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Author(s): **E Vico del Cerro**

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Integrated Waste Management Approach WP 1 Review Report – Italy and Spain

Executive summary

Although Italy has constructed 4 commercial reactors only one (Latina) is graphite moderated. None of these four commercial reactors are now in operation. Ten experimental/test reactors have been constructed about half are still in operation. It is unclear how many of these reactors contain graphite but based on design it is predicted that some may contain a comparative small of graphite. It is estimated that about 2,100 tes of i-grahite will arise from the decommissioning of the Latina reactor. The radionuclide composition of the i-graphite is based on predictions similar to the UK's Magnox reactor values.

There are two graphite moderated reactors in Spain, Vandellos 1 a 460 MWe commercial reactor and a material test reactor JEN-1. Isotopic composition of the graphite sleeves from Vandellos 1 and the thermal shielding of JEN – 1 are reported. The C-14 content of the sleeves is significantly higher than the shield material. This trend is true for all other isotopes measured.

In addition to the i-graphite in these two reactors (2,700 tes) more than 1,000 tes resides in 3 silos.

Revisions

Rev.	Date	Short description	Author	Internal Review	Task Leader	WP Leader
01	26/01/2009	First issue draft	F Madrid, J L Leganés E Vico del Cerro (ENRESA Spain)	E Vico (ENRESA , Spain) A Dodaro (ENEA, Italy)	H Eccles (UK NNL)	H Eccles (UK NNL)
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Chapter contents

1 Introduction

This chapter discusses graphite moderated reactors, the decommissioning strategy and waste management of irradiated graphite for Italy and Spain. The cumulative quantity of i-graphite from these two Member States is of the order 5,500 to 6,000 tes, with about 2,000tes originating from the Latina reactor (Italy) and the remainder from the Vandellos reactor (Spain) and significantly smaller quantity from the JEN 1 MTR (Spain). The Spanish section also describes i-graphite, such as fuel sleeves that has arisen from the operation of these reactors. This waste quantity is not insignificant and may require specific attention.

2 Italy

2.1 The Legislative and Regulatory Framework[1]

Authorisations related to nuclear and radioactive installations are granted by the Ministry for Productive Activities (MAP), on the basis of the technical advice of the National Environmental Protection Agency (ANPA).

ANPA (previous APAT) is the agency which carries out technical and scientific activities of national interest related to the protection of the environment, and the defence of water and territory resources. It has operational and administrative autonomy under the directives and the control of the Ministry of Environment.

For all nuclear activities, the ANPA acts as the technical body of MAP. Its responsibilities for the licensing process of nuclear installations include:

- The assessment of the safety analysis carried out by the operating organisation;
- The inspection of equipment and materials during the design, construction and operational phases for the systematic verification of facility operation safety; and
- The enforcement action to remedy any failure to meet both the licensing conditions and any safety operation criteria.

The Ministry of the Environment is the authority responsible for the decisions in the matter of environmental compatibility of nuclear projects, including the decommissioning of nuclear power stations.

2.1.1 Licensing procedures

Construction, operation and decommissioning of nuclear power plants are governed by:

- Act No 1860/62 on the peaceful use of nuclear energy;
- Act 393 of 23 August 1975 and D.Lgs. 230/95;
- Decree No 230/95 on technical and administrative licensing procedures regarding nuclear safety and radiation protection.

2.1.2 Main national laws and regulations in nuclear power

- Law no. 1860 of 31st December 1962, which is the Basic Act on the peaceful uses of nuclear energy. The Law has been amended by the President's Decree no. 1704 of 30 December 1965, and by the President's Decree no. 519 of 10 May 1975. The Law regulates all activities connected with the peaceful uses of nuclear energy. Excluded from the scope of the Law are the nuclear installations for the generation of electricity, which are governed by the procedure lay down in Legislative Decree no. 230/95.
- Legislative Decree no. 230 of 17 March 1995, which replaced the previous DPR no.185 issued in 1964. Its provisions on nuclear safety and radiation protection apply to all practices involving an ionising risk, including the construction, operation, and decommissioning of nuclear plants.
- Legislative Decree no. 241 of 26th May 2000, which has transposed European Union Directive 96/29/Euratom laying down basic safety standards for the radiation protection of workers and the public, amended in 2001.
- Legislative Decree no. 187 of 26 May 2000 which implements council Directive 97/43/Euratom of 30 June 1997, on health protection of individuals against the dangers of ionising radiation in relation to medical exposure.
- 2003 Decree on Radioactive Waste Management - extension of their validity by 2004 and 2005 ordinances on the same subject.

2.2 Policy Objectives of Decommissioning[1]

2.2.1 Energy policy: current issues and developments on nuclear power

Main nuclear policy issues relate to the decommissioning and waste disposal facilities. The ultimate strategic goal, for the former, is unrestricted site release.

On December 1999 the Ministry of Industry outlined strategic choices and plans to manage the problems connected with the closure of all nuclear activities in the country. The ministry statement outlines three main goals:

- Treatment and conditioning, within a 10 year period, of a liquid and solid radwaste currently in on-site storage, mostly issued from the operation of the plants, with a view to subsequent transport to a national waste repository.
- Site selection and construction of a national repository for low and intermediate level wastes, also within 10 years; the same site would be used for temporary storage for high-level long-lived wastes: the final selection of a site for waste disposal facilities has not been made yet.
- Decommissioning of NPP in about 20 years, proceeding directly to the dismantling stage in order to reach the site release with no radiological constraints.

The target is to reach the complete radiological release of the site within 20 years.

The general programme for all plants has been in principle divided into three phases:

- The first phase will be devoted to all activities that are somewhat independent on the chosen strategy and on the availability of the national repository. This phase was scheduled to be finished by mid-2005.
- The second phase will consider all preparatory activities and include a mock up realisation for the final dismantling. It will start after the beginning of the commencement of the national repository; it is estimated that this phase should finish in 2008-2009.
- The third phase will be devoted to the dismantling of nuclear islands, to the transport off all wastes to the repository and to site restoration. It is estimated that by the end of 2020 all sites should be released.

Of course, the objectives of this programme can be reached only if the construction of a national repository is achieved within the due timeframe.

For all plants, during 2003, a special effort was devoted to the application for the dismantling and other licensing issues, such as the Environmental Impact Assessment (EIA). In addition, many detailed projects were presented to the Safety National Authority, in particular: the evaluation study on graphite removal from Latina reactor.

In the area of nuclear safety and environmental protection, bilateral agreements have been signed with the NRC (USA), NII (UK), CSN (Spain), N.N.S.A. (China), and D.S.I.N. (France).

The main activities already performed on each plant can be summarised as follows:

Garigliano (160MW BWR, operated from 1963 to 1978): to initiate the decommissioning licensing procedure the global decommissioning plan with a new strategy was presented in August 2001. The plant was close to reaching a safe storage condition when the change of strategy occurred. The reactor is defuelled and no fuel is now on site. The radiological characterisation has been completed. Almost all operating wastes have been treated; no radwaste tanks remain to be dismantled and decontaminated. Preliminary work for decommissioning (site logistic and monitoring facility) has been initiated.

Latina (160 MW GCR, operated since 1962 up to 1987): The global decommissioning plan for initiating the licensing procedure was presented in February 2002. The plant has been totally defuelled, the primary circuit has been filled with dry air and blowers, portions of the primary circuit outside the reactor building have been dismantled and the removal of asbestos from the turbine building and from part of the reactor building has been completed.

Trino (260 MW PWR, operated from 1965 to 1987): to initiate the decommissioning licensing procedure the global decommissioning plan with a new strategy was presented in December 2001. The reactor has been defuelled and part of the fuel is now stored in the pool of the plant. The removal of asbestos from the turbine building and the demolition of some conventional buildings have been performed. The dismantling of components from the turbine building and the removal of asbestos from the controlled area has been initiated and the dismantling project plan of the nuclear island is complete and awaiting approval (as part of the global decommissioning plan) by the safety authority.

Caorso (850 MW BWR, operated from 1981 to 1986): the reactor has been defuelled and the fuel is now stored in the pool of the plant. On 4 August 2000 the Ministry of Industry issued a decree authorising specific decommissioning activities (dry storage of irradiated fuel,

dismantling of the turbine and off-Gas, dismantling of the RHR towers, decontamination of the main circuit, treatment of previously produced waste). All these activities are ongoing.
For other activities the global decommissioning plan was presented on 3 August 2001.
Recently SOGIN carried out a deep re-examination of the nuclear power plants decommissioning cost estimates, also with the qualified assistance of international advisors. A total amount of €3500M is estimated as the cost for the total decommissioning of the four nuclear power plants.

2.3 Compliance with International Guidelines and Regulations

Italy as with other Member States recognised and complies with nuclear energy international guidelines.

2.4 Decommissioning of Nuclear Reactors – Current Situation

Italy has constructed four commercial reactors, none of which are now in operation (Table 1) and ten research reactors half of which are still in operation. In addition several research reactor/facilities have also been constructed (Table 2)[1]. Of the commercial reactors only one (Latina) was graphite moderated (Figure 1)

Table 1 Status and trends of nuclear power

Station	Type	Power (Mwe)	Operator	Status	Construction Date	Commercial Date	Shutdown Date
CAORSO	BWR (AMN/GETS)	860	Sogin[1]	Shut Down	01-Jan-70	01-Dec-81	01-jul-90
ENRICO FERMI (TRINO)	PWR (WEST)	260	Sogin[1]	Shut Down	01-jul-61	01-Jan-65	01-jul-90
GARIGLIANO	BWR (GE)	150	Sogin[1]	Shut Down	01-nov-59	01-jun-64	01-mar-82
LATINA	GCR (TNPG)	153	Sogin[1]	Shut Down	01-nov-58	01-Jan-64	01-Dec-87

1- Sogin since November 1st, 1999; before this date the operator was ENEL.

Source: IAEA Power Reactor Information System as of 31 December 2001; Sogin.

It is unclear from the publications if any of these reactors contain graphite and how much. The quantity of graphite in the Latina reactor is of the order 2100 tons (see section 5). There is no evidence in publications to indicate that consideration has been given to retrieving this graphite or for its treatment and subsequent disposal.

Table 2 Research reactor facilities

Site	Power	Operator	Status
Bologna	100 We	ENEA/RB3	In operation
Palermo	20 We	University of Palermo	Not in operation
Roma (Casaccia)	1 MW	ENEA/TRIGA	In operation
Roma (Casaccia)	5 kW	ENEA/TAPIRO	In operation
Pavia	250 kW	University of Pavia	In operation
Ispra		CCR	Not in operation
Ispra		CCR	Not in operation
Milano		CESNEF L54M	Not in operation
Padova		University of Padova	In operation
Pisa		CISAM	Not in operation

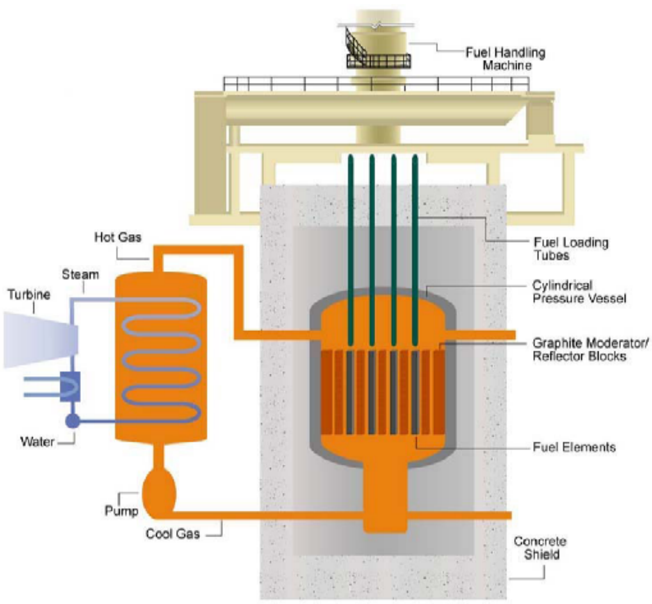


Figure 1: Cutaway schematic of Magnox-type reactor

Currently there is only one reactor in Italy containing some kind of nuclear graphite materials: Nuclear Power Plant of Latina.(Figure 2)
Latina NPP was the first nuclear power plant constructed in Italy. It was ordered in October-1958 by SIMEA and was operative in 1963. It is a Uranium Natural Graphite Gas reactor (UNGG), graphite moderated and cooled by carbon dioxide.

Basic features:

- Gas Cooled Reactor. Model: Magnox
- Net output: 153 MWe
- Operation: 1964-1987 (ENEL and SOGIN)
- Shutdown: 1987
- Energy produced: 25 TWh
- Current situation: permanently shutdown.

  Up to year 2006 removal of primary circuit ducts has been almost completed and works in the Turbine Building is underway. Design of the Magnox sludges conditioning is almost completed. As well as the design for boilers removal and for graphite treatment.

  The global decommissioning plan was presented in Feb-2002.
- Safe enclosure: 20 years.

There is no final disposal facility at the moment for any type of waste but Nucleco operates a temporary storage facility for industrial radwaste at a site near Rome.

All magnox fuel from Latina was sent in the 1990´s to BNFL Sellafield plant in the UK and has all been reprocessed and now activities are been carried out relevant to treatment of its reminder wastes.

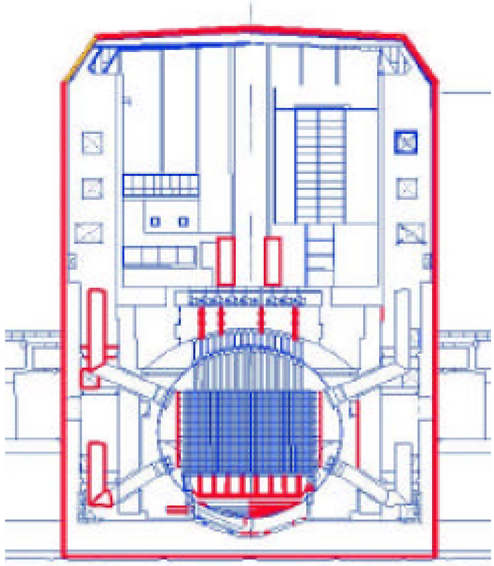


Figure 2: Scheme of the Latina NPP reactor

Wastes returned as vitrified HLW from reprocessing will be temporarily stored in an interim storage facility along with reminder spent fuel.

The baseline decommissioning duration for Latina NPP will be 15 years. Starting at year 2005 complete decommissioning will be done by end of 2019.

Graphite material:

- Amount of graphite in reactor: 2065 tonnes.
- Total number of assemblies: 23210 (1 rod per assembly)

There is no information in the public domain to suggest that the retrieval of graphite from the Latina reactor has been under taken and/or considered. Consequently treatment options for the removal of C-14 (see C-14 data in next section) may not have been assessed and equally true for its subsequent disposal.

2.5 Graphite characteristics

2.5.1 Latina NPP graphite

The graphite was manufactured and supplied by ANSALDO Company. Actually the Latina Nuclear Power Plant keeps closed and is been decommissioned, but the graphite removed from the plant has not been characterized yet.

Due to the fact that the Latina reactor is similar to the Bradwell and Dungness Stations in the UK and many others with the same reactor design, it would be useful, to compare the graphite characteristics for similar design reactors Table 3).

Table 3 Predicted radionulides activities

Reference Magnox Reactor after 10 years decay (assumed graphite mass of 2233 tonnes) (MBq)			
H-3	1,20E+08	Mo-93	8,50E+02
Be-10	7,10E+04	Nb-93m	5,50E+02
C-14	8,50E+07	Nb-94	1,00E-01
Cl-36	9,50E+05	Tc-99	1,70E+02
Ca-41	7,30E+05	Ag-108m	2,30E+04
Mn-54	2,70E+02	Cd-113m	1,00E+04
Fe-55	1,50E+07	Sn-121m	4,50E+04
Ni-59	9,30E+04	Ba-133	5,60E+05
Co-60	2,70E+07	Eu-152	2,20E+05
Ni-63	1,30E+07	Eu-154	5,20E+06
Zn-65	2,10E+02	Eu-155	1,60E+06

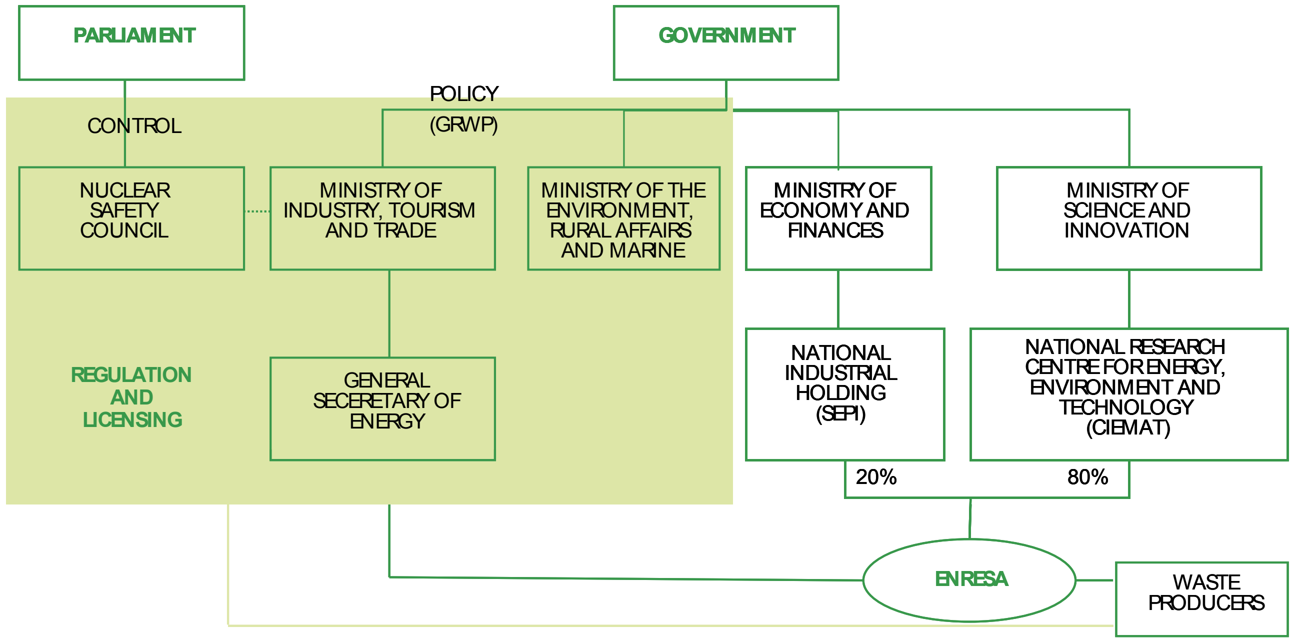
3 Spain

3.1 The Legislative and Regulatory Framework

ENRESA, the Spanish radioactive waste management agency, is responsible for the management of the operations arising as result of the decommissioning of nuclear facilities. According to the Second Spanish National Report on the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (October 2005)[2], Spain possesses the infrastructure required for the management of spent fuel and radioactive waste from the administrative, technical and economic-financial points of view. Regarding the administrative issue, there is an organisation, based on a relatively far-reaching and highly developed framework in keeping with the evolution of the international regulatory requirements that contemplates and includes the main responsibilities of the different parties involved in the process. The Figure 3 shows the institutional framework in Spain for the management of spent fuel and radioactive waste.

The Government establishes the general lines of the national policy on the management of radioactive waste and spent fuel, including the D&D activities of nuclear and radioactive installations through the GRWP. This Plan is an official document drawn up by ENRESA and submitted to the MITYC in compliance with the requirements of the standards in force. It is required to be submitted to the Government by the MITYC and, following its approval, communicated to the Parliament.

The GRWP must contain the actions and technical solutions foreseen for the management of radioactive waste and spent fuel as well as the dismantling and decommissioning of nuclear, and where appropriate radioactive, facilities throughout the timeframe of the Plan, and the economic and financial measures foreseen for the performance of those actions. The GRWP currently in force is the 6th, which was approved in June 2006.



GRWP: General Radioactive Waste Plan

Figure 3: Institutional framework in Spain for the management of spent fuel and radioactive waste

The organisation and responsibilities for dismantling of nuclear and radioactive installations are legally defined. The responsibility for decommissioning is initially to the licensee himself who, prior to the granting of the corresponding authorisation, undertakes the so-called pre-dismantling activities.

The licensee of the operating permit is responsible for conditioning the operating wastes generated during the operation of the plant, in accordance with the acceptance criteria of the disposal facility to which they are to be transferred. Secondly, the licensee of the facility must have unloaded the fuel from the reactor and from irradiated fuel storage pools or, otherwise, for having available a spent fuel management plan approved by the Ministry of Industry, Tourism and Commerce (MITYC), following a report from the Nuclear Safety Council (Spanish Radiation Protection and Nuclear Safety Authority, CSN).

The contract between ENRESA and the NPP licensees, approved by the MITYC, established in greater detail the responsibilities of the licensee and the scope of the work to be performed by him in order to plan the dismantling to be performed by ENRESA.

ENRESA is in charge of submitting the NPP D&D Plan to the MITYC. The acceptance by ENRESA of the responsibility for the dismantling of a NPP requires a transfer of trusteeship, authorised by MITYC simultaneously with the authorisation for dismantling. Following such transfer, ENRESA becomes the licensee of the plant during the performance of the dismantling activities.

The procedure and mechanisms adhered to in carrying out this transfer of trusteeship of the facility are established contractually between the licensee and ENRESA.

3.2 Policy Objectives of Decommissioning

According to the GRWP[3], the decommissioning strategy to be followed by ENRESA is:

- Vandellós I NPP

  This French technology graphite-gas reactor is the only one of its type built in Spain. The plant entered commercial service in 1972 and operated until October 19th 1989 when a fire broke out in its main turbine-alternator group. As a result of this accident, the power operation of the plant was temporarily suspended by a Ministerial Order issued on November 27th 1989. Subsequently, it was agreed to shut down the plant definitively by the Ministerial Order of July 31st 1990. The Resolution of the Directorate General for Energy of November 27th 1992 accepted the alternative proposed by ENRESA for the D&D Plan, which established a partial dismantling (reaching IAEA Level 2) as the first objective. This Plan was submitted in May 1994, the favourable report from the CSN and the Environmental Impact Statement being obtained in 1997 and authorisation for project performance in January 1998.

  On completion of Level 2, this facility has become a passive installation, under ENRESA responsibility, that will remain in this mode for the next 25 years (the dormancy period started in January 2005), until such time as complete dismantling is addressed.

- The other 7 NPPs (8 reactors in operation and 1 PWR in pre-dismantling period)

The strategy is to undertake the complete dismantling of the plant after three years of its definitive shut-down, once the spent fuel has been removed from the pool and the operational LILW transported to El Cabril LILW near-surface disposal facility.

The José Cabrera NPP was shut-down the 30th April 2006, following a decision of the authorities not to renew its operating permit. This PWR (160 MW) was the Spain's first NPP, started up in 1968. The D&D activities are foreseen to start in 2009, the duration of which is estimated at 6 years.

– CIEMAT facilities

The decision to dismantle certain obsolete CIEMAT installations for which no further use was foreseen and take advantage of the space released for performing other activities led to the Integrated Plan for Improvement of CIEMAT Facilities (PIMIC), in the preparation of which ENRESA collaborated as a specialist company with experience in this field.

The Plan, which will extend to the year 2009, is subject to the CSN control and supervision, in accordance with the standards in force, and CIEMAT continues to be the licensee responsible for the installations. ENRESA, since 2006, is undertaking the dismantling operations, based on the contractual agreements with CIEMAT.

The main installations affected by the PIMIC are the JEN-1 experimental reactor and the irradiated fuel reprocessing pilot plant, currently completely dismantled and in the decontamination phase.

3.3 Compliance with International Guidelines and Regulations

Spain complies with international regulations and standards such as International Conventions on Nuclear safety and on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Also, Spain is involved in the IAEA programme for the preparation of IAEA safety standards.

3.4 Decommissioning of Nuclear Reactors – Current Situation

Currently there are 2 reactors in Spain containing some kind of nuclear Graphite materials:

– Nuclear Power Plant of Vandellós 1, located in Hospitalet de l'Infant, Tarragona.
– Material Testing Reactor JEN-1, located in Madrid.

3.4.1 - Vandellós 1 NPP

NPP of Vandellós 1 is a Uranium Natural Graphite Gas reactor (UNGG), it was graphite moderated, cooled by carbon dioxide, and fuelled with natural uranium metal (see Figure 4).

Basic features:

- 460 MWe. 56000 GWh (from 05/06/72 to 10/19/89)
- Licensee:
  - 1971 - 1998 HIFRENSA
  - 1998 – ENRESA
- Operation (1971-1989)
- Shutdown Oct 1989
- Decommissioning Level 2 (1998-2003)
- Safe Enclosure for 25 years

Graphite material:

- 186000 graphite sleeves were used (1000 tons)
- Moderator graphite pile is a 2680 tons cylinder.

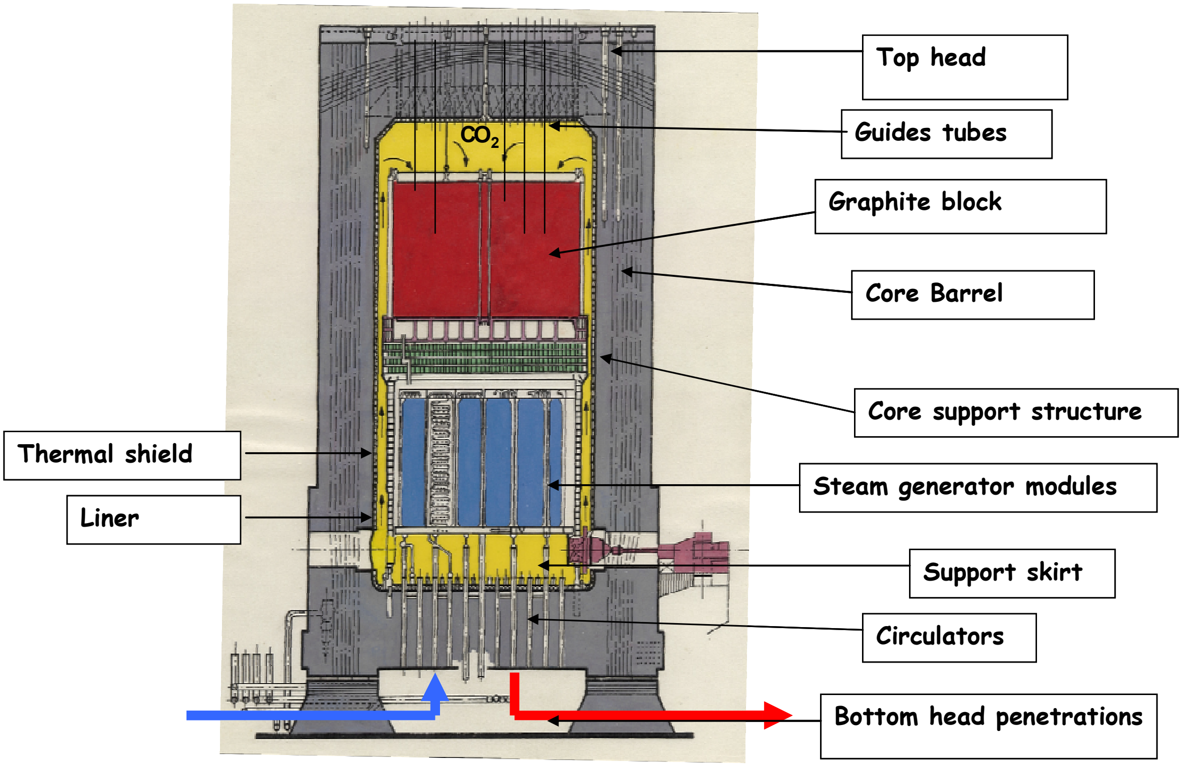


Figure 4: Vandellós 1 Vessel

3.4.2 JEN 1 Material Testing Reactor

JEN 1 Material Testing Reactor is placed inside the Ciemat research centre in Madrid (see Figure 5).

- 19.8% U-235 enrichment, 10,5% burn up and 3 MWe.
- Operation (1959-1984)
- Shutdown Oct 1984
- Currently under dismantling (2007-2009).
- Graphite thermal shield (8 ton).

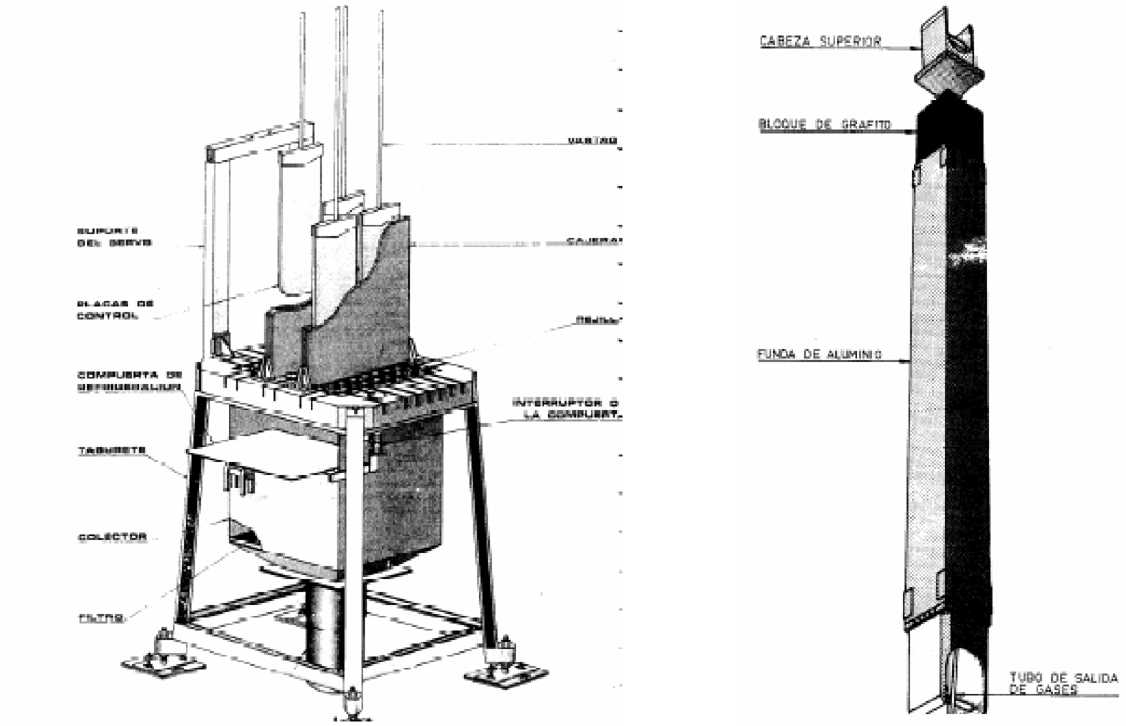


Figure 5: Reactor nucleus (left); Element of Graphite Thermal Shield (right)

3.5 Graphite Characteristics

3.5.1 Vandellós 1 Graphite

This graphite was manufactured by Pechiney SA in France. The manufactured method was not well defined; Pechiney graphite was prepared by baking a paste made of oil coke and pitch, graphitized by electrical heating.

Virgin graphite characteristics:

- Porosity: about 20%
- Density: 1,8 g/cm^3.
- Volatile content: 0.18% mass, mainly H_2O, to a lesser extent CO_2, CO, hydrocarbons (methane, ethane, ethylene, propane, propylene, etc.).
- Water adsorption: 2.2% mass for graphite powder.
- H_2 adsorption: it has not been observed at ambient temperatures. A light weight loss of graphite observed at high temperatures (300º) due to H_2O formation.

– Chemical impurities: N, Li, Mg, Al, Ca, Fe, Ti, Sn, Sr, Cd, Sb, Ag, Cl, Mn. Co, Ni, Be, Mo, Eu.

Irradiate graphite characteristics are included in Table 4 and Table 5[4].

Table 4: Isotopic composition and mean activity of sleeves (01/01/2000)

Isotope	Content (%)	Mean Activity (Bq/g)
H-3	38.44	8.93E+04
C-14	5.81	1.35E+04
Fe-55	11.62	2.70E+04
Co-60	17.22	4.00E+04
Ni-59	0.27	6.25E+02
Ni-63	25.31	5.88E+04
Cs-137	0.16	3.79E+02
Eu-154	0.19	4.35E+02
Pu-241	0.29	6.82E+02
Total	99.31	-

Notice that sleeves were placed inside the reactor pool while removing the fuel; it means that non graphite activation products like Cs-137, TRU, etc., have been absorbed from the pool.

Gasification of virgin graphite, in order to simulate operating conditions, produce a weight loss about 14%-47% with powder graphite, and a 10% in graphite pieces.

Currently the graphite pile is located inside reactor building in a safe enclosure (25 years). The sleeves were extracted as described below and temporally stored at Vandellós 1 in cubic containers.

Table 5: Isotopic composition and mean activity of Graphite pile (01/01/2000)

Isotope	Content (%)	Mean Activity (Bq/g)
H-3	74.97	2.75E+05
C-14	15.32	5.62E+04
Fe-55	2.50	9.15E+03
Co-60	3.65	1.34E+04
Ni-63	2.39	8.77E+03
Pu-241	0.19	6.89E+02
Total	99.02	-

There are 220 cubic containers of m^3 with the crushed graphite obtaining in the separation process carried out in Vandellós 1. The total mass of waste is 1.1E+3tons.

There are 98 cylindrical containers with the wires of the sleeves obtaining in the separation process carried out in Vandellós 1. The total mass of waste is 22 tons.

3.5.2 JEN 1 Graphite

This graphite was manufactured in USA. We do not know the process carried out; there are some differences between Vandellós 1 Graphite described below.

Virgin graphite characteristics:

- Porosity: about 26%
- Density: 1.73 g/cm^3.
- Volatile content: 0.2% mass, mainly H_2O, to a lesser extent CO_2, CO, hydrocarbons (methane, ethane, ethylene, propane, propylene, etc.).
- Water adsorption: from 3.3% mass for graphite powder to 0.5% graphite pieces, intermediate values for grains.
- H_2 adsorption: it has not been observed at ambient temperatures. A light weight loss of graphite observed at high temperatures (300º) due to H_2O formation.

- Chemical impurities: N, Li, Mg, Al, Ca, Fe, Ti, Sn, Sr, Cd, Sb, Ag, Cl, Mn. Co, Ni, Be, Mo, Eu, In.

Irradiate graphite characteristics are included in Table 6[4].

Table 6: Isotopic composition and mean activity of thermal shielding (01/01/2007)

Isotope	Content (%)	Mean Activity (Bq/g)
C-14	1,930	6.30E+00
Fe-55	0,267	8.72E-01
H-3	93,400	3.05E+02
Ni-63	0,512	1.67E+00
Sr-90	0,744	2.43E+00
Co-60	0,008	2.68E-02
Eu-152	3,050	9.96E+00
Eu-154	0,036	1.19E-01
Total	99.95	-

Currently there are 6 cubic containers with graphite from thermal shielding; the total amount of waste is 8 tons.

3.6 I-Graphite / Carbonaceous Wastes Retrieval Options

3.6.1 Vandellós 1 NPP silos retrieval of the radioactive wastes[5]

The technical solution selected by the former licensee for the interim storage of the radioactive waste, generated during the process, consisted of the direct dropping of the wastes inside concrete silos (three silos).

The radioactive wastes contained in the silos (see Table 7) were the ones produced typically during the operations of Vandellós 1: Graphite sleeves, graphite cylinders (rondins), neutron flux absorbents and seat support wires. Additionally two spent fuel elements, with its graphite sleeves, were accidentally dropped inside silo 1.

The main activities of the Vandellós 1 silos retrieval process were:

- Visual selection (CCTV) and retrieval
- Transfer from silos to the conditioning building
- Conditioning (section 3.7.1.)
- Transfer to interim storage (section 3.7.1)

During normal operation the equipment required for retrieval and transfer of the wastes were: waste retrieval manipulator, skip, transfer cask, mobile cabin, handling frame crane.

The mobile cabin was designed as a steel structure to house during the retrieval of the wastes, the manipulator and the transfer cask and provided them with protection against the atmospheric elements action.

Table 7: Waste Inventory

Type	Silo 1		Silo 2		Silo 3	
	Units	Weight(Kg)	Units	Weight (Kg)	Units	Weight(Kg)
Graphite Sleeves	36.123	1.95+5	106.666	5.76+5	43.628	2.35+5
Graphite Cylinders	4.834	4.83+4	54	5.40+2	-	-
Absorbents	210	2.68+3	282	3.60+3	-	-
Seat support wires	36.614	3.07+2	108.165	9.59+2	43.628	3.87+2
Spent fuel elements	2	3.41+0	-	-	-	-

The facility was equipped with several auxiliary system required to support the operation, but due to the characteristics of the wastes (mainly graphite), only the ventilation system and the fire protection system (related to the fire extinction using Argon as extinguisher) are outstanding.

3.7 Treatment / Conditioning Techniques / Process used on Retrieved I-Graphite

3.7.1 Vandellós 1 NPP silos conditioning of the radioactive wastes[5]

The functions of conditioning systems were:

- Processing of the graphite wastes and segregation of the support seal wires.
- Packaging of the different stream of wastes
- Transfer of the waste packages outside of the conditioning building

The process and transfer areas are housed inside a reinforced concrete building which main operating areas are: sorting hot cell, graphite shredding machine and containers loading platform.

The working processes were as follows: the skip from the transfer cask was lowered in a basket elevator and, once lowered down to the graphite conveyor belt level, discharged the waste on it.

The conveyor belt fed the graphite shredding machine and was provided with a gamma detector that stopped the belt in case of absorbent detection allowing the retrieval and buffer storage of the absorbents carried out by means of the wall manipulators.

Total amount of graphite, steel wires and miscellaneous wastes was packaged in 220, CME type boxes "cubic containers" with 1054 ton of graphite, 95 CBE type casks "cylindrical containers" with 19,7 ton of steel cylinders absorbents and wires, and 11 CE2 type container "El Cabril containers" with miscellaneous wastes, plus 875 drums with technological wastes generated during the retrieval process, that were sent to El Cabril LLW repository.

The average surface dose rate in CME was: 3,66 mSv/h contact and 0,68 mSv/h at 1 meter. For CBE casks, average surface dose rate was 0,06 mSv/h and 7,9 mSv/h at 1 meter.

Due to high dose rate of CME boxes an overpack was built to transport from conditioning building to interim storage building. This transport was made with a forklift truck having a lead glass for shielding purposes of drivers cab.

4 References

[1] www. Nuclear Energy Agency Country Profiles - Italy.htm

[2] Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. Second and Third Spanish National Report (www.iaea.org and www.mityc.es)

[3] Sixth General Radioactive Waste Plan. June 2006. (www.mityc.es) and ENRESA web page (www.enresa.es)

[4] Antonio Morales, Gabriel Piña, Angel Gonzalez de la Huebra, Marina Rodríguez, Jose Antonio Suárez, “Characterisation of Radioactive Graphite from NPP Dismantling” RadWap 2002,. September 2002.

[5] Conditions and Approval by the Industry Minister of starting “Extraction, retrieval and conditioning of Silos wastes “. May 1995