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Spent fuel management options for research reactors in Latin America

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FOREWORD

Research reactors (RRs) have been operated in Latin America since the late 1950s, and a total of 23 RRs have been built in the region. At the time of writing (November 2005), 18 RRs are in operation, 4 have been shut down and 1 has been decommissioned. The number of operating RRs in Latin America represents around 6% of the existing operational RRs worldwide and around 21% of the RRs operating in developing countries. Common to all RRs in the region is a consistent record of safe and successful operation.

With the purpose of carrying out a collaborative study of different aspects of the management of spent fuel from RRs, some countries from the region proposed to the IAEA in 2000 the organization of a Regional Project. The project (IAEA TC Regional Project RLA/4/018) that was approved for the biennium 2001–2002 and extended for 2003–2004 included the participation of Argentina, Brazil, Chile, Mexico and Peru.

The main objectives of this project were: (a) to define the basic conditions for a regional strategy for managing spent fuel that will provide solutions compatible with the economic and technological realities of the countries involved; and (b) to determine what is needed for the temporary wet and dry storage of spent fuel from the research reactors in the countries of the Latin American region that participated in the project.

This TECDOC is based on the results of TC Regional Project RLA/4/018. This project was successful in identifying and assessing a number of viable alternatives for RRSF management in the Latin American region. Options for operational and interim storage, spent fuel conditioning and final disposal have been carefully considered.

This report presents the views of Latin American experts on RR spent fuel management and will be useful as reference material for the Latin American RR community, decision making authorities in the region and the public in general.

In publishing this TECDOC, the IAEA wishes to thank all the experts participating in project RLA/4/018. Special thanks are also due to A.M. Bevilacqua and O.E. Novara, who carried out a thorough technical revision of the different versions of the manuscript, and A.J. Soares, who contributed to the final edition and straightening of the text. The project officer for RLA/4/018 was J.M. Guarnizo of the Division for Europe, Latin America and West Asia of the Department of Technical Cooperation. The IAEA technical officers responsible for this report were P. Adelfang and I.G. Ritchie of the Division of Nuclear Fuel Cycle and Waste Technology.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND, PURPOSE AND SCOPE OF THE REPORT

The countries of Latin America, such as Argentina, Brazil, Chile, Mexico and Peru, have the responsibility of safely and securely managing the spent fuel from their research reactors that have been in operation for several decades. These countries share the view that it is essential and opportune to begin the evaluation of options for interim storage and disposal of their spent fuel, including any derivatives, if processing is one of the chosen options. These facts were the driving force for initiating the IAEA Technical Cooperation (TC) Regional Project RLA/4/018, “Management of Spent Fuel from Research Reactors”.

The main objective of TC Regional Project RLA/4/018 was to define the basic conditions for a regional strategy for managing research reactor spent fuel, providing solutions within the economic and technological realities of the countries involved, and in particular, to determine the needs for the temporary wet and dry storage of spent fuel from the research reactors in the countries that participated in the project.

To accomplish these objectives, the project was organized into four main activities: spent fuel characterization, nuclear safety and regulation, options for spent fuel management and public communication strategies.

The spent fuel characterization activity aimed at the implementation of a surveillance programme for research reactor spent fuel in interim storage in every participant country, including methodologies to perform: water quality control, visual inspection, sipping tests, corrosion monitoring by coupon assessment, and non destructive burn-up tests. Besides, within this activity, a common database on inventories of fuel in the region was also developed, for use by the participant countries and the IAEA, and a burn-up exercise was performed in order to intercompare the codes and methodologies used by the reactor physicists who are responsible for fuel management in their respective countries.

The nuclear safety and regulation activity sought the homogenization of safety criteria for research reactor spent fuel management activities, such as transport and interim storage, within the region. It aimed at the formalization of regulatory documents, compatible with the regional realities, to ensure agreement with international standards and guidelines, mainly those of the IAEA.

The option for spent fuel management activity dealt with the identification of technically viable possibilities for research reactor spent fuel management, particularly for operational and interim storage (wet and dry), spent fuel transport (development of a dual purpose cask) and spent fuel conditioning. Moreover, it aimed to be the main source of information for the design and implementation of related facilities.

The public communication strategies activity was assumed to be a way of implementing institutional communication strategies associated with spent fuel management in the region, adapted to the diverse sites and possible scenarios, to improve the public’s perception and to help gain public acceptance that will allow the safe and secure management of spent fuel from the research reactors in the region.

Since the core of TC Regional Project RLA/4/018 was the identification and assessment of options for spent fuel management, including future disposal, this report presents those options that are supported by the technical and scientific background of the nuclear community from the participating countries.

These options were presented, in view of the economic realities, in the national, regional and extra-regional contexts for those countries that participated in TC Regional Project RLA/4/018 and which currently operate nuclear research reactors for radioisotope production, fundamental research in physics and biology, materials irradiation, education and training.

This report describes the expert opinions on all technically feasible options available, without excluding or failing to evaluate any option that might be impossible because of current national laws of the participating countries at the time of writing.

It is understood that it may be approximately 50 years before there will be any deep geological repository available in the region, and in the interim there will certainly be technological advances, political and legal changes, as well as transformation of public attitudes that may become favourable for regional options to be implemented. Therefore, this publication aims to be a technical guide for decision makers of the Latin American region, and it is expected that it could be revised and updated at least every five years.

This report took into account the management of spent fuel only from research reactors in operation in Argentina, Brazil, Chile, Peru and Mexico. Therefore the analysis was restricted to the national, regional and extra-regional context of only these countries.

Critical and subcritical facilities with low power, of 100 W or less, usually are considered separately because they generate a negligible amount of spent fuel due to their very low burn-up. Nevertheless, the fuel from such facilities may be regarded as spent fuel at decommissioning and its ultimate fate, either refabrication into fuel for higher power research reactors after reprocessing or disposal after conditioning, is considered in this report.

1.2. DEFINITIONS

For the purpose of this document, the definitions below will be adopted. Some definitions are adopted from the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [1]. If no appropriate definition is found in the above reference, the definitions are adopted from the IAEA's Radioactive Waste Management Glossary, 2003 Edition [2], the Safety Guide on Classification of Radioactive Waste [3] and the IAEA Safeguards Glossary, 2001 Edition [4]. Some other definitions are adapted from the mentioned documents.

Cask (see Storage cask).

Concrete canister (or Silo). A massive container comprising one or more individual storage cavities. It is usually circular in cross-section, with its long axis vertical. An inner sealed liner and the massive concrete of the canister body provide the necessary containment and shielding of the radioactive material inside the container. Heat removal is accomplished by radiant emission, conduction and convection within the body of the canister, and by natural convection on its exterior surface. Canisters may be located in enclosed or non-enclosed areas.

Disposal. The emplacement of spent fuel or radioactive waste in an appropriate facility without the intention of retrieval.

Dry storage. The storage of fuel assemblies and fuel elements and related components in a gas environment, such as air or an inert gas. Dry storage facilities include the storage of spent fuel in casks, silos and vaults. Dry storage includes the dry canning of fuel subsequently stored underwater for biological shielding (sometimes referred to as semi-wet or semi-dry storage).

Fuel assembly. Fuel elements and associated components that are installed as a single unit in the reactor core and are not disassembled during installation and removal from the reactor core.

Fuel elemen.t A component of the fuel assembly that consists primarily of the nuclear fuel and its cladding materials, e.g. individual rods, tubes or plates of an assembly.

HEU (high enriched uranium). Uranium enriched to a level of at least 20% of the isotope ^{235}U.

HLW (high level waste). The radioactive liquid containing most of the fission products and actinides present in spent fuel — which forms the residue from the first solvent extraction cycle in reprocessing — and some of the associated waste streams; this material following solidification; spent fuel (if it is declared a waste); or any other waste with similar radiological characteristics. Typical characteristics of HLW are thermal powers of about 2 kW/m^3 and long lived radionuclide concentrations exceeding the limits for short lived waste [2].

Interim storage. The storage of spent fuel and related components such that isolation, monitoring, environmental protection and human control are provided until it is retrieved for further processing or direct disposal.

LEU (low enriched uranium). Uranium enriched to a level below 20% of the isotope ^{235}U.

LILW (low and intermediate level waste.- Radioactive waste in which the concentration or quantity of radionuclides is above clearance levels established by the regulatory body, but with a radionuclide content and thermal power below those of high level waste. Low and intermediate level waste are often separated into short lived and long lived wastes. Short lived waste may be disposed of in near surface disposal facilities. Plans call for the disposal of long lived waste in geological repositories [3].

LILW-LL (long lived low and intermediate level waste). LILW with a long lived radionuclide concentration that exceeds the limit for short lived LILW (long lived alpha emitting radionuclides limited to 4000 Bq/g in individual waste packages and to an overall average of 400 Bq/g per waste package) [3].

LILW-SL (short lived low and intermediate level waste). LILW with a limited long lived radionuclide concentration (long lived alpha emitting radionuclides limited to 4000 Bq/g in individual waste packages and to an overall average of 400 Bq/g per waste package) [3].

LWR. Light water (power) reactor.

Radioactive waste. Radioactive material in gaseous, liquid or solid form for which no further use is foreseen and which is controlled as radioactive waste by the regulatory authority under the legislative and regulatory framework.

Research reactor. A nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiations for research and other purposes. In the context of this publication, the term research reactor also includes critical and subcritical facilities.

RRSF. Research reactor spent fuel.

Silo (see Concrete canister).

Spent fuel. Nuclear fuel that has been irradiated in and permanently removed from a reactor core regardless of burn-up.

Spent fuel conditioning. A special processing operation to prepare spent fuel for disposal.

Spent fuel management. All the activities, administrative and operational, that involve the handling, storage, treatment, transport and processing of spent fuel as well as the storage, transport and disposal of its derivatives, performed to guarantee the safety and security of the fuel. It may also involve discharges.

Spent fuel processing. Either spent fuel conditioning or spent fuel reprocessing.

Spent fuel reprocessing. A process or operation, the purpose of which is to extract radioactive isotopes from spent fuel for further use.

Spent fuel storage facilit. An installation used for the interim storage of fuel assemblies, fuel elements and related components after their removal from the reactor pool and before their processing or disposal as radioactive waste.

Storage. The holding of spent fuel, or radioactive waste, in a facility that provides for its containment, with the intention of retrieval.

Storage cask. A massive container that may or may not be transportable. It provides shielding and containment of spent fuel by physical barriers which may include the metal or concrete body of the cask and welded or sealed liners, canisters or lids. Heat is removed from the stored fuel by radiant transfer to the surrounding environment with natural or forced convection. Casks may be located in enclosed or non-enclosed areas.

Vaults. Above ground or below ground reinforced concrete buildings containing arrays of storage cavities suitable for containment of one or more spent fuel units. The containment units or the storage cavities provide the necessary shielding. Spent fuel residual heat removal is done by exhausting the existing air of the vault directly to the outside atmosphere or dissipating it via a secondary heat removal system.

Wet storage. Storage of spent fuel in water. Wet storage facilities for spent fuel are those facilities which store spent fuel in water. The universal mode of wet storage consists of storing spent fuel assemblies or fuel elements in water pools, usually supported on racks or in baskets and/or in canisters which also contain water. The pool water surrounding the fuel provides for heat dissipation and radiation shielding, and the racks, or other devices used, ensure a geometrical configuration to maintain the subcritical condition of the spent fuel distribution.

1.3. MAJOR COMMON FEATURES

Research reactors in Latin America have several common features. The first one is that with the exception of some critical facilities, all of them are water cooled, either Material Testing Reactor (MTR) or Training Research Isotope production General Atomic (TRIGA) reactor types, and, with the exception of the critical facilities and very low power reactors, wet storage is the form used for operational storage of spent fuel.

The second common feature is related to the cladding material. Most fuel assemblies were manufactured using aluminium alloys. This is the case for all MTR and for one of the two TRIGA reactors in the region. The fuel elements for the other TRIGA reactor and for the critical facilities were manufactured with stainless steel cladding. The storage options for both cladding materials are the same.

Currently, a third common feature of every research reactor in the region is that the next step after operational wet storage has not been determined. Thus, spent fuel management is open to a wide range of options on the national, regional or extra-regional scale. Therefore, the main purpose of this report is to examine and describe these options so as to facilitate the work of decision makers dealing with this issue in each country of the region.

1.4. GENERAL CONSIDERATIONS FOR SPENT FUEL MANAGEMENT

In order to avoid generation of secondary waste streams and scrap, as well as occupational exposures, it is common agreement that **spent fuel integrity should be maintained as long as possible**, until an acceptable final physicochemical form is completely and ultimately defined for disposal.

Furthermore, maintenance of fuel integrity in one spent fuel management activity adds flexibility to set the time schedule for each subsequent activity, as it allows better logistical planning. Thus, at any stage it is wise to avoid any transformation of the fuel that might jeopardize or substantially increase the cost, the operational exposure, etc., of any viable, subsequent option. For example, it would be unwise to package a fuel that is intended to be sent for processing, unless it is known that the package is acceptable to the processing facility. In any case, packaging may incur extra handling costs if the package has to be removed later.

For fuel assemblies, modification of the original geometry without exposure of the nuclear material is recommended if the purpose is volume reduction for storage or transport.

1.5. RESEARCH REACTOR SPENT FUEL INVENTORY

Table I presents the summarized RRSF inventory in the countries that participated in TC Regional Project RLA/4/018 (Update: December 2004), with the purpose of having a representative view of the existing amount of spent fuel, mass of irradiated uranium and annual volume of spent fuel arising from the reactors. This information can be used to model various spent fuel management scenarios like the one presented in Section 5.4.

TABLE I. SUMMARIZED RRSF INVENTORY IN THE COUNTRIES THAT PARTICIPATED IN TC REGIONAL PROJECT RLA/4/018

Reactor	Initial ^{235}U mass and enrichment [1]	Fuel elements in core	Spent fuel generation rate (fuel elements/year)	Spent fuel (as of December 2004)			
				At reactor	Away from reactor	Average Burn-up (% ^{235}U)	Total mass (kg)
1	12.2 g 19.7%	183	0	17 Dry	0	~ 0	^{235}U: 0.2 U: 1.0
2	357 g 19.91%	8	0	0	0	—	^{235}U: 0 U: 0
3	183.0 g 45%	22	~4	16	0	43	^{235}U: 1.7 U: 5.3
	218.7 g 19.75%	12	0	0	0	—	^{235}U: 0 U: 0
4	38 g 20%	59	~2	64	0	~10	^{235}U: 2.2 U: 12.2
	133 g 70%	29	0	0	0	—	^{235}U: 0 U: 0
5	280 g 19.75%	29	~2	5	0	53.8	^{235}U: 0.6 U: 6.4
6	278,7 19.9%	24	22 (after 2006)	35 Wet	0	31	^{235}U: 4.3 U: 29.2
7	12.2 g 19.7%	223	0	5 Dry	232 Wet	~10	^{235}U: 2.5 U: 14.2

8	280 g 19.7%	13	0	0	0	—	^{235}U: 0 U: 0
9	135 g 90%	29	0	0	0	—	^{235}U: 0 U: 0
10	12 g 4.3%	660	0	0	0	—	^{235}U: 0 U: 0
11	148.2 g 90%	—	—	—	19 Dry 91 Plates Dry	~0	^{235}U: 3.3 U: 3.6
12	290.7 g 19.7%	25	22	17 Wet	68 Wet	45.6	^{235}U: ~12 U: ~100
13	— 19.7%	—	—	—	—	—	—
14	148.2 g 90%	31	5 (after 2006)	11 Wet	—	~21	^{235}U: 1.3 U: 1.5
15	5.8 g 1.8% 11.6 g 3.6%	— —	0 0	1800 Dry 220 Dry	—	~0	^{235}U: 13.1 U: 654.9
16	37.2 g 19.8%	64	0	0	0	—	^{235}U: 0 U: 0

[1] Corresponding to the most representative fuel.

2. SPENT FUEL STORAGE

For research reactors in Latin America, with only a few exceptions, wet storage is the most common strategy currently used for storage of spent fuel. Critical assemblies and very low power reactors generate small quantities of spent fuel, and have lifetime cores; consequently, dry storage for their irradiated fuel is currently the option used.

The major requirements of a spent fuel management programme are:

- Maintenance of fuel integrity through quality control of the storage environment (the water quality in wet storage, the atmosphere in dry storage, adequate cooling, etc.);
- Use of appropriate shielding;
- Assurance of subcriticality;
- Effective physical security measures;
- Conformance with international safeguards regimes.

When discussing the options selected for operational and interim storage, no distinction regarding uranium enrichment levels is made, because enrichment does not affect and thus modify the spent fuel management strategy.

This, however, is not the case in regard to physical security. Depending on the quantity involved and whether or not the spent fuel has lost its'self-protection' due to the length of storage in water, spent HEU fuel elements are likely to require more stringent protection [5].

Technically, enrichment is taken into account in the evaluation of the options beyond interim storage, i.e. from processing to disposal, including any necessary transportation.

2.1. OPERATIONAL STORAGE

Operational storage is essentially an 'at the reactor' (AR) site activity that takes place immediately after the permanent discharge of the irradiated fuel from the core. Therefore, neither regional nor extra-regional options are applicable for this spent fuel management activity. It will be discussed only within the national context for each country.

2.1.1. Operational wet storage

The major requirement to be met immediately after a fuel assembly (or a fuel element) is permanently discharged from the reactor core is to maintain its integrity, ensuring that it is properly cooled.

Spent fuel cooling is necessary in order to remove the heat produced by the decay of the unstable fission products accumulated within the fuel element, which is a result of the fission process. Removal of the decay heat is necessary to avoid any possible spent fuel blistering, which could arise if the spent fuel were not properly cooled. Moreover, it allows for a sufficient cooling-off period before any transfer of the spent fuel to an away-from-reactor site, if needed.

In the case of research reactors, appropriate cooling can be easily maintained by introducing the spent fuel into a decay pool, either connected to or adjacent to the reactor pool, located at the reactor site. During this phase of the spent fuel management programme, known as operational storage, two additional concerns must be addressed: maintenance of optimum water chemistry for the cladding in question, and compatibility of materials.

Maintenance of optimum water chemistry is necessary in order to avoid fuel cladding degradation that occurs due to the existence of mechanisms that cause corrosion of the fuel cladding when in contact with water. Maintenance of good water quality, and low temperature, reduces the rate at which the corrosion process occurs.

Materials compatibility is also an important issue in operational storage because the contact of different metals within an aqueous medium causes galvanic corrosion of the metal with lower galvanic potential. In particular, contact between aluminium cladding and any other metal with higher galvanic potential, such as stainless steel racks or pool liners, causes localized galvanic corrosion of the aluminium cladding that can propagate and eventually expose the fuel meat, with consequent release of fission products.

It should be pointed out that such corrosion processes, and the associated degradation mechanisms, although controlled, cannot be completely avoided, especially for fuel containing aluminium alloy cladding, which is not thermodynamically stable in water. Therefore, water storage must be seen only as an acceptable option for operational and short periods of interim storage.

For the reasons mentioned above, the need for maintenance of operational wet storage as the first phase of any spent fuel management strategy is easily understood, and the main issues that should be taken into account in this phase of the spent fuel storage programme are as follows:

- There must be an ensured capacity to properly remove the decay heat produced by the fission products accumulated within the fuel element.
- Water chemistry shall be controlled in order to guarantee the spent fuel cladding integrity throughout the planned periods of wet storage.
- Materials compatibility shall be taken into account because different materials will be present in the operational wet storage pool, depending on the construction materials used. In particular, contact between cladding and any dissimilar metal, such as stainless steel racks or pool liners, should be avoided. Furthermore, many additional materials, which can play an important role in initiating localized corrosion, could be introduced into the pool throughout the reactor lifetime due to the experimental activities and/or maintenance or construction activity in the vicinity of the pool.
- Storage capacity shall be large enough in order to smoothly plan the spent fuel transfer to interim storage. It is of particular concern when the next spent fuel management activity is not fully defined, as is currently the case in all of the countries in the Latin American region.

2.1.2. Operational dry storage

The thermal power of the spent fuel shortly after it is discharged from the research reactor core is so high that it precludes operational storage in dry conditions, except as previously noted for critical assemblies and very low power reactors.

The thermal power of critical and subcritical facilities is so low that even immediately after the fuel is permanently discharged from the core, the cooling requirements to maintain integrity can be easily provided by natural convection cooling in air. Therefore, at any time the irradiated or spent fuel from critical facilities and very low power reactors can be properly stored in AR dry storage compartments, with no need for a decay pool.

2.1.3. Operational storage national context

2.1.3.1. Argentina

The main Argentine research reactor, which is dedicated to radioisotope production, is RA-3 (10 MW). It is located at Centro Atómico Ezeiza in Buenos Aires, and its spent fuel is stored in a separate decay pool, physically independent of the reactor pool, located inside the reactor building. Figure 1 shows the decay pool used for operational storage of the spent fuel assemblies of the RA-3 research reactor.

Due to the low nominal power and short operation cycles, the research reactors RA-1 (40 kW) at Centro Atómico Constituyentes in Buenos Aires and RA-6 (500 kW) at Centro Atómico Bariloche in San Carlos de Bariloche do not generate spent fuel. Nevertheless, both installations have a facility to store all the fuel of the reactor core. In RA-1 the irradiated fuel is dry stored in 24 underground concrete tubes located inside the reactor building, and in RA-6 a separate decay pool, physically independent of the reactor pool but located inside the reactor building, is used for wet storage of the fuel assemblies. Figure 2 shows the decay pool of RA-6.

The Argentine critical facilities RA-0 at the University of Córdoba, RA-4 at the University of Rosario and RA-8 at the Pilcaniyeu Technological Complex do not generate spent fuel and store their irradiated fuel in the reactor core or in dry storage racks inside the reactor building.

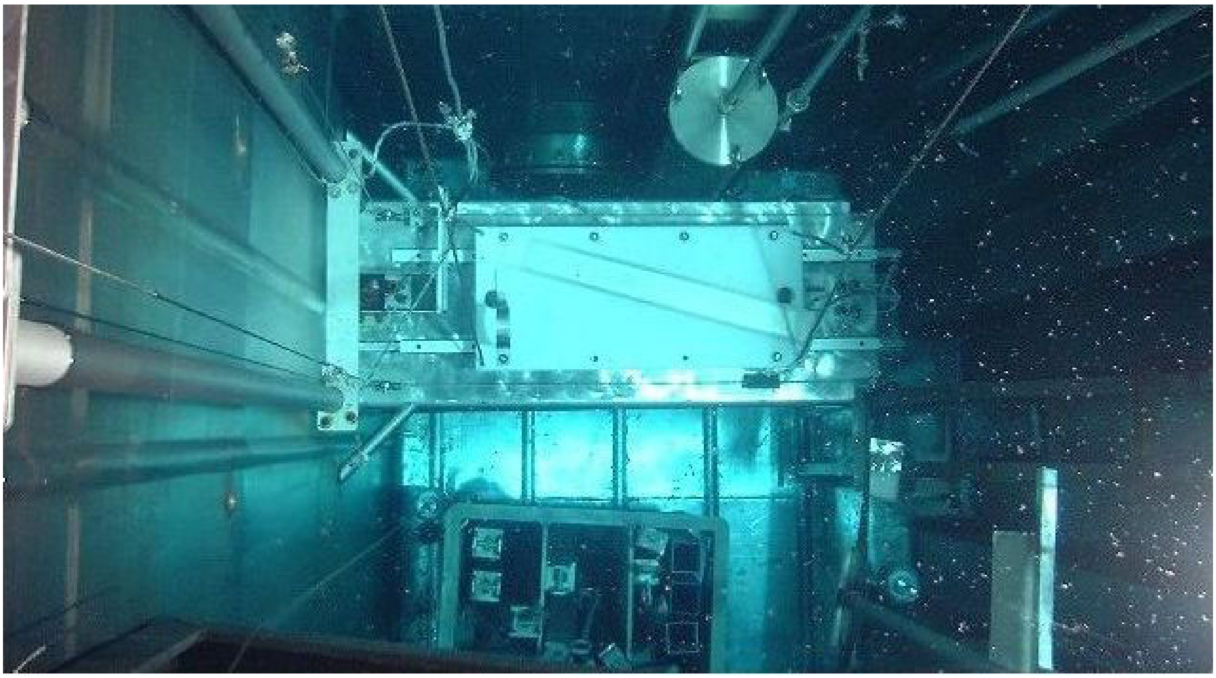


FIG. 1. Decay pool of the RA-3 research reactor (Argentina).

2.1.3.2. Brazil

Operational storage is being carried out at all four operational research reactors in Brazil: IEA-R1, IPR-R1, Argonauta and IPEN/MB-01.

At the present time, the facilities available to store spent fuel at IEA-R1 consist of storage racks located in the same reactor pool, in a decay area adjacent to the reactor. Figure 3 shows the storage racks in the decay area of the reactor pool. The racks, with a total capacity to store 156 fuel assemblies, were emptied in 1999 when 127 spent fuel assemblies were repatriated to the United States of America. The shipment was the result of a joint effort with the US Department of Energy and German companies Nuclear Cargo + Service GmbH (NCS) and Gesellschaft für Nuklear-Service (GNS), within the framework of the US "Research Reactor Spent Nuclear Fuel Acceptance Programme". At present, 35 storage positions are occupied, and 24 are required to store the fuel used in the reactor core in the case of emergency unloading, which means that only 97 positions are available for new spent fuel assemblies removed from the reactor core. Some of the storage racks are made with aluminium, but some are made with stainless steel. In order to avoid galvanic corrosion of the fuel assemblies that could occur as a result of direct contact between stainless steel and aluminium in an aqueous medium, an aluminium basket is introduced into the stainless steel racks prior to insertion of the fuel assemblies.

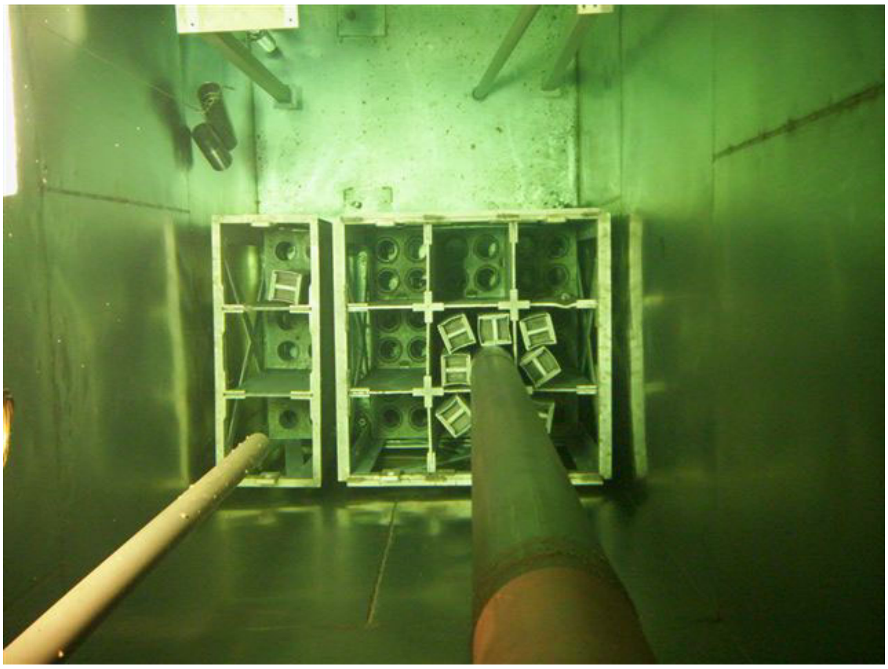
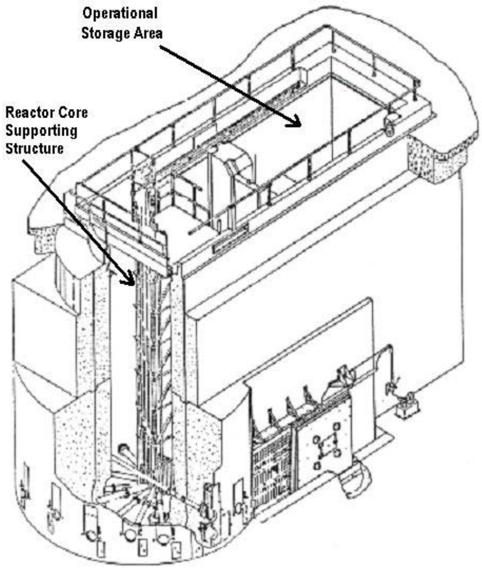
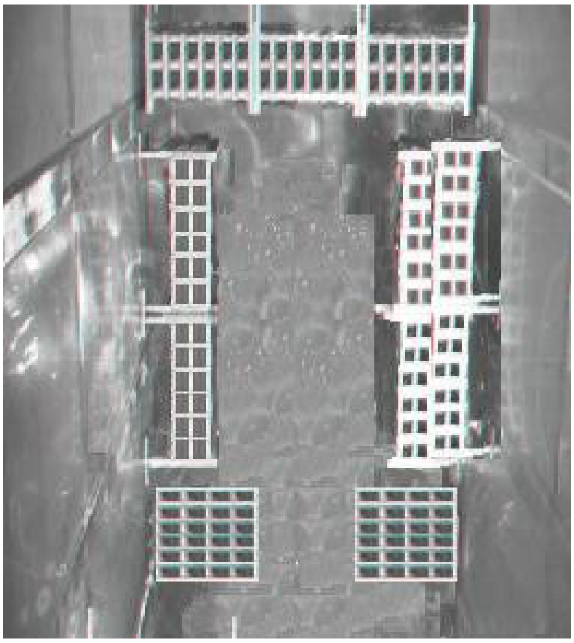


FIG. 2. Decay pool of the RA-6 research reactor (Argentina).



(a) Operational storage area



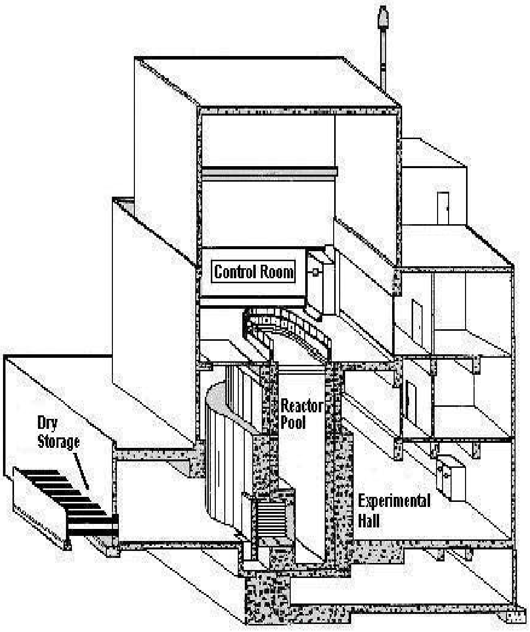
(b) Storage racks

FIG. 3. Spent fuel operational storage area at the IEA-R1 research reactor (Brazil).

In addition to the racks, there are 50 dry horizontal storage tubes (where three standard spent fuel assemblies per tube can be stored) located in the reactor building. However, significant modifications will be required before any decision to store spent fuels in these tubes can be taken. Figure 4 shows a view of the dry horizontal tubes.

As there are plans to extend the operation cycle of the reactor to achieve an expected burn-up rate of 20–22 fuel assemblies per year, it is expected that the storage capacity of the installation will be exhausted in five years of operation under these new conditions.

For IPR-R1, the integrated burn-up of the reactor since its first criticality in 1960 is about 200 MW-days. Due to the low nominal power of the reactor, except for ageing concerns, spent fuel is not considered to be problem. The first fuel replacement of the reactor is expected to occur only after the year 2010, and the reactor pool has 12 positions available, divided into two racks, to store the replaced spent fuel elements. In addition, the installation has 12 pits that were dimensioned to store a total of 72 fuel elements. The pits, made of concrete, are 3040 mm deep, have a stainless steel liner, and need to be filled with water when used for storage of irradiated fuel. To avoid galvanic corrosion that could be caused by contact of the aluminium cladded fuel elements with the stainless steel liner, each pit has an aluminium structure especially designed to receive six fuel elements. Figures 5 and 6 show details of the storage area within the reactor pool and the pits of IPR-R1, respectively.



(a) Location of the dry storage area

(b) Front wall of the dry storage area

FIG. 4. Dry storage area at the IEA-R1 research reactor (Brazil).

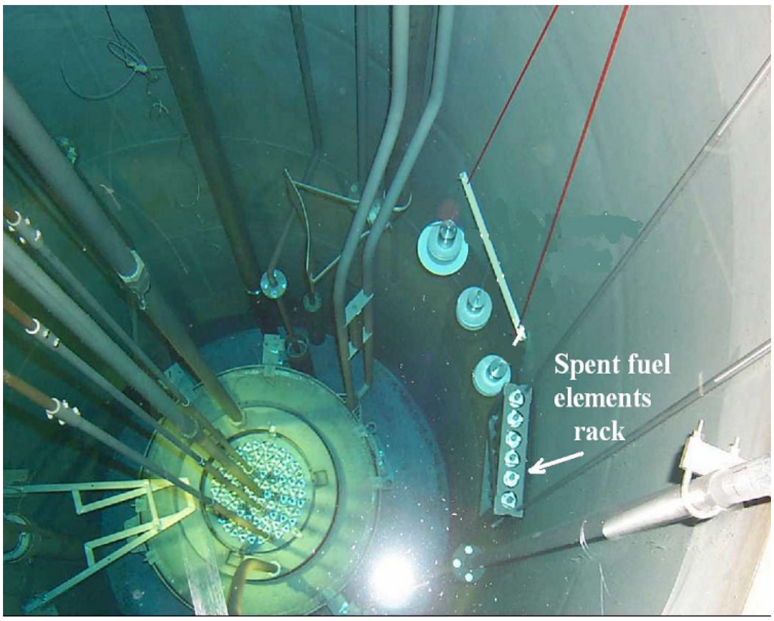
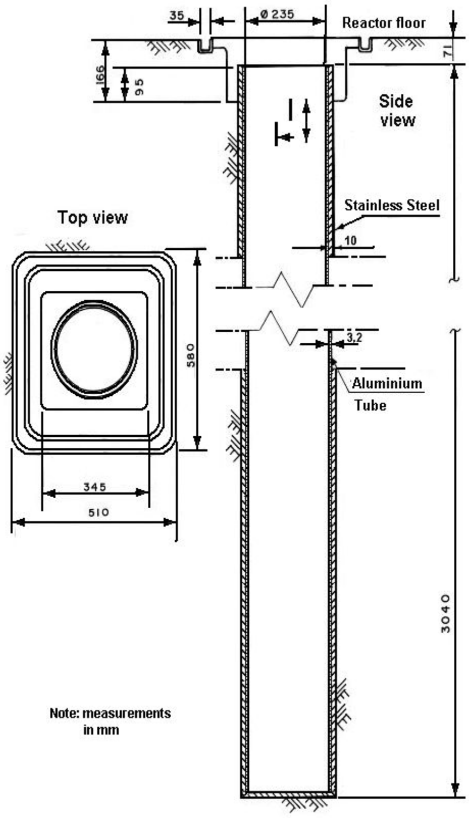
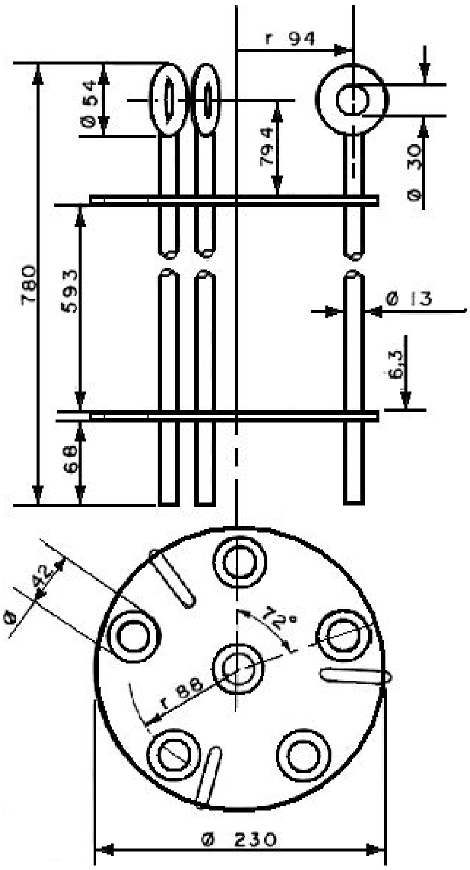


FIG. 5. Operational storage at the IPR-R1 research reactor (Brazil).



(a) Details of the interim storage wells



(b) Details of the interim storage rack

FIG. 6. Wet interim storage available at the IPR-R1 research reactor (Brazil).

For the Argonauta research reactor, the situation is similar to IPR-R1. The accumulated burn-up of the reactor since its first criticality in 1965 is about 0.25 MW-day, and due to the low nominal power of the reactor, spent fuel is not seen as a problem. However, if needed, the installation has 24 pits, located in the reactor room, each one with the capacity to accommodate 17 fuel plates, the equivalent to one fuel assembly. Figure 7 shows the storage pits for the Argonauta reactor. Actually the pits are being used to store 12 fuel plates that are slightly irradiated and conditioned in plastic bags. So far the reactor has no spent fuel to be stored.

IPEN/MB-01 also has a comfortable situation. It has only 680 spent fuel elements (pins) and 3500 positions available for dry storage, shown in Fig. 8. Since it is a critical facility, the expectation is to replace all fuel elements (with very low burn-up) at the same time, to study a new core configuration.

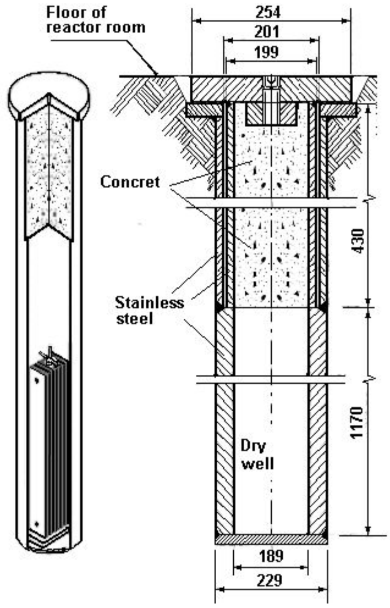


FIG. 7. Details of dry storage available at the Argonauta research reactor (Brazil).

(b) Dry storage well

(a) Dry storage area

(c) Dry storage rack

FIG. 8. Dry storage area available at the IPEN/MB-01 research reactor (Brazil).

2.1.3.3. Chile

In Chile there are two research reactors, RECH-1 and RECH-2.

RECH-1 has two pools connected by a channel, shown in Fig. 9. The first pool contains the reactor core, all associated accessories needed for experiments, and two storage racks that allow the storage of 30 fuel assemblies, 15 on each rack. The racks are mainly used to support manoeuvres during modifications of the core configuration. The second pool is the operational wet storage for the spent fuel assemblies of the reactor. It contains aluminium racks with the capacity to store up to 90 spent fuel assemblies. The storage pool was emptied in December 2000, after shipment to the Savannah River Site, USA, of 58 HEU fuel assemblies made with US enriched uranium. The pool has enough space to accommodate more racks for increasing the storage capacity to at least 250 spent fuel assemblies. Since the present utilization of the RECH-1 reactor generates 3–4 spent fuel assemblies per year, there is enough operational storage capacity for many years of reactor operation, even if there is an increase in reactor utilization. Figure 10 shows the racks available for wet storage at RECH-1.

Research reactor RECH-2 is in the condition of permanent shutdown, and during its short operation period it did not generate any spent fuel. Nevertheless, the reactor pool has four operational storage racks with the capacity to store 10 fuel assemblies on each of them. Besides, the facility has an operational wet storage pool, separate from the reactor pool, with a capacity to store 256 spent fuel assemblies. Figure 11 shows details of the spent fuel storage pool of RECH-2.

At present, the total fuel inventory in Chile is as follows: (a) in RECH-1 there are 40 HEU (45% of ^{235}U) fuel assemblies, 46 LEU (19.75% of ^{235}U) fuel assemblies and 1 experimental LEU fuel assembly; (b) in RECH-2 there are 29 HEU (90% of ^{235}U) fuel assemblies. Thus, the total inventory is 116 fuel assemblies.

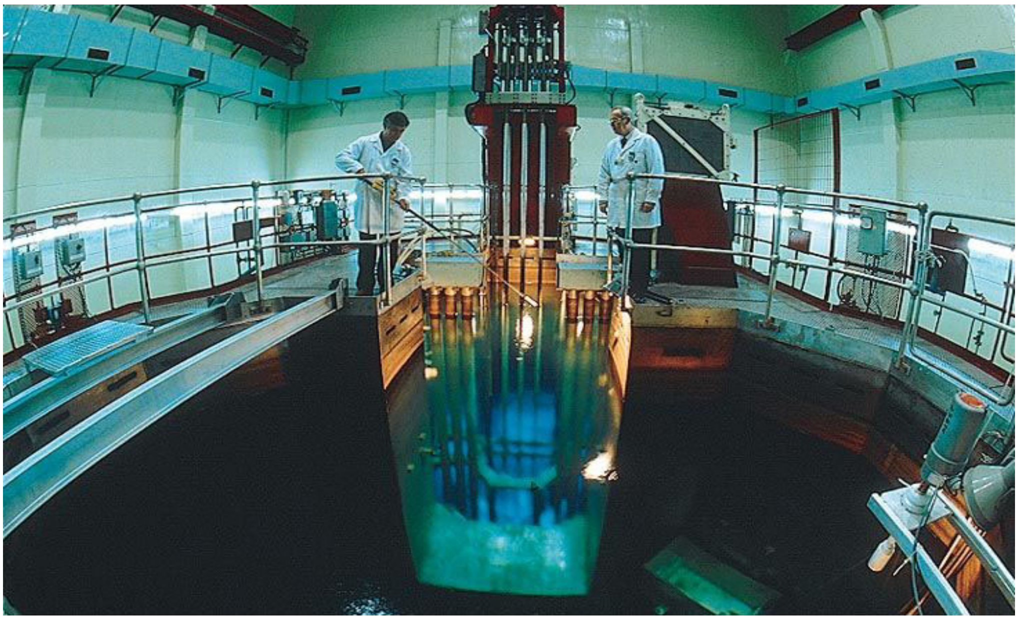
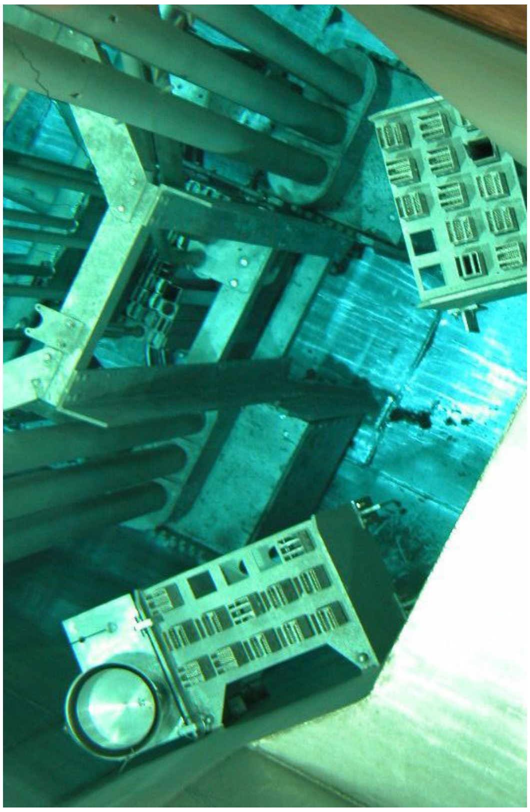
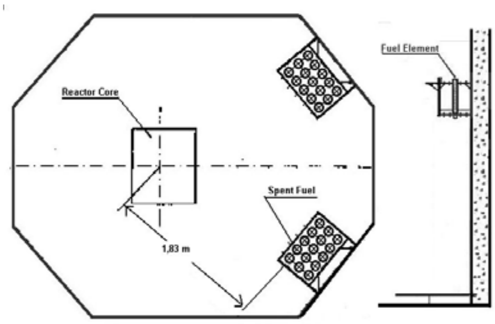


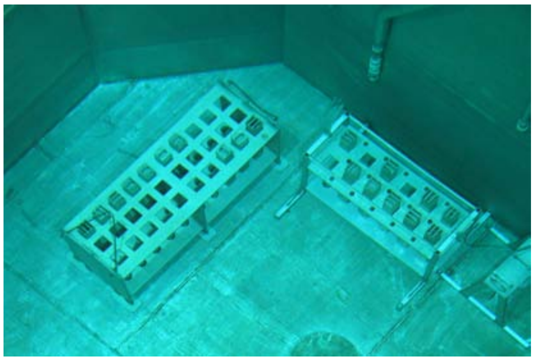
FIG. 9. Storage pool and transfer channel of the RECH-1 research reactor (Chile).



(a) Racks in the reactor pool



(b) Position of the racks in the reactor pool



(c) Racks in the storage pool

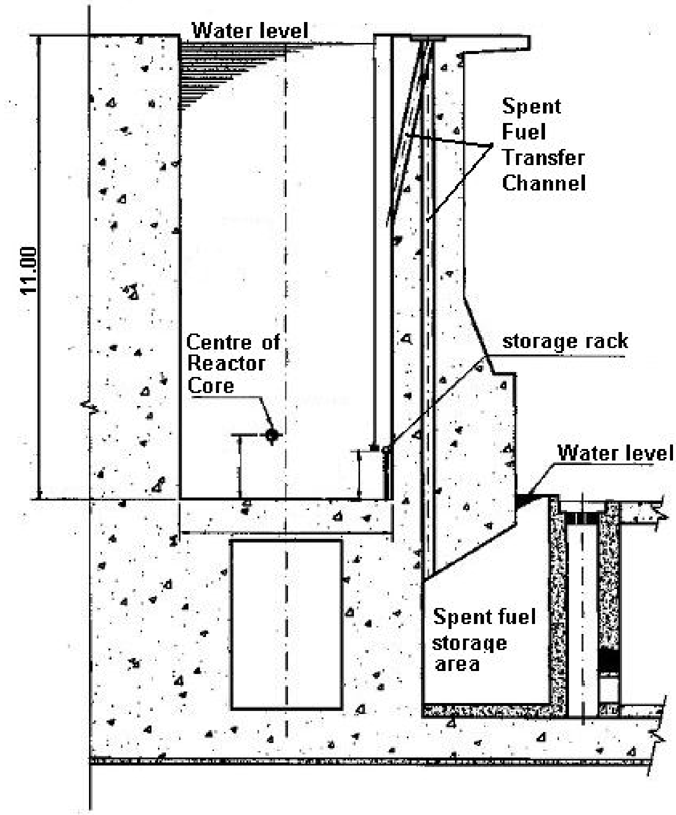
FIG. 10. Details of the spent fuel storage area at the RECH-1 research reactor (Chile).

2.1.3.4. Mexico

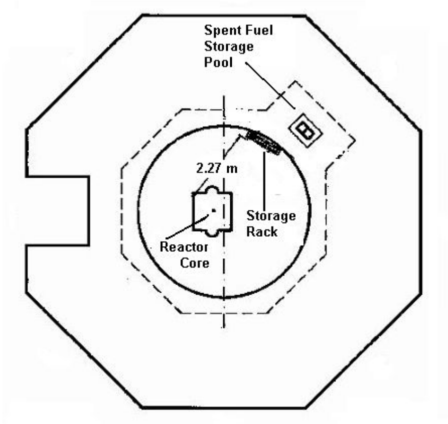
In the TRIGA Mark III nuclear research reactor of Mexico, the spent fuel is located in aluminium storage racks hanging on the reactor pool liner. The racks, shown in Fig. 12, have enough capacity for the total inventory of the installation: 33 fresh, 88 in-core and 64 spent fuel elements.

The fuel burn-up in the reactor is very low. The current reactor core configuration is number 17, which started in December 1991, and it is expected to last until the end of 2006, when some standard fuel elements will be replaced with FLIP type fuel elements (70% enrichment).

The installation also has a fresh fuel storage vault in which there are 4 LEU fuels (3 fuel elements and 1 fuel follower) and 29 HEU fuels (23 fuel elements, 3 instrumented fuels and 3 fuel followers).



(a) Spent fuel storage area at RECH-2



(b) Top view of the reactor pool

(c) Storage rack in the reactor pool

FIG. 11. Details of the spent fuel storage area at the RECH-2 research reactor (Chile).

In Mexico there are two additional sub critical assemblies and a SUR-100 homogeneous reactor.

The spent fuel storage for the subcritical assemblies and the SUR-100 reactor does not represent any complication due to the fact that the fuel burn-up is very low and they do not require cooling or special shielding. For the Nuclear Chicago Mod. 2000 subcritical assembly, the exposure rate on the outer surface is 5.4 mR/h, and at 1 m distance it is 0.4 mR/h.

The subcritical assembly Chicago Modelo 9000 is out of operation since 1973, because of corrosion problems, and the 1400 natural uranium fuel cylinders with aluminium cladding (21 cm long and 3 cm in diameter, weighing 1.8 kg each) that composed the core are, at present, kept in storage boxes in the Regional Centre of Nuclear Studies (Centro Regional de Estudios Nucleares – CREN), which belongs to the University of Zacatecas (Universidad de Zacatecas), in the Mexican state of Zacatecas. The fuel cylinders are in good condition and are periodically inspected thoroughly by visual and X ray inspection.

FIG. 12. Operational storage of spent fuel of the ININ-TRIGA research reactor (Mexico).

The SUR-100 reactor was a solid homogenous reactor with 20% enriched uranium fuel, moderated by polyethylene and using graphite as a reflector; it was shut down in 1984 because utilization was very limited. The core of the reactor, made up of fuel elements in the form of polyethylene discs measuring 24 cm in diameter and stacked to a total height of about 26 cm, was dismantled in 1989 and the fuel was sent to the ININ for surveillance and physical protection, as required by safeguards regulations. The 11 fuel discs have a total mass of 3723.2 g of U, 745.26 g of ^{235}U and are kept in two containers made of aluminium and placed in wooden boxes in the TRIGA fuel storage vault of ININ. Figure 13 shows the boxes where the discs are kept.

FIG. 13. Boxes containing the core of the SUR-1000 critical facility (Mexico).

2.1.3.5. Peru

The main pool of RP-10 is a tank about 4 m in diameter and 11 m high. As shown in Fig. 14**,** in addition to the reactor core it has two racks of aluminium with a capacity for 12 spent fuel assemblies on each. The installation has an auxiliary pool connected by a channel made of stainless steel. The auxiliary pool has a parallelepiped shape of 4 X 5 m and 6 m deep, and has two racks of aluminium with the capacity to store 48 spent fuels each, as shown in Fig. 15. It also contains a stainless steel box especially designed to store up to 16 damaged or leaking fuel assemblies.

According to the actual operating regime of the reactor, three spent fuel assemblies are discharged per year, which means that this installation has enough capacity to store the spent fuel assemblies produced in 35 years.

The fuel consumption in the critical facility RP-0 is very low, and so far it has not generated any spent fuel. The low burn-up of the fuel allows it to be easily handled, and when the facility is not in operation the fuel core is dry stored in an isolated room in the same building, in a wooden rack with capacity for 28 spent fuel assemblies.

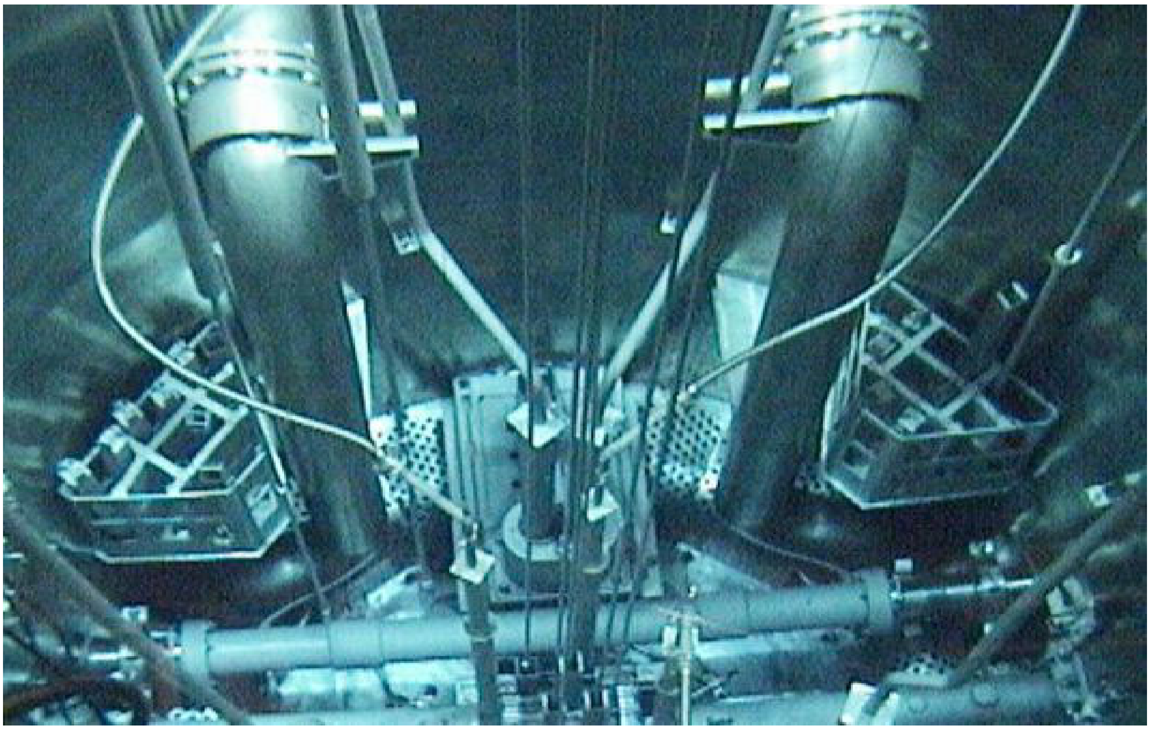
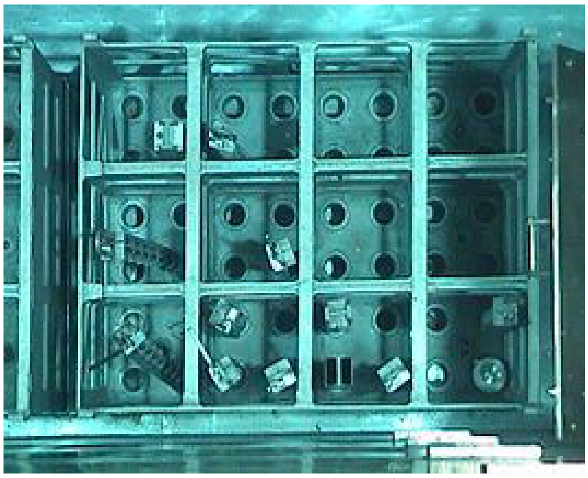


FIG. 14. Operational storage racks in the reactor pool of the RP-10 research reactor (Peru).

(a) Auxiliary pool

(b) Box for damaged/leaking fuel assemblies



(c) Storage rack

FIG. 15. Operational storage racks in the reactor pool of the RP-10 research reactor (Peru).

2.2. INTERIM STORAGE

Currently, circumstances are creating the need to implement the transition from operational to interim storage in order to guarantee the continuous operation of some research reactors in the Latin American region. This is the case for reactors with high nominal power and large utilization cycles, such as RA-3 in Argentina (10 MW, 120 hours/week of operation) and the IEA-R1 in Brazil (also 120 hours/week of operation at 5 MW).

The reasons for the transfer of spent fuel to interim storage with greater capacity are:

- Increased power and load factor of the research reactor: The operational flexibility of research reactors coupled with new needs, such as radioisotope production and fuel qualification programmes, have led in some cases to an increase the reactor power and its load factor. This causes an increase in the generation rate of spent fuel, exhausting in a few years the storage capacity of the operational wet storage area, which is limited.

- Difficulty to implement the foreseen spent fuel management strategy: The current spent fuel management situation is very different from predictions of three decades ago. Many foreseen spent fuel management activities have been impossible to implement for social, political, technical and economic reasons. Many of the originally planned activities are under re-evaluation, but the outcome of this re-evaluation is still unclear.
- Enrichment reduction: The conversion of research reactor cores from HEU to LEU has led to increased spent fuel generation rates under certain operating conditions.
- Longer research reactor lifetimes than foreseen: The continuous modernization programmes of some reactors have permitted life extensions by decades. Therefore, the amount of spent fuel produced is much higher than originally anticipated, surpassing the storage capacity of the operational wet storage area.
- Decommissioning needs could also be seen as an important driving force for the development of an interim storage option.

Many of the above mentioned events were not foreseen during the design phase of some operating research reactors, and thus were not taken into account in establishing the capacity of the spent fuel pools. Therefore, larger interim storage, either wet or dry, must become the next spent fuel management activity, after exhausting operational wet storage capacity, in order to maintain the research reactor operative with adequate spent fuel storage capacities. Both wet and dry technologies have been used worldwide, and could be implemented in the region according to specific evaluations performed by each country. It is understood that drying technology is an issue that has to be successfully solved in the region in order to implement any interim dry storage option

2.2.1. Interim wet storage

A number of features of this technology are:

- The technology is mature and well established.
- Experience has been gained in wet storage of even damaged fuel.
- International experience indicates that aluminium cladded spent fuel may be kept underwater over 50 years in pristine condition, provided that high water quality, proper environmental conditions and a good surveillance programme are maintained. On the other hand, aluminium cladded fuel degrades almost immediately when immersed in poor quality water or suboptimal environmental conditions.
- The resources, both human and financial, required to implement this technology may be greater than for some dry interim storage alternatives.
- Research reactors with large pools can have a part of them adapted to be used as interim wet storage. This option permits the water purification system, used to maintain the quality of the cooling water, to be used to also maintain the quality of the water in the interim storage.

2.2.2. Interim dry storage

Interim dry storage is a good approach to allow the continued use of reactor facilities, as either an alternative or a complementary option to the interim wet storage. Presently, dry storage is likely to be a competitive alternative to interim wet storage and may be the best option to store aluminium based fuel. This is probably the only economical interim storage technology that will provide a country with the time necessary to study options for subsequent spent fuel management activities.

The major features of this technology are:

- It eliminates corrosion degradation of the fuel cladding, provided that the storage is truly dry and the fuel has been properly dried prior to storage. This allows, at least in principle, a much longer period of interim storage, extending considerably the time required for the country's authorities to reach a decision on final options.
- It can be implemented through modular designs, e.g. metal casks, concrete containers, dual purpose (transport/storage) metal casks, and can be constructed or procured as needed, potentially providing considerable economic benefits, especially if the 'dual purpose cask' discussed in Section 6 is successfully developed in the framework of the TC Regional Project RLA/4/018.
- It is an alternative that, once it is selected, can easily be implemented within the reactor building. It requires less modification than the wet storage option and, if properly planned, can be implemented while maintaining the reactor in operation, without having to stop the production of radioisotopes, which could generate problems in fulfilling commercial contracts.
- The need to encapsulate leaking and degraded fuel elements is an important factor when considering dry interim storage options, as it could considerably modify the economic aspects of this option. For this reason, it is necessary to carefully consider the feasibility of developing a portable, automatic fuel packaging machine, to be used in conjunction with the dual purpose cask discussed in Section 6.

2.2.3. Interim storage national context

2.2.3.1. Argentina

The only research reactor that currently generates spent fuel in Argentina is the RA-3 (10 MW, 120 hours/week) at Centro Atómico Ezeiza, with an expected consumption rate of 22 spent fuel assemblies/year. At such a high rate of spent fuel generation, the operational storage capacity at the reactor building is sufficient only for the spent fuel consumed during one year of operation, so the utilization of a complementary wet storage facility is required.

For the RA-3, interim storage is currently being carried out in a wet facility with 198 sub-superficial vertical tubes (2.10 m long, 0.14 m diameter) that can stack up to two spent fuel assemblies each. Figure 16 shows a general view of the installation. Due to difficulties in properly operating the water circulation and purification system of the facility, a new wet storage facility has been proposed, with the objectives of improving the capability of interim storage as well as to provide the complementary cooling to the spent fuel that should be removed early from the reactor decay pool.

Due to the availability of a 16 m deep stainless steel pool in a post-irradiation test facility, known as LAPEP at Centro Atómico Ezeiza, the Argentinean Atomic Energy Commission (CNEA) decided to invest in a centralized 'away-from-reactor' (AFR) interim storage pool. This project has been supported since the beginning by the IAEA, initially through TC

Regional Project RLA/4/018 and afterwards through the national TC Project ARG/3/010, "Interim Storage for Research Reactor Spent Fuel".

This project takes advantage of the unused capacity that CNEA has in one of the pools of the LAPEP facility. With some modifications it will be transformed into an interim wet storage facility. This facility, shown in Fig. 17, will provide the necessary capacity to accommodate the increased spent fuel output of RA-3 for the next 20 years, assuming that it maintains the current discharge rate of 22 SF/year.

This new facility has been proposed and designed as interim storage for the LEU spent fuel inventory from RA-1 and RA-3, and the HEU spent fuel from RA-6. However, CNEA began negotiations with the US Department of Energy for an agreement to ship the full HEU inventory from RA-6 to the USA within the framework of the US acceptance programme (discussed in Section 2.2.5). The shipment of RA-6 spent fuel is expected to occur between 2006 and 2007.

Besides, Argentina and USDOE are negotiating shipment of the fuel inventory from the RA-2 critical assembly, which was shut down in 1983. The fuel inventory consists of 19 HEU fuel assemblies and 91 HEU curved fuel plates that are currently stored in a dry storage facility away from the reactor building. Shipment of the RA-2 fuel is expected to occur in conjunction with the shipment of spent fuel from RA-6.

According to current national planning, after interim wet storage in the new facility, the spent fuel from the Argentine research reactors will be processed and dry stored until a deep geological disposal facility is available.

2.2.3.2. Brazil

The only research reactor that has concerns related to spent fuel interim storage is IEA-R1 at IPEN. These concerns were the driving force for Brazil to define a strategy for managing spent fuel and to provide solutions taking into consideration the economic and technological realities of the country. In particular, it was necessary to determine the basic conditions for the interim storage of the RRSF, including the definition of a transfer cask and a dual transport cask with both possible applications, transport and storage.

During the period 2001–2003, IPEN conducted an internal discussion about the necessity of an interim spent fuel storage facility to be developed in the next ten years, when possibly all positions in the spent fuel area of the IEA-R1 reactor poll will be loaded. Therefore, within the scope of the IAEA supported TC Regional Project RLA/4/018, IPEN started to analyse the technical options for interim storage of RRSFs. For the specific case of the IEA-R1 research reactor, the preliminary results point to a dry storage option, using metallic cask based systems or concrete canisters (silos). Engineers and researchers of IPEN have analysed both options and started producing the technical documents that will help the competent authorities to reach a final decision about the type of dry storage to be implemented. After the decision is made, the Preliminary Safety Analysis Report (PSAR) will be prepared and submitted to the regulatory authority. Also as part of the strategy for long term interim storage, engineers and researchers of IPEN, in a joint project with engineers and researchers from the Centre for Development of Nuclear Technology (CDTN), started the design and submission for licensing of a dual purpose cask for transport and storage. Details of the proposed cask are given in Section 6 of this publication.

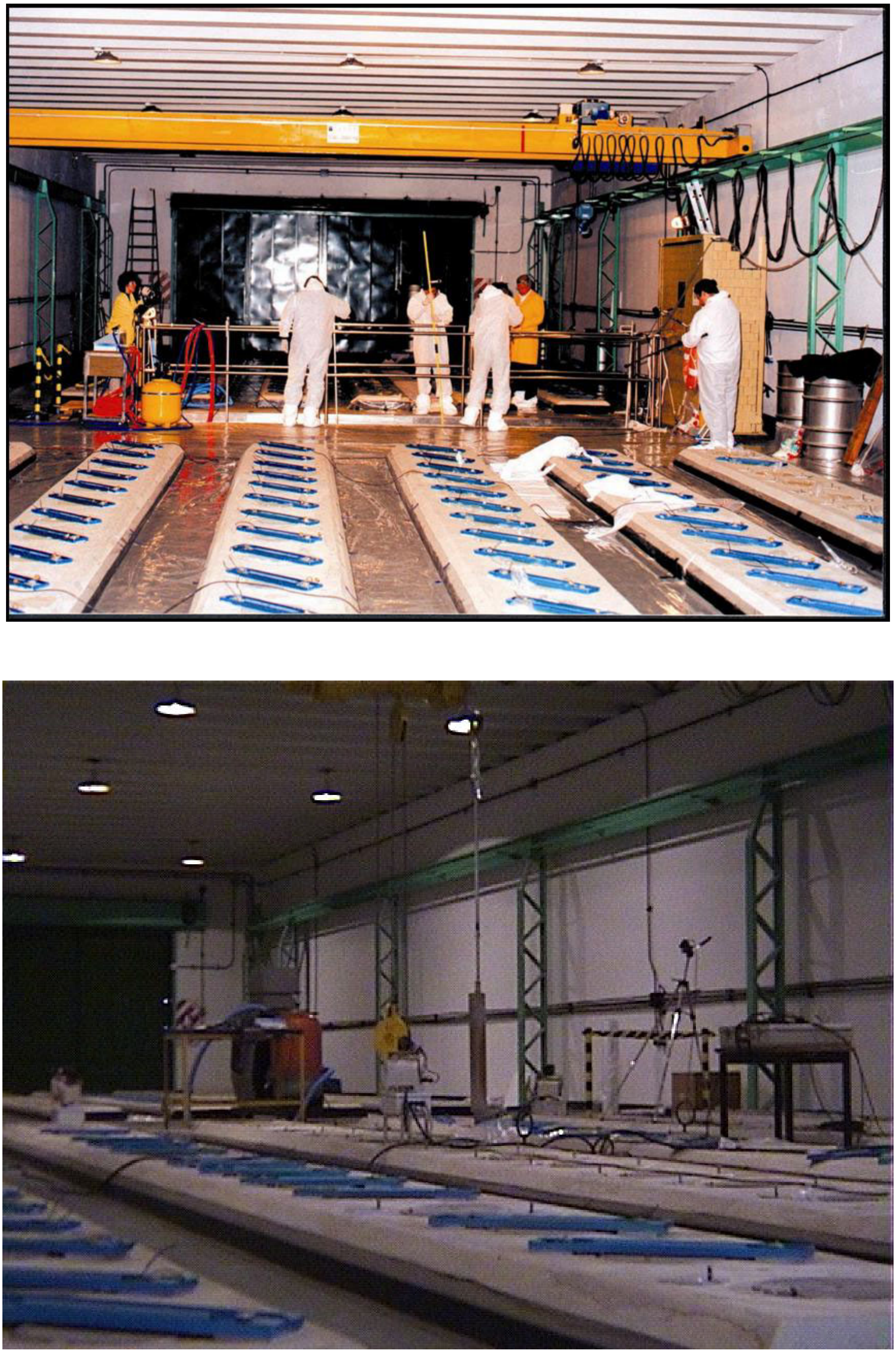
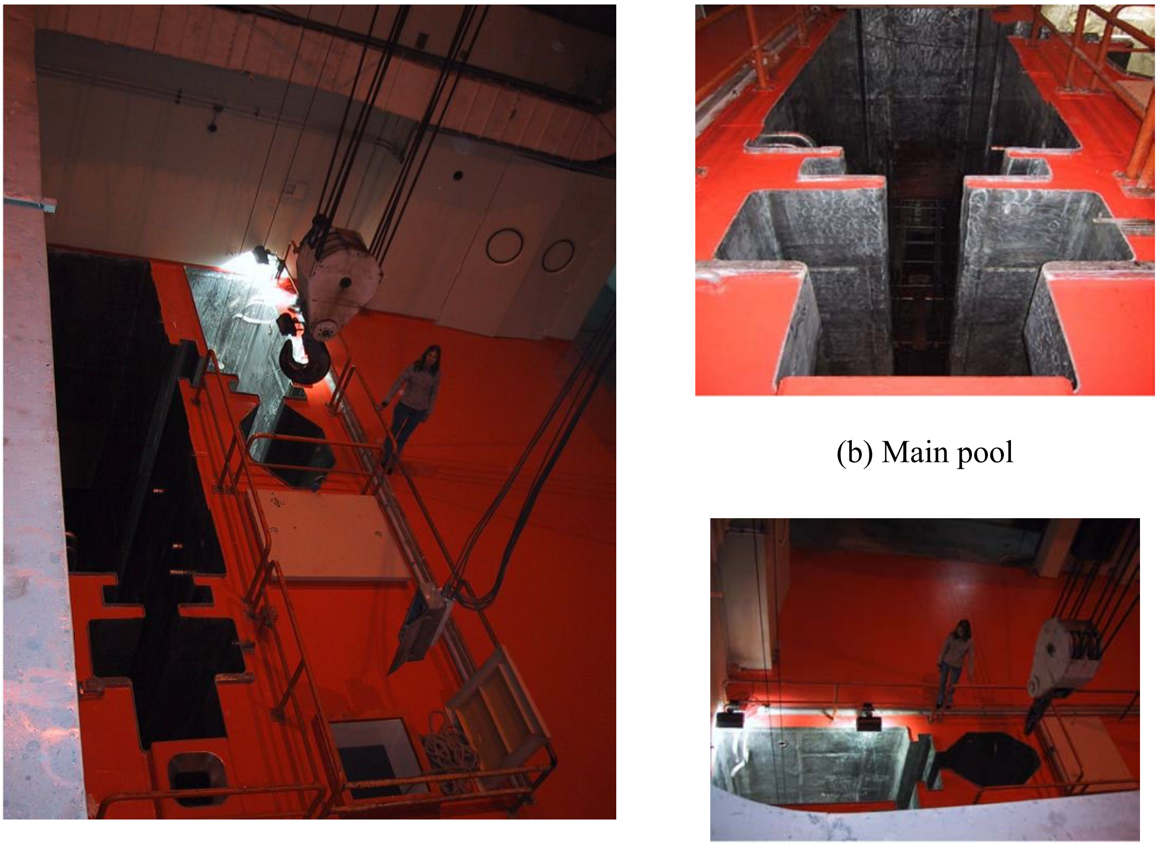


FIG. 16. Central storage facility at Ezeiza (Argentina).



(a) General view

(b) Main pool

(c) Reception and storage pool

FIG. 17. Storage pool at the LAPEP facility, Ezeiza (Argentina).

Parallel to this activity, IPEN will analyse the possibility of increasing the number of racks in the IEA-R1 reactor pool and of refurbishing a section of the reactor building in which 50 dry storage horizontal tubes are located (in which three standard spent fuel assemblies per tube can be stored), as discussed in Section 2.1.3.2.

2.2.3.3. Chile

According to the present spent fuel inventory, and due to the high water quality in both reactors, the Chilean spent fuel assemblies can be maintained in operational wet storage for many years.

With the purpose of finding long term interim storage, the engineers and researchers of CCHEN started the conceptual design of an interim dry storage. This development also started within the framework of the IAEA TC Regional Project RLA/4/018, and for the biennium 2005–2006 the IAEA approved, in the framework of the TC Programme, a Chilean national project entitled “Interim dry storage of spent fuel assemblies for research reactors: Basic engineering”. This was later converted into IAEA TC Regional Project RLA/3/004, “Spent Fuel Management for Research Reactors”, developed in collaboration with Brazilian engineers and researchers.

The fundamental objective of this project is to provide an acceptable and feasible proposal to dry store the spent fuel assemblies at a site away from reactor, after sufficient cooling has been done in the operational wet storage. The spent fuel assemblies will remain in dry storage until an acceptable and final solution has been established for them. In addition, the project is expected to study and develop, as part of the project, the necessary technology to encapsulate failed and damaged spent fuel assemblies.

2.2.3.4. Mexico

As previously mentioned, the total amount of fuel at the TRIGA Mark III reactor can easily be wet stored in the reactor pool. Since the operational experience reported at similar facilities shows that fuel with stainless steel cladding has been maintained in very good condition in wet storage, this option is considered to be an adequate method for interim storage at the Mexican research reactor. However, one point that needs to be considered is the fact that the aluminium liner of the pool has been repaired twice since 1985. It has a galvanic corrosion problem that gave rise to the assumption that the reactor pool may not be the best option for interim storage of the TRIGA spent fuel. Therefore, ININ has recently implemented a project for the design of a dry storage cask and, in parallel, started studies about the possibility of constructing a smaller pool, with the specific objective to store the fuel from the research reactor. One of these options will be utilized for interim storage. If a new wet storage facility is the selected option, the pool will be smaller than the original reactor pool, in order to reduce costs for operation and maintenance. It is understood that ININ has the personnel capable of design and construction of the wet storage pool. For the cask design, engineers of ININ are taking advantage of the dual purpose cask being developed in TC Regional Project RLA/4/018 that can be used also for the TRIGA fuel.

Since the US Foreign Research Reactor Acceptance Programme has been extended, and considering that all TRIGA fuel qualifies for this programme, an attractive option is to send back the fuel to the USA. However, before this decision is taken, the Mexican authorities must first negotiate the replacement of the existing HEU fuel in the reactor core with fresh LEU fuel. The negotiations are necessary because the cost of TRIGA type fuel elements has considerably increased during the past years, making the cost to convert the complete reactor core too high.

Reprocessing of the spent fuel of the ININ reactor is not being considered as an option, as TRIGA spent fuel reprocessing has only been demonstrated on a laboratory scale and no commercial service is currently available. As in other countries, the decision on TRIGA spent fuel disposal will be tied to the plans for spent fuel disposal of the nuclear power plants operating in the country.

For the moment, all spent fuel inventory can easily be accommodated in the reactor pool.

2.2.3.5. Peru

Taking into account the low rate of spent fuel generation in the research reactor RP-10, about three spent fuel assemblies per year, the present operational capacity placed in the reactor tank and in the auxiliary pool is enough to store the spent fuel generated in the next 30 years. So, in order to guarantee the integrity of the aluminium cladding during such a long period, a surveillance programme of environmental conditions such as water quality and quality of hall air has been implemented. Also, the reactor water supply system was revised in order to maintain a high water quality.

However, if RP-10 were to operate at its nominal power, and in accordance with the original design schedule, the capacity of the operational storage will be exhausted in a few years. The most logical option would be to increase the storage capacity by adding a row of aluminium racks, in addition to the racks already installed in the auxiliary pool. An alternative approach would be the construction of a dry storage facility at the reactor site, according to a conceptual design, which is already being developed. This project started with the support of the IAEA through TC Regional Project RLA/4/018. Also, through the regional project, a surveillance programme has been established to maintain the water chemistry in adequate condition for operational storage as well as interim storage.

2.2.4. Interim storage regional context

A dual purpose cask, for transport and storage, is being developed by Brazilian engineers and researchers with the collaborative support of counterparts of the other countries participating in TC Regional Project RLA/4/018. It will constitute another option in Latin America. In other words, the dual purpose cask, which could be located within the premises of the reactor building, is considered an option for the interim storage of spent fuel for the countries of the region. The cask has been designed for 21 MTR type fuel assemblies and 78 TRIGA type fuel elements. In 2005 the engineers from CDTN (Brazil) finalized the design of a half-scale prototype of the cask, to be constructed with the support of the IAEA. After construction the prototype will undergo qualification tests, according to international standards for spent fuel transport casks. More details of the cask are presented in Section 6.

2.2.5. Interim storage extra-regional context

As shown in Table II, the Foreign Research Reactor Spent Fuel Acceptance Programme created by USDOE has allowed a number of countries of the region to ship back to the USA 392 fuel assemblies, an important part of the US-origin RRSF inventory.

However, as can be seen in Table III, there is still a considerable amount of remaining spent fuel inventories eligible to be sent back to the USA. In this sense, the Latin American countries participating in the TC Regional Project RLA/4/018 recognize the convenience of shipping back the existing spent fuel inventories of US-origin LEU and HEU, and very much appreciate the fact that the US Foreign Research Reactor Spent Fuel Acceptance Programme has been extended for a further ten years until 2016/2019[2].

Unfortunately, at the present time the Russian Research Reactor Fuel Return Programme applies only to Russian-origin enriched fuel burnt in reactors designed in the former Soviet Union. However, there are cases in the region of Russian-origin enriched fuel being used in a non-Russian designed reactor; consequently, this fuel is not eligible for the Russian return programme.

Therefore, the countries of the region would like the IAEA to make an effort to broaden the scope of the Russian return programme in order to cover some types of spent fuel not

[2] According to the previous acceptance policy established by the USDOE and the State Department in 1996, the USA could accept certain eligible spent fuel that was irradiated by May 2006 and returned to the USA by May 2009. A revised record of decision, signed by National Nuclear Security Administration (NNSA) Administrator Linton Brooks on 22 November 2004, extended the irradiation deadline to May 2016 and the acceptance deadline to May 2019.

currently covered. This would offer an additional option for management of the spent fuel from some of the research reactors in the region. It is understood that this programme might be financed by third parties, i.e. the RERTR Programme, Global Threat Reduction Initiative (GTRI), etc. Similarly, the possibility that the USA could agree to receive spent fuel with foreign origin enriched uranium under the GTRI should also be explored.

TABLE II. SPENT FUEL ASSEMBLIES SHIPPED TO THE USA WITHIN THE FRAMEWORK OF THE FOREIGN RESEARCH REACTOR SPENT FUEL ACCEPTANCE PROGRAMME

Country	Year	Number and type of spent fuel	Total
Argentina	2000	207, MTR HEU	207
Brazil	1999	127, MTR HEU and LEU	127
Chile	1996	28, MTR HEU	58
	2000	30, MTR HEU	

TABLE III. SPENT FUEL STILL ELIGIBLE TO BE SHIPPED TO THE USA WITHIN THE FRAMEWORK OF THE FOREIGN RESEARCH REACTOR SPENT FUEL ACCEPTANCE PROGRAMME

Country	Number and type of fuel	Total
Argentina	230 SF, Pin LEU	230
	30 SF, MTR HEU and 31 in-core, MTR HEU	61
	91 SF, MTR plates HEU	91
Brazil	16 SF, MTR LEU	16
	128 SF, MTR plates, LEU	128
	64 in-core TRIGA, LEU and 4 fresh TRIGA, LEU	68
Peru	5 SF MTR LEU and 29 in-core fuel, MTR LEU	34
Mexico	64 SF TRIGA, LEU	
	59 in-core fuel, TRIGA LEU	
	29 in-core fuel, TRIGA HEU	185
	4 fresh fuel, TRIGA LEU	
	29 fresh fuel, TRIGA HEU	

2.3. EXTENDED INTERIM STORAGE

The indefinite deferral of or lack of decision about the next activity of RRSF management, spent fuel processing, might lead to the extended interim storage option.

The aim of extended interim storage is to maintain the integrity and retrievability of the spent fuel for a definite or even indefinite time period. It is considered that extended interim storage lasts beyond 20 years.

The decision to water store spent fuel for periods longer than originally expected should be carefully compared with the option of migrating to dry storage. The final decision should be taken comparing the cost of both options.

There is general agreement among the authors of this report that the current technical, economic and institutional status suggests that the countries of the region will likely implement the extended interim storage option for all research reactors.

3. SPENT FUEL PROCESSING

It is clearly understood that at some time in the future it will be necessary to process the RRSF. Moreover, several factors may anticipate the need for processing the spent fuel and making provisions for the subsequent disposal of its derivatives. One reason is the end of the US Foreign Research Reactor Spent Fuel Acceptance Programme in 2019. Another reason could be the reduction in research reactor utilization due to the replacement by alternative technologies to produce neutron fluxes and radioisotopes. This, consequently, could lead to the closure of many research reactors and their associated operational and interim storage facilities.

For spent fuel processing, two main approaches can be considered: conditioning for disposal, and reprocessing. Both processes produce long lived radioactive wastes that need to be disposed of in deep geological repositories. The main differences between them are the type and the final volume of waste to be disposed.

3.1. SPENT FUEL CONDITIONING

Conditioning produces low and intermediate level wastes (LILW), instead of the high level waste (HLW) that results from reprocessing, because the thermal loading of the spent fuel derivatives arising from conditioning is well below 2 kW/m^3. Spent fuel derivatives from both conditioning and reprocessing are long lived as they contain significant levels of radionuclides with half-lives greater than 30 years.

Currently, there are several different spent fuel conditioning technologies, at different levels of development. In this report we will consider only those that are considered potentially applicable in the regional context. The considered technologies are:

Can-in-canister. A critically safe quantity of non-processed spent fuel is placed in a can. This can is placed in a canister into which HLW glass is poured to form a solidified unit.

Press and dilute/poison. To minimize volume, the spent fuel is mechanically compressed and either diluted with depleted uranium or mixed with a neutron poison.

Chop and dilute/poison. The spent fuel is chopped into small pieces and diluted with depleted uranium or mixed with a neutron poison.

Melt and dilute. The spent fuel is melted and diluted with depleted uranium. This process was developed at the Savannah River Technology Center (SRTC), USA, and involves melting of the entire spent fuel assembly, diluting the uranium alloy with depleted uranium for isotopic dilution and aluminium for eutectic formation.

Plasma arc treatment. The spent fuel is placed directly into a plasma centrifugal furnace with depleted uranium and neutron absorbers, where it is melted and converted into a HLW glass waste form.

Glass material oxidation and dissolution system. The spent fuel is placed in a glass melt furnace where it is oxidized by lead dioxide and then converted into a LILW-LL glass waste form.

Dissolve and vitrify. The spent fuel is dissolved and mixed with depleted uranium to reduce the enrichment. The mixture is then fed into a vitrification plant for conversion to a LILW-LL glass waste form.

Electrometallurgical treatment. The spent fuel is melted with silicon and electro-refined. The bulk of the aluminium is electrolytically removed for disposal as low level waste; the residual aluminium, actinides and fission products are vitrified. Pure uranium is recovered.

Chloride volatility techniques. The spent fuel is reacted at high temperatures with chlorine gas and all of the materials converted to a volatile gas. The uranium, actinides and fission products are separated from each other by cooling and distillation.

By looking at these alternatives we can see that they can be classified into two main categories:

- Those in which the final product will comprise an aluminium based metallic matrix: press and dilute/poison, chop and dilute/poison, and melt and dilute;
- Those with a glass or glass ceramic matrix that immobilize the spent fuel: can-in-canister, plasma arc treatment, glass material oxidation and dissolution system, and dissolve and vitrify.

Electrometallurgical treatment and the chloride volatility techniques do not produce a disposable waste form; they are rather prior steps to either conditioning or reprocessing.

Researchers in the region involved in R&D conditioning processes consider that the final products from the processes of the second category, those with a glass or glass ceramic matrix, represent a better guarantee for long term immobilization of the radionuclides, as long as more evidence — either natural or human made — for long term behaviour is available.

Considering the second category, it can be seen that the final product of three of the processes is a glass waste form, where the radionuclides are dissolved and immobilized. The final product of the fourth process (can-in-canister) is a metallic can (with the untreated spent fuel in it) embedded in a glass block.

3.1.1. Spent fuel conditioning national context

3.1.1.1. Argentina

For the aluminium based RRSF, currently in operational and interim wet storage, interim dry storage of the processed spent fuel is being considered.

The diverse fuel types considered are Al + Al–U alloys, Al + Al–uranium oxides, Al + Al–uranium silicide and Al + Al–UMo.

Previous work, carried out during the spent fuel repatriation campaign and post-irradiation examination of spent fuel, consisted in removing structural parts from spent fuel assemblies by cropping and disassembling (Fig. 18). This experience would be most probably applied to use separated fuel plates instead of full fuel assemblies as input for the different fuel processing options.

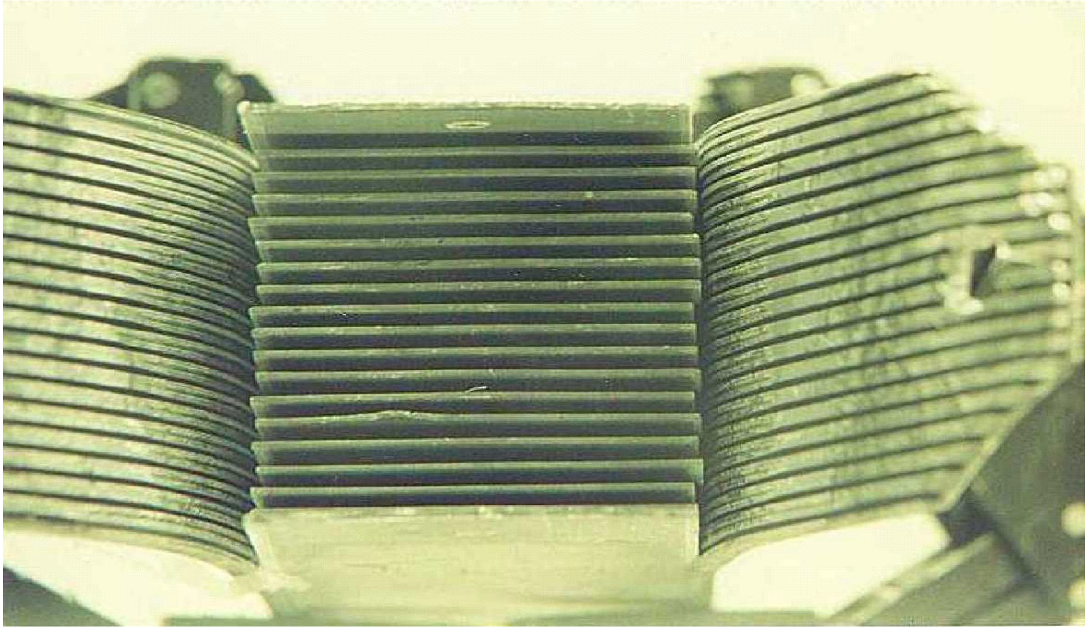


FIG. 18. MTR fuel disassembling (Argentina).

In the framework of the Radioactive Waste Management Strategic Plan, and supported by the Radioactive Waste Management National Programme, the research lines currently being studied in Argentina include:

- Decladding through a dry process: The HALOX process (patent pending) was developed by researchers and engineers of CNEA, and involves the utilization of chlorine gas for chemical treatment of the aluminium cladding. It forms volatile aluminium chloride at a temperature of about 200°C, which can be condensed and turned into aluminium oxide, and afterwards conditioned. The remaining fuel meat (after isotopic dilution) is also oxidized for vitrification. This process has emerged as an innovative, versatile and interesting technology for the conditioning of aluminium spent fuel from research reactors.
- Decladding through a wet process: After dissolution of the fuel plates in a 1M sodium hydroxide solution, aluminium from the cladding can be mostly removed with the liquid short lived low and intermediate level waste (LILW-SL) stream. Although this generates a large amount of LILW-SL, all the chemistry and processes are well known from the current production of ^{99}Mo by fission at CNEA.
- Isotopic dilution through a wet process: In this process, either the cladded or decladded spent fuel plate is dissolved in nitric acid and mixed with a soluble natural or depleted uranium compound in order to isotopically dilute the ^{235}U content present in the spent fuel.
- Immobilization of the spent fuel derivatives in a suitable matrix (glass or ceramic): Technologies currently being studied include glass sintering, glass melting and ceramization (CERUS process). The CERUS process (patent pending) consists of using a U_3O_8 (natural or depleted) powder in both, to dilute and act as an immobilization matrix for the spent fuel derivatives. In this way it is possible to reduce the ^{235}U enrichment of the spent fuel and create a homogeneous material having structural integrity and good corrosion resistance provided by the sintering of the U_3O_8. This method has the advantage

of avoiding the need for a vitrification process, in which the volume of the waste form increases substantially. For the vitrification process, iron phosphate glasses are being currently studied for melting as well as sintering technologies.

Diverse alternatives of integrated processes under consideration are:

- Decladding through the dry process HALOX + isotopic dilution + vitrification or ceramization,
- Decladding through a wet process + isotopic dilution + vitrification or ceramization (after a preconditioning step),
- Complete dissolution by a wet route + isotopic dilution + vitrification or ceramization (after a preconditioning step),
- Direct ceramization of the undecladded spent fuel plate material (CERUS process).

Argentina has the basic infrastructure of hot cells needed to implement the selected conditioning process on a hot laboratory scale and to scale it up to pilot plant scale. When these cells are operative, Argentina will be able to solve its RRSF conditioning needs, and, as long as no legal prohibitions exist for the entrance of foreign spent fuel, it could offer conditioning services to third countries, returning the radioactive wastes to the customer for disposal. However, public opinion and political acceptance have to be carefully addressed in order to allow the implementation of this option.

A foreign technology that could be considered by CNEA is the ‘melt–dilute’ process, provided it complies with the acceptance criteria applied to deep geological disposal. In this sense, the ‘melt–dilute’ process should gain more evidence about the long term durability of the aluminium matrix in comparison to glass and ceramic matrices.

3.1.1.2. Brazil

Conditioning processes for the Brazilian research reactors and critical facilities consider primarily Al + Al–USi alloys. The actions pursued up to now involve two research lines, dissolution and immobilization.

Dissolution is being considered as a method of volume reduction, where it is proposed to separate the aluminium from the remaining compound, which includes the fission products, the remaining uranium silicide and the actinide elements. In a first step the aluminium structure is mechanically separated, then the fuel plates are dissolved through a wet process and the aluminium is separated by an electrochemical process keeping, as much as possible, the fission products insoluble in the media studied. The remaining compound could be processed for re-utilization of the uranium silicide, or immobilized, after isotopic dilution. In this phase of the project, re-utilization of the uranium silicide is being considered as a technical alternative, since it has the advantage of minimizing the amount of high level waste (HLW) to be immobilized and disposed.

Immobilization is also a current research activity. The proposal is to immobilize the resulting high level waste, or the remaining compound, using glass matrices based on niobium phosphate glasses, which can be melted in microwaves or in electrical furnaces. Three different samples of niobium phosphate glasses have been produced and sent for vapour hydration tests (VHT) at the Centro de Estudios Nucleares La Reina, in Chile. The results showed that these matrices are promising.

3.1.1.3. Chile

In Chile the spent fuel assemblies are being stored in the reactor pools as operational storage. The basic engineering design of an interim dry storage facility is being developed, considering that there will be no conditioning processes of the spent fuel assemblies. However, at any time a decision can be taken about initiating studies on the conditioning of spent fuel at laboratory level, primarily with the purpose to develop know-how in this field. As a spin-off, the results of this initiative could be used to immobilize conventional hazardous materials.

As a first approach to investigating conditioning processes, Chile has joined the immobilization team formed by Argentina and Brazil within the framework of TC Regional Project RLA/4/018. During 2004, accelerated corrosion tests were done in laboratories of CCHEN using samples of glasses prepared at the Centro Atómico Bariloche, Argentina, and at IPEN, Brazil.

Due to the experience gained and the good results obtained during development of the accelerated corrosion tests, CCHEN could continue to provide cooperative support to the research groups of Argentina and Brazil. Additionally, CCHEN will evaluate the feasibility of initiating the preparation of glasses or ceramic matrices for spent fuel immobilization under the guidance of the Argentine and Brazilian groups.

3.1.2. Spent fuel conditioning regional context

Currently, conditioning processes are being studied on the laboratory scale in the Latin American region just for aluminium based MTR type fuel. TRIGA fuel conditioning has not yet been considered.

Within the framework of TC Regional Project RLA/4/018, the processes HALOX and electrochemical dissolution — initial steps for conditioning — are open for collaborative studies in the region. Immobilization processes using glass as a final matrix are jointly studied by Argentina and Brazil. These glass materials could be used in any process that involves glass as a final matrix for spent fuel derivatives immobilization.

The maximum MTR spent fuel turnover for the Latin American region has been estimated to be 53 spent fuel assemblies/year. For the calculated turnover and taking into account the backlog awaiting processing, 142 spent fuel assemblies represent 2.8 years at the estimated rate; it is then considered to be reasonable to estimate a rate of 100 spent fuel assemblies/year of processing capacity to cover the needs of the Latin American region. This means that a pilot plant with a capacity to process and condition between two and three spent fuel assemblies per week would be able to process the annual arisings, as well as the backlog.

To summarize the regional context, considering a 30 year period or operation, the spent fuel derivatives arising from the processing plant would have produced 627 t of LILW-LL, corresponding to a volume of 220 m^3 (more detailed information is presented in Section 5.4).

3.1.3. Spent fuel conditioning extra-regional context

Currently, no commercially available conditioning service is being offered to third countries.

3.2. SPENT FUEL REPROCESSING

Currently, spent fuel reprocessing based on the wet process PUREX has been implemented up to commercial scale for MTR type RRSF highly diluted in a stream of LWR spent fuel (COGEMA, La Hague, France). As a result of this kind of reprocessing, the radioactive wastes are typical HLW from LWR spent fuel reprocessing, i.e. with higher transuranium elements content than the original RRSF. Although no commercially available reprocessing process is currently being offered specifically for spent fuel from research reactors, some wet as well as dry technologies are being studied.

3.2.1. Spent fuel reprocessing national and regional context

Argentina, as stated in the Radioactive Waste Management Strategic Plan, has established that by 2030 it will decide whether the spent fuel from its nuclear power plants is to be reprocessed. Reprocessing plans for spent fuel from nuclear power plants could have an impact on RRSF reprocessing if the current commercial strategy of high dilution of RRSF into the nuclear power plant spent fuel stream continues in practice. Furthermore, in order to minimize the amount of long lived intermediate level waste, the separation of uranium — by dissolution of the spent fuel plates in nitric acid — is under consideration as a management option. Such a process avoids the isotopic dilution of uranium that is responsible for a large volume increase.

Brazil, Chile, Mexico and Peru have not established a date to decide if reprocessing will be included as an option for spent fuel management.

No country in the region is developing reprocessing technologies, and, at the time being, it is not expected that such a service could be offered in the region in the near future.

3.2.2. Spent fuel reprocessing extra-regional context

The Latin American countries consider this as an option because COGEMA in France offers commercial services for reprocessing RRSF of the ‘reprocessable type’, i.e. with either aluminium–uranium alloy or aluminium–uranium oxide dispersion as fuel material. Recently, COGEMA included RRSF with uranium silicide as fuel material for such customers that agree reprocessing contracts of an amount of RRSF of reprocessable type that is large enough to achieve a sufficient dilution of the uranium silicide in the aluminium–uranium alloy and/or aluminium–uranium oxide in order to reprocess it with neither plant equipment nor process parameter changes. It should be noted that the implementation of this option requires resolution of a complex array of political issues.

It should also be pointed out that, because spent fuel from research reactors (RRSF) is co-reprocessed with spent fuel from light water reactors (LWRSF) by adding about 1 kg of RRSF per 1000 kg of LWRSF to the feed stream, the returning HLW has a mass ratio of total long lived transuranium elements (mainly neptunium, americium, curium and non-recovered plutonium nuclides) to fission product which is similar to the mass ratio obtained by processing only LWRSF. This mass ratio after a 1 year decay time is about 2.2 times higher than the value obtained by reprocessing only RRSF. In addition, considering current practices, uranium from RRSF co-reprocessed with LWRSF (about 1% enriched) is expected to remain at the reprocessing plant, and the uranium from RRSF, originally about 14% enriched, after dilution down to 1% during the co-reprocessing, cannot be recovered.

As will be detailed in Section 5.4, reprocessing of all spent fuel produced in the research reactors of the Latin American region during a 30 year period would produce a total amount of 1.56 t of HLW, representing 0.56 m^3 that can be accommodated in four COGEMA HLW canisters with a maximum capacity of 450 kg HLW each. It is worth comparing these figures with the 627 t of LILW-LL, corresponding to a total volume of 220 m^3 immobilized spent fuel derivatives, that will be produced if no reprocessing is done. The remarkable difference arises from the isotopic dilution with natural uranium down to 1% enrichment, that is to say, the value of the HLW stream from LWR spent fuel reprocessing.

4. PROCESSED SPENT FUEL STORAGE

LILW from conditioning and HLW from reprocessing impose different requirements on the interim storage.

It is likely that processed spent fuel, i.e. either LILW or HLW, will be stored in extended interim dry storage in each country of the region until a deep geological repository is available.

These materials are most likely to be stored in each individual country. Regional or extra-regional solutions are not foreseen as long as they constitute radioactive wastes for which storage of radioactive waste from third parties is forbidden by law in all countries participating in TC Regional Project RLA/4/018.

5. PROCESSED SPENT FUEL DISPOSAL

LILW from conditioning and HLW from reprocessing have different impacts on deep geological repositories. The most important one relates to thermal effects and radiolysis shortly after disposal. However, the low amount produced in comparison with the co-disposed nuclear power plant spent fuel makes this effect negligible.

It is well known that repositories studied so far have been for power reactor fuel with initial enrichment of 4% or less, being about 1% the remnant enrichment in the spent fuel inventory. Repeating these studies for research reactor fuel, including the environmental impact assessment, would be enormously expensive. Besides, criticality considerations and the susceptibility of the materials to degradation lead to the conclusion that direct disposal of RRSF cannot be considered a viable option.

Therefore, unless the selected option considers the possibility to retrieve the remaining uranium composite existing in the spent fuel from research reactors, the trend is to effectively dilute the concentration of ^{235}U to below 2% to match the remnant enrichment level of nuclear power plant spent fuel.

Regardless of the solution adopted for the RRSF, international consensus suggests that, due to the long lived nuclides that it contains, the disposal of the processed RRSF shall be planned for a deep geological repository.

For countries with spent fuel arising from nuclear power plants, the same deep geological repository is highly likely to be used for the relatively small amount of processed RRSF to be disposed. However, it must be underlined that the IAEA increasingly sees the need for multi-national solutions for countries with small nuclear power programmes or one or more research reactors and no nuclear power programme, since individual geological repositories for these countries seems prohibitively expensive.

Co-disposal of radioactive and non-radioactive waste among other chemical, industrial, toxic and biological wastes should be carefully analysed as long as the volume of non-radioactive wastes to be disposed in a deep geological repository could be large enough to justify the construction of such a facility in each country.

Public opinion is mostly negative regarding any possibility of disposal (particularly geological) of such waste, especially radioactive waste disposal, because such facilities are considered highly dangerous. In this regard, a continuous and effective public information, communication and involvement programme in each country must be implemented in order to counter this perception. This modification of public attitudes is necessary prior to site selection and construction of any disposal facility.

5.1. PROCESSED SPENT FUEL DISPOSAL NATIONAL CONTEXT

Argentina, in the framework of the Radioactive Waste Management Strategic Plan, has defined its schedule for deep geological disposal. The site should be selected by 2030 and a deep geological repository should be in operation by 2050.

During the 1980s, Brazilian researchers performed some geological studies in which about 20 sites were eligible for geological disposal, but so far no decision has been taken. However, it is considered that the HLW or the RRSF derivatives will be disposed on the same site selected for disposal of spent fuel from Brazilian nuclear power plants.

5.2. PROCESSED SPENT FUEL DISPOSAL REGIONAL CONTEXT

In the Latin American region, due to the scale of the nuclear programmes, the quantities of processed RRSF for disposal are small enough that the construction of the deep geological repository would be beyond the possibilities of the countries involved, even for the region as a whole. It is, however, stressed that the responsibility for the RRSF management still remains and it is not related to the amount of spent fuel generated.

As was indicated in Section 3, several factors may anticipate the need of RRSF processing and subsequent disposal of its derivatives in the region for the next decades (i.e. the end of the US acceptance programme in 2019, research reactors closure, etc.).

In considering the technical features of a regional repository, it seems that there are no additional issues to be addressed when compared to the case of a national deep disposal repository. Some of these technical issues are: processing (either conditioning or reprocessing) domestic or abroad, development of substantial knowledge for site selection, characterization, design, transport, development and construction, etc. Furthermore, it is recognized that currently no country in the region has fully developed any RRSF disposal option.

Because the amounts of processed spent fuel involved are too small, the economic burden of developing and operating one deep geological regional repository for the countries of the Latin American region seems to be of the same order of magnitude as in the case of individual disposal options, therefore the savings for the shared costs would not be significant.

Also, experience so far indicates that regulatory issues are difficult to harmonize. Currently, there are no common regulations accepted in the Latin American region for this purpose. Nevertheless, it is assumed that by adopting regulations conforming to relevant IAEA safety standards and guidelines, a certain level of harmonization would be inherent. However, it is important to understand that the national legislation of all the relevant countries prohibits the importation of radioactive wastes, and amendments in order to modify this situation are not foreseen.

Finally, we also have to consider that in all countries public opinion is mostly negative regarding any possibility of radioactive waste disposal (particularly geological), because there is a feeling that such facilities are highly dangerous. In this regard, independent of the selected option, a continuous and effective public information programme in each country must be implemented, in order to counter this opposition.

If the countries in the region cannot resolve the economic, legal, political and public acceptance issues related to a regional repository, an alternative would be to request the IAEA to assess the feasibility of an extra-regional international repository. Since such an international repository would address some of the problems being tackled by the Global Threat Reduction Initiative (GTRI) in the areas of non-proliferation and securing of materials that could be used in radioactive dispersal devices (dirty bombs), it could be pursued jointly by the GTRI and IAEA.

5.3. PROCESSED SPENT FUEL DISPOSAL EXTRA-REGIONAL CONTEXT

Future alternatives could be associated with deep geological repositories in extra-regional countries or international territories under UN administration, more likely in those countries with extensive nuclear programmes and an advanced stage of implementation of this management activity. It should be noted that the implementation of this option requires a significant political effort to resolve the complex array of agreements needed. However, as explained before, an international repository would resolve some of the problems being tackled by the Global Threat Reduction Initiative (GTRI) in the areas of non-proliferation and securing of materials, and it is understood that it could be pursued jointly by the GTRI and IAEA.

5.4. PROCESSED SPENT FUEL DISPOSAL MODEL

The purpose of this model is to represent the amount of spent fuel and its derivatives that should be managed — including processing and disposal — within the framework of a regional strategy, involving the countries that participated in the TC Regional Project RLA/4/018.

The spent fuel turnover is described by the quantities of total U, ^{235}U, total Pu, ^{239}Pu and Al shown in Fig. 19. The output of the regional conditioning plant, presented in Table IV, is described by the quantities of total immobilized spent fuel derivatives (LILW-LL waste form), total U, ^{235}U, total Pu, ^{239}Pu, Al, fission products and minor actinides (Np, Am and Cm) as well as the volume of the LILW-LL waste form. Additionally, the activity of alpha emitters in the immobilized waste form and its thermal load have been calculated. The quantity and the volume of the very low level waste (VLLW) Al stream, from the disassembled structural parts of the fuel assemblies, are also shown.

The main assumptions taken into account for the modelling were as follows:

- The regional countries share a common strategy for spent fuel processing and join in a partnership to install a conditioning plant in one of the partner countries. The plant will process the whole spent fuel inventory (turnover and backlog).
- The annual spent fuel turnover for the countries of the region corresponds to the estimated operation regimes of the research reactors in a realistic scenario, as shown in Table I in Section 1.5.
- The research reactors of the region will operate for 30 years more.
- Mean burn-up values for each country with a 2 year cooling time were considered for the spent fuel, in order to get conservative results for radionuclide quantities, alpha activity and thermal loading calculations.
- The structural components of the spent fuel assemblies — upper and bottom ends, side plates, etc. — are removed before conditioning. Thus, only the fuel elements (plates) are conditioned.
- A uranium isotopic dilution down to 1% enrichment of ^{235}U is necessary in order to match the final enrichment of spent fuel from nuclear power plants intended for direct disposal. Such isotopic dilution is achieved with natural uranium.
- Immobilization is carried out with a phosphate based glass and an assumed waste loading of 15 wt.%, reaching a density of 2.85 kg/dm^3.

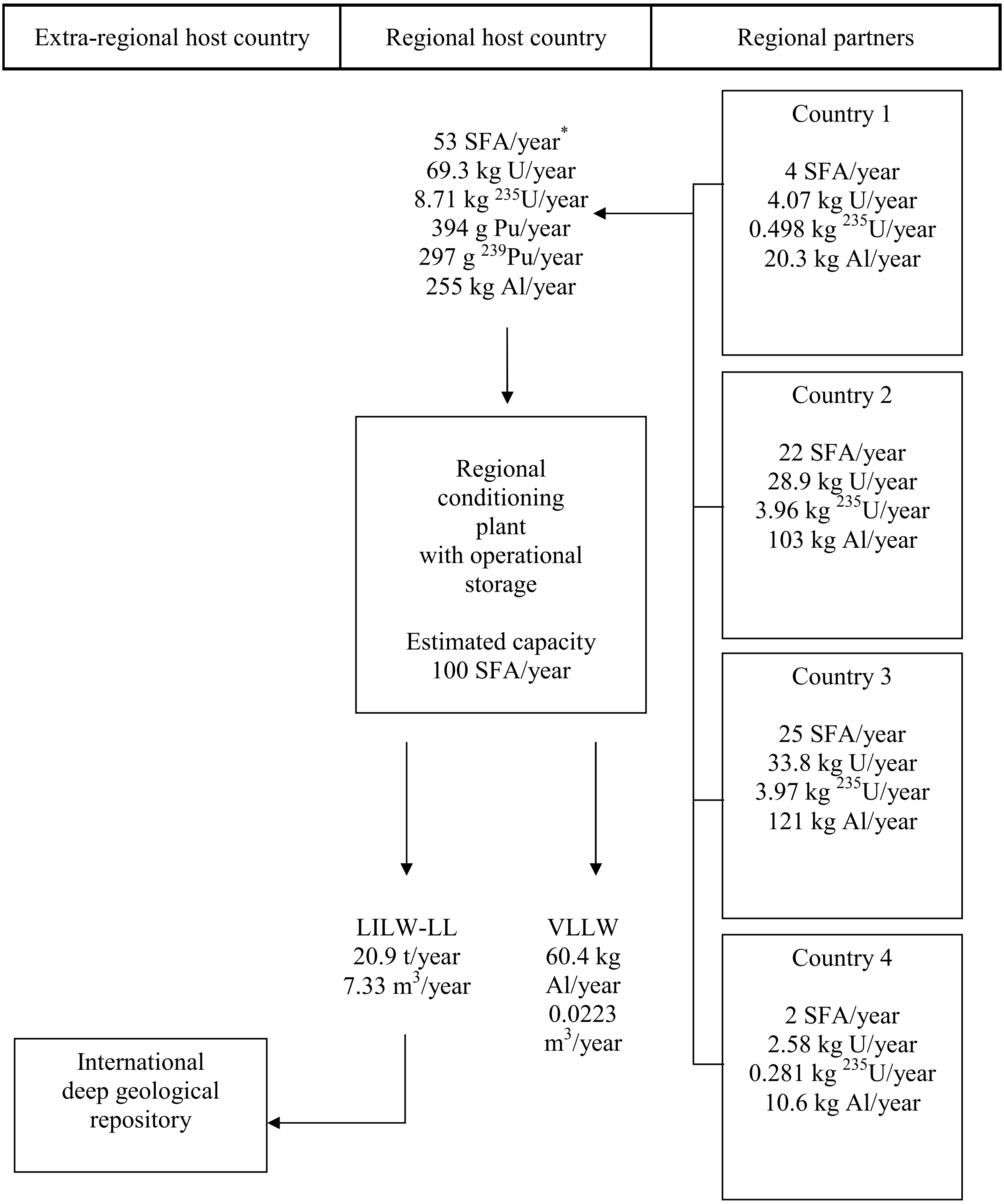


FIG. 19. Flow diagram for a regional RRSF conditioning plant.

* Note: The amounts are based on projected turnover for the reactors of the countries that participated in the IAEA TC Regional Project RLA/4/018.

Considering the above assumptions, then, according to the model, the spent fuel arisings will be those presented in Table IV.

The 627 t of immobilized spent fuel derivative (waste form) shown in Table IV constitutes a long lived low and intermediate level waste stream, as long as it contains long lived alpha emitting radionuclides, which accounts for an activity of the order of 1.4×10^{14} Bq (88% from plutonium, 10% from minor actinides — neptunium, americium and curium — and 2% from uranium, mostly from isotopic dilution with natural uranium). Such alpha activity results in a long lived alpha emitting radionuclide concentration of 220 000 Bq/g in the waste form, which exceeds the limits established for short lived LILW of 4000 Bq/g of alpha emitting radionuclides in individual waste packages and an overall of 400 Bq/g of alpha emitting radionuclides per waste package [3].

TABLE IV. OUTPUT OF A REGIONAL RRSF CONDITIONING PLANT CONSIDERING A 30 YEAR PERIOD OF OPERATION

Materials, elements and nuclides	Mass	Volume
Immobilized spent fuel derivatives (LILW-LL)	627 t	220 m^3
Composition:		
Phosphate-based glass matrix	532.3 t	—
U	87.8 t	—
^{235}U	878 kg	—
Pu	11.8 kg	—
^{239}Pu	8.92 kg	—
Al	5.84 t	—
Fission Products	156 kg	—
Minor actinides (Np, Am, Cm)	1.37 kg	—
Structural Al from SF (VLLW)[3]	1.81 t	0.670 m^3

Furthermore, the waste thermal loading falls well below the 2 kW/m^3 limit as long as those 627 t, i.e. 220 m^3 of waste form, comprises just 28 kW, which means 0.127 kW/m^3, assuming that the spent fuel is processed at least two years after it is removed from the core. It is worth noting that the thermal loading declines to 4.71 kW after a decay period of ten years [6].

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[3] If the structural Al parts of the spent fuel assembly are not removed in the partner countries of the region, immobilization in the main waste stream at the conditioning plant is the preferred approach.

The quantity and volume of the immobilized waste that results from the conditioning process of all spent fuel used in the research reactors of the region in a 30 year period is much smaller than the values arising from the nuclear power plants operating in the region. Moreover, it is significantly less than the quantity and volume of the waste resulting from the processing of RRSF worldwide.

In conclusion, the total amount of waste — LILW-LL as well as VLLW — that results from the conditioning process is too small to justify the design, construction, operation and closure of a deep geological repository, even if it would be for regional use. Additionally, even if the need for disposing of the processing derivatives may be eventually established in the region in the medium term future, political, legal and public acceptance aspects for a regional repository may not be solved. Therefore, the results of this analysis point towards an international repository outside the region.

6. SPENT FUEL TRANSPORT

Transport of RRSF has been considered an important issue to be addressed in the region. In the national and regional context for spent fuel management, some scenarios have been identified, for example, countries that have research reactors at different sites and that are considering the centralized interim storage facility option for their spent fuel or its derivatives, countries that consider the option of modular dry storage by means of a dual purpose cask, the possibility that one country hosts the spent fuel conditioning service for other countries of the region, etc.

According to the reality of these countries, it is recognized that having a spent fuel transport capability in the region strongly contributes to possible solutions for spent fuel management inside the countries (in-country solutions). It also contributes to technically feasible regional solutions that require transport of spent fuel between countries.

For these reasons, the design and development of a dual purpose cask, for transport and dry storage of RRSF, is being carried out with the support of TC Regional Project RLA/4/018. In the first stage, Brazilian researchers and engineers from the CDTN and IPEN research centres started the design of a prototype cask. In the second stage, they were joined by engineers and researchers of the other participant countries, aiming to make the cask versatile enough to be used in the entire region.

In this sense, a major expected output of TC Regional Project RLA/4/018 is to attain the design of a dual purpose cask and its qualification by means of testing a scaled 1:2 prototype. For this activity, an agreement among the participants is the early involvement of the regulatory bodies of each country.

The following is a summary of the cask concept, which is shown in Fig. 20.

- The cask has been designed with a storage capacity to accommodate either 21 MTR spent fuel assemblies (assemblies up to 98.7 cm long with 20% enrichment U material) or 78 TRIGA spent fuel elements with 70% enrichment U material.
- The cask is a cylindrical packaging, vertically transported, provided with a double closure lid and protected by external shock absorbers. The cask body has a sandwich-like shielded wall: stainless steel–lead–stainless steel and metallic gasket seals. An assortment of items necessary for proper cask manipulation, connection to site services (water, gas, etc.), per-use leakage tests, and tie-down to the transport vehicle, are taken into account in the design. The maximum weight (when loaded) is 10 t.
- Criticality and shielding analyses were performed, conservatively for normal operation and accident scenarios, based on models of fuel existing in the region as well as the foreseen burn-up and cooling periods (about 420 g of ^{235}U in fresh fuel, 50% burn-up, 2 year cooling period for MTR fuel, 5 year cooling period for TRIGA, etc.).

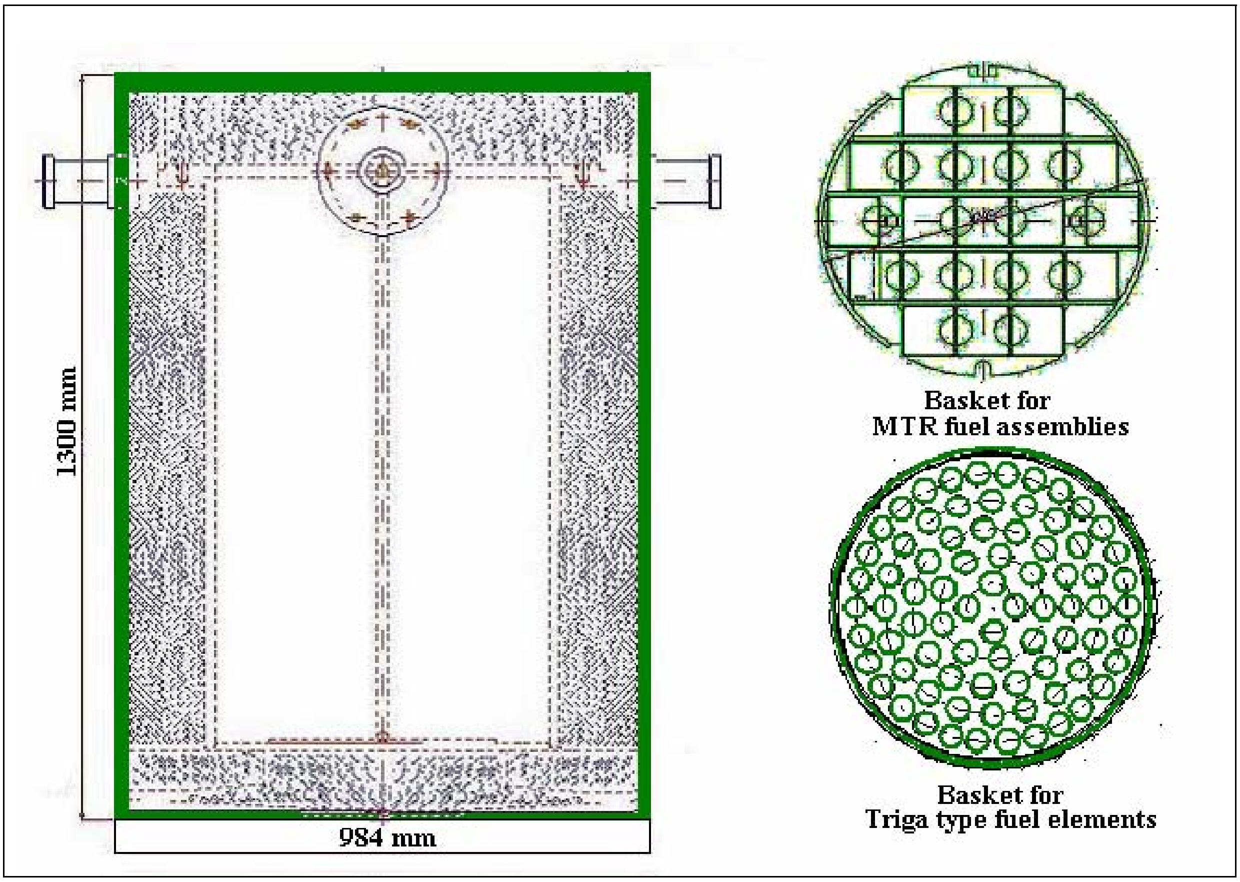


FIG. 20. The dual purpose cask.

7. NATIONAL STRATEGIES FOR RRSF MANAGEMENT

7.1. ARGENTINA

The Radioactive Waste Management Strategic Plan approved by CNEA includes the strategy for RRSF management that is to be periodically reviewed in order to have the opportunity to consider or adopt new concepts and processes.

Current planning considers that the operational wet storage will be followed by interim wet storage in a 16 m deep pool at LAPEP at the Centro Atómico Ezeiza, currently under construction. As regards the spent fuel currently stored at the facility with underground tubes, they will be sent to the storage pool at LAPEP.

The new pool is designed to store spent fuel from all operating research reactors in Argentina: RA-1 (40 kW), RA-3 (10 MW) and RA-6 (500 kW). Nevertheless, the RA-6 HEU fuel inventory will be shipped back to the USA within the framework of the USDOE Foreign Research Reactor Spent Fuel Acceptance Programme. It is also expected that irradiated HEU fuel assemblies and fuel elements from the RA-2 critical assembly will be shipped back to the USA.

Currently, experimental studies of conditioning are being performed involving diverse technologies. After evaluation of the different conditioning processes, with and without decladding, one conditioning route will be chosen for testing in hot cells with radioactive samples. Furthermore, in order to reduce amounts of LILW-LL — avoiding the need for isotopic dilution — a spent fuel processing option that includes separation of uranium is under consideration.

For the RRSF processing options that are being studied, the selected conditioning technology will be implemented up to hot pilot scale for an output of 2–3 MTR SF/week. Such a pilot plant could start operation in 2015 according to present planning.

Currently, aluminium based spent fuels from research reactors are in operational and interim wet storage. However, for an eventual extended storage as well as for disposal, there are some concerns about the suitability of the aluminium as a matrix resistant to water corrosion. Therefore, the preferred alternatives are extended storage in dry conditions and disposal of the derivatives from spent fuel processing instead of its direct disposal.

Current planning for the development of a deep geological repository includes the construction and operation of an underground research laboratory starting in 2030 and the operational start-up of the deep geological repository by 2050.

7.2. BRAZIL

As explained before, Brazilian engineers and researchers are producing the necessary documents to help the competent authorities to reach a final decision on the type of dry storage facility that will be implemented, to store the spent fuel from the IEA-RI research reactor. They also consider that a transfer cask will be necessary to transfer the spent fuel from the IEA-R1 reactor pool to the dry storage facility. Also, as part of the strategy, CDTN and IPEN are continuing the development of a dual purpose cask that can be used to dry store the spent fuels within the reactor building.

For disposal, no decision is foreseen in the next ten years, but it is strongly believed that the HLW, or the derivatives of RRSF, will be disposed of on the same site selected for disposal of spent fuel from Brazilian nuclear power plants. For this purpose, research is being done considering the possibility of dissolution of the spent fuel element, followed by uranium dilution and immobilization of the derivatives using glass matrices.

7.3. CHILE

The Chilean spent fuel assemblies will be maintained intact in wet operational storage at least until the decision to move them to interim dry storage has been taken. For this purpose, a water quality monitoring programme was implemented, and a surveillance programme, including corrosion tests, sipping tests and visual inspections, was established in accordance with approved protocol and procedures developed by experts of the participating countries.

During the biennium 2003–2004, a conceptual design of interim dry storage for spent fuel assemblies was developed, and at present the basic engineering design is being completed. When it is constructed, spent fuel assemblies will be processed for transfer to the facility. At that time the spent fuel assemblies will be stored under the basic criteria of confinement and retrievability.

Additionally, the option of maintaining the spent fuel assemblies intact avoids premature decisions that could affect future options. Maintaining the integrity of the spent fuel permits taking advantage of new technological solutions, more favourable for disposal of the spent fuel, which may be developed in the next decades.

7.4. MEXICO

ININ has recently implemented a project for the design of a dry storage cask and a study of the possibility of constructing a smaller pool, with the specific objective of storing the fuel from the research reactor. One of these options will be utilized for interim storage. If a new wet storage facility is selected, the pool would be smaller than the original reactor pool, in order to reduce costs for operation and maintenance. For the cask design, engineers of ININ are taking advantage of the dual purpose cask being developed in the TC Regional Project RLA/4/018 that can be used also for the TRIGA fuel.

Since the US Foreign Research Reactor Acceptance Programme has been extended, an attractive option is to send back the fuel to the USA, since all TRIGA fuel qualifies for this programme. Due to the fact that the HEU fuel should be sent back to the USA, the Mexican authorities must first negotiate replacement of the existing HEU fuel in the reactor core with fresh LEU fuel. The negotiations are necessary because the cost of TRIGA type fuel elements has considerably increased during the past years, making the cost to convert the complete reactor core high.

Reprocessing of the spent fuel of the ININ research reactor is not being considered as an option, as TRIGA spent fuel reprocessing has only been demonstrated on a laboratory scale and no commercial service is currently available.

As in other countries, the decision on TRIGA spent fuel disposal will be tied to the plans for spent fuel disposal of the nuclear power plants operating in the country.

7.5. PERU

The spent fuel of Peruvian reactors will be maintained in operational storage until a decision is reached about the disposal option. For this purpose, and in the framework of TC Regional Project RLA/4/018, a spent fuel surveillance programme has been implemented. It includes control of the water chemistry, sipping tests and visual inspection activities.

The authorities responsible for the Peruvian reactors have chosen operational wet storage, as, considering the current operational schedule of the reactor, the installation available at RP-10 has enough capacity to wet store all fuel assemblies used in the reactor during the next 35 years. If needed, the storage capacity of the installation can be increased simply by the installation of new storage racks. Activities beyond operational storage are considered to be long term issues.

It is foreseen to start studies on interim storage of spent fuel in the near future as the starting point of a management strategy.

ANNEX I. DESCRIPTION OF RESEARCH REACTORS IN COUNTRIES PARTICIPATING IN TC REGIONAL PROJECT RLA/4/018

Research reactors began to be built in Latin America in the late 1950s, within the US programme “Atoms for Peace”. Since then, 23 reactors were built in the region. The region has now 18 operational research reactors, 5 in shut down condition, and 1 has been decommissioned.

Table V shows the main characteristics of each of the reactors built in the countries that participated in TC Regional Project RLA/4/018.

What follows is a brief description of each reactor listed in Table V. Its purpose is to give the reader an overview of the history and the main characteristics of each reactor. It presents general information before discussing details related to the spent fuel management situation. In order to be complete, all reactors constructed are described, including those that have already been decommissioned. More details of the fuel consumption and spent fuel storage capacity of each installation will be presented in the next sections of this document.

TABLE V. RESEARCH REACTORS BUILT IN THE COUNTRIES THAT PARTICIPATED IN IAEA TC REGIONAL PROJECT RLA/4/018

Country	Facility name	Power (kW)	Reactor type	Status	Fuel type	First criticality
ARG	RA-0	0.01	ZPR - Tank	OPER	Pin	Jan. 1965
ARG	RA-1	40.00	TANK	OPER	Pin	Jan. 1958
ARG	RA-2	0.03	ZPR	SHUT	MTR	July 1966
ARG	RA-3	10 000.00	MTR – Pool	OPER	MTR	May 1967
ARG	RA-4	0.00	ZPR Homog.	OPER	Disk	Jane 1972
ARG	RA-6	500.00	MTR – Pool	OPER	MTR	Sep. 1982
ARG	RA-8	0.01	ZPR - Tank	OPER	Pin	June 1997
BRA	IEA-R1	5 000.00	MTR – Pool	OPER	MTR	Sep.1957
BRA	IPR-R1	250	TRIGA Mark 1	OPER	Rods	Nov. 1960
BRA	ARGONAUTA	0.50	ARG.	OPER	MTR	Feb. 1965
BRA	IPEN-MB	0.10	ZPR Tank	OPER	Pin	Nov. 1988
CHI	RECH-1	5 000.00	MTR - Pool	OPER	MTR	Oct. 1974
CHI	RECH-2	2 000.00	MTR – Pool	SHUT	MTR	Feb. 1977
MEX	CHI-Mod. 9000	0.00	SUBCR	SHUT	Pin	May 1969
MEX	CHI-Mod. 2000	0.00	SUBCR	OPER	Pin	Jan. 1969
MEX	SUR-100	0.00	ZPR Homog.	DEC	Disk	Sep. 1972
MEX	TRIGA Mark III	1 000.00	TRIGA Mark III	OPER	Rods	Nov. 1968
PER	RP-0	0.00	ZPR Tank	OPER	MTR	July 1978
PER	RP-10	10 000.00	MTR – Pool	OPER	MTR	Nov. 1988

I-1. ARGENTINA

Argentina has six operating research reactors that can be divided into two groups. The first group comprises the reactors that operate according to a regular schedule, which includes research reactors RA-1, RA-3 and RA-6. The second group comprises the critical assemblies RA-0, RA-4 and RA-8. Another reactor, RA-2, was shut down in 1983, and currently it is in the process of decommissioning. With the exception of RA-4, all Argentine research reactors use fuel elements fabricated in the country. The fuel elements of RA-4 were donated by Germany. What follows is a brief description of all seven Argentinian research reactors.

RA-0

RA-0 is a critical facility with a nominal power of 10 W. It is located at Córdoba National University (Universidad Nacional de Córdoba), which is responsible for the administration and operation of the reactor. It was developed for training and fundamental research, but with the main objective to study possible alternatives for geometry and configuration of the RA-1 research reactor. It has a LEU core composed of 183 rod elements of UO_2, similar to the core of the RA-1 research reactor. The fuel elements have a diameter of 9.3 mm and an active length of 540 mm, with a graphite segment at each axial end, used as a neutron reflector. The rods are placed in the form of circular rings, forming an annular cylinder. An inner graphite cylinder and an external arrangement of graphite blocks that surrounds the reactor core are used as reflector, and demineralized light water is used as moderator.

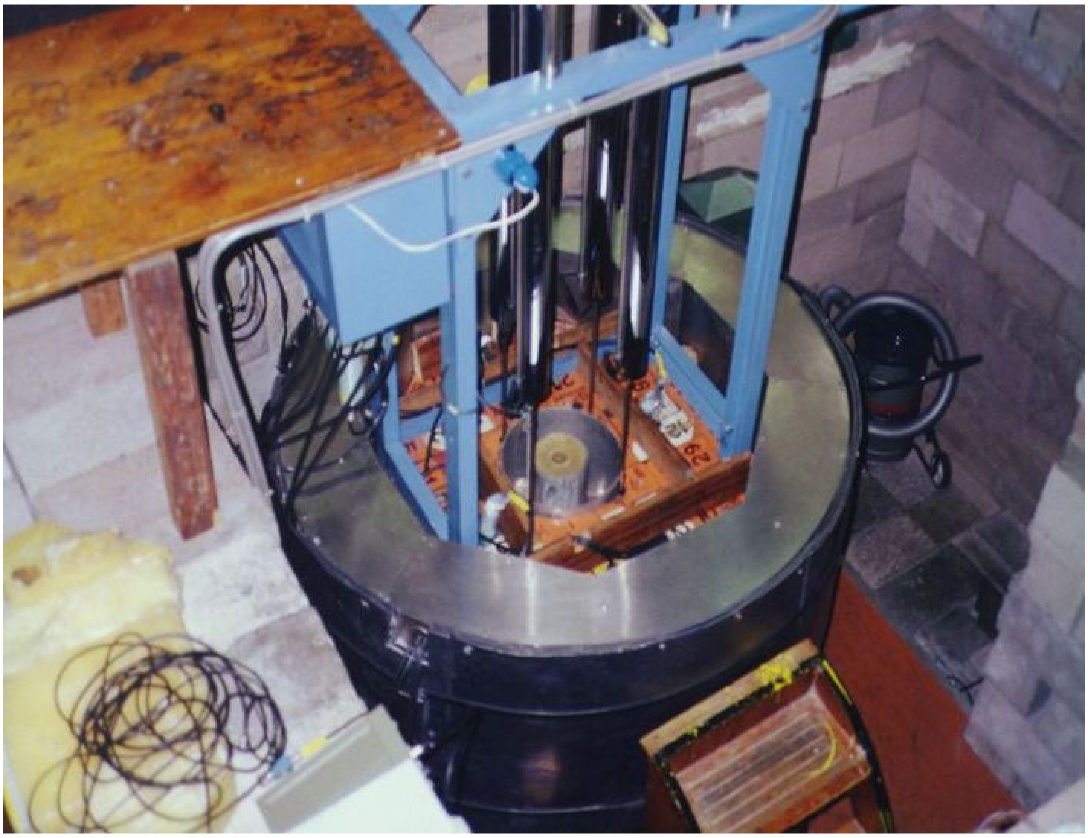
The reactor reached criticality for the first time in January 1965. Since it was designed to study alternatives for RA-1, during the design stage it was assumed that the reactor should be constructed close to RA-1. However, the flexibility of the reactor and the easy accessibility of all major components of the installation led to the conclusion that RA-0 was a research tool of great value. Therefore it was provided by CNEA to the Córdoba National University, in order to be used as a research tool for education purposes and for the further diffusion of nuclear activities in Argentina.

Figure 21 shows details of the RA-0 critical facility.

RA-1

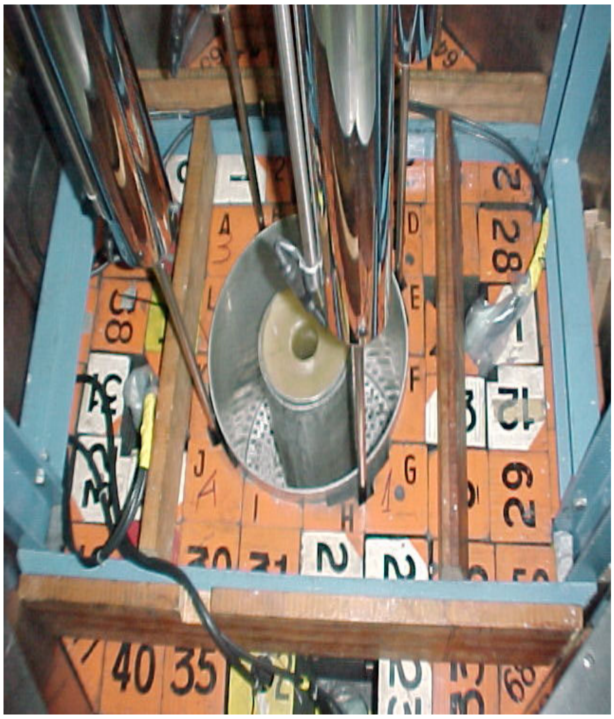
RA-1 is an open-tank type reactor operated at 40 kW, located at the Constituyentes Atomic Centre (Centro Atómico Constituyentes) in Buenos Aires. The Argentinian National Commission of Atomic Energy (Comisión Nacional de Energía Atómica — CNEA) is responsible for the administration and operation of the reactor. The core, originally designed as an Argonauta type, was later modified, and is formed by 228 cylindrical rods filled with low enriched UO_2 fuel. The rods are distributed into five concentric circles, forming an annular cylinder. The inner and outer diameters of the annular cylinder are 153 mm and 335 mm, respectively. The fuel elements are similar to the ones used in RA-0, i.e. fuel rods with a diameter of 9.3 mm, active length of 540 mm, and with a graphite segment at each axial end, to work as a reflector. The core is internally and externally reflected with graphite blocks.

Demineralized light water is used as refrigerant and moderator for the fission neutrons. First criticality of the reactor was reached in January 1958, and it is mainly used for research purposes, including studies of boron neutron capture therapy, studies of radiation damage of materials, and activation analysis.



(a) Reactor structure

(b) Control rod mechanism



(c) Top view of the reactor core

FIG. 21. Critical facility RA-0 (Argentina).

Figure 22 shows the RA-1 research reactor.

RA-2

RA-2 is a 30 W tank type critical assembly located at the Constituyentes Atomic Centre. The facility went critical for the first time in July 1966 and used MTR plate type fuel assemblies in the reactor core. Standard fuel assemblies had 19 fuel plates, and control assemblies 15. Four stainless steel cadmium control rods were used to control the reactor power. The core, with a cross-section of about 305 mm × 380 mm and an effective length of 65.5 cm, was surrounded by a graphite reflector, and demineralized light water was used as moderator and refrigerant. Cooling was done by natural circulation of the water through the reactor core. The reactor was shut down in September 1983, after occurrence of a criticality accident caused during a core reconfiguration sequence. The installation is in the process of decommissioning.

RA-3

RA-3 is an open tank MTR type research reactor located at the Ezeiza Atomic Centre (Centro Atómico Ezeiza), about 40 km from Buenos Aires. As in the case of RA-1, CNEA is responsible for the administration and operation of the reactor. The core of RA-3 is composed of 23 standard fuel assemblies and 4 partial fuel assemblies. The partial fuel assemblies (control assemblies) have fewer fuel plates than the standard fuel assemblies, in order to allow introduction of the control bars. The reactor is refrigerated and moderated using demineralized light water, and went critical for the first time in August 1968.

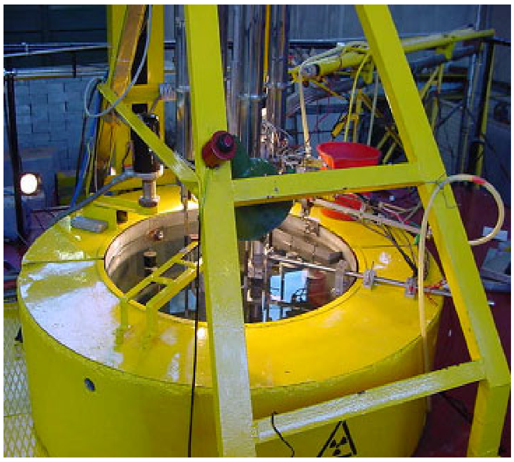
Until 1987 the reactor had fuel assemblies containing flat plates with high enriched uranium (HEU) in the form of U_3O_8 dispersed in aluminium, and the nominal power of the reactor was 2.8 MW. In 1988 a conversion was initialized, with the objective to increase the nominal power of the reactor and to convert the fuel to low enriched uranium (LEU). The modification, made within the framework of the Reduced Enrichment for Research and Test Reactors (RERTR) programme, ended in 1989, when the reactor returned to normal operation with a LEU core, with a thermal power of 5 MW and 120 hours/week of operation.

In 2002, while maintaining the LEU core and the same operation regime, the reactor power was increased to 10 MW, which is the current authorized power for the reactor. The reactor, constructed in a joint programme between CNEA and private industry in Argentina, is used for production of radioisotopes and for research applications like material testing, activation analysis, neutron radiography and testing of prototype fuel assemblies. Since the time of the reactor start-up the fuel assemblies have been manufactured in Argentina, primarily at CNEA, and also at the associated local companies.

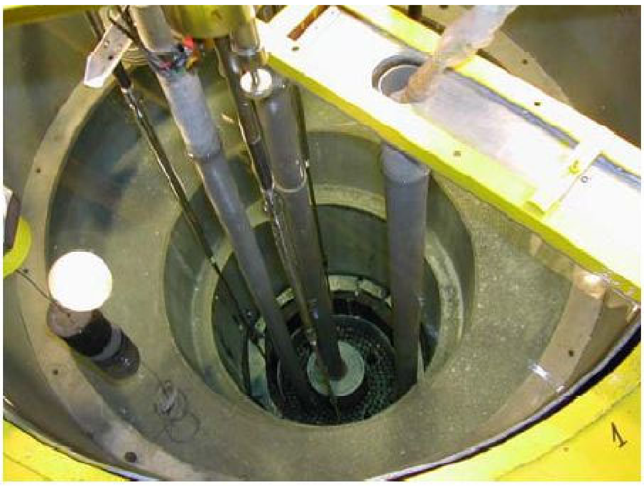
Figure 23 shows the RA-3 research reactor.

(a) Reactor structure

(b) Control rod mechanism



(c) Top of the reactor



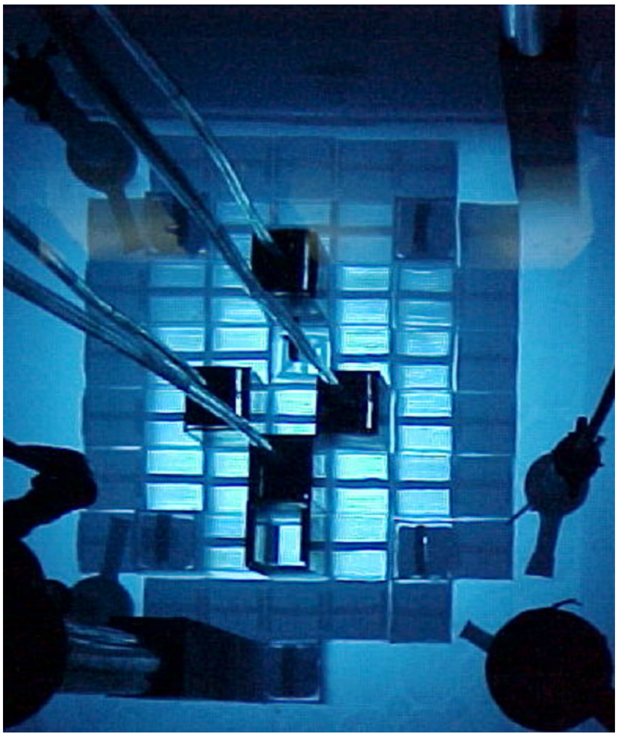
(d) Top view of reactor core

(e) Reactor core

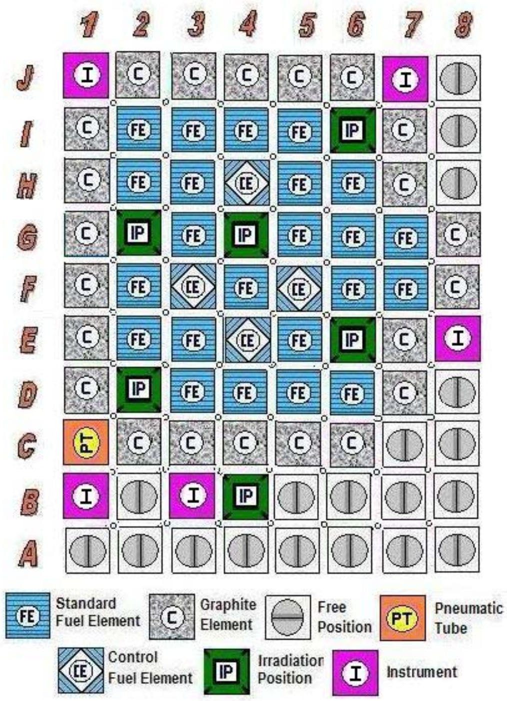
FIG. 22. Research reactor RA-1 (Argentina).

(a) Reactor building

(b) Reactor operation hall



(c) Reactor core



(d) Typical core configuration

FIG. 23. Research reactor RA-3 (Argentina).

RA-4

RA-4 is a critical facility with a nominal power of 100 mW. The reactor, a type SUR-100 (Siemens-Unterrichts-Reaktor 100 mW), reached criticality for the first time in January 1972, and is located on the campus of Rosario National University (Universidad Nacional de Rosario), in the province of Santa Fé. The university is responsible for the administration and operation of the reactor, which has a homogeneous core composed of LEU dispersed in the form of U_3O_8 in polyethylene disks, forming an annular arrangement with an internal diameter of 200 mm and an external diameter of 240 mm. The polyethylene is used as moderator, and graphite, surrounding the core, is used as reflector.

As in the case of research reactor RA-0, the installation was provided by CNEA for Rosario National University to be used as a research tool for graduate programmes, and for diffusion of nuclear activities in Argentina.

Figure 24 shows the RA-4 critical facility.

(a) Reactor tank

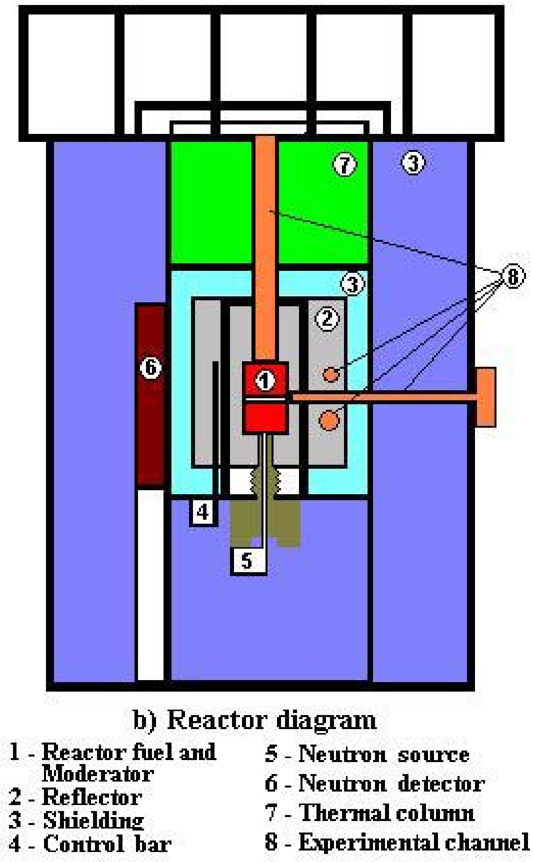


FIG. 24. Critical facility RA-4 (Argentina).

RA-6

RA-6 is located at the Bariloche Atomic Centre (Centro Atómico Bariloche), in the city of San Carlos de Bariloche, in the southern part of Argentina. It is an open vessel pool MTR type reactor rated at 500 kW. Since its start-up in 1982 the reactor core has used spent HEU fuel assemblies from the RA-3 reactor. The original core started as an arrangement of 25 HEU fuel assemblies and currently has 30 HEU fuel assemblies which are arranged into an

8 × 10 grid plate, within a 10.4 m cylindrical stainless steel tank with a 2.40 m diameter. The reactor core is kept about 6.60 m below water level.

The reactor, conceived as a tool for education, training and development in the field of nuclear engineering, went critical for the first time in September 1982. It is used for the development of technology related to boron neutron capture therapy (where the first treatment of a patient was carried out in 2003), neutron activation analysis, training and research (mainly material irradiation). At present, RA-6 is the only Argentinian reactor using HEU fuel elements. Studies have been conducted to convert the reactor core to LEU. At the end of 2004, CNEA and USDOE began negotiations to convert the core to LEU.

Figure 25 shows details of the RA-6 research reactor.

RA-8

RA-8 is a critical facility located at the Pilcaniyeu Technological Complex (Centro Tecnológico de Pilcaniyeu), about 60 km from San Carlos de Bariloche. The reactor, which reached criticality for the first time in June 1997, can be operated during steady state conditions at any constant power up to 10 W, or at 100 W in short transients. The reactor was developed with the specific objective to validate codes used for neutronic core calculations, and to study the nuclear design parameters of a modular Argentinian LWR power reactor named CAREM.

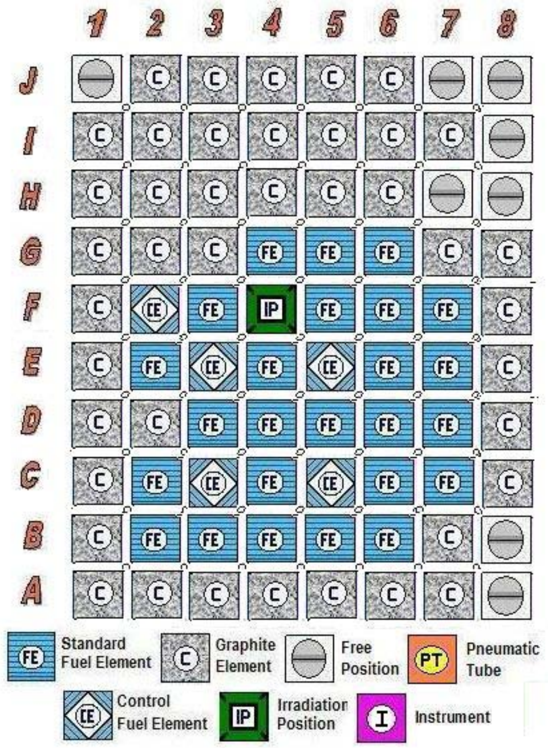
The fuel is composed of 1500 cylindrical rods made with zircaloy containing pellets of UO_2 with enrichment variable from 1.8 to 3.4%. The length of the rods is 1050 mm and the diameter 9 mm. Water is used as moderator and reflector. Natural convection of the water is used to remove the heat generated in the core. Control rods, containing an alloy of Ag–In–Cd, are used to regulate the reactor power and to shut down the reactor. Due to the versatility of the reactor, which allows realization of many academic experiments, the facility is used mainly for training purposes in the field of power reactor engineering, and for determination of nuclear parameters of the CAREM LWR power reactor.

Figure 26 shows details of the RA-8 critical facility.

(a) Reactor building

(b) Reactor pool

(c) Reactor core



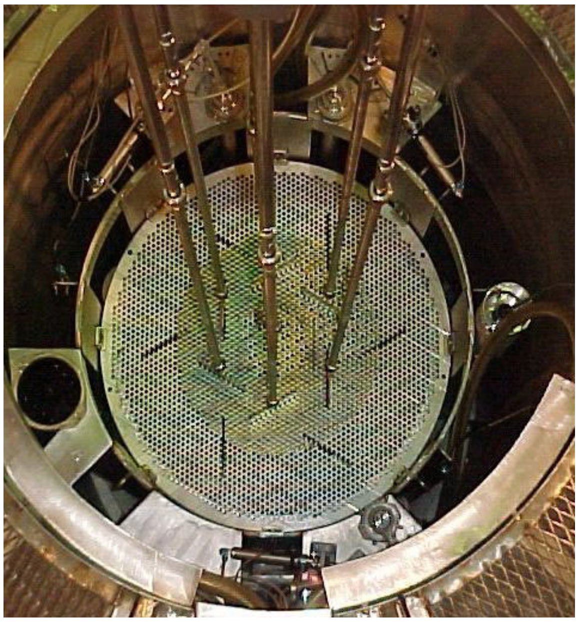
(d) Typical core configuration

FIG. 25. Research reactor RA-6 (Argentina).

(a) Pool structure

(b) Assembling the reactor core structure

(c) Top view of the reactor



(d) Typical core configuration

FIG. 26. Critical facility RA-8 (Argentina).

I-2. BRAZIL

In Brazil there are four operational research reactors, IEA-R1, IPR-R1, Argonauta and IPEN/MB-01. All are operated by the Brazilian National Commission of Nuclear Energy (Comissão Nacional de Energia Nuclear — CNEN).

IEA-R1

IEA-R1 is the oldest research reactor in the southern hemisphere. Located at the Energy and Nuclear Research Institute (Instituto de Pesquisas Energéticas e Nucleares — IPEN), on the campus of São Paulo University, in São Paulo, it reached criticality for the first time on 16 September 1957. It is an MTR pool type reactor designed by Babcock & Wilcox Co., cooled and moderated using demineralized light water. The original core consisted of a 6 × 5 lattice with 26 standard fuel assemblies and 4 control assemblies, and was converted to a 5 × 5 lattice with 20 standard fuel assemblies, 4 control assemblies and one special irradiation device at the centre of the lattice. Standard and control fuel assemblies have 18 and 12 flat plates, respectively. The core is surrounded with beryllium and aluminium canned graphite in an 8 × 10 grid plate, which is suspended by an aluminium structure. A layer of water 8 m thick covers the reactor and is used to cool the reactor, and as radiation shielding.

Original fuel assemblies were made of plates containing high enriched U–Al alloy. In the mid-1980s, with increasing international restrictions on obtaining HEU, a programme was started to convert the core to LEU. It began with five fuel assemblies of UAl_x–Al dispersion type, bought from Germany. A decision was then made to locally produce and qualify the fuel elements necessary to continue operation of the reactor. Local production started with two prototypes of partial fuel assemblies with LEU U_3O_8–Al fuel meat, which in 1988 were introduced into the core for performance and qualification tests. After that, an average of two new nationally produced fuel assemblies were introduced into the core each year, including control assemblies, implementing a gradual process to convert the core from HEU to LEU. The process was concluded in 1997, and since then all fuel assemblies introduced into the core are locally made using LEU (19.75% of ^{235}U). The fuel elements were constructed using U_3O_8–Al dispersion fuel plates with densities of 1.9 gU/cm^3 from 1988 until 1996, and 2.3 gU/cm^3 from 1996 to 1999. In September 1999 IPEN started using U_3Si_2–Al dispersion fuel plates with 3.0 g U/cm^3.

Although designed to operate at 5 MW, from 1957 until 1961 operation of the reactor was mainly for commissioning tests and some nuclear physics experiments, with a power level between 200 kW and 2 MW. In 1961 a programme was established to produce ^{131}I, and the reactor began to be operated at a constant power of 2 MW, 8 hours per day, 5 days per week. In 1995 a new programme was established to increase the national production of radioisotopes, and the operating regime was changed to continuous 64 hours per week, from Monday through Wednesday, keeping the reactor power at 2 MW. Also, at that time, some modifications took place to bring the reactor into compliance with new national legislation to allow the reactor to operate continuously during 120 hours per week at 5 MW. The modifications were concluded in 1997, coincidently with the conclusion of the conversion process of the reactor core from HEU to LEU.

Currently, the reactor operates at 4 MW and is used mainly for radioisotope production, neutron activation analysis, neutron radiography, and research studies, including nuclear physics experiments and boron neutron capture therapy. A further power increase to 5 MW is expected to occur after replacement of the original heat exchanger used in the reactor cooling primary system.

Figure 27 shows the IEA-R1 research reactor.

IPR-R1

IPR-R1 was the second research reactor built in Brazil. It is located at the Centre for Development of Nuclear Technology (Centro de Desenvolvimento de Tecnologia Nuclear), on the campus of the Federal University of Minas Gerais, in Belo Horizonte. It is a TRIGA Mark I type reactor supplied by General Atomics.

The original core of the reactor consisted of 59 low enriched U–Zr–H rods, 3 control rods, a central irradiation tube and an outer cylindrical irradiation ring. In 2002, four additional fuel elements were introduced into the reactor, to compensate for the loss of reactivity due to fuel burn-up. The hydrate of zirconium existing in the fuel rod acts as moderator for fission neutrons, and demineralized light water is used to remove the heat produced in the core. The 59 original fuel rods have aluminium cladding, and the four fuel elements introduced in 2002 have stainless steel cladding.

The fuel rods, 724 mm in length and 37.6 mm in diameter, together with some graphite rods, a neutron source, three control rod guide thimbles and one irradiation tube, are located within a cylindrical aluminium structure surrounded by a graphite ring that works as reflector for the fission neutrons. The internal and external diameters of the graphite ring are 441 mm and 1090 mm, respectively.

The fuel–graphite assembly and the aluminium supporting structure sit at the bottom of the pool, about 6 m below the water surface.

Originally the nominal power of the reactor was 100 kW, and since 2000, after completion of a modernization process, some experiments have been performed, with the authorization of the Brazilian regulatory authority, in order to increase the nominal power of the reactor to 250 kW. Since its first criticality, in November 1960, the reactor has been used for research purposes, mainly involving neutron activation analysis.

Figure 28 shows the research reactor IPR-R1.

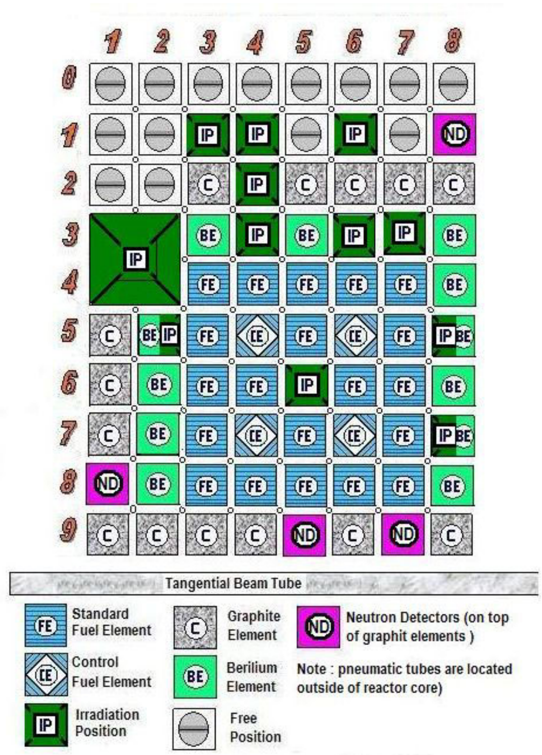
ARGONAUTA

Argonauta is the third Brazilian research reactor. Located at the Nuclear Engineering Institute (Instituto de Engenharia Nuclear), on the campus of the Federal University of Rio de Janeiro, in Rio de Janeiro, it reached criticality for the first time in February 1965. It is an Argonauta type reactor, built by Brazilian engineers and researchers, according to a project supplied by Argonne National Laboratory. The design power of the reactor is 5 kW, but the authorized operating power is only 500 W continuously, or 1 kW during 1 h of operation.

(a) Reactor building

(b) Reactor pool

(c) Reactor core

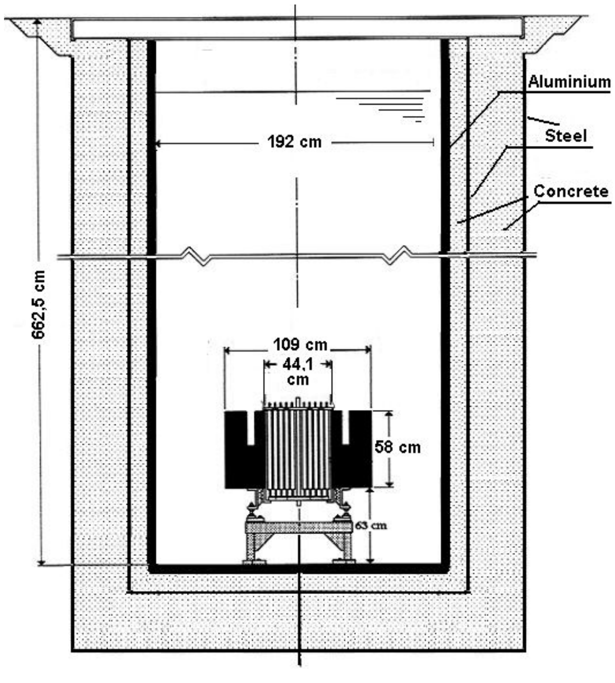


(d) Core configuration

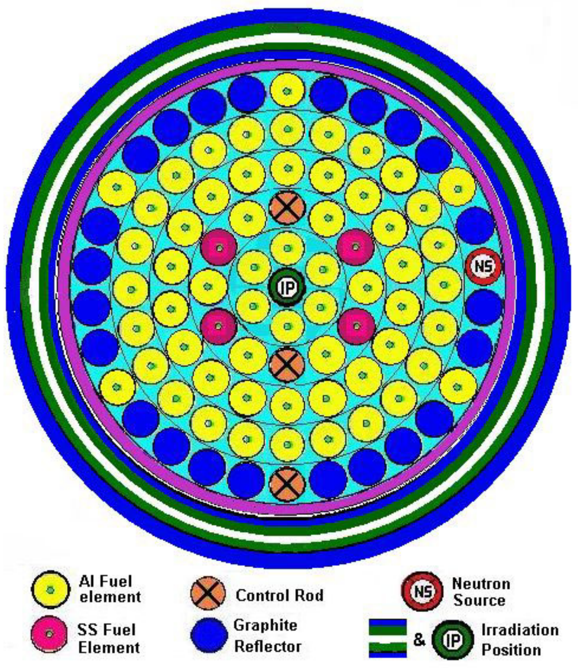
FIG. 27. Research reactor IEA-R1 (Brazil).

(a) Reactor building

(b) Reactor pool



(c) Reactor core



(d) Core configuration

FIG. 28. Research reactor IPR-R1 (Brazil).

The core is composed of LEU in the form of a mixture of U_3O_8–Al dispersed in aluminium, forming aluminium clad fuel plates specially arranged to form fuel assemblies. In its actual configuration the reactor has eight fuel assemblies as follows: four assemblies of type STD; two assemblies of type EMX and two assemblies of type EGP. STD assemblies have 17 fuel plates containing 21 g of ^{235} U each; EMX assemblies have 11 fuel plates containing 21 g of ^{235}U each and 6 fuel plates containing 10 g of ^{235}U each; and EGP assemblies have only 7 fuel plates containing 10 g of ^{235}U each.

The assemblies are arranged together with wedge shaped graphite elements to form a sector of an annular cylinder. The core is surrounded with graphite blocks to work as a reflector. A cylindrical block of graphite is also inserted into the inner part of the annular cylinder, with the same objective. Demineralized water is used as moderator and to remove the heat produced in the core, by natural circulation. The reactor is used for training, research and some minor radioisotope production.

Figure 29 shows the Argonauta research reactor.

IPEN/MB-01

IPEN/MB-01, the newest Brazilian research reactor, is also located at IPEN, close to IEA-R1. The reactor, a water tank type critical facility, with a nominal power of 100 W, is the result of a national joint programme developed by the Brazilian National Commission of Atomic Energy and the Brazilian Navy. The reactor core consists of up to 680 stainless steel fuel pins with UO_2 pellets inside. The diameter of the pin is 9.8 mm, and its length is 1194 mm. The pins have an active length of 546 mm, filled with 4.3% enriched UO_2 pellets. The remainder of the pin is filled with Al_2O_3 pellets.

The pins are manually inserted into a perforated matrix plate, making any desired experimental arrangements within a 28 × 26 matrix. The control and safety assemblies are composed of a total of 48 pins that contain absorbing material. Each safety/control assembly has 12 pins. Detectors around the structure that sustains the matrix plate complement the critical arrangement, which is maintained within a stainless steel tank. Demineralized water is used as moderator and cooling refrigerant. The reactor reached criticality for the first time in November 1988, and since then it is used for training, validation of neutronic codes, and for determination of nuclear parameters for small LWR power reactors.

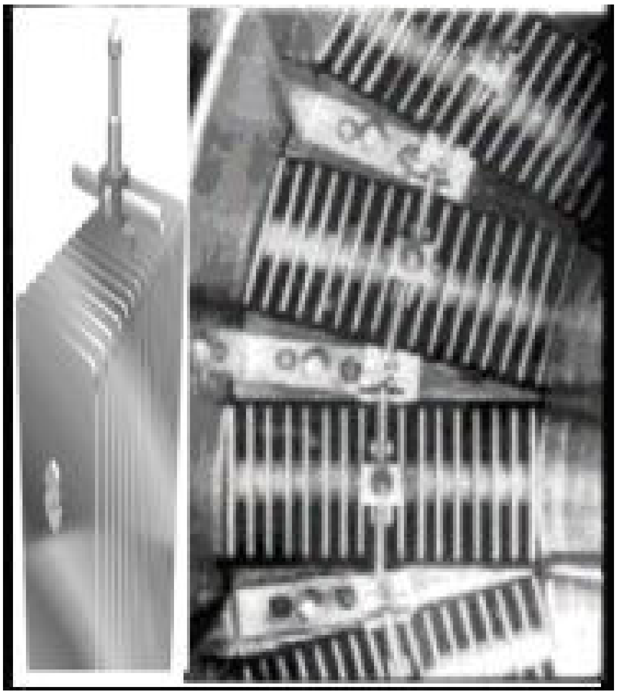
Figure 30 shows the IPEN/MB-01 critical facility.

I-3. CHILE

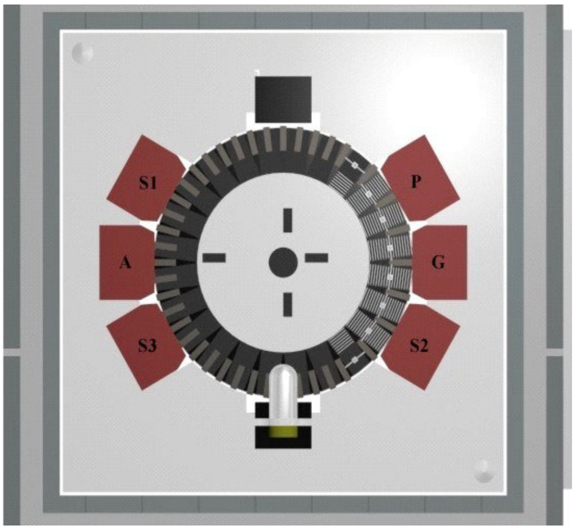
In Chile there are two research reactors, RECH-1 and RECH-2, both administrated and operated by the Chilean Commission of Nuclear Energy (Comisión Chilena de Energía Nuclear — CCHEN).

(a) Reactor building

(b) Reactor block



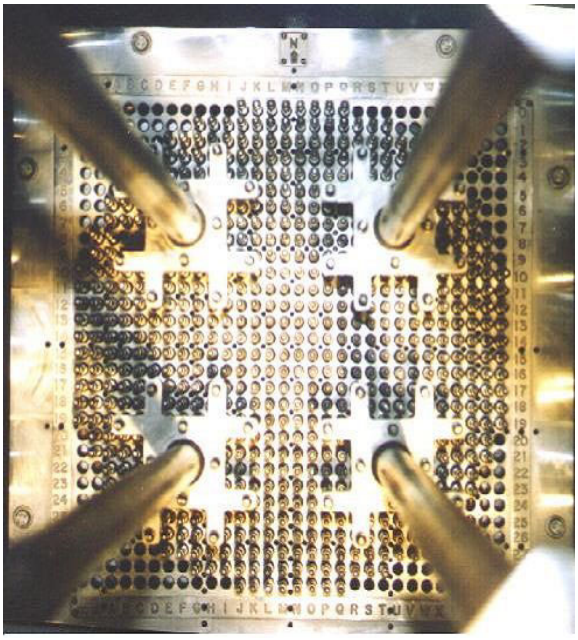
(c) Reactor core



(d) Core configuration

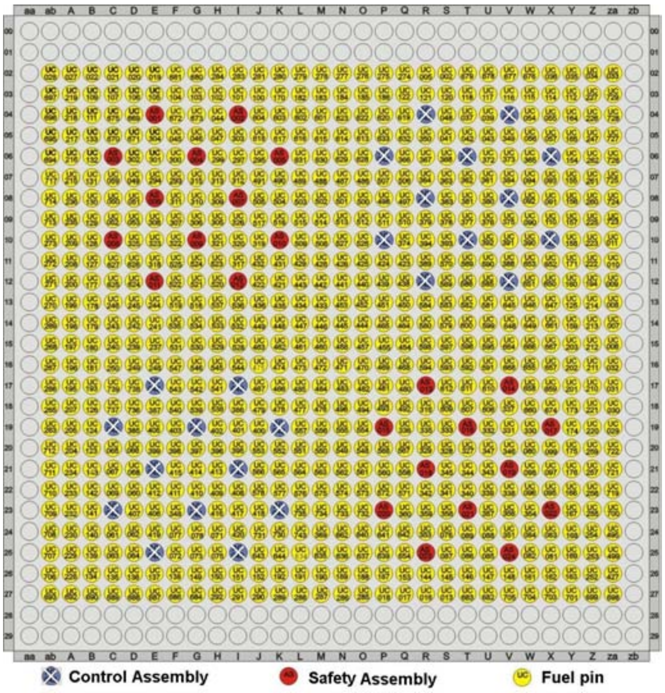
FIG. 29. Research reactor Argonauta (Brazil).

(a) Reactor building



(b) Reactor core (grid plate)

(c) Reactor structure



(d) Core configuration

FIG. 30. Research reactor IPEN/MB-01 (Brazil).

RECH-1

RECH-1 is a 5 MW open pool type research reactor cooled and moderated by light water, reflected by beryllium and using MTR type fuel. The reactor is located at the La Reina Nuclear Centre (Centro de Estudios Nucleares La Reina), in Santiago, and went critical for the first time in October 1974, using HEU (80% of ^{235}U) plate type fuel assemblies fabricated by the United Kingdom Atomic Energy Authority (UKAEA) at Dounreay, Scotland, with uranium enriched in the USA. A second load of fuel assemblies was also fabricated by UKAEA, using British HEU (45% of ^{235}U).

From 1985 to 1998, the reactor operated with a mixed core configured with HEU fuel assemblies of two different enrichments, 80% and 45% of ^{235}U. In 1998, CCHEN started the conversion of the reactor core, using LEU (19.75% of ^{235}U) and fuel assemblies fabricated by the Chilean Fuel Fabrication Plant. Since 1998 national fuel assemblies have been gradually delivered to the reactor, and at the time of writing (November 2005) the core of RECH-1 is a mixed core configured with 10 HEU fuel assemblies and 24 LEU fuel assemblies. The total conversion of the reactor core is expected to be completed in 2006.

Six stainless steel clad cadmium blades, inserted between the fuel assemblies, are used for control and safety purposes. Beryllium blocks are used as reflector, and demineralized light water is used as moderator and coolant. The reactor is used for radioisotope production, neutron activation analysis, geological samples irradiation, training, and research involving material irradiation, neutron radiography and neutron physics.

Figure 31 shows the RECH-1 research reactor.

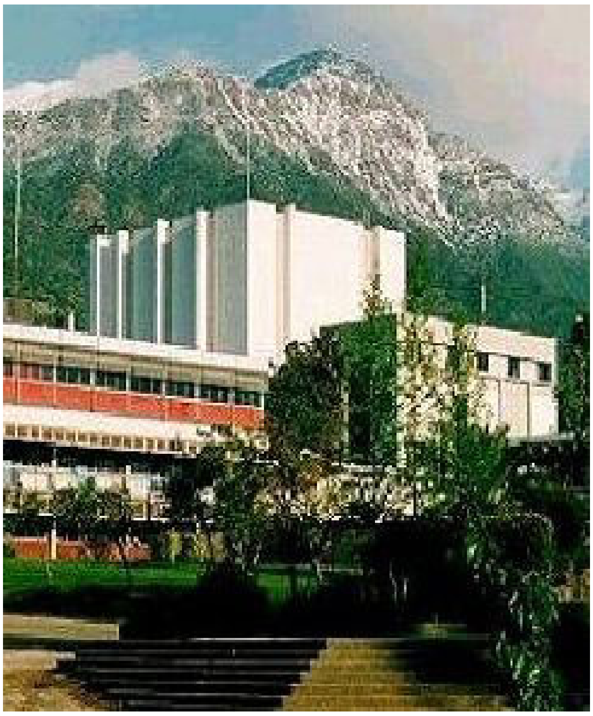
RECH-2

RECH-2 is also an MTR open pool type research reactor and is the result of a technical cooperation agreement signed in 1972 between Spain and Chile. The reactor is located at the Lo Aguirre Nuclear Centre (Centro de Estudios Nucleares Lo Aguirre), and it reached criticality for the first time in February 1977. According to the original design, the nominal power of the reactor is 10 MW. The original 31 fuel assemblies, made of French HEU (90% of ^{235}U), were produced in Spain, by the Junta de Energía Nuclear de España (JEN).

In 1986, a detailed design revision for improving the reactor was made in a joint effort between CCHEN and JEN. During the major modification of the reactor, which ended in 1989, the initial 31 fuel assemblies had to be repaired and only 29 could be reassembled, resulting in a limitation imposed on the reactor power. Due to the limitations in the amount of fuel assemblies available, the operating reactor power was established as 2 MW.

The 29 fuel assemblies are surrounded with graphite elements, used as reflector for the neutrons. Demineralized light water is used as moderator and coolant for the reactor. Due to the limitation in the power level, the reactor is not being operated. Studies have been made in order to convert the core to the use of LEU silicide fuel, but so far no decision has been taken.

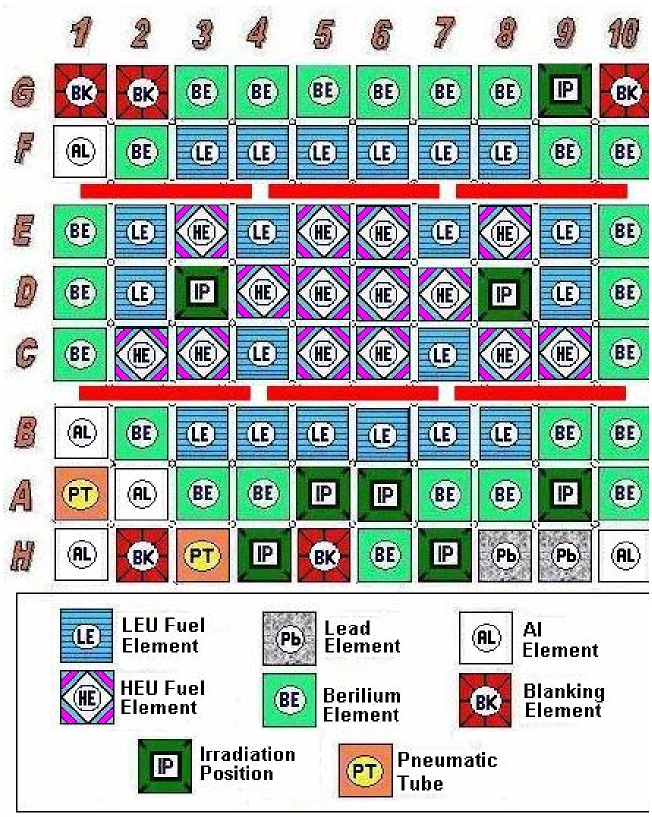
Figure 32 shows the RECH-2 research reactor.



(a) Reactor building

(b) Reactor platform

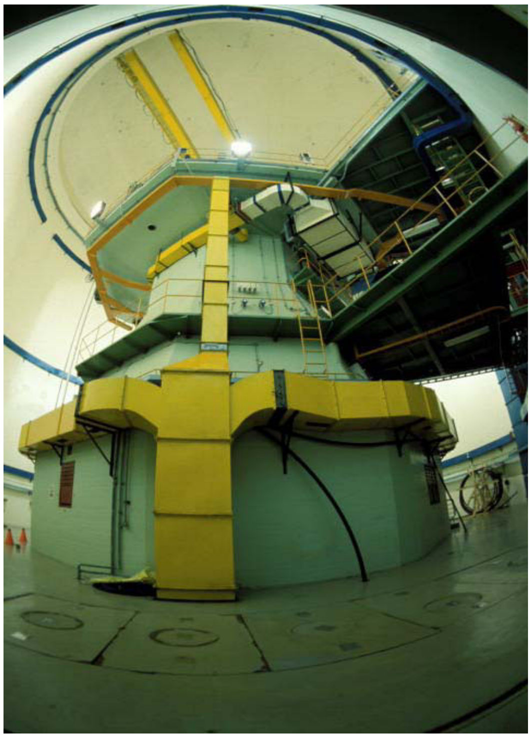
(c) Reactor core



(d) Core configuration

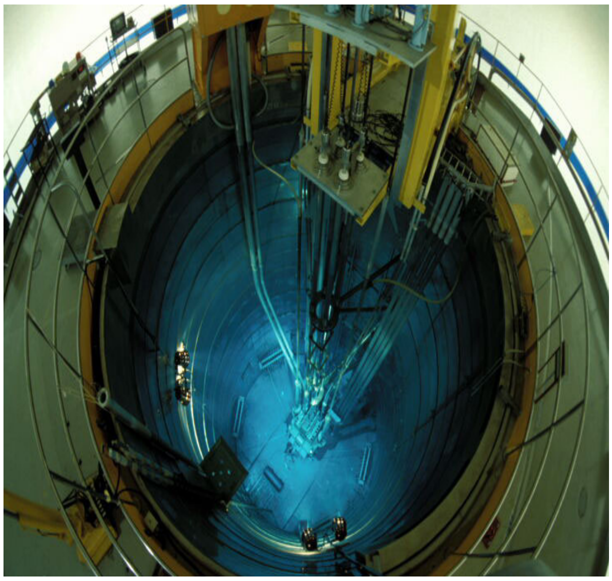
FIG. 31. Research reactor RECH-1 (Chile).

(a) Reactor building



(b) Reactor pool

(c) Reactor work platform



(d) Reactor supporting structure

FIG. 32. Research reactor RECH-2 (Chile).

I-4. MEXICO

In Mexico there is one operational research reactor, a 1 MW TRIGA reactor, and one operational subcritical facility. Another reactor, a SUR-100 homogeneous reactor, was decommissioned after shutdown in 1984, and another subcritical facility is in the condition of 'shut down'.

TRIGA MARK III

The principal research reactor of Mexico is a TRIGA MARK III pool type reactor that is located at the Mexican National Institute of Nuclear Research (Instituto Nacional de Investigaciones Nucleares) and which went critical for the first time in November 1968. The reactor has a steady state nominal power of 1 MW, and in the pulsed mode it can reach is 2000 MW for 10 milliseconds.

The reactor core geometry is formed with six concentric rings. The rings are named from B (the innermost one) to G (the outermost one). Each ring has a multiple of 6 positions available for placement of fuel, instrument or irradiation devices: the first ring has 6 positions, the second 12, and so on. The original core of the reactor consisted of 79 of the so-called standard TRIGA[4] type fuel elements, and three standard control rods with fuel follower[5]. The outer ring G was filled with graphite elements and used as reflector. In October 1978, six additional standard fuel elements were added to the core. In November 1988 a mixed core configuration was adopted, with 59 standard fuel elements, 26 FLIP[6] type fuel elements, three FLIP type control rods with fuel follower, and one fourth control rod, called transient rod, which can be 'fired' pneumatically to generate the 2000 MW transient. The FLIP elements are made using 70% HEU.

As explained for IPR-R1, the TRIGA fuel is a mixture of zirconium hydride moderator homogeneously combined with partially enriched uranium. The active section of this fuel moderator element is 381 mm in length and 36 mm in diameter, and it contains approximately 8.5 ^{w}wt% of uranium. For the standard TRIGA type fuel element the enrichment is 20%, and for the FLIP type fuel element it is 70%. The fuel section, together with two top and bottom graphite slugs, are contained within an aluminium structure, forming a cylinder of 2.1 m height and 53 cm in diameter, with several perforations to allow water to enter and flow through the core, in order to provide the necessary cooling of the reactor. The reactor core structure is immersed in a pool with an aluminium liner and a high density concrete structure. The dimensions of the pool are: 7.6 m length, 3.1 m width and 7.6 m depth, with a total volume of 150 m^3. The active core is kept about 6 m below the water surface of the pool.

Figure 33 shows the ININ TRIGA research reactor of Mexico.

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[4] TRIGA is an abbreviation for "Training, Research, Isotopes, General Atomics". It is a trademark of General Atomics.

[5] The control rod with follower has, at the bottom, a section with fuel, which enters the core as the control rod is removed.

[6] FLIP is an abbreviation for "Fuel Life Improvement Programme".

SUBCRITICAL ASSEMBLY NUCLEAR CHICAGO MOD. 2000

The subcritical assembly Nuclear Chicago Mod. 2000 is located in the northern part of Mexico City, at the National Polytechnic Institute (Instituto Politécnico Nacional — IPN), and is operated by the Nuclear Engineering Department of the Institute. The assembly consists of 1400 natural uranium fuel cylinders with aluminium cladding (21 cm long and 3 cm diameter, weighting 1.804 kg each) placed in 280 cylindrical aluminium tubes arranged in hexagonal shape in a stainless steel tank which is filled with water. The centre of the tank has provision for allocating a Pu–Be neutron source.

The facility was operated for the first time in January 1969, and it is basically used for training students enrolled in the nuclear engineering programme. The training activities include radiation protection surveys, neutron flux measurements and material activation analysis, as well as measurements of reactor parameters.

Figure 34 shows the subcritical assembly Nuclear Chicago Mod. 2000.

SUBCRITICAL ASSEMBLY CHICAGO MODELO 9000

Subcritical assembly Chicago Modelo 9000 is located at the Regional Centre of Nuclear Studies (Centro Regional de Estudios Nucleares — CREN) that belongs to the University of Zacatecas (Universidad de Zacatecas), in the Mexican State of Zacatecas. The assembly is similar to the subcritical assembly Nuclear Chicago Mod. 2000 located at the National Polytechnic Institute. It consists of 1400 natural uranium fuel cylinders with aluminium cladding (21 cm long and 3 cm diameter, weighting 1.804 kg each) placed in 280 cylindrical aluminium tubes arranged in hexagonal shape in a stainless steel tank which is filled with water. The centre of the tank has provision for allocating a Pu–Be neutron source.

The facility was operated for the first time in May 1969, and since 1973 it is out of operation because of corrosion problems in the structural aluminium tubes and in the spacing plates, one of them particularly highly corroded. At present, the fuel is kept in storage boxes, and the tank is empty. The neutron source is in its storage device, which is appropriately shielded. The fuel cylinders are in good condition and are periodically inspected thoroughly by visual and X ray inspection.

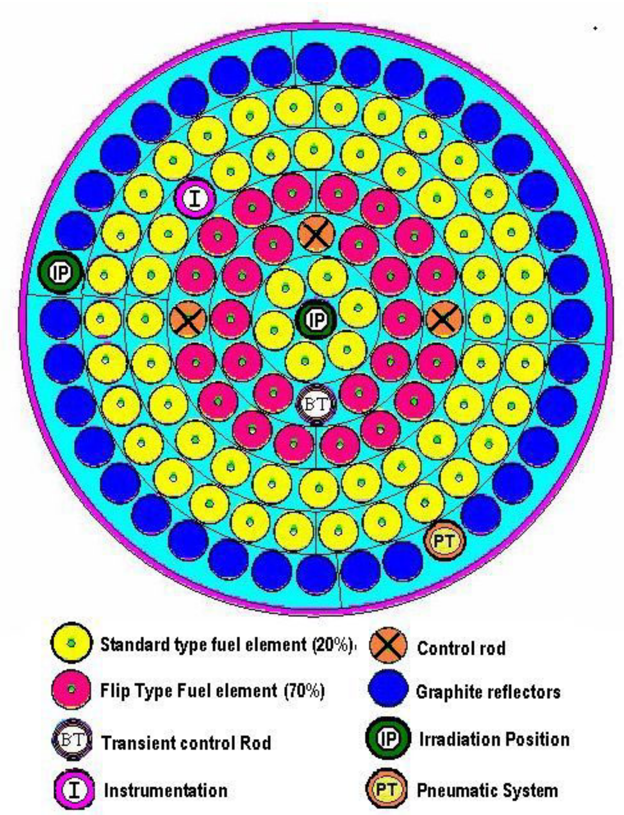
SUR-100

The SUR-100 reactor is a solid homogenous reactor with 20% enrichment uranium fuel, moderated by polyethylene and using graphite as reflector. The core has the form of a cylinder, with 24 cm in diameter and 26 cm high, and is cooled by natural circulation of air. It is made up of fuel elements in the form of polyethylene discs of 24 cm diameter and stacked to a total height of about 26 cm divided into two symmetrical blocks; the upper block is fixed and the lower one is movable. Two cadmium plates outside the core — in the reflector region —– are used as control rods. The reactor also has a Ra–Be neutron source.

(a) Reactor building

(b) Reactor pool

(c) Reactor core



(d) Core configuration

FIG. 33. The ININ TRIGA research reactor (Mexico).

FIG. 34. Subcritical facility Nuclear Chicago Mod. 2000 (Mexico).

The reactor was operated by the Nuclear Studies Centre, which belongs to the National Autonomous University of Mexico (Universidad Nacional Autónoma de México — UNAM) and was mainly used for training. The reactor was offered to UNAM by Siemens in 1971 and reached criticality in September 1972. In 1984 the reactor was shut down because utilization was very limited. The reactor was dismantled in 1989 and the fuel was sent to the ININ for surveillance and physical protection, as required by safeguards regulations.

I-5. PERU

In Peru there are two research reactors, RP-0 and RP-10, both operated by the Peruvian Institute of Nuclear Energy (Instituto Peruano de Energía Nuclear — IPEN).

RP-0

The RP-0 reactor is located at IPEN headquarters in the San Borja district, close to downtown Lima. It is a critical facility that reached criticality for the first time in July 1978 using extruded rod type fuel elements made with 20.09% enriched UO_2 mixed with graphite and aluminium cladding. In 1991 the core was converted to 19.75% enriched fuel. In June of that year the reactor reached criticality using MTR type fuel elements supplied by the Argentinian National Commission of Atomic Energy (Comisión Nacional de Energía Atómica — CNEA).

Until 1991, RP-0 was used for personnel training in the fields of reactor operation, reactor maintenance and reactor physics (calculus and experimental). Currently it is used as a mock-up of the RP-10 research reactor, for experiments to measure nuclear parameters, and for teaching and training. The reactor has a nominal power of 10 W and it is operated three days per week, during four hours each day. The facility has a dry storage room that is used to store the fresh fuels that are not in the reactor core. The capacity of the room is to store 28 fuel assemblies.

Figure 35 shows the critical facility RP-0.

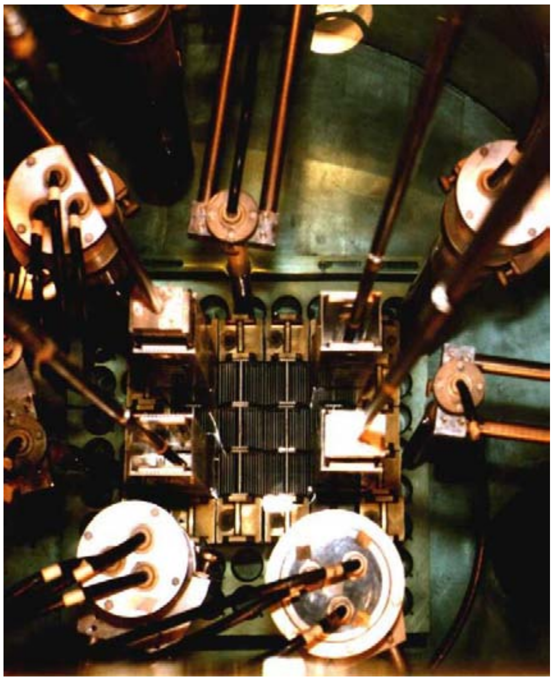
RP-10

RP-10 is a 10 MW research reactor designed and constructed by Argentinian engineers, with the main objectives to produce neutrons for research and for radioisotope production. Located at the RACSO Nuclear Centre, about 30 km from Lima, it reached criticality for the first time in November 1988.

The reactor is a typical MTR pool type reactor, using standard LEU plate type fuel assemblies. The core has a rectangular shape formed by a 5 × 6 lattice. It has 24 standard fuel assemblies, five control assemblies, and one irradiation position at the centre of the lattice. The standard fuel assemblies have 16 plates containing LEU in the form of U_3O_8 dispersed in aluminium; the control fuel assemblies have 12 plates. The core is surrounded with eight beryllium elements and 13 graphite elements that work as reflector for the fission neutrons. Demineralized light water is used as moderator and coolant.

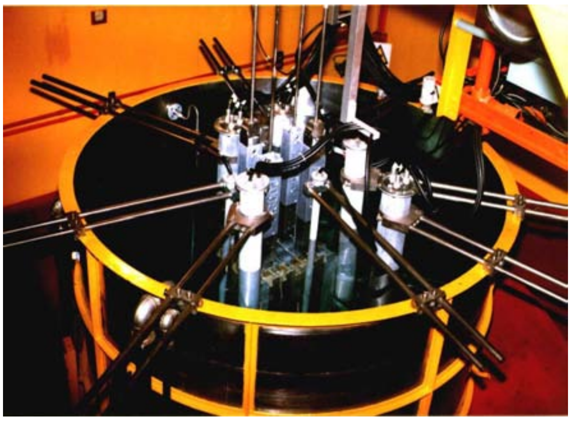
Figure 36 shows details of the RP-10 research reactor.

(a) Reactor building

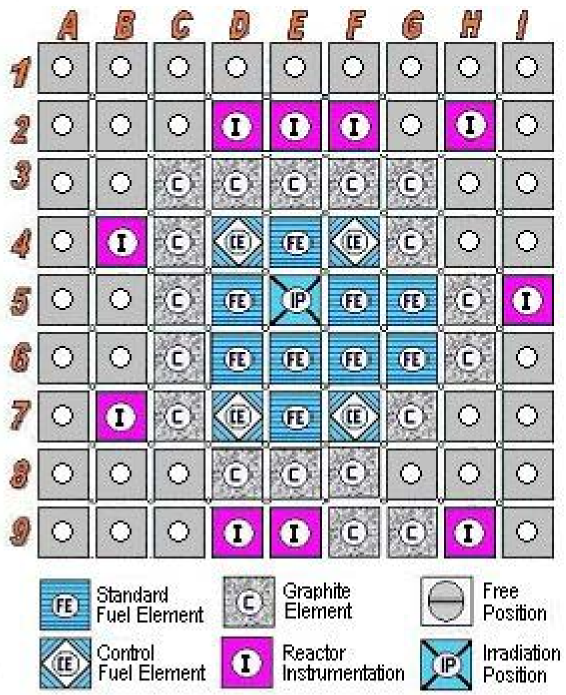


(b) Reactor core

(c) Reactor hall



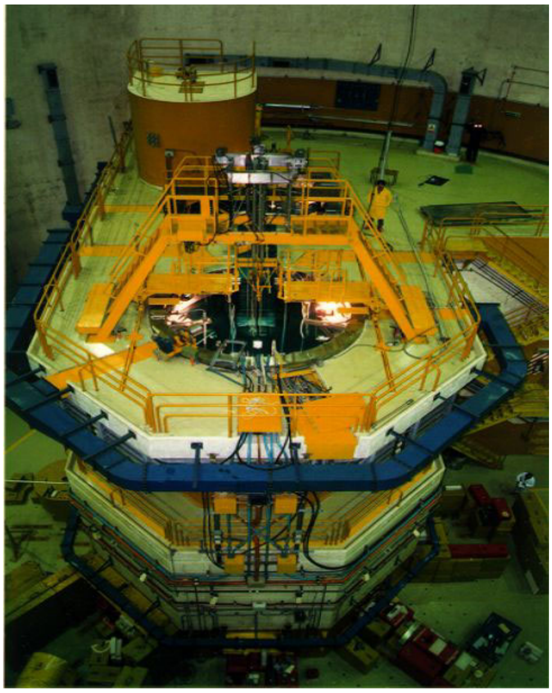
(d) Reactor pool



(e) Core configuration

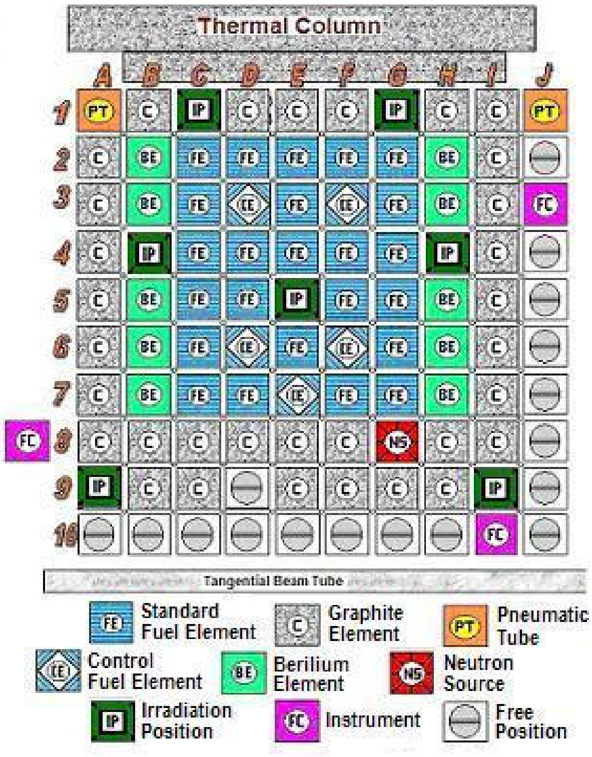
FIG. 35. Critical facility RP-0 (Peru).

(a) Reactor building



(b) Reactor pool

(c) Reactor core



(d) Core configuration

FIG. 36. Research reactor RP-10 (Peru).

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