolant activities monitoring, sipping tests, visual examination). The complementary analysis methodology proposed includes a modeling approach to characterization of the physical effects of the individual chemistry mechanisms that constitute the incubation phase of degradation phenomenon after primary failure that are integrated in the reactor operational history under stationary operational regime, and normal power transients. The computational program DEGRAD-1 was developed based on this modeling approach. The practical outcome of the program is to predict cladding regions susceptible to massive hydriding. The applications presented demonstrate the validity of proposed method and models by actual cases simulation, which (primary and secondary) defects positions were known and formation time was estimated. By using the modeling approach, a relationship between the hydrogen concentration in the gap and the inner cladding oxide thickness has been identified which, when satisfied, will induce massive hydriding. The novelty in this work is the integrated methodology, which supplements the traditional analysis methods (using data from non-destructive techniques) with mathematical models for the hydrogen evolution, oxidation and hydriding that include refined approaches and criteria for PWR fuel, and using the FRAPCON-3 fuel performance code as the basic tool.

Evaluation of Debris fretting failures on PWR fuel by post-irradiation examination and modeling in the DEGRAD-1 Code

One of the major recognized causes of fuel rod failures is fretting of the clad due to the entrapment of debris in a fuel rod spacer that inadvertently dropped into the primary system during maintenance operations. Debris, such as metal cuttings, electrical connectors, and metal fittings, pieces of wire, and small nuts, bolts can become trapped between fuel rods in a spacer where hydraulically induced vibrations can cause fretting failure of the fuel rod. Efforts are made by Fuel Engineering Group to evaluate debris-fretting failures on PWR fuel. The inquiries on fuel rods failures are based on results of analysis using post-irradiation non-destructive examination. The complementary analysis includes a modeling approach by DEGRAD-1 code to characterize the degradation phenomenon after primary failure integrated in the reactor operational history.

Evaluation of irradiation damage of structural materials

It is a know fact that radiation damage of PWR structures is a time limiting factor on its operation, the study of detrimental radiation effects on materials structures is one of the objects of research at IPEN. The R&D activities are basically performed in two fields:

- Establishment of an experimental infrastructure, by designing and fabricating the materials irradiation device CIMAT (Materials Irradiation Capsule), with controlled atmosphere and temperature (from 25 to 700OC) allowing for in-pile irradiation and on-line acquisition of parameters of interest. Priority is given to post irradiation evaluation to be performed by means of mechanical methods: creep, hardness, micro hardness, strain-stress, etc., at fluence of 1019 nvt, which represents a critical dose for the reactor pressure vessel (RPV) steels. Operational tests of CIMAT have been completed and a modified device is under construction. Another device CIMAT 2 is under completion;

- Pre-irradiation characterization has been performed on samples of RPV steel AS 508 cl.3, of two origins: one from ELETROMETAL (Brazil AE) and the other from VITKOVICE (Czech AV), in as delivered condition. The samples were studied by means of metallographic, TEM, SEM, hardness, internal friction and X-ray diffraction. The samples show martensitic (AE) and bainitic (AV) structures, due mainly to different heat treatments. Rockwell (60 fkg) and Vickers (1 fkg/mm2) hardness data provide a maximum toughness of 600-630 MPa, which is in good agreement with the suppliers information. This characterization will be repeated after an annealing at 300OC, corresponding to the time internal of irradiation of the samples at 300OC (approx. 300 h), to equalize the thermal state of the material.

Effect of alloying and heat treatment on structure and mechanical properties of cobalt-free corrosion-resistant maraging steel

Effect of alloying and heat treatment on structure and mechanical properties was investigated in economic alloyed cobalt-free maraging steel with increased corrosion-resistant. These R&D activities counted on a technical cooperation between the Fuel Engineering Group and the St-Petersburg State Marine Technical University researchers. The developed and tested experimental steel present characteristics of a martensite class steel with increased strength provoked for the presence of carbon and intermetallics. The optimal regimen of thermal treatment regarding to the maximum strength is attained when the samples are submitted to quenching at 1050°C for 1 hour and aging to 520°C for 5 hours. Triple-austenitizing provides sufficient homogeneous grain structure. High-temperature microscopy verifies this observation. At 560°C for times higher than 3 hours, the presence of reverted austenite in the microstructure, confirmed by X-ray Diffraction measurements, causes a continuous decrease in the strength (hardness). Scanning Electron Microscopy of the fracture surface gives additional evidence about reduction deformation capacity of the steel. The present results constitute the initial phase of the study. In the next one, the steel will be radiated with neutrons fast in IEA-R1 Reactor, in high temperature and controlled atmosphere, simulating the conditions of work of power reactors, and using the device CIMAT.

Structural analysis & integrity assessment and life management of nuclear components

The research in this area was focused mainly in the Steam Generator Tubing Integrity Assessment, some aspects of the project by analysis covered in the nuclear standards and Life Management and Life Extension of PWR NPP. The continuous safe operation of a SG requires a tube integrity analysis at every outage with, in general, 100% of the tubes being inspected NDE techniques like the Eddy Current one. All uncertainties, like those in the material strength, defect dimensions, etc., should be considered to assess the structural limits of the SG tubes in terms of critical defect dimensions. Usually this is done using a Monte Carlo statistical approach. To assure that the operation did not challenge the adopted criteria based on regulatory limits, some defect can be tested in situ and if it passes the test the criteria remains valid for future assessments. Otherwise the criteria should be revised. Within the scope of this research area, several studies have been performed. Among them can be mentioned: the Structural Analysis of the main components of a Nuclear Power Plant, as well as Research Reactors, like the inlet nozzle of the new IEA-R1 Heat Exchanger; the evaluation of the stress concentration factor in Y fittings, the evaluation of stress intensity factors KI and KII in an postulated inclined crack within a weld for a particular geometric pressure vessel configuration; the continuous development of in-house programs to establish the critical defect dimensions in Angra-1 SG tubing.

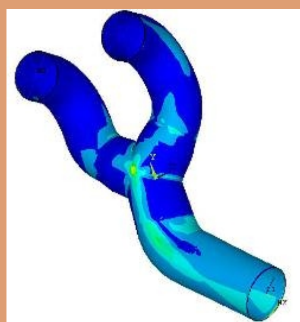


Figure 7. Stress analysis of a Y fitting according to ASME Section III, Division 2

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

The requirements for the qualification of transportation and/or storage packages for nuclear research reactors spent fuel elements are quite severe. Among the postulated accident conditions to be analyzed there is the 9 m free drop on a rigid surface. To avoid high decelerations in the package content, which could jeopardize the required nuclear safety standards, it is allowed to use impact limiters made with energy absorbing materials which will absorb the kinetic energy by means of a great amount of deformation. Their destruction is allowed during this process. The package itself has several components connected to each other in different ways (welded parts, flanged connections, surface contacts, etc.). This research involves other groups in Brazil and abroad (mainly, Argentina) and it is sponsored by AIEA. In its first phase a scale model was designed, it is being constructed and will be tested. Some material tests and properties characterization were already performed as well as some numerical analyses. All analyzes, were and will be performed, using the finite element method and explicit algorithms to simulate the 9 m free drop test condition, to show the worst drop position (the one that causes the maximum damage in the package). Some of the considered aspects are: the materials properties nonlinear modeling, the different stiffness of the package materials, 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.

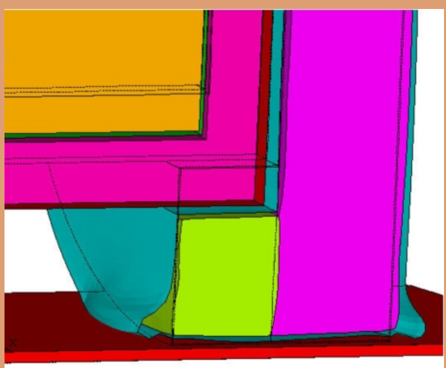


Figure 8. Final deformed for the corner impact (at 45º)

Monitoring and diagnosis of reactor systems and components

Since 2000, the Nuclear Engineering Center is developing techniques for the monitoring and diagnosis of nuclear systems and components using advanced methods for data processing and analysis. This work has been supported by two IAEA technical cooperation contracts being one of them a Latin American regional cooperation project and three CNPq research projects being one of them a bilateral agreement in partnership with the National Science Foundation/University of Tennessee. The total budget up to date amounts to about a million reais. The general idea of this project is to improve reliability, safety and availability of critical components and systems of a nuclear power plant thus also allowing for the plant life extension. Well establish techniques were used as well as advanced methods for data processing such as artificial intelligence, time-frequency domain transforms and higher order statistics were studied. The main activities are summarized bellow:

Plant condition monitoring and fault detection
During the last years, model based fault diagnostics methods have received increasing attention in both research and application. This approach is based on the concept of analytical redundancy as opposed to physical redundancy (by hardware) that uses measurements from redundant sensors for fault diagnosis purpose. In this work, the system analytical model is obtained by using the Group Method of Data Handling (GMDH) which uses data from the reactor instrumentation to generate a normal condition model for different operational conditions, a fault free database. The GMDH is a form of artificial intelligence and is based on a sequential learning networks which are networks of mathematical functions that characterize complex, nonlinear relationships in a compact and rapidly executable form. The method was tested successfully in the IEA-R1 research reactor instrumentation and control system.

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.

Leak detection and localization in pressurized tubes and vessels
Leaks in pressurized tubes and vessels generate acoustic waves that propagate through the walls of these tubes, which can be captured by

accelerometers or by acoustic emission sensors. In this work an analytic model was implemented, through the solution of the equations of motion of a cylindrical shell, with the objective to understand the behavior of the tube surface excited by a point source. The theoretical results were validated through measures accomplished in an experimental setup composed of a steel tube ending in sand boxes, simulating the condition of an infinite tube. To determine the location of the point source on the surface, the process of inverse solution was adopted.

Monitoring and automated diagnosis of rotating machinery
Rotor unbalance, shaft misalignment, and bearings faults are usually detected using conventional frequency domain Fourier transform of the accelerometers signals. In this work we tested the use of Wavelet transforms to obtain new characteristics for the defects with improved diagnosis capabilities. The ability to differentiate localized defect from distributed corrosion type defect is one of the advantages. Also, in order to automate the diagnosis decision a para consistent fuzzy logic engine were developed to process the several characteristics and to indicate a defect automatically. The methodology was tested in a rotating machine defect simulator and the results are quite satisfactory when compared with actual data from IEAR1 research reactor.

Development of an artificial neural network for monitoring and diagnosis of sensor fault and detection in the IEA-R1 research reactor at IPEN-CNEN/SP

The increasing demand on quality in production processes has encouraged the development of several studies on Monitoring and Diagnosis Systems in industrial plant, where the interruption of the production due to some unexpected change can bring risk to the operator's security besides provoking economic losses, increasing the costs to repair some damaged equipment. Because of these two points, the economic losses and the operator's security, it becomes necessary to implement Monitoring and Diagnosis Systems. In this work, a Monitoring and Diagnosis Systems was developed based on the Artificial Neural Networks methodology. This methodology was applied to the IEA-R1 research reactor at IPEN. The development of this system was divided in three stages: the first was dedicated to monitoring, the second to the detection and the third to diagnosis of failures. In the first stage, several Artificial Neural Networks were trained to monitor the temperature variables, nuclear power and dose rate. Two databases were used: one with data generated by a theoretical model and another one with data to a typical week of operation of the IEA-R1 reactor. In the second stage, the neural networks used to monitor the variables were tested with a fault database. The faults were inserted artificially in the sensors signals. As the value of the maximum calibration error for special thermocouples is ± 0,5^{o}C, it had been inserted faults of ± 1^{o}C in the sensors for the reading of the variables T3 and T4. In the third stage a Fuzzy System was developed to carry out the faults diagnosis, and it was considered three conditions: a normal condition, a fault of -1^{o}C, and a fault of +1^{o}C. This system will indicate which thermocouple is faulty.

Time response measurements of temperature and pressure sensors of Angra I nuclear power plant

The Instrumentation, Monitoring and Diagnosis group of IPEN-CNEN/SP currently performs time response measurements of temperature and pressure sensors of Angra-I nuclear power plant protection system. Response time test is a regulatory requirement and it need to be performed in each outage for refueling or in case of one transmitter need to be replaced. Sensor time response measurement represents an important requirement to be observed in the maintenance of a nuclear reactor protection system. This measurement has to be performed periodically to ensure that the protection reactor limits are respected.

Loop Current Step Response - LCSR
The methodology for temperature time response measurements is the Loop Current Step Response LCSR, used to determine time constant for Resistance Temperature Detector-RTD sensors. The test consists in applying a small current to the RTD leads that heats the sensor filament and the temperature transient due to a step change is analyzed to determine the response time that would have followed a fluid temperature change. The LCSR data gives the sensor response of an internal heating perturbation, but the response of interest is the one that results from a fluid temperature perturbation. An analytical transformation was developed to predict the response to a fluid temperature perturbation by using information from the LCSR data record.

Noise Analysis
The methodology used for pressure sensor time response measurements is the Noise Analysis methodology. Random noise techniques for measurements on nuclear reactor systems have been developed as a tool for system surveillance or to analyse dynamic behaviour with a minimum of interference during normal operation. Random variations in the neutron flux density, temperature, steam flow or pressure may be used to derive useful information about the system dynamics or to monitor sensor characteristics. The frequency domain method consists of evaluating the Power Spectral Density (PSD) function. The information needed for time constant estimation can be obtained by fitting an all-pole transfer function to this power spectral density. If the system has only one dominant pole, then the time constant of interest can be obtained from the break frequency of the Power Spectral Density curve. Sensors' signals are filtered and amplified before acquired using a data acquisition system connected to a PC (Figure 9).

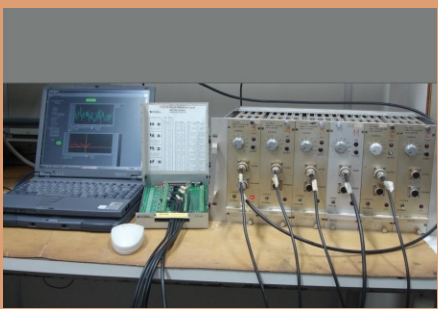


Figure 9. Isolating amplifiers used to data pre-processing in the noise analysis methodology

Development of a wide-range reactor power measuring channel

The main objective of this research is the development of a digital wide-range reactor power measuring channel (hardware and software) to be used in control and protection systems for nuclear research reactors. The system is composed of two neutron detectors (pulse and current modes) arranged to cover 10 decades of reactor neutron flux and associated electronics. The conditioned signals are processed by distributed microcontrollers. Hardware and software tests are currently running and the estimated date for experimental installation in IPEN/MB-01 research reactor is set for end of 2008.

Using cumulants and spectra to model nuclear radiation detectors

A general mathematical methodology is presented to model a nuclear radiation detector. This is accomplished by using a proposed generalization of Campbell's theorem, which employs nth-order cumulants and spectra analysis and a vector of random parameters to describe the current pulses. This allows a more elaborate, higher order statistical characterization of the radiation detection process, as compared to the usual second order treatment. To demonstrate the method, it is applied to a neutron ionization chamber (fission or /sup 10/B). A deterministic model for the chamber current pulses is developed and their probability density functions are used to define the characteristics of the vector of random parameters. Computer simulations of the proposed theoretical methodology were developed and potential applications suggested.

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 CEN-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 CEN-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 utilization of Reactor IPEN-MB/01

The first criticalization of the IPEN/MB-01 reactor (Figure 6) was obtained on November 9th, 1988. Since this date more than 2150 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 configurations (Benchmarks to OECD - NEA). 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. Beyond experimental purposes the IPEN/MB-01 reactor is used also to reactor operators training and educational purposes (Graduated and Post-graduated).

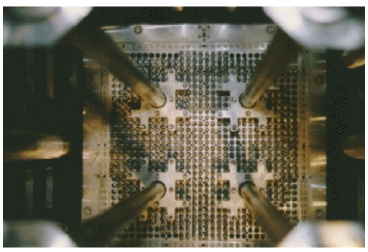


Figure 10. IPEN/MB-01 Reactor core

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 using economics resources of the Centro de Engenharia Nuclear (CEN). Normally are reserved until 15 weeks by year to maintenance of the IPEN/MB-01 Reactor. Basically the reactor has two kinds of activities: Operational and Experimental. 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 partners (CTMSP-Brazilian Navy).

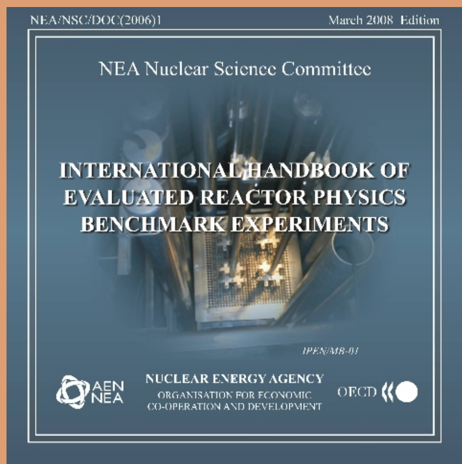


Figure 11. International Handbook of Evaluated Reactor Physics Benchmark Experiments Containing IPEN/MB-01 Reactor Experiments

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 other IPEN centers 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 with other institutions that already have tradition in this area. During the last three 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. In fact, the origin of the phenomenon of the global warming (intensive energy use), and the global social deficit, is the incentive for the demand of products and services that are not necessary to a worthy life of the humanity (consumerism). It is observed that the benefits for these products and services go to a lowermost part of the world-wide population, widening each time more the abyss that separates the supplied populations of the great majority of the poor persons and that, in not rare times, do not make use of the minimum for the proper survival. Another point is the study of the emergy methodology and its scientific fundamentals that takes into account the embodied energy during a transformation process, making possible to measure the nature contribution, until now considered free, in the same calculation basis, such method comes to increase the resources utilized to evaluate the natural systems sustainability. 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 considering activities directly related to cores of nuclear reactors, waste management, thermal and hydraulics and safety analysis of plants and systems. The technical expertise of the professionals in the above specialties is directed to solve problems related to conceptual design, operation, nuclear fuel design and performance evaluation, life extension, reliability and safety analysis. The following areas were covered: thermal and hydraulics, nuclear fuels, safety assessment, reactor physics, training of operators, and plant engineering. Complementing the services described in the nuclear engineering field, other specialized technological services were also performed in design and analysis of systems and equipments of nuclear reactors, fuel cycle installations and cogeneration power plants. The services were directed to solve technical problems related to conceptual design, operation, aging of components, in service inspection, and so on. The background acquired is also applied to conventional thermal and general process plants. The following areas were covered: instrumentation and control, electricity, structural mechanics, conventional power plants, calibration and metrology, and fuel cell. All the services and specialized services described above were conducted by a team of 40 (forty) professionals that are working at the Nuclear Engineering Center. A total of approximately 30.000-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), the Brazilian nuclear utility (named Eletronuclear) and the Brazilian navy technological centre (name CTMSP). All the engineering services provided by the Nuclear Engineering Center are certified with the ISO 9001:2000.

In the triennial 2005/2007 the IEA-R1 Research Reactor has been operated most of the time at a reactor power of 3.5 MW and the reactor operation schedule of 62 hours per week. The Operation and Maintenance Area of the IEA-R1 Research Reactor is composed by three groups as follows:

- operation group
- maintenance group
- technical support group

Operational group has a crew of 23 operators responsible for the reactor operation, placement and disposal of irradiation samples in the reactor core, tests, experiments and control of operational documents. Maintenance group has 9 technicians for the preventive and corrective reactor systems maintenance, study of ageing and maintenance registrations. Technical support group, formed by two persons, is dedicated to neutronics and termohidraulics calculations. In this triennial, to given an example of previous years, several changings in order to achieve reactor modernization has been done. These works are listed below:

New heat exchanger acquisition and installation
The heat exchanger new project was developed by Ipen nuclear engineering group. The construction and installation was carried by Brazilian company IESA. The old equipment removal and assembly of the new one were followed by the reactor operation and maintance group, leased by the Ipen's Radiological Protection.

Figure 12. New heat exchanger

New irradiation pneumatic system construction
The old system has presented infiltration and contamination by the swimming pool water. The new system was projected and constructed by the Ipen technicians.

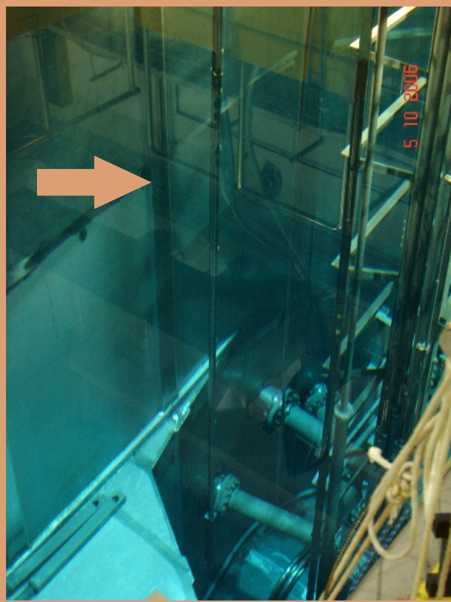


Figure 13. New irradiation pneumatic system

The start of the construction of pneumatic system between the reactor and radiopharmacy center
This system will allow to send samples from the reactor swimming pool hall to the radiopharmacy center cells of processing, avoiding the use of shields. With this, time will be saved and the level of radioisotope activity will be increased.

Restoration of the building reactor wall
After 50 years, the wall concrete has presented craks, hardware and internal humidity. The wall restauration allowed the correction of the problems in the concrete, waterproofing and painting.

Figure 14. IEA-R1 Research Reactor

Transport of 33 reactor spent fuel elements to United States
This task has continued the transportation of spent fuel elements to Savannah River Site Laboratory in South Carolina In 1999, it has been transported 127 fuel elements to United States. In november 2007 has been transported 33 elements. In this operation it has been used one LWT cask, developed by an american company NAC.

Figure 15. transportation of spent fuel elements

Acquired equipaments by AIEA program
- Spectrometer of HPG;
- 2 fission chambers;
- Digital personal dosemeters;
- SPND detector (flux detection);
- Gateway radiation protection monitor;
- 2 ducts radiation detectors (cintilator);
- 2 area radiation detectors (gieger muller);
- Lead viewfinder;
- 2 ionization chambers (flux detection);
- Surface contamination monitor;
- Contamination detector;
- Vibration system (primary pump monitoration).

Monthly, the Reactor Operation and Maintenance Area issues a report containing the activities of the reactor. This report encloses the number of reactor operation schedule, energy dissipated in this period, reactor core configurations, chemical and physical characteristics of the water systems, 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. The main operational parameters of the triennial 2005/2007 were:

- Reactor power: 3,5 MW
- Time of reactor operation in hours: 7850.91
- Energy dissipation in MWh: 24823.33
- Time of low power operation in hours: 76.06
- Number of samples irradiated in the grid plate: 3309
- Number of samples irradiated in the pneumatic system: 319
- Number of new core configurations: 14

In 2002 the Research Reactor Center (CRPq) started the process of implementation of a quality management systems (QMS) in compliance to NBR ISO 9001 version 2000. This process resulted in the accreditation of the Operation and Maintenance of the IEA-R1 Reactor and Irradiation Services by Fundação Carlos Alberto Vanzolini on December 13, 2002. Since then, the reactor has operated with this certification, which is renovated every year by external and internal audit.

Nuclear Reactors Program Team

Research Staff

Dr. Adonis Marcelo S. Silva; Dr. Aucyone Augusto da Silva; Dr. Adimir dos Santos; Dr. Álvaro Luiz G. Carneiro; Dr. Antonio Teixeira e Silva; Dr. Benedito D. Baptista Filho; Dr. Carlos Alexandre J. Miranda; Dr. Celso Antonio Teodoro; Dr. Daniel Kao Sun Ting; Dr. Delvonei A. de Andrade; Dr. Eduardo Lobo L. Cabral; Dr. Eduardo Winston Pontes; Dr. Elita Urano de C. Frajndlich; Dr. Fábio Branco Vaz de Oliveira; Dr. Gaianê Sabundjian; Dr. Hélio Yoriaz; Dr. Iraci Martinez P. Gonçalves; Dr. José Rubens Maiorino; Dr. Luiz Antonio A. Terremoto; Dr. Luiz Antonio Mai; Dr. Michelangelo Durazzo; Dr. Miguel Mattar Neto; Dr. Myrthes Castanheira; Dr. Paulo Henrique F. Masotti; Dr. Paulo Rogério P. Coelho; Dr. Paulo Tarso D. Siqueira; Dr. Ricardo Diniz; Dr. Roberto N. de Mesquita; Dr. Thadeu das Neves Conti; Dr. Tufic Madi Filho; Dr. Ulysses d´Utra Bitelli; Dr. Valdemir Gutierrez Rodrigues; MSc. Alfredo José A. de Castro; MSc. Alvaro Ikuta; MSc. Antonio S. Vieira Neto; MSc. Arivaldo Vicente Gomes; MSc. Carlos Alberto de Oliveira; MSc. Custodio A. Guimarães; MSc. Eduardo Maprelian; MSc. Gerson Antonio Rubin; MSc. Gerson Fainer; MSc. Gilberto Hage Marcondes; MSc. Glaucia Regina T. Santos; MSc. Graciete Simões de A. Silva; MSc. Jean Claude Bozzolan; MSc. José Eduardo R. da Silva; MSc. José Patrício Náhuel Cárdenas; MSc. José Roberto Berretta; MSc. Leda Cristina C.B. Fanaro; MSc. Leslie de Molnary; MSc. Luiz Alberto Macedo; MSc. Marcos Rodrigues de Carvalho; MSc. Margaret de Almeida Damy; MSc. Maria Alice M. Ribeiro; MSc. Maria Eugênia L. J. Sauer; MSc. Miriam Aparecida Cegalla; MSc. Mitsuo Yamaguchi; MSc. Nicolau Dirjawoj; MSc. Patricia da S. P. Oliveira; MSc. Paulo Roberto B. Monteiro; MSc. Pedro Ernesto Umbehaun; MSc. Rita Izabel Ricciardi; MSc. Roberto Frajndlich; MSc. Rosane Napolitano Raduan; MSc. Rosani Maria L. da Penha; MSc. Sérgio Marcelino; MSc. Wageeh Sidrak Bassel; MSc. Walmir Máximo Torres; MSc. Walter Pereira; Msc. Walter Ricci Filho; Tech. Aristeu Florêncio da Silva; Tech. Davilson G. da Silva; Tech. Edeval Vieira; Tech. Edvaldo Dal Vecchio; Tech. Eliezer Silas Bertelini; Tech. Eneas Tavares de Oliveira; Tech. Felipe B. Jaldin Ferrufino; Tech. Ilson Carlos Martins; Tech. Ivo Oliveira de Jesus; Tech. João Batista da Silva Neto; Tech. João Lopes de Araújo; Adilson Aboláfio; Tech. Jorge Clementino dos Santos; Tech. José Carlos de Carvalho; Tech. José Marcos F. da Silva; Tech. José Maria Fidelis; Tech. José Vicente Pereira; Tech. Marinete Nóbrega da Silva Moraes; Tech. Olair dos Santos; Tech. Raimundo Rodrigues da Silva; Tech. Reinaldo Aparecido da Costa; Tech. Sebastião Silva Macedo; Tech. Sérgio Rabello; Tech. Valdeci Ap. F. da Costa; Altair Antonio Faloppa; Antonio Rodrigues de Lima; Ary Pereira Junior; Cesar Veneziani; Cristina Oscrovani Leandro; Eduardo Pinto Kurazumi; Elza de Fátima Pinto; Fernando Fornarolo; Francisco Felix de Figueiredo; Gelson Toshio Otani; Haruyuki Otomo; José Antonio Batista de Souza; José Carlos de Almeida; José Francisco Bistulfi; Marcio Simioni; Marcos Afonso Bissa; Maria Aparecida S. Macedo; Orlando Nogueira da Silva; Paulo Alves Costa; Roberto Carlos dos Santos; Rogério Jerez; Samuel Carracioli Santos; Sérgio Ricardo P. Perillo; Sílvio Carlos Menzel.

Graduate Students

Alberi Anunes; Aretha Sanchez; Carlos Eduardo Maria de Bedia; Claudio Souza do Nascimento; Daniel Massami Hirata; Fernando Luiz de Oliveira; Fabio Raia; Ivan Dionísio Arone; José Gomes Neto; José Luís Martines Morales; Lucas Batista Gonçalves; Maria da Penha Sanches Martins; Maria Ely de Fatima Carbonari Zana; Miréia Florêncio Maio; Osmar Conceição Junior; Paulo Roberto de Andrade Marchesini; Paulo Sadocco de Camargo; Pedro Carlos Russo Rossi; Robert Joseph Didio; Rui Yoshio Nishizawa; Seung Min Lee; Silvia Regina Vanni; Thiago Carluccio; Valdeci Carneiro Junior; Wagner Luiz Andreassa; Fabio Paolini; Juan Y. Zevallos Chaves; Maria Clara Carvalho da Silva; Renato Yoschi Ribeiro Kuramoto.

Undergraduate Students

Anderson Hiroshi Kuwahara; Cesar Augusto Tafner Bitelli; Dayane Faria Silva; Djalma Rodrigues Filho; Diogo Satoru Aihara; Felipe Vieira Aburaya; Fernanda Nunes Lima da Rocha; Francine Menzel; Gabriel Rolim de Melo; Gabriel Paiva Fonseca; Gregório Soares de Souza; Jefferson Francisco do Nascimento; José Ribeiro de Almeida; José Sergio Bleckmann Reis Junior; Luis Felipe Liambos Mura; Mauro Ferreira da Silva Filho; Pedro de Campos Costa; Rafael Cavalcanti de Souza; Rafael de Oliveira Rondon Muniz; Ricardo Separovic Seerban; Vinicius Alexandre de Castro.