CONTENTS

A. Destructive Fuel assembly Characterization

Destructive FA characterization methods are fundamental as well as no-destructive ones in nuclear field even if with different time-steps and scopes. In fact DA are used to characterize FA once it had been discharged from the last fuel power cycle from the reactor and not inside refuelling operation as NDA because damage the fuel pins definitively. They are applied inside hot cells due to the high activity of the spent fuel and the hazardous related to these operations. There are not present at all NPP whereas it can be easily found in a nuclear research centres such as for instance a CEA Cadarache[1]. For the last affirmation derives that DA are not always used in NPP whereas non-destructive in-core and out-core ones yes.

The major targets of destructive techniques are [1]:

- The evaluation the accuracy of BU nondestructive correlation and the goodness of depletion codes ,as ORIGEN-S, due to the their better precision than non-destructive assays
- Determination of spent fuel features which are not possible to evaluate with no destructive assays
- Better understanding of the process of formation and destruction of fission products
- The study the behaviour of new fuel typology.

Commonly they are included inside the post irradiation examinations or briefly PIE [32] which consist of:

- Nondestructive methods:
  - Visual examination
  - Gamma scanning
  - Profilometry
  - Oxide thickness measurements
- Destructive methods:
  - Puncturing and fission gas analysis
  - Metallography/ceramography on cut fuel pin samples
  - Hydrogen hot gas extraction
  - Electron probe micro-analysis (EPMA)
  - Secondary ion mass spectroscopy (SIMS)
  - Scanning and transmission electron microscopy (SEM and TEM)

- Laser ablation inductively coupled plasma mass spectroscopy (LA-ICPMS)
- X-ray absorption spectroscopy, and mechanical testing facilities

A very useful data of PIE methods is the INFCIST IAEA database where you can find a lot of information of both NDA or DA applied all over the word.

In this final thesis project will be resumed two techniques which allow determining BU and fissile amount material and a description of a typical hot cell.

A.1 Radiochemical analysis

A radiochemical analyses allows [2] to the determination of isotopic compositions and concentrations of actinides and fission products in different types of industrial (UO_2, MOX) and experimental nuclear fuels (UAlx, U_3Si_2, UMo, …). From the amount of specified fission products an accurate Bu can be obtained. In SCK-CEN program for example twenty-one actinides and fission products are analysed with a possibly future increase to up to fifty isotopes.

Basically the method is composed by three steps:

1. Dissolution of the fuel sample
2. a, b and g-spectrometry
3. Determination of BU and fission product utilizing a correlation made by a depletion code

As said, a first requirement for a successful radiochemical analysis is a quantitative dissolution of the spent fuel. Some fuel samples, typically ranging from 1 to 25 g, are dissolved in a lead-shielded facility. For industrial UO_2 and MOX fuels with homogeneous fuel pellets that are packed in a cylindrical Zircaloy cladding, a two steps procedure using nitric acid as main solvent is applied resulting in a dissolution of the fuel but leaving the cladding intact. Dispersion fuels, where the fuel particles (UAlx, U_3Si_2, UMo) are mixed with Al-powder and confined as a thin wafer between Aluminium plates, require the dissolution of the fuel together with the Al matrix and cladding. After the acid dissolution steps a small amount of residue is possibly left that contains primarily metallic fission products such as Mo, Tc, Ru, Rh, Ag and Sb-isotopes.

The actinides and fission products in the fuel solutions are analysed using Thermal Ionization Mass spectrometry (TIMS and Inductively Coupled Plasma Mass Spectrometry-Dynamic Reaction Cell and ICPMS-DRC [3]) and radio analytical techniques (a, b- and g-spectrometry). These are all techniques

with analyses methodologies under the scope of QA accreditation according to the international ISO/IEC 17025 standard.

Prominent α and β-emitters and most of the metallic isotopes can be directly measured on a diluted spent fuel solution. However the majority of the isotopes need to be separated before they can be properly analysed. As shown in destructive techniques all source of radiation (a, band g) are useful where in the nondestructive one no: in this way you obtain more information reaching a better accuracy [1,2,4].

In radiochemistry burn-up is expressed as % FIMA, i.e. the number of Fissions that have occurred per Initial 100 heavy Metal Atoms (U and/or Pu). The number of fissions that occurred during irradiation and so the BU can be derived from the concentration of selected key fission products in the spent fuel sample under investigation comparing them to a correlation created by a depletion code as ORIGEN-S. The principal requirements for a useful element are [4]:

- No migration in irradiating in fuel matrix
- Low destruction cross-section and low formation cross-section from mass chains
- Well established fission yield that preferably is constant for all fissile elements and it is independent of the neutron energy
- Goof emission characteristics for radiation spectrometry
- Sufficient amount in the fuel sample

At SCK-CEN, the stable Nd-isotopes ^{143}Nd, ^{144}Nd, ^{145}Nd, ^{146}Nd, ^{148}Nd (ASTM method E 321-69 [5]), ^{150}Nd [5]and the g-emitters ^{137}Cs and ^{144}Ce are selected as fission product monitors. Both groups of fission products are analysed with different analysis techniques, i.e. the Nd-isotopes by isotopic dilution TIMS after a complex separation procedure and the g-emitters by gamma-spectrometry directly on a diluted spent fuel solution. The initial number of heavy metal atoms can then be calculated by summing the number of fissions and the analysed number of actinides after irradiation, usually predominantly U and Pu isotopes. The complete procedure is illustrated in the following figure.

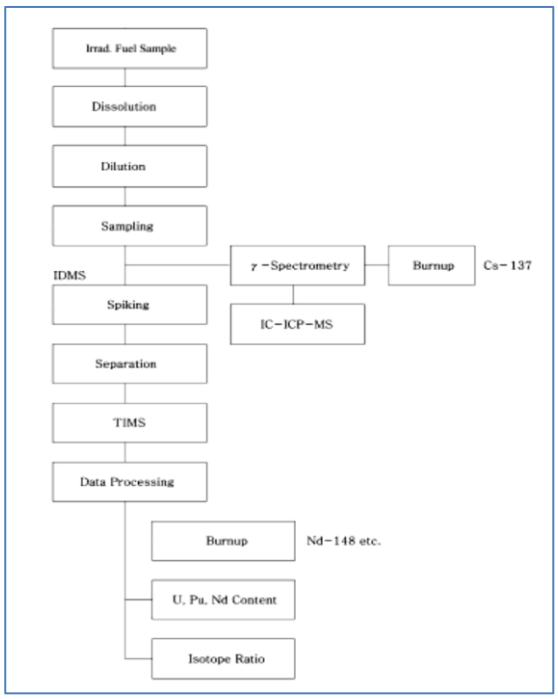


Figure A-1 Radiochemistry characterization process (Ref.6)

To conclude, the accuracy of this technique is reported following [2]:

- Determination of U, Pu isotopic: <±1.0%
- Fission product isotopes: ±2%
- Elemental analysis: ±2-5%

A.1.1 Released fission gas analysis

As a part of destructive examination [7], fuel elements are punctured for the measurement of the amount of released fission gases. Volume of the released fission gas and the void volume in the fuel pin are measured to arrive at the pressure inside the fuel pin. A gas chromatograph and a mass spectrometer are used to analyse the respective chemical and isotopic composition of the collected gases. The FP gas release rate (mostly composed by Xe, Kr, I) increased with increased BU and a correlation can be made by a use of depletion code. Further correlations between BU and fission product, i.e. ^{238}Pu, can be made and so the amount of some elements can be obtained indirectly.

Fig. A-2 show one of this installation inside a hot cell when you can denote the chamber and the fuel pin.

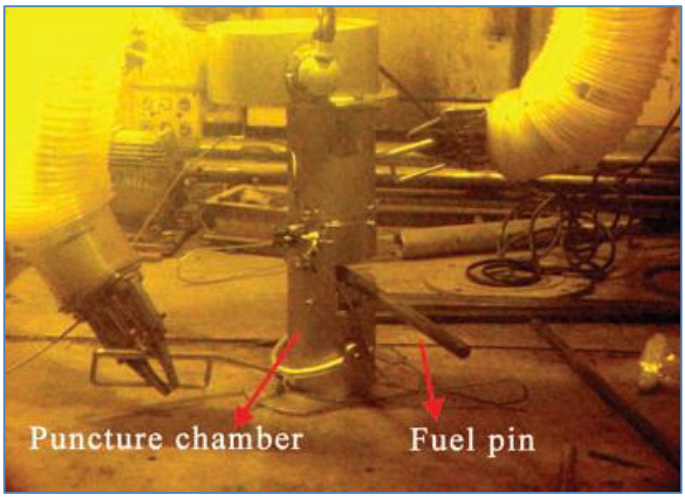


Figure A-2: Hot Cell (Ref. 8)

The accuracy of this technique is quite low compared to the previous one: in fact the determined value of BU is within the ±10% of the core-follow one and so additional dependent correlations are also bad [3].

A.1.2 Hot cell

Shielded nuclear radiation containment chambers are commonly referred to as hot cells when the word "hot" refers to radioactivity. Hot cells are used in both the nuclear-energy and the nuclear-medicines industries. They are a very important step also for the MOX fuel because the cutting of the used fuel, the dissolving of the fuel and the first extraction cycle of a nuclear reprocessing PUREX process would need to be done in a hot cell.

They are required to protect individuals from radioactive isotopes by providing a safe containment box in which they can control and manipulate the equipment required in remote way by robots. Moreover normally an air-recirculating and purifying system and a video control system is present.

They are constituted mostly by concrete and lead [8]. They are of several dimensions which depend on the use of the device.

A.2 References

[1] IAEA-TECDOC-CD-1635, *Post-Irradiation Examination and In-Pile Measurement Techniques for Water Reactor Fuels*, IAEA,2009

[2] Bruce A. Hilton, Ph.D., *Postirradiation Examination Analysis and Capabilities, Idaho National Laboratory* , ATR National Scientific User Facility Users Week 1-5 June 2009

[3] IAEA-TECDOC-1343, *Spent fuel performance assessment and research Final report of a Cordinated Research Project on Spent Fuel Performance Assessment and Research (SPAR)* 1997–2001, IAEA, March 2003

[4] John G. Williams, *Reactor Dosimetry: Radiation Metrology and Assessment,* Edizione 1398, STP,2001

[5] American Society for Testing and Materials Standard Method E 321-69, *Atom Percent Fission in Uranium and Plutonium Fuel(Neodymium-148 Method)*, 1045 (1969).

[6] Mireille Gysemans, *Radiochemical analysis of nuclear fuel burn-up and spent fuel key nuclides*, SKF-CEN, Scientific report 2005

[7] S. Anantharaman, E. Ramadasan, J.L. Singh, J.S. Dubey and Prerna Mishra*, Post Irradiation Examination of Thermal Reactor*, Post Irradiation Examination Division, Barc Newsletter Agust 2011-No.315

[8] Maurizio Cumo*, Impianti nucleari*, Università la Sapienza, 2008

B. Safeguard on Nuclear Material

B.1 Introduction

The Euratom Safeguards Office performs safeguards on all civil nuclear material in the European Union. Its legal basis and scope is defined in the Treaty establishing the European Atomic Energy Community signed in 1957. Co-operation with the IAEA assures effective and efficient safeguards in the European Union. Due to this reason all nuclear materials inside a NPP, LEU and spent fuel, have to be accounted and reported to EURATOM following specific laws. Moreover, to each State is required to establish and maintain a State system of accounting for and control of nuclear material (SSAC) subject to safeguards under the agreement.

The task of the Euratom Safeguards Office is to ensure that within the European Union nuclear material is not diverted from its intended use and that safeguarding obligations assumed by the Community under an agreement with a third state or an international organization are complied with.

Safeguards is the set of measures performed by the controlling authority to verify that nuclear material and equipment are not diverted from their intended (peaceful) uses, e.g. are not used to produce nuclear weapons. The aim is to allow the use of nuclear energy whilst ensuring that civil nuclear material remains in peaceful nuclear programs.

The controls performed are done by the combination of:

- Nondestructive techniques
- Destructive techniques
- Visual techniques

A very useful database for nuclear material account is the Nuclear Fuel Cycle Information System (NFCIS) which covers civilian nuclear fuel cycle facilities around the world. It contains information on operational and non-operational, planned, and cancelled facilities. All stages of nuclear fuel cycle activities are covered, starting from uranium ore production to spent fuel storage facilities.

B.2 Collaboration IAEA-Euratom Safeguards Office

The IAEA and the Euratom Safeguards Office co-operate in the thirteen Non-Nuclear Weapon States following the arrangements laid down under the New Partnership Approach (NPA) as agreed in 1992 between the European Commission and the IAEA. Co-operation in the UK and France is performed under the so-called Joint-Team arrangements. Under the NPA and the Joint Team arrangements, inspection activities of the IAEA and the Euratom Safeguards Office are executed jointly. Inspection activities carried out by the Euratom Safeguards Office are taken into account by the IAEA in drawing its own conclusions and vice-versa.

B.3 Basic Technical Characteristics

Utilities have to provide the Commission with the (BTC) of their installation following a detailed questionnaire. The information required includes a description of the nuclear material used and how it is handled, as well as the system of nuclear material accountancy control. Any changes to the BTC must be communicated to the Commission. The operators must establish and maintain a system of nuclear material accounts when they start handling such material. Features of this material accounting system are that all parts of the installation in which nuclear material may be found, have to be allocated to a series of Material Balance Areas (MBA). The operators also have to notify regularly their program of activities, including the program for the taking of physical inventories. In addition, operators have to notify in advance certain transfers, imports, and exports of nuclear material.

An area in a nuclear installation defined in such a way that the nuclear material quantity in each transfer into or out of this area can be determined, and the physical inventory of nuclear material in this area can be determined when necessary, in order that the material balance for safeguards purposes can be established.

B.4 Inspection inside LWR Nuclear Power Plant

LWRs are inspected during this outage period when all the fuel is accessible for verification. In situ verification techniques, involving expeditious testing for attributes of fuel, are employed, on fuel discharged from the core. When such verifications do not lead to conclusive results, other more intrusive techniques, which may involve handling of the spent fuel, may be used. In addition to these

direct verification techniques, cameras are installed and seals are used at some LWRs in order to facilitate oversight of the fuel and fuel handling activities. Intermittent inspections, to the extent that resources permit, are carried out typically at quarterly intervals between outages.

A number of LWRs use MOX fuel elements to consume the plutonium produced under their reprocessing contracts. Inspections dedicated to verify these fuel elements in the Non-Nuclear Weapon States have been assigned a higher priority over the past two years with the result that safeguards implementation in these installations was significantly improved.

For the verification, the control of the nuclear material and the consistency of MBA with the reports on inventory, the Euratom Safeguards Office can draw on a range of technological resources. Non-destructive assay (NDA) instruments are available for direct measurements of the quantity of plutonium and uranium in different forms. Many such instruments are permanently installed at nuclear installations. For calibration purposes, special reference materials prepared for EURATOM are used. Finally, containment and surveillance measures ensure that the continuity of knowledge relating to specific nuclear material or a place of work is maintained. To this end, optical surveillance units are installed which, for example, take photographs at determined intervals. The images are extracted periodically, and the records reviewed. Extensive use is made of seals, particularly for material that may stay unchanged between verifications. The Euratom Safeguards Office operates a secure computer system for the storage, retrieval, and analysis of reported data, for the preparation of the reports for the IAEA, for on-site evaluation of measurement results, and for the verification and evaluation of operators' data.

Moreover, the Commission may impose sanctions in the event of an infringement of the safeguards obligations to a NPP and Member States shall ensure that they are enforced.

B.5 References

[1] *Report from the Commission to the European Parliament and the Council - Operation of the Euratom safeguards office* ,1999-2000

C. Industrial devices

C.1 Introduction

In this annex several nondestructive out-core FA characterization devices will be presented. It have been chosen to describe four of the most important and innovative systems in the word. The following devices will be analysed:

- The fork
- NAJA
- PHYTON^{TM}
- Cerenkov radiation monitor device

Unfortunately due to patents over this device and the industrial secrets only a summary description can be given of them because the lack of specific information.

C.2 The fork

"The fork" is a detector system and it was developed by Los Alamos National Laboratory during nineties. With proper calibration and corrections, the fork detector can determine the BU of individual fuel assemblies at an average accuracy of about 5% [1] of plant records. The observed data must be corrected for both by a parameter which takes into account the cooling times and initial enrichment. It is not an on-line device as NAJA whereas a spent fuel pool device.

The fork system has been used at several utility spent fuel pools like for instance:

- Arkansas Nuclear One (ANO) (Russellville, United states of America)
- Oconee Nuclear Station (South Carolina, United States of America)
- Surry Power Station (Virginia, United States of America)

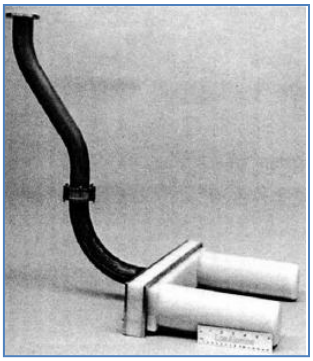


Figure C-1: The fork detector system.(Ref. 1)

The NPP changed some features of that device to obtain some particular characteristics useful for their application.

Next the application at ANO Unit 1 will be examined.

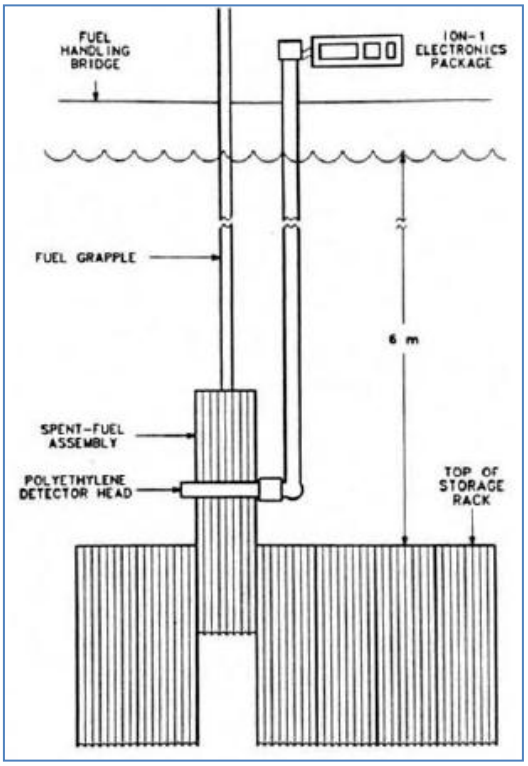


Figure C-2: Schematic showing the method for making measurements in a spent fuel pool using the fork detector system.(Ref. 1)

The Fork system was used to verify utility reactor records. It provided a non-destructive means to characterize the fuel assemblies following several months of cooling time in the spent fuel pools.

The neutron and gamma-ray emissions in the examined assemblies were measured raising each assembly partially out of the storage rack, as shown in figure 2, and performing the measurement near the centre of the assembly. In fact these measurements were taken at 1 foot below the assembly centreline because of the presence of a steel band located at the centre of each assembly.

The count time on each assembly was 100 s. The fork detector system used two arms that were placed on either side of the fuel assembly facing one another. Each arm of the device contained two fission chambers for neutron detection and an ionization chamber for gamma-ray detection to provide information on the neutron and gamma radiation fields emanating from the SNF assembly examined. Because of the features of the gamma-ray detector (it is less sensitive to variations in burn-up) it used to confirm the burn-up with approximately 15% uncertainty and for this reasons it was applied only as a confirmation of assembly cooling time and burn-up determined by neutron passive count.

Because measurements were taken only near the assembly centreline and compared with a best-fit calibration curve, the true assembly full-length average burn-up values were not actually measured.

The average burn-up values for the assemblies were developed as calculated numbers based on:

- The fork detector response (measured counts) at each assembly centreline
- The comparison of the detector output with the calibration standard curve
- Adjustment of the result for other factors such as for instance the cooling times

Finally, a few words about calibration method of fork device.

In contrast with the most of out-of-core measurement equipment which use a reference assembly of known burn-up and cooling time and of similar geometry for the calibration, ANO's Fork device uses a self-calibration (best-fit curve) shown in the next figure.

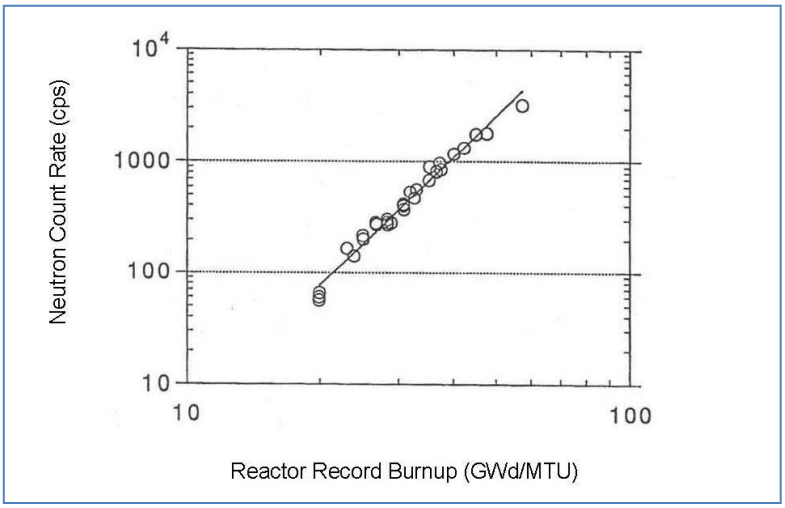


Figure C-3: Log-Log best-fit plot of neutron signal versus reactor record burn-up at ANO. (Ref. 1)

The approach used was to accumulate measurements from a number of assemblies and generate an internal calibration by comparing each assembly with the best-derived fit of all the site data. Thanks to this approach the uncertainties and complications that were introduced by external calibration techniques were eliminated while the accuracy of the measurement was not perturbed.

The self-calibration curve was the power law best-fit to the data and is given by:

$$N = CB^{3.83} \qquad Equ.C.1$$

where N is the neutron count rate in counts per second, B is the burn-up in units of GWd/MTU, and C is a fitted constant. The average deviation of the burn-up measurements from the best-fit curve of reactor record burn-up versus neutron signal was 2.7% at ANO-1, with a maximum deviation for a single assembly of 9.1%. This was consistent with the 2to 3% random variation among the reactor records for average assembly burn-up.

C.3 NAJA

NAJA [2] is a non-destructive automatic on-line device for fuel assembly characterization and core lore conformity control for PWR. This device uses both active and passive technique. Human interaction is not necessary. NAJA is also the name of an R&D project which is being carried out by CEA (Commissariat à l'énergie atomique et aux energies alternatives) and EDF. Nowadays, the feasibility study is over (both nuclear and optical aspects) and the work continues by a mechanical setting-up study and a prototype specification.

It consists of a developing a measurement device which combines nuclear methods and video control in order to evaluate the physical characteristics of each fuel assembly and to validate automatically the final core loading.

The device would be placed on the passage of the fuel assembly between the storage pond and the reactor building, in the fuel transfer tube. It should be useful for core loading conformity control and on-line core monitoring.

The NAJA device is able to determine automatically for each assembly:

- The nature of the fuel element (fresh or irradiated, UOX or MOX);
- The presence and the kind of neutron absorber;
- The initial enrichment in 235U for fresh UOX assembly with active neutron interrogation
- The identification number by an Optical Character Recognition
- Accurate burn-up measurement with passive neutron counting
- Burn-up profile by gamma spectrometry
- The activities of different fission and activation products thanks to gamma spectrometry

Following, the equipment needed for all the measurements is resumed:

- High efficiency gamma detector for gamma spectrometry
- Fission chamber detector for neutron counting
- A source of ^{252}Cf for active neutron interrogation
- Two video systems: the first one is linked to the NAJA device in the storage pool and the second one is linked to the loading machine in the reactor building. These two video systems are necessary in order to be sure of the good appropriateness

- between the position X,Y of the fuel assembly in the core and its physical characteristics

Next, it is shown the theoretical NAJA device setting-up principle but in a storage pool.

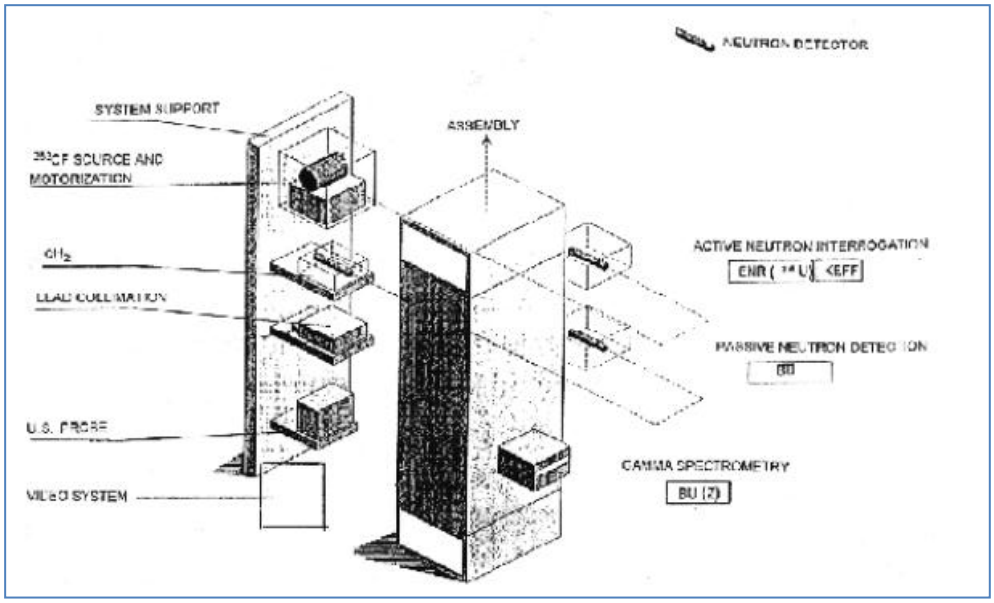


Figure C-4: NAJA device setting-up principle in a storage pool. (Ref. 2)

Latter, you an idea about the operational scheme of active neutron interrogation has been given.

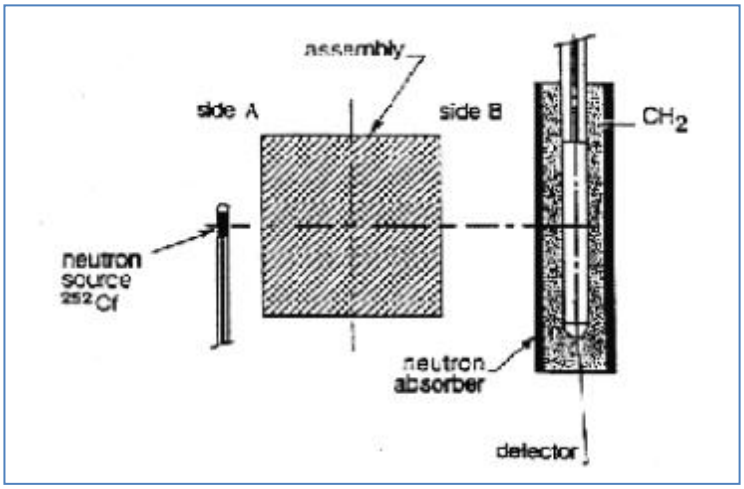


Figure C-5: physical principle of active neutron interrogation (Ref. 2)

All the information determined allow to:

- Characterize the fuel assemblies BU accurately : ±2% as a global uncertainty at two standard deviations on the absolute average burn-up evaluation
- Have a core loading conformity without human factor hazard because the device controls each assembly which goes to the reactor building (loading) or which goes to the storage building (unloading) storing all the information
- Increase the safety of the plant
- Rise the reactor availability

- Improve the operating margins due to a reduction of calculus / measure uncertainties because of the better knowledge of the fuel burn-up
- Improve the reshuffling itself because of the knowledge of the measured fuel burn-up
- Have a perfect of nuclear core code by the comparison of in- core and out-core BU measurement

Finally NAJA device has been reviewed in some characteristics and consequently can be also applied in:

- Dry storage or wet storage
- Spent fuel pool
- Cask loading for transportation.

C.4 PHYTON™

PHYTON™ [3] device have been developed by CEA and EDF. It mains objectives are to measure the average and extremity BU for safety critical purposes and the cooling time determination. It is a combination of various under water measurements as:

- A passive neutron measurement
- A collimated total gamma measurement
- An on line evolution code

To perform these measurements this device employs:

- Ionisation chamber
- High efficiency fission chamber

Figure 4, shows a schematic view of the two measurement heads that operate on the top of the storage racks in the NPP ponds.

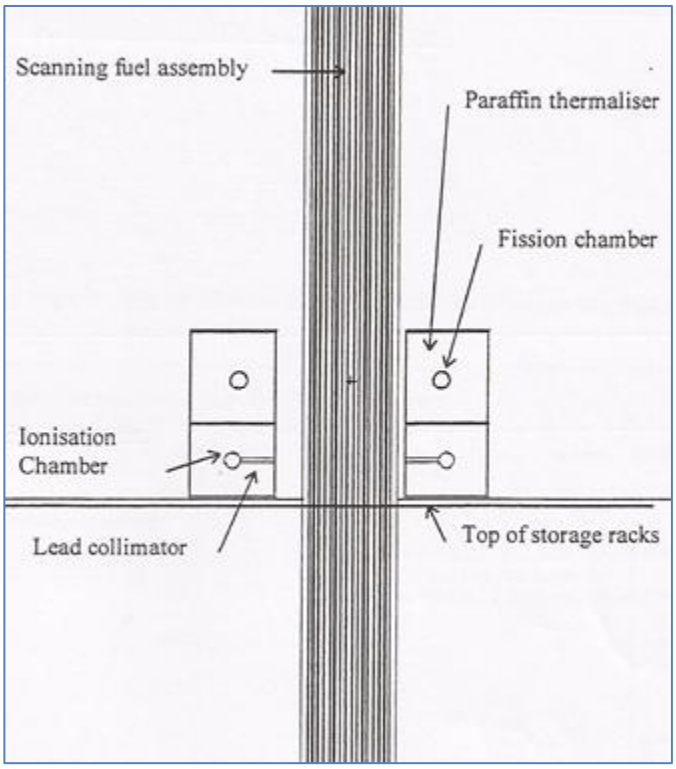


Figure C-6: The two detection heads of PHYTON (Ref. 3)

Since the PHYTON^{TM} device is intended to measure the average BU, the neutron signal has to be representative of the entire fuel assembly. This means that contributions to the average signal have to originate from the whole fuel. You have to keep in mind that contrary to gamma emissions that are mostly adsorbed when cross fuel pins, the neutrons detected by the fission chamber originate from almost all the fuel section. In addition, since the two head signals are averaged, the sensitiveness to radial gradient for the BU measurement is very low.

To take into account the neutron emission axial profile, fuel is scanned between the two heads and signal is averaged. However, despite neutron signals are acquired along the fuel, it is not possible to get directly from them BU profile. As a result, the BU profile has to be measured with the collimated total gamma detector and extremity BU calculated using both gamma profile and the average BU applying the next relation which assumes that TGE is mainly composed of BU proportionally produced gamma emitters:

$$EBU_{(0,z_0)} = BU\frac{z}{z_0} * \frac{\int_0^{z_0} TGE(z)dz}{\int_0^{z} TGE(z)dz} * F(CT) \qquad Equ.C.2$$

where EBU is the extremity BU, TGE is the total gamma emission, and F(CT) is a correction factor which has to be used for short cooling times CT, because short live fission products do not induce proportionality. Specifically for PWR reactors the follow relation can be convenient:

$$F(CT) = 0.5 * CT^{0.1} \qquad Equ.C.3$$

where the BU measurement is performed by a passive neutron technique the axial importance of the fission chamber cannot be overlooked. The following figure shows that almost a length of several tens of centimetres contributes significantly to the signal.

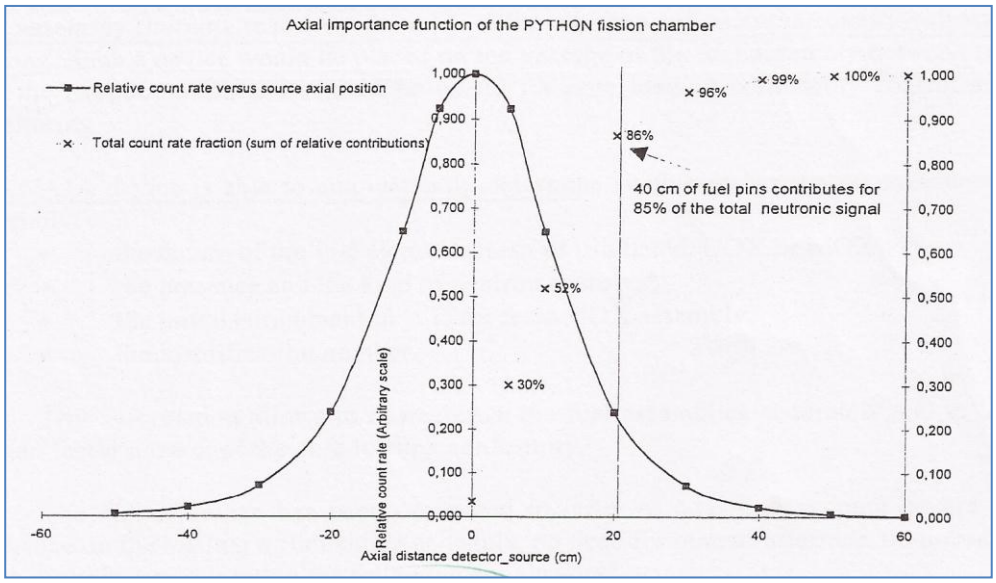


Figure C-7: Axial importance function of the PHYTON fission chamber (Ref. 3)

Besides, the PHYTON™ device can be also used for safety critically purpose: in this case the irradiation histories are supposed to be known as well the initial components of the fuel. They are used as input data by the on-line evolution code (i.e. ORIGEN-S) that determines a correlation law (BU=a NE^b) [3] for each IE. As a result, no standardization is required to determine the relevant correlation law to apply. Only a calibration is necessary to measure the detector yields. In order to avoid mistakes, normally the measure yields values are confirmed by MCNP calculations.

Furthermore, the PHYTON^{TM} system has intensively been qualified using a prototype device with active mode capability at Tricastin NPP and with industrial system manufactured by EURISYS MESURES. The accuracy evaluated by comparison with declared values for BU, CT and k_{eff} evaluation is as follow:

- On average BU: 2%
- On CT: 15%
- On k_{eff} : 3 %

Nowadays, R&D projects have been focused on extensions of the PHYTON^{TM} capabilities in order to use BU and reactivity estimation for core loading checks.

C.5 Cerenkov radiation monitoring devices

Cerenkov light measurement is a very simple and nonintrusive method for verifying the effectiveness of spent fuel safeguards compared to other methods because it involves only the viewing of spent fuel assemblies using a Cerenkov viewing device (CVD), and no movement of the stored spent fuel is required. Nowadays IAEA's inspector test CVD for those measurements.

Cerenkov radiation is emitted whenever a charged particle passes through a medium at a velocity exceeding the phase velocity of light in that medium. In water, the phase velocity of light is about 75% of its value in a vacuum. An electron passing through water and having a kinetic energy greater than approximately 0.26 MeV will produce a Cerenkov radiation. In irradiated fuel, these electrons include Compton electrons produced by gamma radiation, beta rays that escape directly into the water, and the interactions of high-energy neutron capture gamma rays that produce electrons from Compton scattering and pair production. The intensity of Cerenkov light generated by irradiated fuel is proportional to the radiation field intensity in the vicinity of the irradiated fuel. This field intensity is proportional to the burn-up of the fuel and inversely proportional to the cooling time. Intensity is measured by selecting the brightest pixels of the recorded image. A Swedish study report indicated that it may be possible to determine the BU of an assembly by a DCVD (or developed Cerenkov viewing device) knowing its cooling time even if theoretical calculations of photon intensities as a function of BU and CT for spent fuel are not available [3].

Current CVDs can detect Cerenkov glow images without interference from normal lighting. As Cerenkov emissions range from the ultraviolet (UV) to the infrared ones, the CVD uses a UV light

image intensifier, which has good sensitivity to UV light due to the use of a Cs-Te photocathode material. The CVD also uses a conventional night viewing device with an optical filter to minimize interference from lights in the facility. The use of modern CVD equipment allows the detection of a weak Cerenkov glow image that may result from fuel that has a low burnup or a relatively long cooling time.

Considerable work remains to be done, specifically with respect to the precision of readings and the stability of the CVD detector. Some parameters have to be kept under control:

- The concentration of boron: in spent fuel pools it is an important factor, because it allows absorbing neutrons emitted from the spent fuel causing the reduction of the light intensity of the Cerenkov glow.
- The "near-neighbour" effect of spent fuel assemblies in proximity to the assembly to be examined (either adjacent to or diagonal): It can affect the meaningfulness of the results.

To conclude it may be very difficult to distinguish between old, moderately burnt fuel and new, very lightly burnt fuel by the Cerenkov glow measurement. Nevertheless, a Cerenkov glow examination may be capable of providing "go/no-go" information for imprecise screening and the big IAEA's effort can lead in a short period to improve the DCVD capability.

C.6 References

[1] B.B. Bevard, J.C. Wagner and C.V. Parks (ORNL) M. Aissa (NRC), *Review of Information for Spent Nuclear FuelBurn-up Confirmation*, US NRC, 2009

[2] US NRC, 2009 G.Bignan, D.Janvier, *NAJA: A new non-destructive automatic on-line device for fuel assembly characterization and core lore conformity control*, ANS meeting "Advance in nuclear fuel management II", Myrtle 1997

[3] A. Lebrun, G.Bignan, H. Recroix, *Characterizations of spent fuel assemblies for storage facilities using nondestructive assay*, IAEA-SM-352/33, 1999

D. Confidential database analysis

D.1 Introduction

In this annex a confidential PWR database about FA will be analysed, nine hundred and four in total. It is owned a Spanish Nuclear Power Plant whose name cannot be revealed to preserve national security. You can find important information, as for example the BU [MWd/tU], the download time (when the FA is discharged and is sent to the spent fuel pool) and also the IE.

Because of the large amount of data and information it has been chosen to examine it with statistical software: MINITAB 16.

The analysis has been divided in topics.

D.2 Initial enrichment

A histogram helps us to understand the variety of FA's IE.

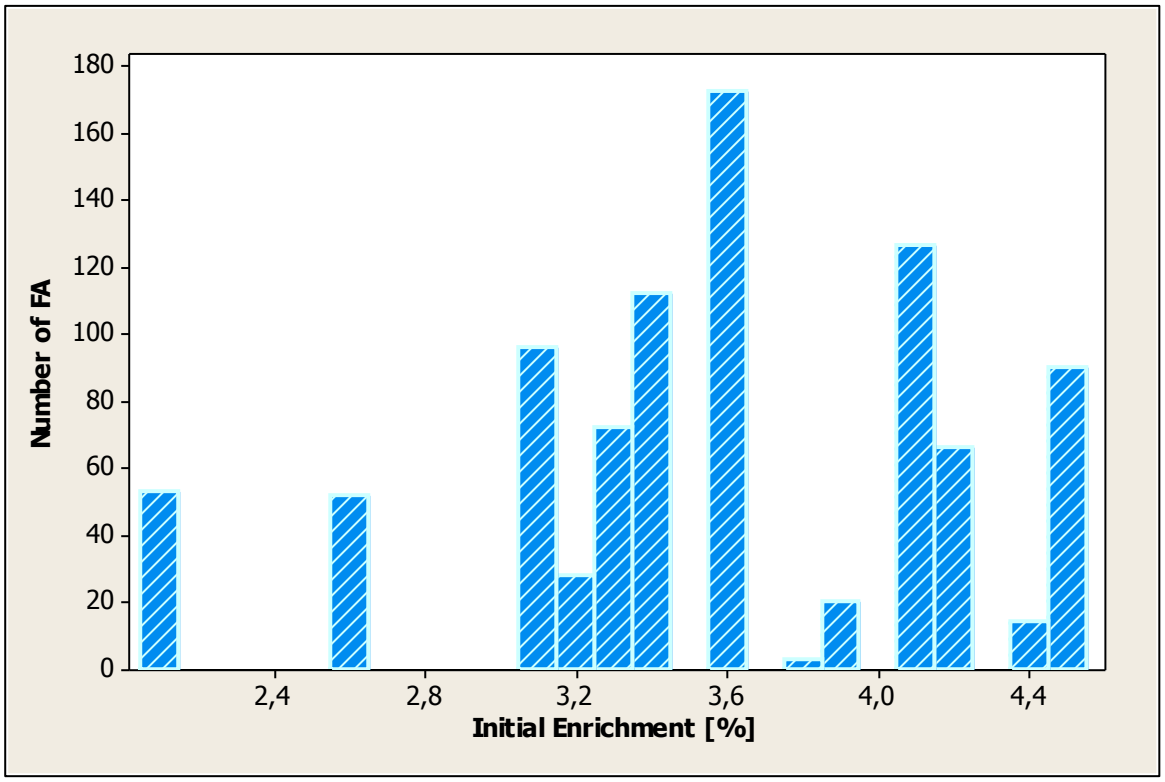


Figure D-1: Number of FA vs. IE

You can observe the FA used in this NPP has different IE. This is caused by the improvement of nuclear fuel management that has required different quantity of U^{235} along the time. In general, the recharge operation is becoming more complex than the standard "centripetal operation" to obtain higher BU and so major plant efficiency, a smaller leakage of neutron due to neutron flux slope. The IE equal to 3.6% is the most used one. It is a normal IE for a PWR.

Moreover, the FA has been divided in groups depending on the IE with the maximum range of 1% from the bottom to the top end of each. This apparently strange operation will be helpful later.

The rate has been calculated in the follow way:

$$[\%] = \frac{IE_{max} - IE_{min}}{IE_{min}} \qquad Equ.D.1$$

Sixteen groups have been obtained:

BU [MWd/tU]		Group	Number of FA
From	*Up to*		
2,1149	2,1151	A	53
2,5829	3,1031	B	52
3,1029	3,1031	C	52
3,1484	3,1486	D	44
3,2287	3,2591	E	100
3,3500	3,3665	F	73
3,3965	3,4085	G	39
3,5881	3,5999	H	38
3,6000	3,6266	I	134
3,8476	3,8567	L	23
4,0667	4,1088	M	28
4,1213	4,1588	N	117
4,1603	4,2120	K	47
4,4381	4,4695	P	41
4,4699	4,4996	O	39
4,5159	4,5496	Q	24
		Total	
			904

Figure D-2: fuel assembly initial enrichment groups

Now it is interesting to note how the IE has been changed along the NPP life.

In the following graph, it is shown the relation of IE with the moment of FA discharge.

It is a linear dependence as shown by the blue fit line: nowadays this NPP use a higher IE than in the past.

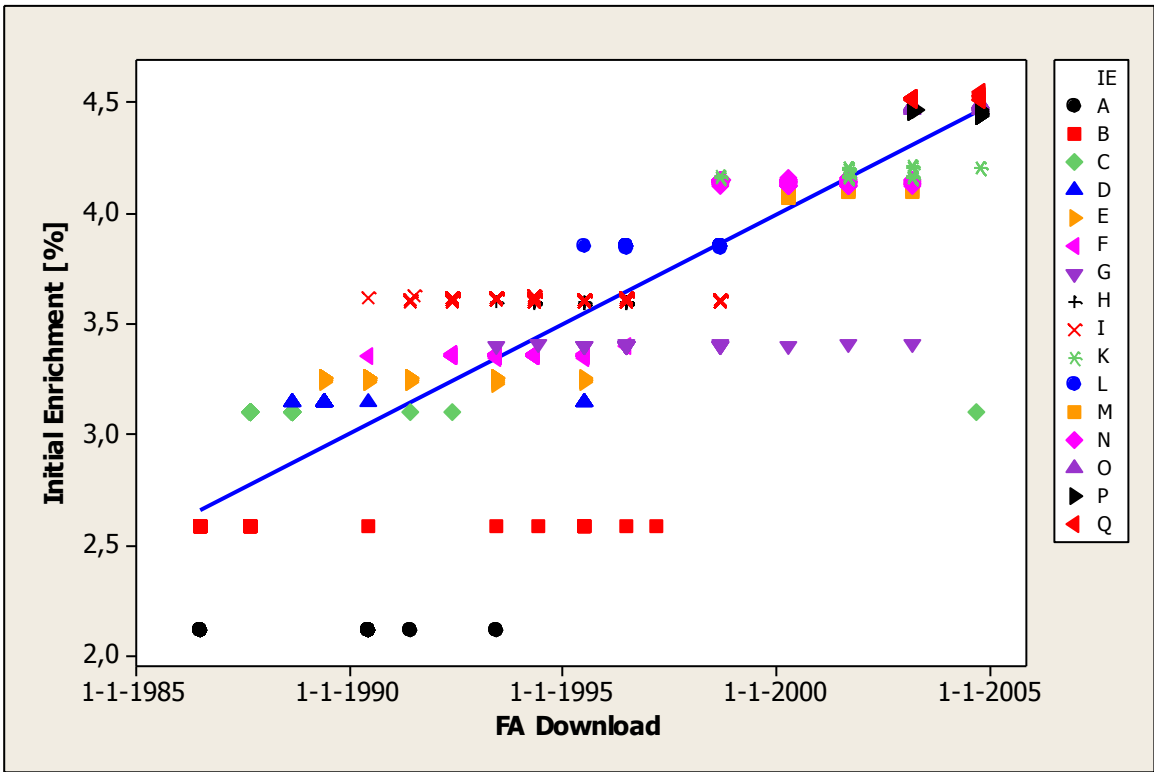


Figure D-3: IE vs. FA download

D.3 FA discharged during recharge operation

The next histogram indicates the number of FA discharged during each recharge operation. You can observe how the FA of the same group has ended its work in different moments because the reactor core is not composed by FA with the same initial amount of U^{235}. A standard usual PWR you use fuel with three types of different IE.

Furthermore you can assume that the kinds of fuel downloaded and sent to spent fuel pool give us some information about the composition of the reactor core. In this way you can presume that the core is formed by "neighbour" groups and so not very different IE.

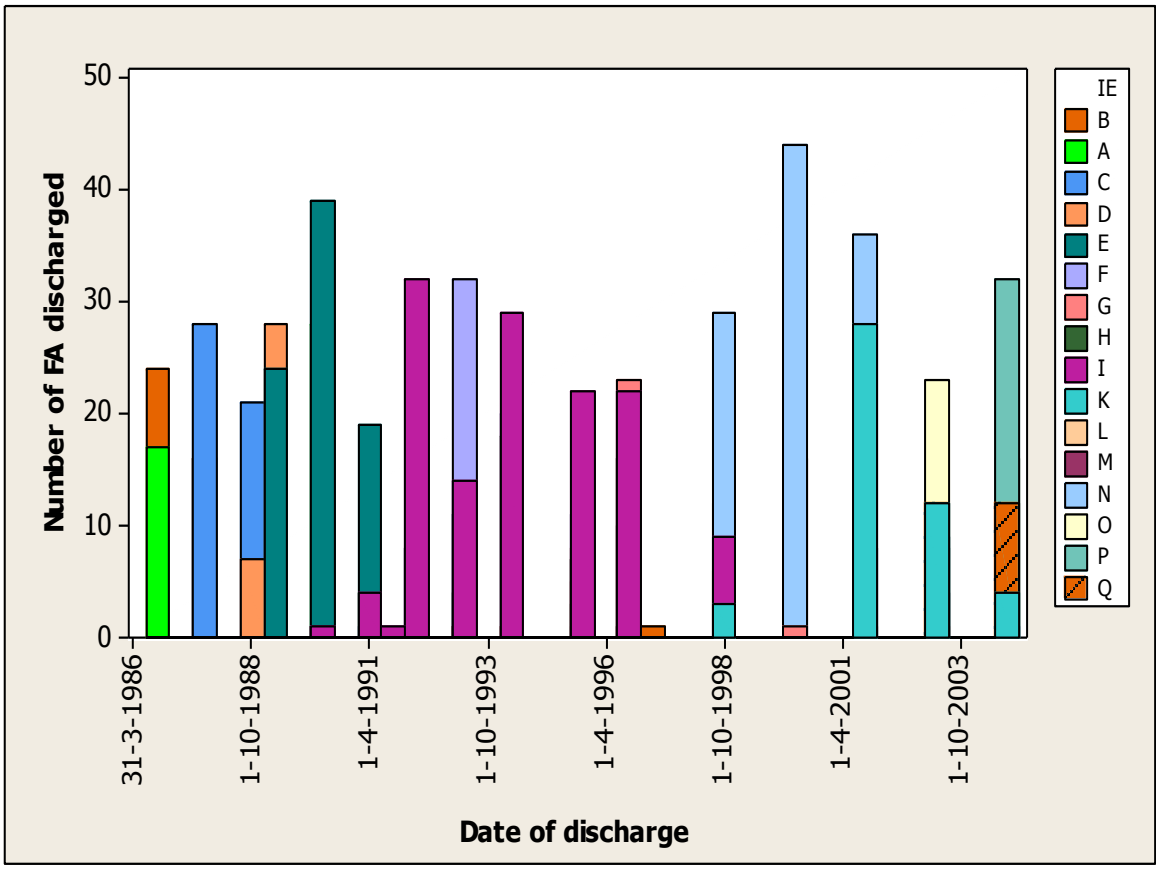


Figure D-4: date of discharge vs. number of FA discharged

D.4 Fuel cycle time

Both the previous histogram and the following elaborated table from the database, give us the possibility to analyse the fuel cycle time.

Time of download	FA discharged	Fuel Cycle Time [Month]
1-7-1986	41	n.d.
1-9-1987	44	14
1-9-1988	28	12
1-6-1989	52	9
1-6-1990	71	12
1-6-1991	28	12
1-7-1991	1	1
1-6-1992	41	11
1-6-1993	64	12
1-5-1994	67	11
1-6-1994	2	1
1-7-1995	64	13
1-7-1996	73	12
1-3-1997	1	8
1-9-1998	58	18
1-10-1998	1	1

1-4-2000	61	18
1-9-2001	73	17
1-3-2003	69	18
1-9-2004	1	18
1-10-2004	64	1

Figure D-5: fuel cycle time

At first, you can presume that there are four stops of the reactor non-scheduled (highlighted in red) because the fuel cycle time and the number of discharged FA are very different compared with other programmed recharge operations: you can observe only two of these stops on the previous graph because of the round and the scale of MINITAB. They could be caused by a broken fuel rod pin and in particular of the cladding.

In order to obtain the best efficiency and load factor of the NPP the nuclear engineering teams work hard on the nuclear core management, especially on the IE, the BU and the movement of FA inside the core.

Secondly, examining the previous table you can notice that the maximum number of FA changed is 73; supposing that in a PWR with 3000MW_t there are about 160 FA, it follows behind that the utility inserts at most 42% of fresh fuel in each recharge.

The coming graph shows that the number of FA discharged in recharge operation and the fuel cycle time has increased along the time: that can be confirmed with MINITAB computing the correlation coefficients:

- Correlation of Time of download and FA discharged = 0,633
  P-Value = 0,009
- Correlation of Time of download and Fuel Cycle Time = 0,694
  P-Value = 0,004

These P-values confirm our hypothesis because they are less than 0.05, the usual risk assumed.

However calculating the correlation coefficient between number of FA discharged and Fuel Cycle Time you obtain 0,244 and a P-Value equal to 0,381. You can affirm that in this NPP there is no relation between the duration of the fuel cycle and the numbers of FA discharged.

You would like to remember that correlation coefficient is dimensionless, it goes from -1 to +1, and can be computed as follow:

$$r_{x,y} = \frac{\text{covariance}\,(x,y)}{s_x * s_y} \qquad Equ.D.2$$

where s_x and s_y are the standard deviations of the variables x and y.

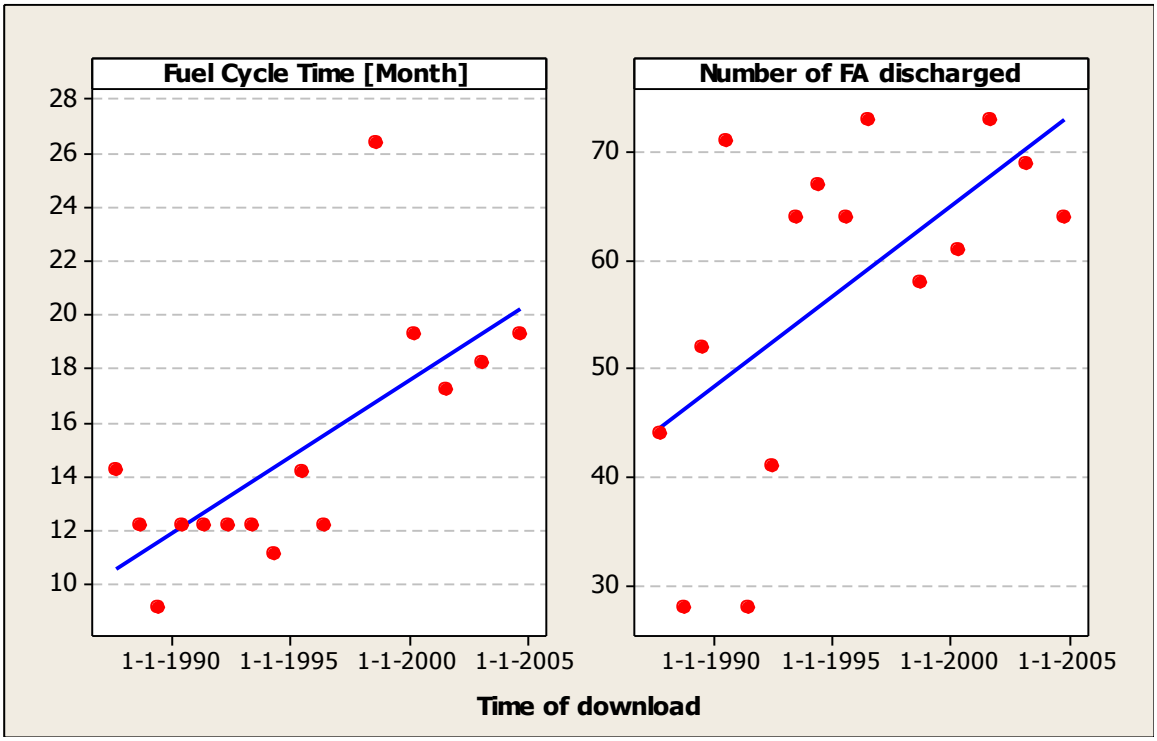


Figure D-6: time of download vs. fuel cycle time and number of FA discharged

D.5 Burnup

Now the behaviour of BU along the time will be examined. The results have been reported as follows.

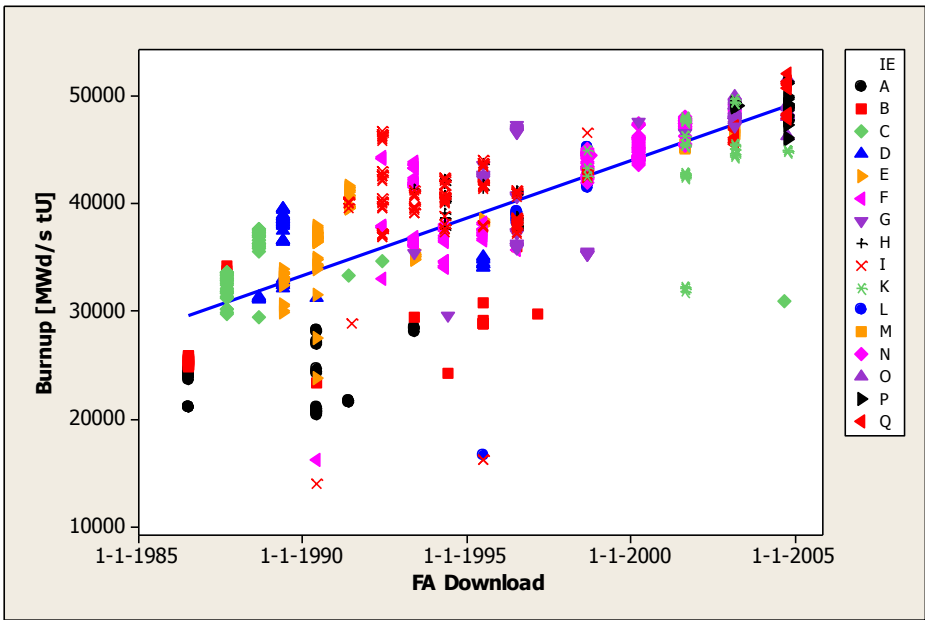


Figure D-7: burnup vs. time of download

It can be affirmed that BU increases during the time: of course this mainly depends on the IE, but also by the fuel management in core. Indeed plotting the IE versus the BU a positive linear correlation is obtained as shown by the fit line in the next graph.

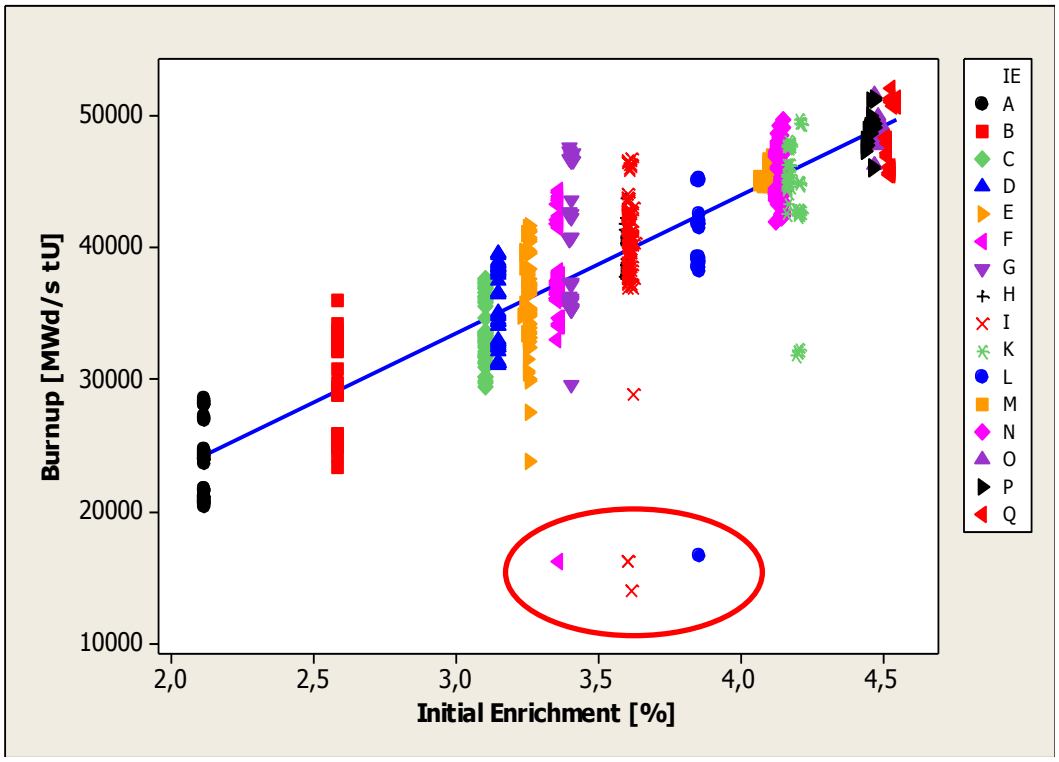


Figure D-1: burnup vs. initial enrichment

The previous figure shows how for the same group of IE the BU is different that may be caused by:

- Different irradiation history
- Dissimilar initial condition of IE in the group
- The randomness of fission
- Mechanical problems on the FA, i.e. the breaking of the cladding, that causes the advanced change of it in comparison with the programmed fuel cycle

Moreover, you can presume that the FA with such a different BU has some problems and can cause the stop of NPP. This has been resumed in the next table:

FA Code	FA Download	Burnup [MWd/s tU]	Storage time [day]	Enrichment [%]	Group
J-I	1-6-1990	16276,9	7930	3,3564	F
ER-10	1-7-1995	16173,9	6074	3,6028	I
H-46	1-6-1990	14066,6	7930	3,6164	I
ER-11	1-7-1995	16690,2	6074	3,8507	L
AG-55	1-9-2001	31828,53	3820	4,1945	K
AG-52	1-9-2001	32073,36	3820	4,1999	K

Table D-1: problematic fuel assembly

Comparing these download times of with those of the non-scheduled unloading operations you can assert that these six FA did not cause the NPP stops. The real reasons for these strange BU values cannot be exactly defined but surely they were not caused by a mechanical problem and they were a combination of those proceeding listed.

D.6 Burnup study for initial enrichment equal to 3.60%

It has been chosen to study one group and examine the behaviour of BU. In particular, the examination has been performed "group I" because both it contains the largest amount of FA and the range of IE is very small ([%] = 0.738%).

Using all information in our possession, the following dot-plot and the boxplot have been elaborated.

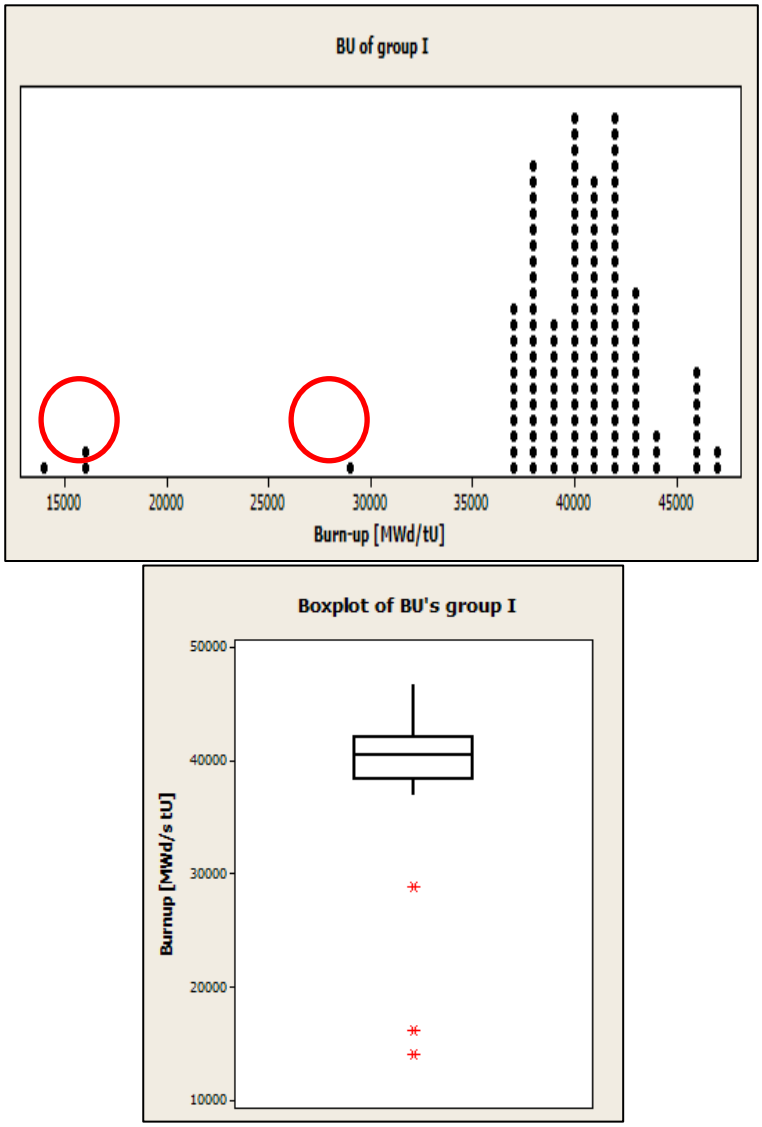


Figure D-9: statistical analysis of group I

Examining the previous two figures you can report that four FA have quite different values of BU compared with to the others, but only three of them have significantly different statistical value: these are highlighted by red stars in the boxplot.

At the end, taking out those three FA, a Normal distribution of BU has been obtained with a mean of 40616 MWd/tU and a standard deviation of 2396 MWd/tU.

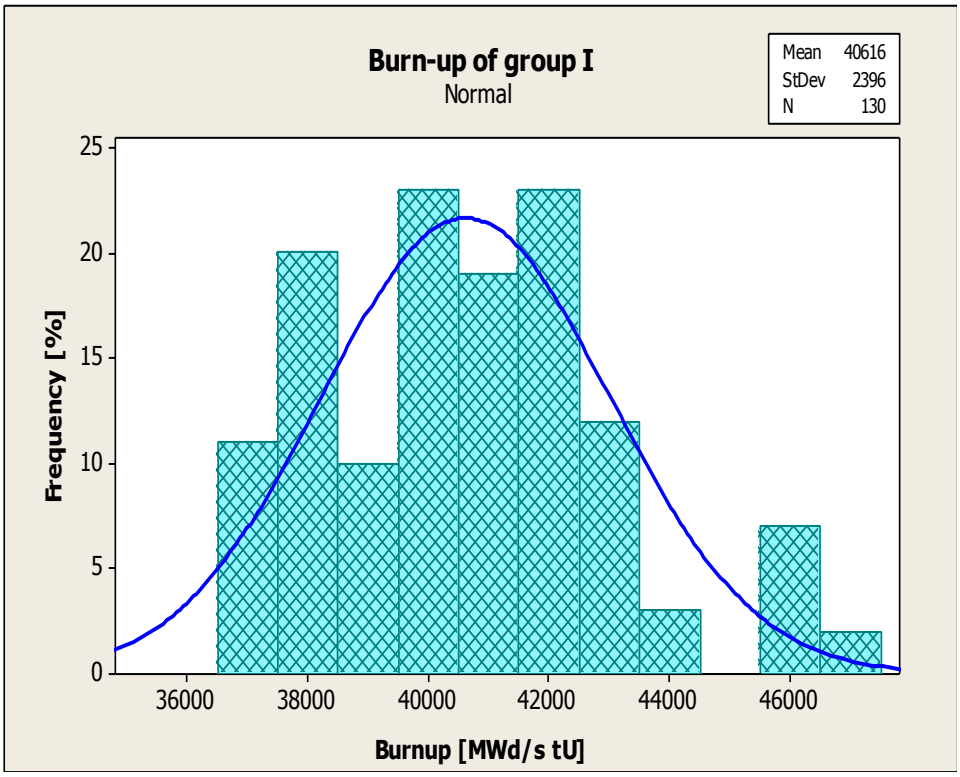


Figure D-10: Burnup histogram of group I

Of course this behaviour does not surprise us and respects the central limit theorem of probability theory applied to neutronics. CLT says that the mean of a sufficiently large number of independent random variables (as for instance in this case the microscopic fission cross section of materials), each with finite mean and variance, will be approximately normally distributed.

D.7 Summary of the analysis

The confidential database of a Spanish NPP contains 904 FA in total mostly of group I (IE $\in$ [3.6000: 3.6266]). Along the time the NPP has increased the amount of U^{235} in FA in a linear way: indeed the last downloaded ones contain the largest amount of IE and belong to groups Q, O and P.

Due to both the increasing IE and the better fuel in core management, the BU is increased in a linear way. Besides a more elevated IE has permitted to rise the fuel cycle time according to a higher load factor.

Furthermore in the database you have noticed five non-scheduled NPP stops caused by FA with a "normal" BU.

Finally the behaviour of BU in "group I" has been studied noticing that three FA have significantly different statistical values of it. Taking out those three ones from the analysis the BU can be represented by a Normal distribution in line with a Central Limit Theorem.

D.8 References

[1] UPC edition,*55 Respuestas a Dudas Típicas de Estadística*, Behar, Grima. Díaz de Santos, 2004.

[2] UPC edition, Métodos Estadísticos, *Control y Mejora de la Calidad, Prat, Tort-Martorell, Grima*, Pozueta, Sole. Edicions UPC, 2004

[3] UPC edition, Prat, *Estadística Práctica con MINITAB* , Tort-Martorell. Prentice Hall, 2004

E. Fuel handling operation

E.1 Introduction

In this annex are going to be described fuel handling operations and the related equipment of ASCO-I PWR nuclear power plant.

E.2 Description of a refuelling operation

The recharge operation is composed by a lot of detailed operations which lead a secure and efficient process that has to be in compliance with the standards For example, NRC requires:

- An effective multiplication factor of core less than 0.95 which can be reached by chemical shim and control rods.
- At least three meters of water above the active length of FA during all fuel handling operation to maintain radiation level below acceptable criteria.
- Minimal handling of fuel in order to avoid misloading operation and possible handling incidents.

A refuelling operation can be divided in four main phases:

a) Preparation
b) Reactor's disassembling
c) Fuel handling
d) Reactor's assembling

E.2.1 Preparation

The reactor is shut down with all control fully inserted to maintain the k_{eff} less than 0.95. All refuelling operations can begin only when the reactor has been subcritical for 34 hours[1].

Entering the reactor building is permitted only when radiation levels are considered acceptable after air cleaning operations n and activity measure tests. The moderation level in the vessel is carried down just below the head flange. The refuelling machine is testes .The reactor pool is prepared taking off the ballistic slab; the fuel transfer facility is checked.

E.2.2 Reactor disassembling

The control drive mechanisms, cooling air ducts, reactor vessel head insulation and all in-core instrumentation are detached from the reactor vessel head and moved to storage. The blind flange of the fuel transfer tube is removed. Once the reactor pool is ready to be flooded the reactor-vessel-to-pool seal is bolted down and the closure head assembly is lifted less or more 30cm. Subsequently the flooding operation is started by the by residual heat residual system's pumps which introduce borated water (2600 ppm, 50°C [2]) into primary loops from internal recharge water tank o IRWT, forcing it to overflow into reactor pool. The reactor head vessel is lifted up as the water rises, keeping it just below the reactor head. When the height of water is sufficient to provide a safety shielding, the vessel head is stored is in a dry area. The reactor vessel upper internals are lifted out of the vessel and stored in the refuelling pool, thus providing access to the assemblies.

E.2.3 Fuel handling

The various handling and transfer operations performed during refuelling are described below:

- The refuelling machine is positioned over FA to be removed.
- Alignment of the crane to the top of the fuel assembly is accomplished by using a digital readout system, and the process is monitored by closed-circuit television.
- The FA is raised high enough to pass above the vessel whilst remaining submerged in water, in order to minimise any risk of exposure of operations staff to radiation.
- After removal from the core, the spent fuel assembly is moved underwater to the transfer area of the pool while the transfer system container is placed in a vertical position by the winging chassis of the fuel transfer facility.
- The refuelling machine introduces the fuel assembly into the fuel transfer facility container
- The fuel transfer facility container is placed in a horizontal position by the swinging chassis of the fuel transfer facility.
- The conveyor trolley carrying the container is moved through the fuel transfer tube towards the transfer pool of the fuel building.
- The container is placed in the vertical position by the swinging chassis of the fuel transfer facility.
- The fuel assembly is removed from the container by the spent fuel mast bridge and moved from transfer pool to storage fuel pool through an opening. This is kept close using a door and a sluice gate, except refuelling operation.

- The fuel assembly is inserted into a cell in the underwater fuel storage rack or, if necessary, in a storage cell for defective assemblies.
- When all the fuel assemblies have been transferred into the underwater fuel storage rack, the control clusters and the thimble plugs are changed between the fuel assemblies using the spent fuel mast bridge.

E.2.3 Reactor's assembling

Refuelling core with new and irradiated FA consists essentially on performing the above operation in reverse. FA introduced in the core are some fresh and some irradiated. Generally one third of the core is composed by fresh fuel which in "centripetal refuelling movements" is positioned in the outer zone whereas the irradiated one is positioned in the inner zone. In the next figure this type of setup and one more developed (three-region checkerboard) is shown.

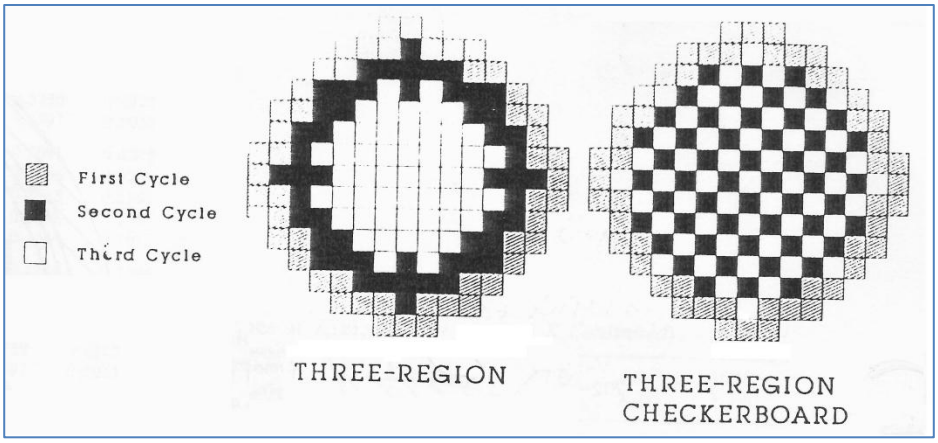


Figure E-1: Fuel loading patterns (based on information from Westinghouse Electric corporation Ref. 7)

Besides a lot of studies are carried out to improve the BU and homogenize it by a "special and innovative" configuration. Moreover *Southern Nuclear Operating Company* [4] has developed a very particular reshuffling operational to decrease the time of the recharge. This technique limits the FA movements. In fact in this case not all FA are moved to the SFP but only a sufficient amount to allow the FA reshuffling directly in the core.

E.3 Description and characteristics of equipment

The fuel handling system comprises the equipment needed for refuelling the reactor core, i.e.: refuelling machine, fuel transfer facility and spent fuel mast bridge. The areas associated with the fuel handling equipment are the reactor pool, the transfer pool in the reactor building and the fuel building.

E.3.1 Spent fuel pool

The spent fuel pool has to provide wet storage and protection of 1.421 FA, with a maximum IE of 4.95% of U^{235}. Its volume is $1059m^3$.

The major part of FA, 1.245, have to be irradiated FA with a minimum BU to maintain the k_{eff} less than 0.95 also in the worst conditions of accidents whereas the remaining FA , 176, can be fresh fuel or FA which do not reach the minimum BU[1].

Spent fuel racks can storage different types of FA (i.e. 17x17 STD, OFA 17x17, AEF 17x17 and MAEF) which have been changed along the plant life to improve the in-core nuclear management.

Spent fuel racks design allows borated water to flow inside FA , to eliminate the residual heat and to ensure subcritical geometry. They have been also projected to resist both to basis and design earthquake.

A fully loaded PWR storage pool contains approximately 1.3×10^5 kg uranium (0.8 per cent enriched U^{235}) and 1200 kg plutonium (all isotopes)[3].

E.3.2 Fresh fuel pool

Fresh fuel pool is placed on fuel building and there can be stored 72 new FA in a dry way with a maximum enrichment of ^{235}U equal to 4.95.

The FA are stored in vertical cells. The cells are arranged in two groups of two rows each with 18 cells per row. In each group the separation between cell centres is 533 mm. Cells of the two central rows (18 x 2) have a stainless steel plate 4 mm of thickness for shielding neutron on its outer face (central passage) . The surfaces of the cells in contact with the fuel elements are austenitic stainless steel.

The fresh fuel pool has internal dimensions of 9.7 m by 2.59 m a depth of 4.85 m from the level of operation, concrete slabs have to be closed at the top and bottom drainage sufficient to ensure elimination of any accidental spillage.

E.3.3 Reactor pool

The reactor pool is placed in reactor building between the vessel and the fuel building. It is filled by water only during refuelling operation. The rector pool is located in connection with the storage pool through a transfer tube.

E.3.4 Transfer pool

The transfer pool is situated in the fuel building and connects fuel tunnel transfer to spent fuel pool. It is deep, at average, 10 m and contains 208m^3 of borated water during refuelling operation.

E.3.5 Fuel transfer facility

The fuel transfer facility is used to transport fuel assemblies between the reactor building and the fuel building and vice versa, through the containment penetration formed by the fuel transfer tube. A container used to transport a fuel assembly is mounted on a conveyor trolley. The trolley is moved horizontally on runway tracks by a rigid pusher stainless steel chain driven by an electric motor located on the service floor. When the reactor is on power the trolley is stored inside the fuel tunnel transfer.

At each end of the transfer tube, a swinging chassis is used to place the container in the horizontal or vertical position. The fuel assemblies are placed into and removed from the container using the refuelling machine or the spent fuel mast bridge.

The chassis is swung by electrical winches located at the service floors and is controlled locally from two control consoles: one in the fuel building, which also controls the lateral movement of the conveyor trolley, and the other in the reactor building.

During operation of the reactor, the transfer tube is isolated on the fuel building side by a manual valve and on the reactor building side by a blind flange with rapid opening and closing features. It turns out that the fuel tunnel transfer is empty when the reactor is on power whereas in refuelling operation is filled with borated water which an amount of Borum equal to 2600 ppm as the spent fuel pool.

An improvement of this system and in particular of the blind flange locking system has been done by Westinghouse Nuclear Division [5].

E.3.6 New fuel elevator

The new fuel elevator is located in a side of the transfer pool and comprises a basket of stainless steel travelling on two vertical rails using an electric winch located on the service floor. Its maximum velocity is 3m/min. It is used to take new UO2 fuel assemblies from the surface of the storage pool to the bottom of the pool. Specifically in Ascò nuclear power plant the new fuel elevator cannot transport irradiated FA whereas in the EPR yes. In fact in that innovative PWR reactor an out-core FA characterization composed by active and passive techniques will be placed in this section of the transfer channel.

E.3.7 Spent fuel mast bridge

The spent fuel mast bridge is used to handle the fuel assemblies, underwater in the fuel building, between the following equipment: new fuel elevator, underwater storage rack, fuel transfer facility transfer pool. The spent fuel mast bridge is also used to change the rod control clusters.

The spent fuel mast bridge is a bridge equipped with a trolley and a hoisting mast.

It is equipped with the following devices:

- A control console linked to all the instrumentation and control systems needed to operate the spent fuel mast bridge via two operating PLCs
- Sensors and positioning devices which provide the two PLCs with information such as the position (X, Y, Z) of the hoisting mast
- A system for permanent video control of the handled load, with automatic shutdown if there is excess load or under-load

E.3.8 Refuelling machine

The main function of the refuelling machine is to handle the new or irradiated fuel assemblies under water in the reactor building. . It can be moved in automatic or manual way. The refuelling machine comprises a bridge, a trolley and two fuel hoisting masts.

The gripper, located on the lower end of the first hoisting mast, may grasp a fuel assembly and move it in three directions (X, Y, Z) in the reactor pool. The second mast, dedicated to handling fuel clusters, allows changing of rod cluster control assemblies (RCCA) and thimble plug assemblies (TPA) in the reactor building.

The refuelling machine is also equipped with the following devices:

- A control console placed on the service floor of the reactor building linked to the instrumentation and control devices needed to operate the refuelling machine via two Programmable Logic Controllers (PLC)
- Sensors and positioning devices which provide the two PLCs with information such as the position (X, Y, Z) of the fuel mast
- A system for permanent video control of the handled load with automatic shutdown if there is excess load or under-load

Nowadays improvements of the video control are carried out by Westinghouse [6]. This advanced technique can also validate automatically the finally core loading.

E.3.9 Sluice gate

The fuel storage pool has two sluice gates which link to transfer pool and to cask loading pit. The first one is closed except in refuelling operation when the transfer pool is full of borated water whereas the second is opened only during fuel handling with the loading pit. Those opening are kept closed using a door and a sluice gate. They are made of stainless steel and designed to support both basis and design earthquake.

E.3.10 Auxiliary crane

The fuel building auxiliary crane travels along a runway track attached to cantilevers located at the top of the building walls.

The auxiliary crane is used during the construction phase to install the main equipment inside the fuel building (e.g. the underwater fuel storage rack). During the operational phase, the auxiliary crane is used to handle the following:

- The new UO_2 fuel assemblies
- The sluice gate of the loading pit or the transfer pool
- The fuel assemblies and control rod clusters in the event of failure of the spent fuel mast bridge hoisting function

E.4 References

[1] EPR UK, *Refuelling operation*, 2009

[2] ANAV, *Informe seguridad final de la central de Asco*, 1988

[3] John W. Moyer, Cecil S. Sonnier, *Baseline description for LWR spent fuel storage*, handling, and transportation, June 1978

[4] W.R. Sutton III, J.T. Van Bloois, P.H. Wakker, John W. Moyer, *Reducing refuelling outage duration by optimizing core design and shuffling sequence,* NRG & Southern Nuclear Operating Company, ANS, 2003

[5] Westinghouse Nuclear Division, *Fuel Transfer Tube Bolt Reduction*, April 2009

[6] Westinghouse Nuclear Division, *Refuelling for PWR/BWR TELESCOPE Sipping Services*, April 2009

[7] Westinghouse Nuclear Division, *Appunti Impianti a Fissione*, Prof. Bruno Pannella, POLITO, 2010

F. MatLab programs and functions

Due to the great amount of simulations done with ORIGEN-S, around three hundreds, MatLab-17 have been utilized to import and elaborate those file and afterwards create an Excel database.

Thanks to the programs created, import files are very easy and require introducing the filename only. The big effort has been translated to the MatLab programing and so the amount of simulations done is not a problem. In this way future development of this database can be done straightforwardly. Moreover the curve fitting MatLab toolbox has been used obtain the best fit of the BU/NE and PU/NE correlations.

In this annex the following will be described and reported:

- A Mundus operandi which describes the methodologies of data treatment
- The MatLab programs and functions created
- An Excel database description
- Best fit

It is worth remembering that the whole database is not printed in this annex because of its enormous extension but it is consultable in digital format inside the CD.

F.1 Modus Operandi

Fundamentally in this master thesis project, four major groups of ORIGEN-ARP simulations have been done to study parameter sensitivities, varying one of the base-case factors whereas the others were fixed. They are listed as follows:

- Simulation in function of IE and BU
- Simulation in function of different IH and BU
- Simulation in function of diverse FCL and BU
- Simulation in function of fuel pellet porosity and BU

Basically thanks to OPUS each ORIGEN-S simulation has been exported in 4 different text-files:

- Composition of fuel in function of time during irradiation time
- Composition of fuel in function of time during refuelling time
- Gamma spectra at each out-core measurement
- Neutron spectra at each out-core measurement

The files created by SCOPUS cannot be directly loaded into MatLab because they alternate text rows to data rows. All files need to be modified manually in order to create matrixes with text column or data column and loaded to MatLab. Once modified they can be loaded by the special function created explicitly by *Import Wizard*, if there are text and data columns, or by the function *fscanf* if only data columns are present.

Each SCOPUS text-file has been called with a predefined name according to the parameters sensitiveness studied. Following a resume of predefined names has been reported. They have been divided them into the four major group simulations: the first hash key has to be substituted with the number of IE and the second one with the number of ORIGEN-S simulation. Moreover an increasing number of ORIGEN-S simulations correspond to a larger BU.

NE=NE(IE,BU)	
Name	**File content**
E#O#C	Fuel composition during irradiation period
E#O#D	Fuel composition during refuelling time or decay time
E#O#N	Neutron spectra at each NDA operated
E#O#G1	Gamma spectra of group energy 1 at each NDA operated
E#O#G2	Gamma spectra of group energy 2 at each NDA operated
E#O#G3	Gamma spectra of group energy 3 at each NDA operated

NE=NE(IE,BU,FCL)	
Name	**File content**
F1E1O#C	Fuel composition during irradiation period; IE1; FCL equal to 10 months
F1E1O#D	Fuel composition during refuelling time or decay time; IE1; FCL equal to 10 months

F1E1O#N	Neutron spectra at each NDA operated; IE1; FCL equal to 10 months
F1E1O#G	Gamma at each NDA operated; IE1; FCL equal to 10 months
F2E1O#C	Fuel composition during irradiation period; IE1; FCL equal to 14 months
F2E1O#D	Fuel composition during refuelling time or decay time; IE1; FCL equal to 14 months
F2E1O#N	Neutron spectra at each NDA operated; ; IE1; FCL equal to 14 months
F2E1O#G	Gamma at each NDA operated; IE1; FCL equal to 14 months
F1E5O#C	Fuel composition during irradiation period; IE5; FCL equal to 16 months
F1E5O#D	Fuel composition during refuelling time or decay time; IE5; FCL equal to 16 months
F1E5O#N	Neutron spectra at each NDA operated; IE5; FCL equal to 16 months
F1E5O#G	Gamma at each NDA operated; IE5; FCL equal to 16 months
F2E5O#C	Fuel composition during irradiation period; IE5; FCL equal to 20 months
F2E5O#D	Fuel composition during refuelling time or decay time; IE5; FCL equal to 20 months
F2E5O#N	Neutron spectra at each NDA operated; IE5; FCL equal to 20 months
F2E1O#G	Gamma at each NDA operated; IE5; FCL equal to 20 months

NE=NE(IE,BU,IH)	
Name	**File content**
D#E#O#C	Fuel composition during irradiation period
D#E#O#D	Fuel composition during refuelling time or decay time
D#E#O#N	Neutron spectra at each NDA operated
D#E#O#G	Gamma at each NDA operated

NE=NE(IE,BU,FPD)	
Name	**File content**
P1E1O#C	Fuel composition during irradiation period; IE1; fuel pellet density equal to 93% of theoretical
P1E1O#D	Fuel composition during refuelling time or decay time; IE1; fuel pellet density equal to 93% of theoretical
P1E1O#N	Neutron spectra at each NDA operated; IE1; fuel pellet density equal to 93% of theoretical
P1E1O#G	Gamma at each NDA operated; IE1; ; fuel pellet density equal to 93% of theoretical
P2E1O#C	Fuel composition during irradiation period; IE1; fuel pellet density equal to 97% of theoretical
P2E1O#D	Fuel composition during refuelling time or decay time; IE1; fuel pellet density equal to 97% of theoretical
P2E1O#N	Neutron spectra at each NDA operated; ; IE1; fuel pellet density equal to 97% of theoretical

P2E1O#G	Gamma at each NDA operated; IE1 fuel pellet density equal to 97% of theoretical
F1E5O#C	Fuel composition during irradiation period; IE5 fuel pellet density equal to 93% of theoretical
P1E5O#D	Fuel composition during refuelling time or decay time; IE5; fuel pellet density equal to 93% of theoretical
P1E5O#N	Neutron spectra at each NDA operated; IE5; fuel pellet density equal to 93% of theoretical
P1E5O#G	Gamma at each NDA operated; IE5; fuel pellet density equal to 93% of theoretical
P2E5O#C	Fuel composition during irradiation period; IE5; fuel pellet density equal to 97% of theoretical
P2E5O#D	Fuel composition during refuelling time or decay time; IE5 fuel pellet density equal to 97% of theoretical
P2E5O#N	Neutron spectra at each NDA operated; IE5 fuel pellet density equal to 97% of theoretical
P2E1O#G	Gamma at each NDA operated; IE5; fuel pellet density equal to 97% of theoretical

Table F-1: ORIGEN-S code simulation

Once all OPUS text-files have been elaborated and exported by MatLab with *Program_database* then *Program_plot* allows creating a lot of graph and results. This second program does not require any input but loads all automatically.

Moreover some function, such as for example *p_3,* have been created to reorganize SCOPUS outputs and *Program_database* outputs to be easily plotted and analysed.

Next all the created and used MatLab scripts have been resumed.

Typology:	Name:	Scope:
Program	Program_database	Import Scope's file (i.e. fuel composition, gamma and neutron spectra) , elaborate then export them to Excel and to texts-file for subsequent analysis
Program	Program_plot	Import texts-file created by Program_database and produce a series of plot of NE and BU in function of several factors
Function	p_3	Reorganize neutron spectra
Function	p_4	Reorganize gamma spectra
Function	p_7	Restructure composition's file of major neutron emission elements

Table F-2: List of MatLab programs

For more information about these programs and functions look at the correspondent subsequent chapter.

Besides once all file have been loaded and elaborated thanks to curve fitting MatLab toolbox the BU/NE and fission products/NE correlations have been determined by best fitting.

F.2 MatLab script

F.2.1 Program_database

F.2.1.1 Purpose of the "Program_database"

Once all ORGINEN-ARP simulations were done, all the SCOPUS modified text-files have to be loaded to MatLab. We are talking about a large amount of files, more or less one thousand which can require months to be imported manually in a database. To accelerate and atomise this repetitive operation a specific program has been created: *Program_database.*

Basically, this program in order:

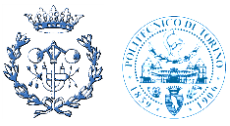
1. Asks which group of simulations you need to load
2. Waits for an input of your choice " 1 to 4" or exits typing "0", as a result for the script:

```
Display ('1-->Composition and NE(FCL,IE)')
display('2-->Composition and NE (Bu,IE)')
display('3-->Composition and NE (Bu,IH)')
display('4-->Composition and NE (Bu,FPD)')
display('0-->To close the program')
```

3. Goes to the correspondent for cycle according to the group simulation type
4. Asks which type of simulation you want to load
5. Waits for the simulation code
6. Loads the fuel composition text-file during irradiation
7. Loads the fuel composition text-file during refuelling operation
8. Orders the matrix rows which hold the fuel composition during irradiation time in a predefined scheme comparing string matrixes: one of the output and one user defined
9. Orders the matrix rows which hold the fuel composition during decay time in a predefined structure comparing string matrixes: one of the output and one user defined
10. Joins the two ordered composition matrix in a proper way
11. Loads gamma spectra
12. Loads neutron spectra
13. Selects the time of the simulation depend on the IE group
14. Plots gamma-spectra of different out-core measurement in the same figure
15. Plots neutron-spectra of different out-core measurement in the same figure
16. Elaborates gamma-spectra to prepare it to the database by function p5
17. Elaborates neutron-spectra to prepare it to the database by function p4
18. Integrates neutron –spectra over all energies to obtain the NE
19. Adds this value in a *sumN* matrix
20. Adds the fit time in the first row to the total composition matrix
21. Inserts value of some predefined element at each out-core measurement in *E#_Pu_Cs* matrix
22. Exports the composition matrix to Excel database generating a new worksheet
23. Exports elaborated gamma-spectra to Excel database
24. Exports elaborated neutron-spectra to Excel database
25. Exports *E#_Pu_Cs* matrix in a text file
26. Exports *sumN* matrix in a text file

Running Program_database once is sufficient to load a group of ten simulations of the same type. In fact the steps from four to twenty-three are performed ten times thanks to a cycle for; the other steps are done once. Moreover introducing the simulation code at step five and the typology of study at step two, all files related to that specific simulation are loaded immediately and the additionally “for cycles” performed. To clarify the complex idea you will be presented two examples.

Example 1: you want to load the E1O1 in function of the FCL. Type one at step two and type E1O1 at step five. Automatically it will be loaded: F1E1O1 and F2E1O1 and whole inherent files such as composition and spectra file. For cycle will be done twice.



Example 2: you want to load the E5O6 in function of the IH. Type one at step three and type E5O6 at step five. Automatically it will be loaded: D1E5O6, D2E5O6, D3E5O6 and D4E5O6 and whole inherent files such as composition and spectra file. For cycle will be done four times.

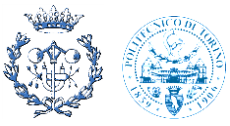
As a result to loading all the simulations in MatLab you need only thirteen runs of Program_database with a great saving of time.

In order to give an idea, the duration of a for cycle (4→ 23 steps) is around 15 seconds and the duration of a complete run is about 3 minutes. It turns out that the big effort was building the program which required several weeks and not loading the simulations. Future database extensions are not a problem and can be easily done with few changes only.

F.2.1.2 Script

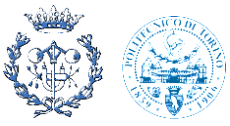
```
% Start

close all

clear all

clc

format long

%Choose the operation to perform

display('Choose the group of simulation to load and elaborate:')

display('1-->Composition and NE (FCL,IE)')

display('2-->Composition and NE (Bu,IE)')

display('3-->Composition and NE (Bu,IH)')

display('4-->Composition and NE (Bu,FPD)')

display('0-->To close the program')

performe=input('Input:');

if performe==0

    display('You have got out from the programme')

    break

end
```

```
 %Load files and join them
importfileText('RadionuclideList.txt')
Nuclide_list=RadionuclideList;
importfileText('nuclideML.txt')
importfileText('nuclideML_IH.txt')
[n,m]=size(nuclideML);
CC=1:3:30;
 %% Composition in functiomn of Fuel cycle time
if performe==1
    for ww=1:10
        clear dataCO
        clear dataDO
        clear dataC
        clear dataD
     filename=input('Input the case [E#O#] (without .txt):','s');
    filenameout2=('IE5FC');
    filename_out=strcat(filenameout2,'.xls');
    filename_tipoC_F1=strcat('F1',filename,'C.txt');
    filename_tipoC_F2=strcat('F2',filename,'C.txt');
    filename_tipoGF1=strcat('F1',filename,'G.txt');
    filename_tipoNF1=strcat('F1',filename,'N.txt');
    filename_tipoGF2=strcat('F2',filename,'G.txt');
    filename_tipoNF2=strcat('F2',filename,'N.txt');
    filenameGF1=strcat('F1',filename,'G');
    filenameNF1=strcat('F1',filename,'N');
    filenameGF2=strcat('F2',filename,'G');
```



```
filenameNF2=strcat('F2',filename,'N');
%Choose the time steps in base of the IE
Time_12=(37.2:37.2:(372*3));                    % Timesteps [days]
Time_12=[Time_12 Time_12(end)+1.5];             % Add decay time[days]
Time_18=(55.8:55.8:(18*31*3));                  % Timesteps [days]
Time_18=[Time_18 Time_18(end)+1.5];             %Add decay time[days]
 % Choose the correct time for the simulation
if str2num(filename(2))>3
       time=Time_18;
else
       time=Time_12;
end
% Reorganize the matrix of composition F1
importfileC(filename_tipoC_F1);
dataC_F1=data;
dataTC_F1=textdata;
[nn,mm]=size(dataTC_F1);
for i=1:60
    for j=1:60
        val=strcmp(dataTC_F1(j),nuclideML_IH(i));
        if val==1;
            dataCO_F1(i,:)=dataC_F1(j,:);
        end
    end
end
% Reorganize the matrix of composition F2
```

```
importfileC(filename_tipoC_F2);
dataC_F2=data;
dataTC_F2=textdata;
[nn,mm]=size(dataTC_F2);


for i=1:60
    for j=1:60
        val=strcmp(dataTC_F2(j),nuclideML_IH(i));
        if val==1;
            dataCO_F2(i,:)=dataC_F2(j,:);
        end
    end
end
% Reorganize decay matrix F1
clear textdata
clear data
filename_tipoD=strcat('F1',filename,'D.txt');
importfileD(filename_tipoD);
dataD_F1=data(:,end);
dataTD_F1=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTD_F1(j),nuclideML_IH(i));
        if val==1;
            dataDO_F1(i,:)=dataD_F1(j,:);
         end
```

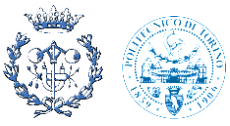


```
    end
end
% Reorganize decay matrix F2
clear textdata
clear data
filename_tipoD=strcat('F2',filename,'D.txt');
importfileD(filename_tipoD);
dataD_F2=data(:,end);
dataTD_F2=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTD_F2(j),nuclideML_IH(i));
        if val==1;
            dataDO_F2(i,:)=dataD_F2(j,:);
         end
    end
end
labelC_F1=[dataCO_F1(1:60,:) dataDO_F1(1:60,:)];
labelC_F2=[dataCO_F2(1:60,:) dataDO_F2(1:60,:)];
% Load and elaborate gamma and neutron spectra
clear data
fid1=fopen(filename_tipoGF1,'r');
GF1=fscanf(fid1,'%e%e',[2 inf]);
GF1=GF1';
fclose('all');
clear data
```

```
fid2=fopen(filename_tipoGF2,'r');
GF2=fscanf(fid2,'%e%e',[2 inf]);
GF2=GF2';
fclose('all');
[uu,ll]=size(GF1);
for i=1:uu
    if GF1(i,end)<1
       GF1(i,end)=1;
    end
    if GF2(i,2)<1
       GF2(i,end)=1;
    end
end
GR1_F1=GF1(1:82,:);
GR2_F1=GF1(83:164,:);
GR3_F1=GF1(165:end,:);
GR1_F2=GF2(1:82,:);
GR2_F2=GF2(83:164,:);
GR3_F2=GF2(165:end,:);
clear data
nomefileN=fopen(filename_tipoNF1,'r');
NF1=fscanf ( nomefileN , '%e %e' , [2 inf] );
NF1=NF1';
clear data
nomefileN=fopen(filename_tipoNF2,'r');
NF2=fscanf ( nomefileN , '%e %e' , [2 inf] );
```

```
    NF2=NF2';
    NR1_F1=NF1(1:88,:);
    NR2_F1=NF1(89:176,:);
    NR3_F1=NF1(177:264,:);
    NR1_F2=NF2(1:88,:);
    NR2_F2=NF2(89:176,:);
    NR3_F2=NF2(177:264,:);
%% Plots
    figure(1)
    set(gca,'fontsize',14)
    semilogy(NR1_F1(:,1),NR1_F1(:,2),'c','linewidth',2)
    hold on
    semilogy(NR2_F1(:,1),NR2_F1(:,2),'g','linewidth',2)
    semilogy(NR3_F1(:,1),NR3_F1(:,2),'r','linewidth',2)
    semilogy(NR1_F2(:,1),NR1_F2(:,2),'b-','linewidth',2)
    hold on
    semilogy(NR2_F2(:,1),NR2_F2(:,2),'y-','linewidth',2)
    semilogy(NR3_F2(:,1),NR3_F2(:,2),'m-','linewidth',2)
    xlim([0 10]);
    xlabel('Energy [MeV]')
    ylabel('Intensity [n/MeV/s]')
    legend('R1-F1','R2-F1','R3-F1','R1-F2','R2-F2','R3-F2' )
    title(strcat('Neutron Spectrum of simulation',filename,' changing FC));
    figure(2)
    set(gca,'fontsize',14)
    semilogy(GR1_F1(:,1),GR1_F1(:,2),'c','linewidth',2)
```

```
hold on
semilogy(GR2_F1(:,1),GR2_F1(:,2),'g','linewidth',2)
semilogy(GR3_F1(:,1),GR3_F1(:,2),'r','linewidth',2)
semilogy(GR1_F2(:,1),GR1_F2(:,2),'b','linewidth',2)
hold on
semilogy(GR2_F2(:,1),GR2_F2(:,2),'y','linewidth',2)
semilogy(GR3_F2(:,1),GR3_F2(:,2),'m','linewidth',2)
xlabel('Energy [MeV]')
ylabel('Intensity [photons/MeV/s]')
title(strcat('Gamma Spectrum of simulation',filename,'changing FCL'));
legend('R1-F1','R2-F1','R3-F1','R1-F2','R2-F2','R3-F2')
% Organize Neutron Spectrum Output
NOR1_F1=p3(NR1_F1);
NOR2_F1=p3(NR2_F1);
NOR3_F1=p3(NR3_F1);
NOR1_F2=p3(NR1_F2);
NOR2_F2=p3(NR2_F2);
NOR3_F2=p3(NR3_F2);
GOSR1_F1=p4(GR1_F1);
GOSR2_F1=p4(GR2_F1);
GOSR3_F1=p4(GR3_F1);
GOSR1_F2=p4(GR1_F2);
GOSR2_F2=p4(GR2_F2);
GOSR3_F2=p4(GR3_F2);
NNR1_F1=NOR1_F1(:,end);
NNR2_F1=NOR2_F1(:,end);
```

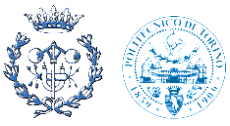


```
NNR3_F1=NOR3_F1(:,end);

NNR1_F2=NOR1_F2(:,end);

NNR2_F2=NOR2_F2(:,end);

NNR3_F2=NOR3_F2(:,end);

[jj,uu]=size(NOR1_F1);

[hh,tt]=size(GOSR1_F1);

 if ww==1

    compositionIE1_F1=zeros(60,10);

    neutron_IE1_F1=zeros(jj,uu*15);

    gamma_IE1_F1=zeros(hh,tt*15);

    compositionIE1_F2=zeros(60,10);

    neutron_IE1_F2=zeros(jj,uu*15);

    gamma_IE1_F2=zeros(hh,tt*15);

end

compositionIE1_F1(:,ww)=labelC_F1(1:60,end);

compositionIE1_F2(:,ww)=labelC_F2(1:60,end);

% Export content of some important elements

E1_Pu_Cs_F1(1:5,CC(ww):(CC(ww)+2))=

    [labelC_F1(5:9,10) labelC_F1(5:9,20)    labelC_F1(5:9,31)];

E1_Pu_Cs_F1(6:7,CC(ww):(CC(ww)+2))=

          [labelC_F1(55,10) labelC_F1(55,20) labelC_F1(55,31);

           labelC_F1(57,10) labelC_F1(57,20) labelC_F1(57,31)];

E1_Pu_Cs_F1(8:9,CC(ww):(CC(ww)+2))=

          [labelC_F1(2,10) labelC_F1(2,20) labelC_F1(2,31);

           labelC_F1(4,10) labelC_F1(4,20) labelC_F1(4,31)];

E1_Pu_Cs_F1(10:14,CC(ww):(CC(ww)+2))=
```

```
            [labelC_F1(11,10) labelC_F1(11,20) labelC_F1(11,31);
             labelC_F1(14,10) labelC_F1(14,20) labelC_F1(14,31);
             labelC_F1(16,10) labelC_F1(16,20) labelC_F1(16,31);
             labelC_F1(37,10) labelC_F1(37,20) labelC_F1(37,31);
             labelC_F1(59,10) labelC_F1(59,20) labelC_F1(59,31)];
E1_Pu_Cs_F2(1:5,CC(ww):(CC(ww)+2))=
           [labelC_F2(5:9,10) labelC_F2(5:9,20) labelC_F2(5:9,31)];
E1_Pu_Cs_F2(6:7,CC(ww):(CC(ww)+2))=
           [labelC_F2(55,10) labelC_F2(55,20)labelC_F2(55,31);
            labelC_F2(57,10) labelC_F2(57,20) labelC_F2(57,31)];
E1_Pu_Cs_F2(8:9,CC(ww):(CC(ww)+2))=
           [labelC_F2(2,10) labelC_F2(2,20) labelC_F2(2,31);
            labelC_F2(4,10) labelC_F2(4,20) labelC_F2(4,31)];
E1_Pu_Cs_F2(10:14,CC(ww):(CC(ww)+2))=
           [labelC_F2(11,10) labelC_F2(11,20) labelC_F2(11,31);
            labelC_F2(14,10) labelC_F2(14,20) labelC_F2(14,31);
            labelC_F2(16,10) labelC_F2(16,20) labelC_F2(16,31);
            labelC_F2(37,10) labelC_F2(37,20) labelC_F2(37,31);
            labelC_F2(59,10) labelC_F2(59,20) labelC_F2(59,31)];
labelC_F1=[time;labelC_F1(1:60,:)];
labelC_F2=[time;labelC_F2(1:60,:)];
neutron_IE1_F1(:,CC(ww))=NOR1_F1(:,end);
neutron_IE1_F1(:,CC(ww)+1)=NOR2_F1(:,end);
neutron_IE1_F1(:,CC(ww)+2)=NOR3_F1(:,end);
neutron_IE1_F2(:,CC(ww))=NOR1_F2(:,end);
neutron_IE1_F2(:,CC(ww)+1)=NOR2_F2(:,end);
```

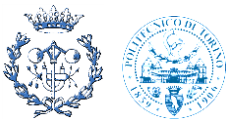
```
neutron_IE1_F2(:,CC(ww)+2)=NOR3_F2(:,end);

gamma_IE1_F1(:,CC(ww))=GOSR1_F1(:,end);

gamma_IE1_F1(:,CC(ww)+1)=GOSR2_F1(:,end);

gamma_IE1_F1(:,CC(ww)+2)=GOSR3_F1(:,end);

gamma_IE1_F2(:,CC(ww))=GOSR1_F2(:,end);

gamma_IE1_F2(:,CC(ww)+1)=GOSR2_F2(:,end);

gamma_IE1_F2(:,CC(ww)+2)=GOSR3_F2(:,end);

sumN_F1(1,ww)=sum(NNR1_F1);

sumN_F1(2,ww)=sum(NNR2_F1);

sumN_F1(3,ww)=sum(NNR3_F1);

sumN_F2(1,ww)=sum(NNR1_F2);

sumN_F2(2,ww)=sum(NNR2_F2);

sumN_F2(3,ww)=sum(NNR3_F2);

[uu,ll]=size(GOSR1_F1);

for i=1:uu

    if GOSR1_F1(i,end)==1

       GOSR1_F1(i,end)=0;

    end

    if GOSR2_F1(i,end)==1

       GOSR2_F1(i,end)=0;

    end

    if GOSR3_F1(i,end)==1

       GOSR3_F1(i,end)=0;

    end

    if GOSR1_F2(i,end)==1

       GOSR1_F2(i,end)=0;
```

```
        end
        if GOSR2_F2(i,end)==1
            GOSR2_F2(i,end)=0;
        end
        if GOSR3_F2(i,end)==1
            GOSR3_F2(i,end)=0;
        end
    end
    NR=[NOR1_F1 NOR2_F1(:,end) NOR3_F1(:,end)...
        NOR1_F2(:,end) NOR2_F2(:,end) NOR3_F2(:,end) ];
    GR=[GOSR1_F1 GOSR2_F1(:,end) GOSR3_F1(:,end)...
        GOSR1_F2(:,end) GOSR2_F2(:,end) GOSR3_F2(:,end) ];
    % Export neutron and gamma spectrum
    xlswrite(filename_out,GR,filename,'AJ6');
    xlswrite(filename_out,NR,filename,'AU6');
    % Case F1:export fuel composition
   xlswrite(filename_out,labelC_F1,filename,'C2')
    xlswrite(filename_out,Nuclide_list,filename,'B2')
    % Case F2:export fuel composition
    xlswrite(filename_out,labelC_F2,filename,'C68')
    xlswrite(filename_out,Nuclide_list,filename,'B68')
   if ww==10
        save('NE_FC_F1E5.txt','sumN_F1','-ascii')
        save('compositionIE5_F1.txt','compositionIE1_F1','-ascii')
        save('neutron_IE5F1.txt','neutron_IE1_F1','-ascii')
        save('gamma_IE5F1.txt','gamma_IE1_F1','-ascii')
```



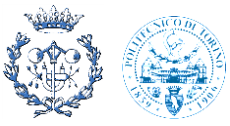
```
        save('E5_Pu_Cs_F1.txt','E1_Pu_Cs_F1','-ascii')
        save('NE_FC_F2E5.txt','sumN_F2','-ascii')
        save('compositionIE5F2.txt','compositionIE1_F2','-ascii')
        save('neutron_IE5F2.txt','neutron_IE1_F2','-ascii')
        save('gamma_IE5F2.txt','gamma_IE1_F2','-ascii')
        save('E5_Pu_Cs_F2.txt','E1_Pu_Cs_F2','-ascii')
    end
    end
end
%% Composition in function of BU and IE
if performe==2
    for ww=1:10
        clear dataCO
        clear dataDO
        clear dataC
        clear dataD
    filename=input('Input the case [E#O#] (without .txt):','s');
    filenameout2=('IE2BU');
    filename_out=strcat(filenameout2,'.xls');
    filename_tipoC=strcat(filename,'C.txt');
    importfileC(filename_tipoC);
    dataC=data;
    dataTC=textdata;
    [nn,mm]=size(dataTC);
% Choose the time steps in base of the IE
    Time_12=(37.2:37.2:(372*3));                % Timesteps [days]
```

```
Time_12=[Time_12 Time_12(end)+1.5];          % Add decay time[days]
Time_18=(55.8:55.8:(18*31*3));               % Timesteps [days]
Time_18=[Time_18 Time_18(end)+1.5];          % Add decay time[days]
if str2num(filename(2))>3
       time=Time_18;
else
       time=Time_12;
end
% Reorganize the matrix of composition
for i=1:59
    for j=1:59
        val=strcmp(dataTC(j),nuclideML(i));
        if val==1;
            dataCO(i,:)=dataC(j,:);
        end
    end
end
dataCO=[dataCO;dataC(end,:)];
clear textdata
clear data
filename_tipoD=strcat(filename,'D.txt');
importfileD(filename_tipoD);
dataD=data(:,end);
dataTD=textdata;
for i=1:60
    for j=1:60
```



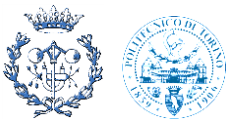
```
            val=strcmp(dataTD(j),nuclideML_IH(i));
            if val==1;
                dataDO(i,:)=dataD(j,:);
             end
        end
    end
    labelC=[dataCO dataDO];
    filename_tipoGS1=strcat(filename,'G1.txt');
    filename_tipoGS2=strcat(filename,'G2.txt');
    filename_tipoGS3=strcat(filename,'G3.txt');
    filename_tipoN=strcat(filename,'N.txt');
    filenameG=strcat(filename,'G');
    filenameN=strcat(filename,'N');
    clear data
    fid1=fopen(filename_tipoGS1,'r');
    GS1=fscanf(fid1,'%e%e',[2 inf]);
    GS1=GS1';
    fclose('all');
    clear data
    fid2=fopen(filename_tipoGS2,'r');
    GS2=fscanf(fid2,'%e%e',[2 inf]);
    GS2=GS2';
    fclose('all');
    fid3=fopen(filename_tipoGS3,'r');
    GS3=fscanf(fid3,'%e%e',[2,inf]);
    GS3=GS3';
```

```
fclose('all');
GSR1=[GS3(1:88,:) ; GS2(1:118,:) ; GS1(1:88,:)];
GSR2=[GS3(89:176,:) ; GS2(119:236,:) ; GS1(89:176,:)];
GSR3=[GS3(177:end,:) ; GS2(237:end,:) ; GS1(177:end,:)];
[uu,ll]=size(GSR1);
 for i=1:uu
    if GSR1(i,end)<1
       GSR1(i,end)=1;
    end
    if GSR2(i,2)<1
       GSR2(i,end)=1;
    end
    if GSR3(i,2)<1
       GSR3(i,end)=1;
    end
end
clear data
nomefileN=fopen(filename_tipoN,'r');
N= fscanf ( nomefileN , '%e %e' , [2 inf] );
N=N';
NR1=N(1:88,:);
NR2=N(89:176,:);
NR3=N(177:264,:);
figure(1)
set(gca,'fontsize',14)
semilogy(NR1(:,1),NR1(:,2),'c','linewidth',2)
```



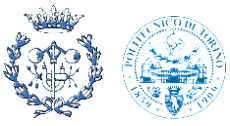
```
hold on
semilogy(NR2(:,1),NR2(:,2),'g','linewidth',2)
semilogy(NR3(:,1),NR3(:,2),'r','linewidth',2)
xlabel('Energy [MeV]')
ylabel('Intensity [n/MeV/s]')
title(strcat('Neutron Spectrum of simulation-   ',filename));
legend('R1','R2','R3','location','northeast')
figure(2)
set(gca,'fontsize',14)
semilogy(GSR1(:,1),GSR1(:,2),'c','linewidth',2)
hold on
semilogy(GSR2(:,1),GSR2(:,2),'g','linewidth',2)
semilogy(GSR3(:,1),GSR3(:,2),'r','linewidth',2)
xlabel('Energy [MeV]')
ylabel('Intensity [photons/MeV/s]')
title(strcat('Gamma Spectrum of simulation-   ',filename));
legend('R1','R2','R3','location','southwest')
figure(3)
set(gca,'fontsize',14)
semilogy(GSR1(:,1),GSR1(:,2),'c','linewidth',2)
hold on
semilogy(GSR2(:,1),GSR2(:,2),'g','linewidth',2)
semilogy(GSR3(:,1),GSR3(:,2),'r','linewidth',2)
axis([0.2 0.832 10^12 10^20])
xlabel('Energy [MeV]')
ylabel('Intensity [photons/MeV/s]')
```

```
title(strcat('Gamma Spectrum of simulation zoom-   ',filename));
legend('R1','R2','R3','location','southwest')
% Organize Neutron Spectrum Output
NOR1=p3(NR1);
NOR2=p3(NR2);
NOR3=p3(NR3);
GOSR1=p4(GSR1);
GOSR2=p4(GSR2);
GOSR3=p4(GSR3);
NNR1=NOR1(:,end);
NNR2=NOR2(:,end);
NNR3=NOR3(:,end);
[jj,uu]=size(NOR1);
[hh,tt]=size(GOSR1);
 if ww==1
    compositionIE1=zeros(60,10);
    neutron_IE1=zeros(jj,uu*15);
    gamma_IE1=zeros(hh,tt*15);
end
 compositionIE1(:,ww)=labelC(1:60,end);
% Export content of some important elements
E1_Pu_Cs(1:5,CC(ww):(CC(ww)+2))=
      [labelC(5:9,10) labelC(5:9,20) labelC(5:9,31)];
E1_Pu_Cs(6:7,CC(ww):(CC(ww)+2))=
      [labelC(55,10) labelC(55,20) labelC(55,31);
       labelC(57,10) labelC(57,20) labelC(57,31)];
```



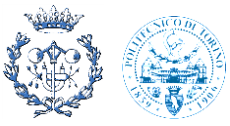
```
E1_Pu_Cs(8:9,CC(ww):(CC(ww)+2))=
        [labelC(2,10) labelC(2,20) labelC(2,31);
         labelC(4,10) labelC(4,20) labelC(4,31)];
E1_Pu_Cs(10:14,CC(ww):(CC(ww)+2))=
         [labelC(11,10) labelC(11,20) labelC(11,31);
labelC(14,10) labelC(14,20) labelC(14,31);
labelC(16,10) labelC(16,20) labelC(16,31);
labelC(37,10) labelC(37,20) labelC(37,31);
labelC(59,10) labelC(59,20) labelC(59,31)];
labelC=[time;labelC(1:60,:)];
neutron_IE1(:,CC(ww))=NOR1(:,end);
neutron_IE1(:,CC(ww)+1)=NOR2(:,end);
neutron_IE1(:,CC(ww)+2)=NOR3(:,end);
gamma_IE1(:,CC(ww))=GOSR1(:,end);
gamma_IE1(:,CC(ww)+1)=GOSR2(:,end);
gamma_IE1(:,CC(ww)+2)=GOSR3(:,end);
sumN(1,ww)=sum(NNR1);
sumN(2,ww)=sum(NNR2);
sumN(3,ww)=sum(NNR3);
[uu,ll]=size(GOSR1);
for i=1:uu
    if GOSR1(i,end)==1
       GOSR1(i,end)=0;
    end
    if GOSR2(i,end)==1
       GOSR2(i,end)=0;
```

```
        end
        if GOSR3(i,end)==1
            GOSR3(i,end)=0;
        end
    end

    %To export parameters to Exel
    GR=[GOSR1(1:49,1:end) GOSR2(1:49,end) GOSR3(1:49,end) ...
        GOSR1(50:98,1:end) GOSR2(50:98,end) GOSR3(50:98,end)...
        GOSR1(99:end,1:end) GOSR2(99:end,end) GOSR3(99:end,end)];
    xlswrite(filename_out,labelC,filename,'C2') e
    xlswrite(filename_out,Nuclide_list,filename,'B2')
    xlswrite(filename_out,GR,filename,'AJ3');
    xlswrite(filename_out,NOR1(1:22,:),filename,'BC3')
    xlswrite(filename_out,NOR2(1:22,end),filename,'BG3')
    xlswrite(filename_out,NOR3(1:22,end),filename,'BH3')
    xlswrite(filename_out,NOR1(23:44,:),filename,'BI3')
    xlswrite(filename_out,NOR2(23:44,end),filename,'BM3')
    xlswrite(filename_out,NOR3(23:44,end),filename,'BN3')
    if ww==10
        save('NE_RC_E2.txt','sumN','-ascii')
        save('compositionIE2.txt','compositionIE1','-ascii')
        save('neutron_IE2.txt','neutron_IE1','-ascii')
        save('gamma_IE2.txt','gamma_IE1','-ascii')
        save('E2_Pu_Cs.txt','E1_Pu_Cs','-ascii')
    end
```



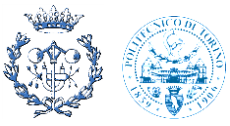
```
    end
end
%% Composition in function of BU and Irradiation History
if performe==3
    for ww=1:10
    clear dataCO_D1
    clear dataCO_D2
    clear dataCO_D3
    clear dataCO_D4
    clear dataDO_D1
    clear dataDO_D2
    clear dataDO_D3
    clear dataDO_D4
    filename=input('Input the case [E#O#] (without .txt):','s');
    filenameout2=('IE1IH');
    filename_out=strcat(filenameout2,'.xls');
    filename_tipoC_D1=strcat('D1',filename,'C.txt');
    filename_tipoC_D2=strcat('D2',filename,'C.txt');
    filename_tipoC_D3=strcat('D3',filename,'C.txt');
    filename_tipoC_D4=strcat('D4',filename,'C.txt');
    filename_tipoD_D1=strcat('D1',filename,'D.txt');
    filename_tipoD_D2=strcat('D2',filename,'D.txt');
    filename_tipoD_D3=strcat('D3',filename,'D.txt');
    filename_tipoD_D4=strcat('D4',filename,'D.txt');
    filename_tipoG_D1=strcat('D1',filename,'G.txt');
    filename_tipoG_D2=strcat('D2',filename,'G.txt');
```

```
filename_tipoG_D3=strcat('D3',filename,'G.txt');

filename_tipoG_D4=strcat('D4',filename,'G.txt');

filename_tipoN_D1=strcat('D1',filename,'N.txt');

filename_tipoN_D2=strcat('D2',filename,'N.txt');

filename_tipoN_D3=strcat('D3',filename,'N.txt');

filename_tipoN_D4=strcat('D4',filename,'N.txt');

filenameG=strcat(filename,'G');

filenameN=strcat(filename,'N');

% Time

% E1

T_E1I=[3.720E+01 7.440E+01 1.116E+02 1.488E+02 1.860E+02 2.232E+02...

       2.604E+02 2.976E+02 3.348E+02 3.720E+02 4.092E+02 4.464E+02...

       4.836E+02 5.208E+02 5.580E+02 5.952E+02 6.324E+02 6.696E+02...

       7.068E+02 7.440E+02 7.812E+02 8.184E+02 8.556E+02 8.928E+02...

       9.300E+02 9.672E+02 1.004E+03 1.042E+03 1.079E+03 1.116E+03];

T_E1D=[1.000E-01 3.000E-01 1.000E+00 1.500E+00 3.000E+00 1.000E+01...

       3.000E+01 1.000E+02 3.000E+02 4.460E+02];

T_E1DD=[1.000E-01 3.000E-01 1.000E+00 1.500E+00 3.000E+00 1.000E+01...

        3.000E+01 1.000E+02 3.000E+02 8.550E+02];

T_E1D1=[T_E1I(1:10) T_E1I(10)+T_E1D T_E1D(end)+T_E1I(11:end)];

T_E1D2=[T_E1I(1:10) T_E1I(10)+T_E1DD T_E1D(end)+T_E1I(11:end)];

T_E1D3=[T_E1I(1:20) T_E1I(20)+T_E1D T_E1D(end)+T_E1I(21:end)];

T_E1D4=[T_E1I(1:20) T_E1I(20)+T_E1DD T_E1D(end)+T_E1I(21:end)];

% E5

T_E5I=[5.580E+01 1.116E+02 1.674E+02 2.232E+02 2.790E+02 3.348E+02...

       3.906E+02 4.464E+02 5.022E+02 5.580E+02 6.138E+02 6.696E+02...
```



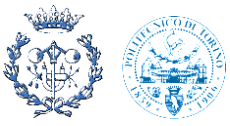
```
        7.254E+02 7.812E+02 8.370E+02 8.928E+02 9.486E+02 1.004E+03...
        1.060E+03 1.116E+03 1.172E+03 1.228E+03 1.283E+03 1.339E+03...
        1.395E+03 1.451E+03 1.507E+03 1.562E+03 1.618E+03 1.674E+03];
T_E5D=[1.000E-01 3.000E-01 1.000E+00 1.500E+00 3.000E+00 1.000E+01...
        3.000E+01 1.000E+02 3.000E+02 6.320E+02];
T_E5DD=[1.000E+00 1.500E+00 3.000E+00 1.000E+01 3.000E+01 1.000E+02...
         3.000E+02 1.000E+03 1.227E+03];
T_E5D1=[T_E5I(1:10) T_E5I(10)+T_E5D T_E5D(end)+T_E5I(11:end)];
T_E5D2=[T_E5I(1:10) T_E5I(10)+T_E5DD T_E5D(end)+T_E5I(11:end)];
T_E5D3=[T_E5I(1:20) T_E5I(20)+T_E5D T_E5D(end)+T_E5I(21:end)];
T_E5D4=[T_E5I(1:20) T_E5I(20)+T_E5DD T_E5D(end)+T_E5I(21:end)];
% Reorganize the matrix of fuel composition D1-irradiation
clear dataC
importfileC(filename_tipoC_D1)
dataC=data;
dataTC=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTC(j),nuclideML_IH(i));
        if val==1;
            dataCO_D1(i,:)=dataC(j,:);
        end
    end
end
% Reorganize the matrix of fuel composition D2-irradiation
clear dataC
```

```
importfileC(filename_tipoC_D2)
dataC=data;
dataTC=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTC(j),nuclideML_IH(i));
        if val==1;
            dataCO_D2(i,:)=dataC(j,:);
        end
    end
end
% Reorganize the matrix of fuel composition D3-irradiation
clear dataC
importfileC(filename_tipoC_D3)
dataC=data;
dataTC=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTC(j),nuclideML_IH(i));
        if val==1;
            dataCO_D3(i,:)=dataC(j,:);
        end
    end
end
% Reorganize the matrix of fuel composition D4-irradiation
clear dataC
```



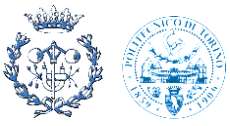
```
importfileC(filename_tipoC_D4)
dataC=data;
dataTC=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTC(j),nuclideML_IH(i));
        if val==1;
            dataCO_D4(i,:)=dataC(j,:);
        end
    end
end
% Reorganize the matrix of fuel composition D1-decay
clear textdata
clear data
importfileD(filename_tipoD_D1)
dataD_D1=data;
dataTD_D1=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTD_D1(j),nuclideML_IH(i));
        if val==1;
            dataDO_D1(i,:)=dataD_D1(j,:);
         end
    end
end
% Reorganize the matrix of fuel composition D2-decay
```

```
clear textdata
clear data
importfileD(filename_tipoD_D2)
dataD_D2=data;
dataTD_D2=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTD_D2(j),nuclideML_IH(i));
        if val==1;
            dataDO_D2(i,:)=dataD_D2(j,:);
         end
    end
end
% Reorganize the matrix of fuel composition D3-decay
clear textdata
clear data
importfileD(filename_tipoD_D3)
dataD_D3=data;
dataTD_D3=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTD_D3(j),nuclideML_IH(i));
        if val==1;
            dataDO_D3(i,:)=dataD_D3(j,:);
         end
    end
```



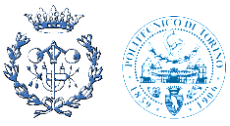
```
end
% Reorganize the matrix of fuel composition D4-decay
clear textdata
clear data
importfileD(filename_tipoD_D4)
dataD_D4=data;
dataTD_D4=textdata;
for i=1:60
    for j=1:60
        val=strcmp(dataTD_D4(j),nuclideML_IH(i));
        if val==1;
            dataDO_D4(i,:)=dataD_D4(j,:);
         end
    end
end
% Composing decay and irradiation fuel compositions
Composition_D1=[dataCO_D1(:,1:10) dataDO_D1 dataCO_D1(:,11:end)];
Composition_D2=[dataCO_D2(:,1:10) dataDO_D2 dataCO_D2(:,11:end)];
Composition_D3=[dataCO_D3(:,1:20) dataDO_D3 dataCO_D3(:,21:end)];
Composition_D4=[dataCO_D4(:,1:20) dataDO_D4 dataCO_D4(:,21:end)];
% Load and elaborate gamma spectra
clear data
fid1=fopen(filename_tipoG_D1,'r');
G_1=fscanf(fid1,'%e%e',[2 inf]);
G_1=G_1';
fclose('all');
```

```
clear data

fid2=fopen(filename_tipoG_D2,'r');

G_2=fscanf(fid2,'%e%e',[2 inf]);

G_2=G_2';

fclose('all');

clear data

fid3=fopen(filename_tipoG_D3,'r');

G_3=fscanf(fid3,'%e%e',[2,inf]);

G_3=G_3';

fclose('all');

clear data

fid4=fopen(filename_tipoG_D4,'r');

G_4=fscanf(fid4,'%e%e',[2,inf]);

G_4=G_4';

fclose('all');

G_D1R1=G_1(1:82,:);

G_D1R2=G_1(83:164,:);

G_D1R3=G_1(165:end,:);

G_D2R1=G_2(1:82,:);

G_D2R2=G_2(83:164,:);

G_D2R3=G_2(165:end,:);

G_D3R1=G_3(1:82,:);

G_D3R2=G_3(83:164,:);

G_D3R3=G_3(165:end,:);

G_D4R1=G_4(1:82,:);

G_D4R2=G_4(83:164,:);
```



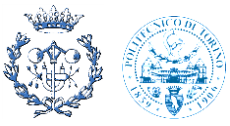
```
G_D4R3=G_4(165:end,:);

[uu,dd]=size(G_D1R1);

% Modify gamma spectra for plot

for i=1:uu

    if G_D1R1(i,end)==0

       G_D1R1(i,end)=1;

    end

    if G_D1R2(i,end)==0

       G_D1R2(i,end)=1;

    end

    if G_D1R3(i,end)==0

       G_D1R3(i,end)=1;

    end

    if G_D2R1(i,end)==0

       G_D2R1(i,end)=1;

    end

    if G_D2R2(i,end)==0

       G_D2R2(i,end)=1;

    end

    if G_D2R3(i,end)==0

       G_D2R3(i,end)=1;

    end

    if G_D3R1(i,end)==0

       G_D3R1(i,end)=1;

    end

    if G_D3R2(i,end)==0
```

```
        G_D3R2(i,end)=1;
    end
    if G_D3R3(i,end)==0
        G_D3R3(i,end)=1;
    end
    if G_D4R1(i,end)==0
        G_D4R1(i,end)=1;
    end
    if G_D4R2(i,end)==0
        G_D4R2(i,end)=1;
    end
    if G_D4R3(i,end)==0
        G_D4R3(i,end)=1;
    end
end
% Load and elaborate neutron spectra
clear data
fidN_1=fopen(filename_tipoN_D1,'r');
N_1= fscanf (fidN_1, '%e %e' , [2 inf] );
N_1=N_1';
clear data
fidN_2=fopen(filename_tipoN_D2,'r');
N_2= fscanf (fidN_2, '%e %e' , [2 inf] );
N_2=N_2';
clear data
fidN_3=fopen(filename_tipoN_D3,'r');
```



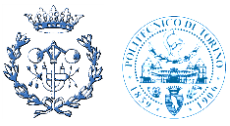
```
N_3= fscanf (fidN_3, '%e %e' , [2 inf] );

N_3=N_3';

clear data

fidN_4=fopen(filename_tipoN_D4,'r');

N_4= fscanf (fidN_4, '%e %e' , [2 inf] );

N_4=N_4';

N_D1R1=N_1(1:88,:);

N_D1R2=N_1(89:176,:);

N_D1R3=N_1(177:264,:);

N_D2R1=N_2(1:88,:);

N_D2R2=N_2(89:176,:);

N_D2R3=N_2(177:264,:);

N_D3R1=N_3(1:88,:);

N_D3R2=N_3(89:176,:);

N_D3R3=N_3(177:264,:);

N_D4R1=N_4(1:88,:);

N_D4R2=N_4(89:176,:);

N_D4R3=N_4(177:264,:);

% Organize Neutron Spectrum for database

NO_D1R1=p3(N_D1R1);

NO_D1R2=p3(N_D1R2);

NO_D1R3=p3(N_D1R3);

NO_D2R1=p3(N_D2R1);

NO_D2R2=p3(N_D2R2);

NO_D2R3=p3(N_D2R3);

NO_D3R1=p3(N_D3R1);
```

```
NO_D3R2=p3(N_D3R2);

NO_D3R3=p3(N_D3R3);

NO_D4R1=p3(N_D4R1);

NO_D4R2=p3(N_D4R2);

NO_D4R3=p3(N_D4R3);


% Organize gamma Spectrum for database

GO_D1R1=p4(G_D1R1);

GO_D1R2=p4(G_D1R2);

GO_D1R3=p4(G_D1R3);

GO_D2R1=p4(G_D2R1);

GO_D2R2=p4(G_D2R2);

GO_D2R3=p4(G_D2R3);

GO_D3R1=p4(G_D3R1);

GO_D3R2=p4(G_D3R2);

GO_D3R3=p4(G_D3R3);

GO_D4R1=p4(G_D4R1);

GO_D4R2=p4(G_D4R2);

GO_D4R3=p4(G_D4R3);

NN_D1R1=NO_D1R1(:,end);

NN_D1R2=NO_D1R2(:,end);

NN_D1R3=NO_D1R3(:,end);

NN_D2R1=NO_D2R1(:,end);

NN_D2R2=NO_D2R2(:,end);

NN_D2R3=NO_D2R3(:,end);

NN_D3R1=NO_D3R1(:,end);
```



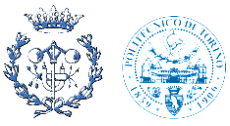
```
NN_D3R2=NO_D3R2(:,end);

NN_D3R3=NO_D3R3(:,end);

NN_D4R1=NO_D4R1(:,end);

NN_D4R2=NO_D4R2(:,end);

NN_D4R3=NO_D4R3(:,end);

[jj,uu]=size(NO_D1R1);

[hh,tt]=size(GO_D1R1);

if ww==1

    compositionIE1_D1=zeros(59,10);

    compositionIE1_D2=zeros(59,10);

    compositionIE1_D3=zeros(59,10);

    compositionIE1_D4=zeros(59,10);

    neutronIE1_D1=zeros(jj,uu*15);

    neutronIE1_D2=zeros(jj,uu*15);

    neutronIE1_D3=zeros(jj,uu*15);

    neutronIE1_D4=zeros(jj,uu*15);

    gammaIE1_D1=zeros(hh,tt*15);

    gammaIE1_D2=zeros(hh,tt*15);

    gammaIE1_D3=zeros(hh,tt*15);

    gammaIE1_D4=zeros(hh,tt*15);

    E1D1_Pu_Cs=zeros(14,30);

    E1D2_Pu_Cs=zeros(14,30);

    E1D3_Pu_Cs=zeros(14,30);

    E1D4_Pu_Cs=zeros(14,30);

end

compositionIE1_D1(:,ww)=Composition_D1(1:59,end);
```

```
    compositionIE1_D2(:,ww)=Composition_D2(1:59,end);
    compositionIE1_D3(:,ww)=Composition_D3(1:59,end);
    compositionIE1_D4(:,ww)=Composition_D4(1:59,end);
E1D1_Pu_Cs(1:5,CC(ww):(CC(ww)+2))=
  [Composition_D1(5:9,10) Composition_D1(5:9,30) Composition_D1(5:9,40)];
    E1D1_Pu_Cs(6:7,CC(ww):(CC(ww)+2))=...
    [Composition_D1(55,10) Composition_D1(55,30) Composition_D1(55,40);
     Composition_D1(57,10) Composition_D1(57,30) Composition_D1(57,40)];
    E1D1_Pu_Cs(8:9,CC(ww):(CC(ww)+2))=...
    [Composition_D1(2,10) Composition_D1(2,30) Composition_D1(2,40);
     Composition_D1(4,10) Composition_D1(4,30) Composition_D1(4,40)];
    E1D1_Pu_Cs(10:14,CC(ww):(CC(ww)+2))=...
       [Composition_D1(11,10) Composition_D1(11,30) Composition_D1(11,40);
        Composition_D1(14,10) Composition_D1(14,30) Composition_D1(14,40);
        Composition_D1(16,10) Composition_D1(16,30) Composition_D1(16,40);
        Composition_D1(37,10) Composition_D1(37,30) Composition_D1(37,40);
        Composition_D1(59,10)Composition_D1(59,30) Composition_D1(59,40)];

  E1D2_Pu_Cs(1:5,CC(ww):(CC(ww)+2))=
   [Composition_D2(5:9,10) Composition_D2(5:9,29) Composition_D2(5:9,39)];
   E1D2_Pu_Cs(6:7,CC(ww):(CC(ww)+2))=...
    [Composition_D2(55,10) Composition_D2(55,29) Composition_D2(55,39);
     Composition_D2(57,10) Composition_D2(57,29) Composition_D2(57,39)];
   E1D2_Pu_Cs(8:9,CC(ww):(CC(ww)+2))=...
    [Composition_D2(2,10) Composition_D2(2,29) Composition_D2(2,39);
     Composition_D2(4,10) Composition_D2(4,29) Composition_D2(4,39)];
```



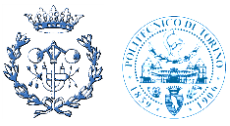
```
E1D2_Pu_Cs(10:14,CC(ww):(CC(ww)+2))=...
    [Composition_D2(11,10) Composition_D2(11,29) Composition_D2(11,39);
     Composition_D2(14,10) Composition_D2(14,29) Composition_D2(14,39);
     Composition_D2(16,10) Composition_D2(16,29) Composition_D2(16,39);
     Composition_D2(37,10) Composition_D2(37,29) Composition_D2(37,39);
     Composition_D2(59,10)Composition_D2(59,29) Composition_D2(59,39)];

E1D3_Pu_Cs(1:5,CC(ww):(CC(ww)+2))=
 [Composition_D3(5:9,10) Composition_D3(5:9,30) Composition_D3(5:9,40)];
 E1D3_Pu_Cs(6:7,CC(ww):(CC(ww)+2))=...
  [Composition_D3(55,10) Composition_D3(55,30) Composition_D3(55,40);
   Composition_D3(57,10) Composition_D3(57,30) Composition_D3(57,40)];
 E1D3_Pu_Cs(8:9,CC(ww):(CC(ww)+2))=...
  [Composition_D3(2,10) Composition_D3(2,30) Composition_D3(2,40);
   Composition_D3(4,10) Composition_D3(4,30) Composition_D3(4,40)];
 E1D3_Pu_Cs(10:14,CC(ww):(CC(ww)+2))=...
    [Composition_D3(11,10) Composition_D3(11,30) Composition_D3(11,40);
     Composition_D3(14,10) Composition_D3(14,30) Composition_D3(14,40);
     Composition_D3(16,10) Composition_D3(16,30) Composition_D3(16,40);
     Composition_D3(37,10) Composition_D3(37,30) Composition_D3(37,40);
     Composition_D3(59,10)Composition_D3(59,30) Composition_D3(59,40)];

 E1D4_Pu_Cs(1:5,CC(ww):(CC(ww)+2))=
 [Composition_D4(5:9,10) Composition_D4(5:9,29) Composition_D4(5:9,39)];
 E1D4_Pu_Cs(6:7,CC(ww):(CC(ww)+2))=...
  [Composition_D4(55,10) Composition_D4(55,29) Composition_D4(55,39);
```

```
  Composition_D4(57,10) Composition_D4(57,29) Composition_D4(57,39)];
E1D4_Pu_Cs(8:9,CC(ww):(CC(ww)+2))=...
 [Composition_D4(2,10) Composition_D4(2,29) Composition_D4(2,39);
  Composition_D4(4,10) Composition_D4(4,29) Composition_D4(4,39)];
E1D4_Pu_Cs(10:14,CC(ww):(CC(ww)+2))=...
    [Composition_D4(11,10) Composition_D4(11,29) Composition_D4(11,39);
     Composition_D4(14,10) Composition_D4(14,29) Composition_D4(14,39);
     Composition_D4(16,10) Composition_D4(16,29) Composition_D4(16,39);
     Composition_D4(37,10) Composition_D4(37,29) Composition_D4(37,39);
     Composition_D4(59,10)Composition_D4(59,29) Composition_D4(59,39)];
neutronIE1_D1(:,CC(ww))=NO_D1R1(:,end);
neutronIE1_D1(:,CC(ww)+1)=NO_D1R2(:,end);
neutronIE1_D1(:,CC(ww)+2)=NO_D1R3(:,end);
neutronIE1_D2(:,CC(ww))=NO_D2R1(:,end);
neutronIE1_D2(:,CC(ww)+1)=NO_D2R2(:,end);
neutronIE1_D2(:,CC(ww)+2)=NO_D2R3(:,end);
neutronIE1_D3(:,CC(ww))=NO_D3R1(:,end);
neutronIE1_D3(:,CC(ww)+1)=NO_D3R2(:,end);
neutronIE1_D3(:,CC(ww)+2)=NO_D3R3(:,end);
neutronIE1_D4(:,CC(ww))=NO_D4R1(:,end);
neutronIE1_D4(:,CC(ww)+1)=NO_D4R2(:,end);
neutronIE1_D4(:,CC(ww)+2)=NO_D4R3(:,end);
gammaIE1_D1(:,CC(ww))=GO_D1R1(:,end);
gammaIE1_D1(:,CC(ww)+1)=GO_D1R2(:,end);
gammaIE1_D1(:,CC(ww)+2)=GO_D1R3(:,end);
gammaIE1_D2(:,CC(ww))=GO_D2R1(:,end);
```



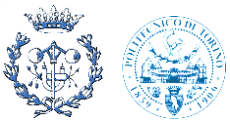
```
gammaIE1_D2(:,CC(ww)+1)=GO_D2R2(:,end);

gammaIE1_D2(:,CC(ww)+2)=GO_D2R3(:,end);

gammaIE1_D3(:,CC(ww))=GO_D3R1(:,end);

gammaIE1_D3(:,CC(ww)+1)=GO_D3R2(:,end);

gammaIE1_D3(:,CC(ww)+2)=GO_D3R3(:,end);

gammaIE1_D4(:,CC(ww))=GO_D4R1(:,end);

gammaIE1_D4(:,CC(ww)+1)=GO_D4R2(:,end);

gammaIE1_D4(:,CC(ww)+2)=GO_D4R3(:,end);

sumN_D1(1,ww)=sum(NN_D1R1);

sumN_D1(2,ww)=sum(NN_D1R2);

sumN_D1(3,ww)=sum(NN_D1R3);

sumN_D2(1,ww)=sum(NN_D2R1);

sumN_D2(2,ww)=sum(NN_D2R2);

sumN_D2(3,ww)=sum(NN_D2R3);

sumN_D3(1,ww)=sum(NN_D3R1);

sumN_D3(2,ww)=sum(NN_D3R2);

sumN_D3(3,ww)=sum(NN_D3R3);

sumN_D4(1,ww)=sum(NN_D4R1);

sumN_D4(2,ww)=sum(NN_D4R2);

sumN_D4(3,ww)=sum(NN_D4R3);

[uu,ll]=size(GO_D1R1);

% Convert to original value for database

for i=1:uu

    if GO_D1R1(i,end)==1

       GO_D1R1(i,end)=0;

    end
```

```
if GO_D1R2(i,end)==1
   GO_D1R2(i,end)=0;
end
if GO_D1R3(i,end)==1
   GO_D1R3(i,end)=0;
end
if GO_D2R1(i,end)==1
   GO_D2R1(i,end)=0;
end
if GO_D2R2(i,end)==1
   GO_D2R2(i,end)=0;
end
if GO_D2R3(i,end)==1
   GO_D2R3(i,end)=0;
end
if GO_D3R1(i,end)==1
   GO_D3R1(i,end)=0;
end
if GO_D3R2(i,end)==1
   GO_D3R2(i,end)=0;
end
if GO_D3R3(i,end)==1
   GO_D3R3(i,end)=0;
end
if GO_D4R1(i,end)==1
   GO_D4R1(i,end)=0;
```



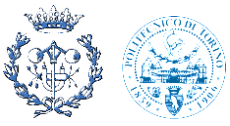
```
        end
        if GO_D4R2(i,end)==1
            GO_D4R2(i,end)=0;
        end
        if GO_D4R3(i,end)==1
            GO_D4R3(i,end)=0;
        end
    end
    %To export parameters
    G_R1=[GO_D1R1 GO_D2R1(:,end) GO_D3R1(:,end) GO_D4R1(:,end)];
    G_R2=[GO_D1R2 GO_D2R2(:,end) GO_D3R2(:,end) GO_D4R2(:,end)];
    G_R3=[GO_D1R3 GO_D2R3(:,end) GO_D3R3(:,end) GO_D4R3(:,end)];
    N_R1=[NO_D1R1 NO_D2R1(:,end) NO_D3R1(:,end) NO_D4R1(:,end)];
    N_R2=[NO_D1R2 NO_D2R2(:,end) NO_D3R2(:,end) NO_D4R2(:,end)];
    N_R3=[NO_D1R3 NO_D2R3(:,end) NO_D3R3(:,end) NO_D4R3(:,end)];
    % Plots
    figure(1)
    set(gca,'fontsize',14)
    semilogy(N_D1R1(:,1),N_D1R1(:,2),'k','linewidth',2)
    hold on
    semilogy(N_D2R1(:,1),N_D2R1(:,2),'g','linewidth',2)
    semilogy(N_D3R1(:,1),N_D3R1(:,2),'r','linewidth',2)
    semilogy(N_D4R1(:,1),N_D4R1(:,2),'c','linewidth',2)
    xlabel('Energy [MeV]')
    ylabel('Intensity [n/MeV/s]')
    title(strcat('Neutron Spectrum of simulation-    ',filename));
```

```
    legend('D1','D2','D3','D4','location','northeast')
    figure(2)
    set(gca,'fontsize',14)
    semilogy(G_D1R1(:,1),G_D1R1(:,2),'k','linewidth',2)
    hold on
    semilogy(G_D2R1(:,1),G_D2R1(:,2),'g','linewidth',2)
    semilogy(G_D3R1(:,1),G_D3R1(:,2),'r','linewidth',2)
    semilogy(G_D4R1(:,1),G_D4R1(:,2),'c','linewidth',2)
    xlabel('Energy [MeV]')
    ylabel('Intensity [photons/MeV/s]')
    title(strcat('Gamma Spectrum of simulation-   ',filename));
    legend('D1','D2','D3','D4','location','southwest')
if ww==10
    save('NE_RC_E1D1.txt','sumN_D1','-ascii')
    save('NE_RC_E1D2.txt','sumN_D2','-ascii')
    save('NE_RC_E1D3.txt','sumN_D3','-ascii')
    save('NE_RC_E1D4.txt','sumN_D4','-ascii')
    save('compositionIE1_D1.txt','compositionIE1_D1','-ascii')
    save('compositionIE1_D2.txt','compositionIE1_D2','-ascii')
    save('compositionIE1_D3.txt','compositionIE1_D3','-ascii')
    save('compositionIE1_D4.txt','compositionIE1_D4','-ascii')
    save('neutronIE1_D1.txt','neutronIE1_D1','-ascii')
    save('neutronIE1_D2.txt','neutronIE1_D2','-ascii')
    save('neutronIE1_D3.txt','neutronIE1_D3','-ascii')
    save('neutronIE1_D4.txt','neutronIE1_D4','-ascii')
    save('gammaIE1_D1.txt','gammaIE1_D1','-ascii')
```



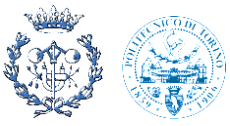
```
    save('gammaIE1_D2.txt','gammaIE1_D2','-ascii')
    save('gammaIE1_D3.txt','gammaIE1_D3','-ascii')
    save('gammaIE1_D4.txt','gammaIE1_D4','-ascii')
    save('E1D1_Pu_Cs.txt','E1D1_Pu_Cs','-ascii')
    save('E1D2_Pu_Cs.txt','E1D2_Pu_Cs','-ascii')
    save('E1D3_Pu_Cs.txt','E1D3_Pu_Cs','-ascii')
    save('E1D4_Pu_Cs.txt','E1D4_Pu_Cs','-ascii')
end
     Composition_D1=
    [T_E1D1;dataCO_D1(:,1:10)dataDO_D1 dataCO_D1(:,11:end)];
    Composition_D2=[T_E1D2;dataCO_D2(:,1:10)dataDO_D2 dataCO_D2(:,11:end)];
    Composition_D3=[T_E1D3;dataCO_D3(:,1:20)dataDO_D3 dataCO_D3(:,21:end)];
    Composition_D4=[T_E1D4;dataCO_D4(:,1:20)dataDO_D4 dataCO_D4(:,21:end)];
    % Export composition fuel along time in EXCEL
    xlswrite(filename_out,Composition_D1,filename,'C2')      % D1
    xlswrite(filename_out,Composition_D2,filename,'C66')     % D2
    xlswrite(filename_out,Composition_D3,filename,'C130')    % D3
    xlswrite(filename_out,Composition_D4,filename,'C194')    % D4
     xlswrite(filename_out,Nuclide_list,filename,'B2')        % D1
    xlswrite(filename_out,Nuclide_list,filename,'B66')       % D2
    xlswrite(filename_out,Nuclide_list,filename,'B130')      % D3
    xlswrite(filename_out,Nuclide_list,filename,'B194')      % D4
     % Export gamma spectrum in EXCEL
    xlswrite(filename_out,G_R1,filename,'AT3');             % R1
    xlswrite(filename_out,G_R2,filename,'BB3');             % R2
    xlswrite(filename_out,G_R3,filename,'BJ3');             % R3
```

```
     % Export neutron spectrum in EXCEL

    xlswrite(filename_out,N_R1,filename,'AT51');            % R1

    xlswrite(filename_out,N_R2,filename,'BB51');            % R2

    xlswrite(filename_out,N_R3,filename,'BJ51');            % R3

     end

end

%% Composition in function of fuel pellet density
% CHANGE NAME EXCEL, CHANGE FILE OUTPUT
if performe==4
    for ww=1:10
        clear dataCO
        clear dataDO
        clear dataC
        clear dataD
    filename=input('Input the case [E#O#] (without .txt):','s');
    filenameout2=('IE5P');
    filename_out=strcat(filenameout2,'.xls');
    filename_tipoC_F1=strcat('P1',filename,'C.txt');
    filename_tipoC_F2=strcat('P2',filename,'C.txt');

    filename_tipoGF1=strcat('P1',filename,'G.txt');
    filename_tipoNF1=strcat('P1',filename,'N.txt');

    filename_tipoGF2=strcat('P2',filename,'G.txt');
    filename_tipoNF2=strcat('P2',filename,'N.txt');

    filenameGF1=strcat('P1',filename,'G');
    filenameNF1=strcat('P1',filename,'N');

    filenameGF2=strcat('P2',filename,'G');
    filenameNF2=strcat('P2',filename,'N');

    %Choose the time steps in base of the IE
    Time_12=(37.2:37.2:(372*3)); % timesteps [days]
    Time_12=[Time_12 Time_12(end)+1.5]; %Add decay time[days]
    Time_18=(55.8:55.8:(18*31*3)); % timesteps [days]
    Time_18=[Time_18 Time_18(end)+1.5]; %Add decay time[days]

    if str2num(filename(2))>3
           time=Time_18;
    else
           time=Time_12;
    end

    % Orden the matrix of composition F1
    importfileC(filename_tipoC_F1);
    dataC_F1=data;
    dataTC_F1=textdata;
    [nn,mm]=size(dataTC_F1);
```



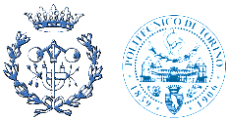
```
    for i=1:60
        for j=1:60
            val=strcmp(dataTC_F1(j),nuclideML_IH(i));
            if val==1;
                dataCO_F1(i,:)=dataC_F1(j,:);
            end
        end
    end
%     dataCO_F1=[dataCO_F1;dataC_F1(end,:)];

    % Orden the matrix of composition F2
    importfileC(filename_tipoC_F2);
    dataC_F2=data;
    dataTC_F2=textdata;
    [nn,mm]=size(dataTC_F2);

    for i=1:60
        for j=1:60
            val=strcmp(dataTC_F2(j),nuclideML_IH(i));
            if val==1;
                dataCO_F2(i,:)=dataC_F2(j,:);
            end
        end
    end
%     dataCO_F2=[dataCO_F2(1:59,:);dataC_F2(end,:)];

    % Orden decay matrix F1
    clear textdata
    clear data
    filename_tipoD=strcat('P1',filename,'D.txt');
    importfileD(filename_tipoD);

    dataD_F1=data(:,end);
    dataTD_F1=textdata;

    for i=1:60
        for j=1:60
            val=strcmp(dataTD_F1(j),nuclideML_IH(i));
            if val==1;
                dataDO_F1(i,:)=dataD_F1(j,:);
             end
        end
    end
%     dataDO_F1=[dataDO_F1;dataD_F1(end,:)];

    % Orden decay matrix F2
    clear textdata
    clear data
    filename_tipoD=strcat('P2',filename,'D.txt');
    importfileD(filename_tipoD);

    dataD_F2=data(:,end);
    dataTD_F2=textdata;

    for i=1:60
```

```
        for j=1:60
            val=strcmp(dataTD_F2(j),nuclideML_IH(i));
            if val==1;
                dataDO_F2(i,:)=dataD_F2(j,:);
             end
        end
    end
%     dataDO_F2=[dataDO_F2;dataD_F2(end,:)];

    labelC_F1=[dataCO_F1(1:60,:) dataDO_F1(1:60,:)];
    labelC_F2=[dataCO_F2(1:60,:) dataDO_F2(1:60,:)];

    % Load and elaborate gamma and neutron spectra
    clear data
    fid=fopen(filename_tipoGF1,'r');
    GF1=fscanf(fid,'%e %e',[2 inf]);
    GF1=GF1';
    fclose('all');

    clear data
    fid2=fopen(filename_tipoGF2,'r');
    GF2=fscanf(fid2,'%e %e',[2 inf]);
    GF2=GF2';
    fclose('all');

    [uu,ll]=size(GF1);

    for i=1:uu
        if GF1(i,end)<1
           GF1(i,end)=1;
        end
        if GF2(i,2)<1
           GF2(i,end)=1;
        end
    end

    GR1_F1=GF1(1:82,:);
    GR2_F1=GF1(83:164,:);
    GR3_F1=GF1(165:end,:);

    GR1_F2=GF2(1:82,:);
    GR2_F2=GF2(83:164,:);
    GR3_F2=GF2(165:end,:);

    clear data
    nomefileN=fopen(filename_tipoNF1,'r');
    NF1=fscanf ( nomefileN , '%e %e' , [2 inf] );
    NF1=NF1';
    fclose('all');

    if ww<10
        clear data
        nomefileN=fopen(filename_tipoNF2,'r');
        NF2=fscanf ( nomefileN , '%e %e' , [2 inf] );
        NF2=NF2';
        fclose('all');
```



```
    end
    if ww==10
        A=NF2;
    end
    NR1_F1=NF1(1:88,:);
    NR2_F1=NF1(89:176,:);
    NR3_F1=NF1(177:264,:);

    NR1_F2=NF2(1:88,:);
    NR2_F2=NF2(89:176,:);
    NR3_F2=NF2(177:264,:);

    % subplot(1,2,2)
    figure(1)
    set(gca,'fontsize',14)
    semilogy(NR1_F1(:,1),NR1_F1(:,2),'c','linewidth',2)
    hold on
    semilogy(NR2_F1(:,1),NR2_F1(:,2),'g','linewidth',2)
    semilogy(NR3_F1(:,1),NR3_F1(:,2),'r','linewidth',2)
    semilogy(NR1_F2(:,1),NR1_F2(:,2),'b-','linewidth',2)
    hold on
    semilogy(NR2_F2(:,1),NR2_F2(:,2),'y-','linewidth',2)
    semilogy(NR3_F2(:,1),NR3_F2(:,2),'m-','linewidth',2)

    xlim([0 10]);
    xlabel('Energy [MeV]')
    ylabel('Intensity [n/MeV/s]')

    legend('R1-P1','R2-P1','R3-P1','R1-P2','R2-P2','R3-
P2','location','northeast')
    title(strcat('Neutron Spectrum of simulation-   ',filename,' changing
fuel pellet density'));

    figure(2)
    set(gca,'fontsize',14)
    semilogy(GR1_F1(:,1),GR1_F1(:,2),'c','linewidth',2)
    hold on
    semilogy(GR2_F1(:,1),GR2_F1(:,2),'g','linewidth',2)
    semilogy(GR3_F1(:,1),GR3_F1(:,2),'r','linewidth',2)
    semilogy(GR1_F2(:,1),GR1_F2(:,2),'b','linewidth',2)
    hold on
    semilogy(GR2_F2(:,1),GR2_F2(:,2),'y','linewidth',2)
    semilogy(GR3_F2(:,1),GR3_F2(:,2),'m','linewidth',2)
    xlabel('Energy [MeV]')
    ylabel('Intensity [photons/MeV/s]')
    title(strcat('Gamma Spectrum of simulation-   ',filename,'changing
fuel pellet density'));
    Legend('R1-P1','R2-P1','R3-P1','R1-P2','R2-P2','R3-
P2','location','northeast')

    % Organize Neutron Spectrum Output
    NOR1_F1=p3(NR1_F1);
    NOR2_F1=p3(NR2_F1);
    NOR3_F1=p3(NR3_F1);
    NOR1_F2=p3(NR1_F2);
    NOR2_F2=p3(NR2_F2);
```

```
NOR3_F2=p3(NR3_F2);
GOSR1_F1=p4(GR1_F1);
GOSR2_F1=p4(GR2_F1);
GOSR3_F1=p4(GR3_F1);
GOSR1_F2=p4(GR1_F2);
GOSR2_F2=p4(GR2_F2);
GOSR3_F2=p4(GR3_F2);
NNR1_F1=NOR1_F1(:,end);
NNR2_F1=NOR2_F1(:,end);
NNR3_F1=NOR3_F1(:,end);
NNR1_F2=NOR1_F2(:,end);
NNR2_F2=NOR2_F2(:,end);
NNR3_F2=NOR3_F2(:,end);
[jj,uu]=size(NOR1_F1);
[hh,tt]=size(GOSR1_F1);

if ww==1
    compositionIE1_F1=zeros(60,10);
    neutron_IE1_F1=zeros(jj,uu*15);
    gamma_IE1_F1=zeros(hh,tt*15);
    compositionIE1_F2=zeros(60,10);
    neutron_IE1_F2=zeros(jj,uu*15);
    gamma_IE1_F2=zeros(hh,tt*15);
end
compositionIE1_F1(:,ww)=labelC_F1(1:60,end);
compositionIE1_F2(:,ww)=labelC_F2(1:60,end);

% Pu_all,U 235-238,Cs 134 137,Am 241,Cm 242 244,Ce 144
E1_Pu_Cs_F1(1:5,CC(ww):(CC(ww)+2))=
     [labelC_F1(5:9,10) labelC_F1(5:9,20) labelC_F1(5:9,31)];
E1_Pu_Cs_F1(6:7,CC(ww):(CC(ww)+2))=
     [labelC_F1(55,10) labelC_F1(55,20) labelC_F1(55,31);...
      labelC_F1(57,10) labelC_F1(57,20) labelC_F1(57,31)];
E1_Pu_Cs_F1(8:9,CC(ww):(CC(ww)+2))=
     [labelC_F1(2,10) labelC_F1(2,20) labelC_F1(2,31);
      labelC_F1(4,10) labelC_F1(4,20) labelC_F1(4,31)];
E1_Pu_Cs_F1(10:14,CC(ww):(CC(ww)+2))=
        [labelC_F1(11,10) labelC_F1(11,20) labelC_F1(11,31);
         labelC_F1(14,10) labelC_F1(14,20) labelC_F1(14,31);
         labelC_F1(16,10) labelC_F1(16,20) labelC_F1(16,31);
         labelC_F1(37,10) labelC_F1(37,20) labelC_F1(37,31);
         labelC_F1(59,10) labelC_F1(59,20) labelC_F1(59,31)];
E1_Pu_Cs_F2(1:5,CC(ww):(CC(ww)+2))=
        [labelC_F2(5:9,10) labelC_F2(5:9,20) labelC_F2(5:9,31)];
E1_Pu_Cs_F2(6:7,CC(ww):(CC(ww)+2))=
        [labelC_F2(55,10) labelC_F2(55,20) labelC_F2(55,31);...
         labelC_F2(57,10) labelC_F2(57,20) labelC_F2(57,31)];
E1_Pu_Cs_F2(8:9,CC(ww):(CC(ww)+2))=
        [labelC_F2(2,10) labelC_F2(2,20) labelC_F2(2,31);
         labelC_F2(4,10) labelC_F2(4,20) labelC_F2(4,31)];
E1_Pu_Cs_F2(10:14,CC(ww):(CC(ww)+2))=
        [labelC_F2(11,10) labelC_F2(11,20) labelC_F2(11,31);
         labelC_F2(14,10) labelC_F2(14,20) labelC_F2(14,31);
         labelC_F2(16,10) labelC_F2(16,20) labelC_F2(16,31);
         labelC_F2(37,10) labelC_F2(37,20) labelC_F2(37,31);
         labelC_F2(59,10) labelC_F2(59,20) labelC_F2(59,31)];
labelC_F1=[time;labelC_F1(1:60,:)];
```



```
    labelC_F2=[time;labelC_F2(1:60,:)];
    neutron_IE1_F1(:,CC(ww))=NOR1_F1(:,end);
    neutron_IE1_F1(:,CC(ww)+1)=NOR2_F1(:,end);
    neutron_IE1_F1(:,CC(ww)+2)=NOR3_F1(:,end);
    neutron_IE1_F2(:,CC(ww))=NOR1_F2(:,end);
    neutron_IE1_F2(:,CC(ww)+1)=NOR2_F2(:,end);
    neutron_IE1_F2(:,CC(ww)+2)=NOR3_F2(:,end);
    gamma_IE1_F1(:,CC(ww))=GOSR1_F1(:,end);
    gamma_IE1_F1(:,CC(ww)+1)=GOSR2_F1(:,end);
    gamma_IE1_F1(:,CC(ww)+2)=GOSR3_F1(:,end);
    gamma_IE1_F2(:,CC(ww))=GOSR1_F2(:,end);
    gamma_IE1_F2(:,CC(ww)+1)=GOSR2_F2(:,end);
    gamma_IE1_F2(:,CC(ww)+2)=GOSR3_F2(:,end);
    sumN_F1(1,ww)=sum(NNR1_F1);
    sumN_F1(2,ww)=sum(NNR2_F1);
    sumN_F1(3,ww)=sum(NNR3_F1);
    sumN_F2(1,ww)=sum(NNR1_F2);
    sumN_F2(2,ww)=sum(NNR2_F2);
    sumN_F2(3,ww)=sum(NNR3_F2);
    [uu,ll]=size(GOSR1_F1);

    for i=1:uu
        if GOSR1_F1(i,end)==1
           GOSR1_F1(i,end)=0;
        end
        if GOSR2_F1(i,end)==1
           GOSR2_F1(i,end)=0;
        end
        if GOSR3_F1(i,end)==1
           GOSR3_F1(i,end)=0;
        end
        if GOSR1_F2(i,end)==1
           GOSR1_F2(i,end)=0;
        end
        if GOSR2_F2(i,end)==1
           GOSR2_F2(i,end)=0;
        end
        if GOSR3_F2(i,end)==1
           GOSR3_F2(i,end)=0;
        end
    end

    %To export parameters
    NR=[NOR1_F1 NOR2_F1(:,end) NOR3_F1(:,end)...
        NOR1_F2(:,end) NOR2_F2(:,end) NOR3_F2(:,end) ];
    GR=[GOSR1_F1 GOSR2_F1(:,end) GOSR3_F1(:,end)...
        GOSR1_F2(:,end) GOSR2_F2(:,end) GOSR3_F2(:,end) ];
    xlswrite(filename_out,GR,filename,'AJ6');    % export gamma spectrum
    xlswrite(filename_out,NR,filename,'AU6');   % export neutron spectrum

    % Case F1
    xlswrite(filename_out,labelC_F1,filename,'C2') % export composition
fuel along time
    xlswrite(filename_out,Nuclide_list,filename,'B2')

    % Case F2
```

```
    xlswrite(filename_out,labelC_F2,filename,'C68') % export composition
fuel along time
    xlswrite(filename_out,Nuclide_list,filename,'B68')

    if ww==10
        save('NE_FPD_P1E5.txt','sumN_F1','-ascii')
        save('compositionIE5_P1.txt','compositionIE1_F1','-ascii')
        save('neutron_IE5P1.txt','neutron_IE1_F1','-ascii')
        save('gamma_IE5P1.txt','gamma_IE1_F1','-ascii')
        save('E5_Pu_Cs_P1.txt','E1_Pu_Cs_F1','-ascii')
        save('NE_FPD_P2E5.txt','sumN_F2','-ascii')
        save('compositionIE5_P2.txt','compositionIE1_F2','-ascii')
        save('neutron_IE5P2.txt','neutron_IE1_F2','-ascii')
        save('gamma_IE5P2.txt','gamma_IE1_F2','-ascii')
        save('E5_Pu_Cs_P2.txt','E1_Pu_Cs_F2','-ascii')
    end
    end
end
```

F.2.2 Program_plot

F.2.2.1 Purpose of the "Program_plot"

This program allows loading the text-file output previously created by Program_database and creating several graphs of:

- The NE correlations in function of several factors
- The gamma spectrometry-BU correlations
- The amount of some important elements in function of IH, BU, FCL and fuel pellet density

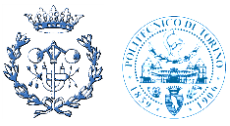
F.2.2.2 Script

The *Program_plot* script has not been reported here because it does not include particular cycles or specific methodologies. Anyway it is accessible and can be consulted in digital format.

F.2.3 Function: p3

F.2.3.1 Purpose of the function p3

As discussed before for each simulation, OPUS creates a text-file with a neutron spectra discretized by energy. The first column contains energies whereas the second one contains the neutron emitted for second and for MeV. Each line contains the same value of the precedent energy to plot a fragmented line. This neutron spectra format is utilized by *Program_database* to recreate neutron spectra plot as those generated by SCOPUS but in a better format.



The neutron spectra SCOPUS text-file format has been restructured in order to obtain a better and helpful setup for the Excel database. Thanks to function *p3* the new arrangement was created: now it contains four columns. *The first column* has the group's number: to the greatest energy range corresponds one while the other have been numerated in increasing way up to the lowest one like in neutronics. *The central two columns* enclose the extremity bounder energies of the interval whereas in the last one you can find the value of NE for that range of energy.

F.2.3.2 Script

```
function NO=p3(N)
[ll,mm]=size(N);
NO=zeros(44,3);         % pre-allocation
k=1;
for i=1:2:ll
    NO(k,1)=N(i,1);
    NO(k,2)=N(i+1,1);
    NO(k,3)=N(i,2);
    k=k+1;
end
n_group=(44:-1:1)';
NO=[n_group NO];
```

F.2.4 Function: p4

F.2.4.1 Purpose of the function p4

Similar to the neutron spectra text-file, the gamma one has to be reorganized to be exported to the Excel database. The new structure has been given by function *p4* . The format generated is the same of that prearranged by function *p3.*

F.2.4.2 Script

```
function GO=p4(G)
[ll,mm]=size(G);
GO=zeros(ll/2,3); % pre-allocation
k=1;
for i=1:2:ll
    GO(k,1)=G(i,1);
    GO(k,2)=G(i+1,1);
    GO(k,3)=G(i,2);
    k=k+1;
end
g_group=((ll/2):-1:1)';
GO=[g_group GO];
```

F.2.5 Function: p7

F.2.5.1 Purpose of function p7

Running *Program_database* a lot of parameters and matrix are exported to text-file to be loaded, elaborated and plotted by *Program_plot.*

One of this matrix is called *E#_Pu_Cs* when hat key needs to be substituted by the IE group number. It contains the mass in gram of the next elements at each out-core NDA measurement operated:

- Fissile elements: ^{235}U, ^{239}Pu and ^{241}Pu
- Fissionable elements: ^{241}Am, ^{238}Pu, ^{240}Pu and ^{242}Pu
- Mainly neutron emitter: ^{242}Cm, ^{244}Cm
- Helpful element for gamma/BU correlation: ^{154}Eu, ^{134}Cs, ^{137}Cs and ^{137}Ce

Following the scheme of the *E#_Pu_Cs* matrix is illustrated .

	E1O1			***E1O2***			**…**
	R1	***R2***	***R3***	***R1***	***R2***	***R3***	**…**
U 235							**…**
U 238							**…**
..	…	…	…	…	…	…	…

Table F-3: *E#_Pu_Cs* matrix scheme

when:

- E1O1 is the simulation code
- R1, R2 and R3 are the three recharge operation when the element's amount is calculated by ORIGEN-ARP
- The light blue area is the only one really memorized by MatLab

Essentially, the function *p7* allows reorganizing the previous matrix using a combination of two for cycles in the subsequent new format easier to be plotted.

	E1O1	E1O2	E1O3	…	E1O1	E1O2	…
	R1	R1	R1	…	R2	R2	…
U 235				…			…
U 238				…			…
..	…	…	…	…	…	…	…

Table F-4: p7 matrix function

F.2.5.2 Script

```
function ttt=p7(xxx)
[n,m]=size(xxx);
r=1:3:30;
y=1:10;
h=11:20;
k=21:30;
ttt=zeros(n,m); % pre-allocation
for j=1:n
    for i=1:length(r)
        ttt(j,y(i))=xxx(j,r(i));
        ttt(j,h(i))=xxx(j,r(i)+1);
        ttt(j,k(i))=xxx(j,r(i)+2);
    end
end
```

F.3 Database

As previously discussed, thanks to *Program_database* a large *Excel database* has been created and structured for future possible amplifications.

For each simulation the following items have been exported in the Excel database:

- The fuel composition development from the first irradiation to the last NDA out-core measurement
- The elaborated gamma-spectra discretized in energy
- The elaborated neutron-spectra discretized in energy

All the elaborated ORIGEN-ARP belonging to the same simulation data has been stored in the identical worksheet which has been automatically called with the correspondent basic simulation code. For instance the simulation F1E5O3 and F2E5O3 (*Annex F*) have been stored in the E1O3 worksheet. This operation allows finding all the ORGINEN-ARP information in the same place and not in different text-files as the SCOPUS output. In this way the analysis has been easier and quicker.

The design of each worksheet is shown in the following figure. If there are more than one simulations corresponding to the variation of the same case, the successive data are stored below the last one in the same worksheet.

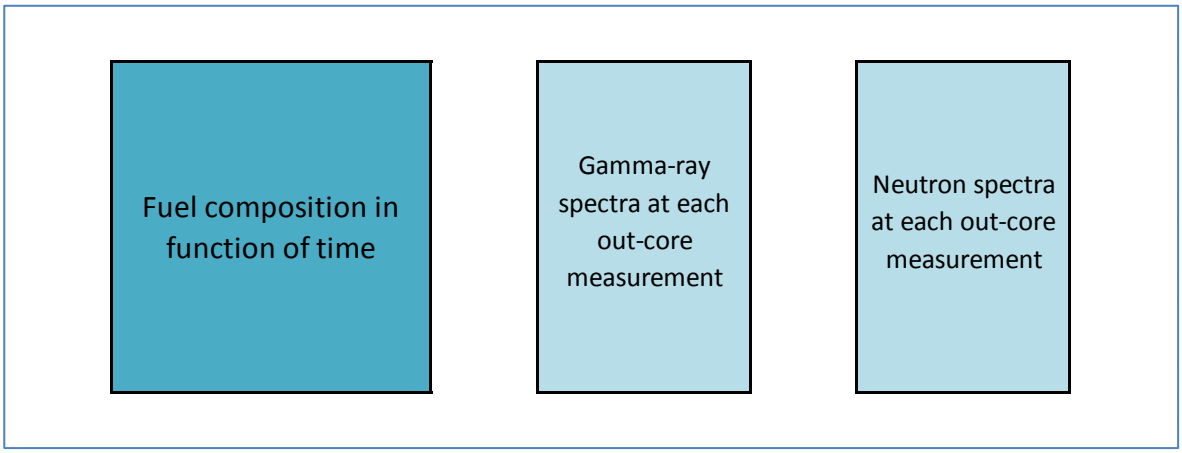


Figure F-1: Layout of Excel database

Moreover, importing the fuel composition using the *Program_database* allows having the predefined elements in the same order and not in a particular order as SCOPUS gives us. In fact the default isotopes hold in SCOPUS text-files are listed in the descending order of the integral of the logarithm of the mass in a random interval of time which ORIGEN-ARP automatically decided.

Not all worksheets have been stored in the same file but they have been divided by typology of sensitive parameters studied and IE. In the next label all the Excel file which compose the Excel database are listed.

Filename	*Content*
IE1BU	Simulation in function of BU, IE of group one
IE2BU	Simulation in function of BU, IE of group two
IE3BU	Simulation in function of BU, IE of group three
IE4BU	Simulation in function of BU, IE of group four
IE5BU	Simulation in function of BU, IE of group five
IE1FCL	Simulation in function of BU and FCL, IE of group one
IE5FCL	Simulation in function of BU and FCL, IE of group five
IE1IH	Simulation in function of BU and IH, IE of group one
IE5IH	Simulation in function of BU and IH, IE of group five
IE1PDP	Simulation in function of BU and FPDD, IE of group one
IE5PDP	**Simulation in function of BU and FPD, IE of group five**

Table F-5: List of file present in the Excel database

To conclude a final example to explain the database structure will be reported.

The simulation D1E1O3, D2E1O3, D3E1O3 and D4E1O3 have been stored in the E1O3 worksheet which belongs to IE1IH because that runs in order to study the sensitivities to the irradiation history or IH.

F.4 Best Fita

Once all experimental data have been obtained, software for the best fitting is fundamental. *MatLab curve fitting tool box* has been chosen to calculate the best fits of the ORIGEN-S data and the MCNP data due to its tested accuracy.

Curve Fitting Toolbox provides graphical tools and command-line functions for fitting curves and surfaces to data. The toolbox lets you perform exploratory data analysis, pre-process and post-process data, compare candidate models, and remove outliers. You can conduct regression analysis using the library of linear and nonlinear models provided or specify your own custom equations. The library provides optimized solver parameters and starting conditions to improve the quality of your fits. The toolbox also supports nonparametric modelling techniques, such as splines, interpolation, and smoothing.

After creating a fit, you can apply a variety of post-processing methods for plotting, interpolation, and extrapolation, estimating confidence intervals and calculating integrals and derivatives.

Once some typologies of different functions have been tried, you can use for the following parameters to select that BF equation, which represents in the best way the experimental one:

- The sum of squares due to error (SSE)
- R-square
- Adjusted R-square
- Root mean squared error (RMSE)

Next, all these parameters will be described and commented in their use.

Moreover you can display the residuals to decide which BF equation is the more suitable for you representation. For instance, in the following figure, the BF study of the ORIGEN-S data of NE in function of BU for IE equal to 4.53%, IE5, is reported. In the above plot you can appreciate the two BF functions tested, one exponential (blue) and one power (red) law, and the experimental data represented by light blue boxes. In the second plot you can note the residual values of the BF equations. Of course in the analysis the power relation has to be preferred to the exponential one.

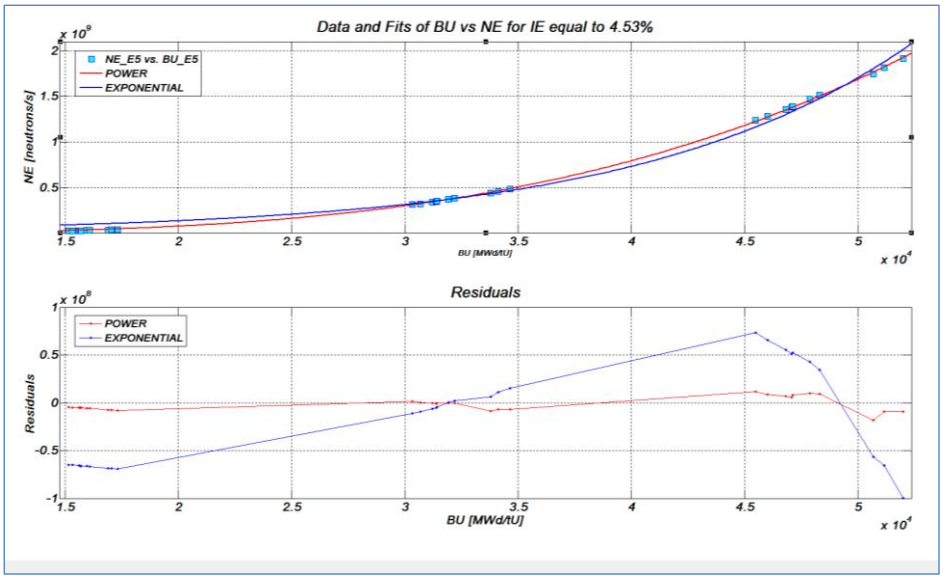


Figure F-2: Best fit residual error

F.4.1 Sum of Squares Due to Error

This statistic measures the total deviation of the response values from the fit to the response values. It is also called the summed square of residuals and is usually labelled as SSE.

$$SSE = \sum_{i=-1}^{n} w_i(y_i - \hat{y}_i)^2 \qquad Equ.F.1$$

A value closer to 0 indicates that the model has a smaller random error component, and that the fit will be more useful for prediction.

F.4.2 R-Square

This statistic measures how successful the fit is in explaining the variation of the data. Put another way, R-square is the square of the correlation between the response values and the predicted response values. It is also called the square of the multiple correlation coefficients and the coefficient of multiple determinations.

R-square is defined as the ratio of the sum of squares of the regression (SSR) and the total sum of squares (SST). SSR is defined as

$$SSR = \sum_{i=-1}^{n} w_i(y_i - \hat{y}_i)^2 \qquad Equ.F.2$$

SST is also called the sum of squares about the mean, and is defined as

$$SST = \sum_{i=-1}^{n} w_i(y_i - \overline{y})^2 \qquad Equ.F.3$$

where SST = SSR + SSE. Given these definitions, R-square is expressed as

$$R - square = \frac{SSR}{SST} = 1 - \frac{SSE}{SST} \qquad Equ.F.4$$

R-square can take on any value between 0 and 1, with a value closer to 1 indicating that a greater proportion of variance is accounted for by the model. For example, an R-square value of 0.8234 means that the fit explains 82.34% of the total variation in the data about the average.

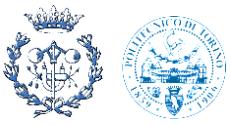
If you increase the number of fitted coefficients in your model, R-square will increase although the fit may not improve in a practical sense. To avoid this situation, you should use the degrees of freedom adjusted R-square statistic described below.

Note that it is possible to get a negative R-square for equations that do not contain a constant term. Because R-square is defined as the proportion of variance explained by the fit, if the fit is actually worse than just fitting a horizontal line then R-square is negative. In this case, R-square cannot be interpreted as the square of a correlation. Such situations indicate that a constant term should be added to the model.

F.4.3 Degrees of Freedom Adjusted R-Square

This statistic uses the R-square statistic defined above, and adjusts it based on the residual degrees of freedom. The residual degrees of freedom is defined as the number of response values n minus the number of fitted coefficients m estimated from the response values.

$$v = m - n \qquad Equ.F.5$$



v indicates the number of independent pieces of information involving the n data points that are required to calculate the sum of squares. Note that if parameters are bounded and one or more of the estimates are at their bounds, then those estimates are regarded as fixed. The degrees of freedom are increased by the number of such parameters.

$$Adjusted\ \ R-square = 1 - \frac{SSE\ (n-1)}{SST\ (v)} \qquad Equ.F.6$$

The adjusted R-square statistic is generally the best indicator of the fit quality when you compare two models that are nested — that is, a series of models each of which adds additional coefficients to the previous model.

The adjusted R-square statistic can take on any value less than or equal to 1, with a value closer to 1 indicating a better fit. Negative values can occur when the model contains terms that do not help to predict the response.

F.4.4 Root Mean Squared Error

This statistic is also known as the fit standard error and the standard error of the regression. It is an estimate of the standard deviation of the random component in the data, and is defined as

$$RMSE = s = \sqrt{MSE} \qquad Equ.F.7$$

where MSE is the mean square error or the residual mean square

$$MSE = \frac{SSE}{v} \qquad Equ.F.8$$

Just as with SSE, an MSE value closer to 0 indicates a fit that is more useful for prediction.

F.4.5 ORIGEN-S Best Fit

The best fit has not been applied to the whole experimental data whereas only to the main useful device's correlations. Basically it concerns:

- Parameter sensitiveness of BU: IH, FPD, IE, FCL
- Correlation between Plutonium isotopes and BU

- Correlation between ^{134}Cs /^{137} Cs mass ration and BU
- Correlation total fissile material and BU

The correlations indicated by IAEA were always inserted in the functions proved and always they were the best choice. In most of the table only two of the best fit functions have been reported even if much more has been tested. The yellow one highlighted is the equation choice thanks to its best characteristics.

Moreover, when the correlation takes into account more than one IE it one best fit function has been preferred than more ones if the goodness of the fit is more or less the same. This has been done to limit the number of simulations needed in the future out-core innovative device.

You have to remember that these are best fits computed by the ORIGEN-S which describe FA emissions. To compute the true best fit of the correlation between neutron detector counts and the BU a Monte Carlo code, in this case MCNP is required.

Finally, when an approximate equality of the goodness of two fits has been encountered, it has been preferred the one actuating the best fit for greater values of Y because it allows to reduce more the relative error.

F.4.5.1 RESULTS

A legend of the best fit model applied and the results has been reported as follows:

- *Power-1*

$$y = a * x^b \qquad Equ.F.8$$

- *Power-2*

$$y = a * x^b + c \qquad Equ.F.9$$

- *Poly-1*

$$y = a * x + b \qquad Equ.F.10$$

- *Poly-2*

$$y = a * x^2 + x + b \qquad Equ.F.11$$

- *Exp-1*

$$y = a * \exp^{x*b} \qquad Equ.F.12$$

- *Exp-2*

$$y = a * \exp^{x*b} + c * \exp^{x*d} \qquad Equ.F.13$$

F.4.5.1.1 BU-NE

		Model	a	b	SSE	R-square	Adjusted R-square	RMSE
X --> BU Y --> NE	*E1*	POWER-1	5,98E-06	3,166	1,38E+15	0,9991	9,99E-01	7,03E+06
		EXP-1	1,28E+07	0,0001447	2,07E+16	0,9781	9,87E-01	2,72E+07
	E2	POWER-1	5,51E-07	3,339	2,56E+15	0,9995	0,9995	9,56E+06
		EXP-1	2,04E+07	0,0001034	6,02E+16	0,9881	0,9876	4,71E+07
	E3	POWER-1	2,42E-07	3,401	3,25E+15	0,9997	0,9996	1,08E+07
		EXP-1	2,34E+07	9,44E-05	8,90E+16	9,90E-01	9,90E-01	5,64E+07
	E4	POWER-1	3,27E-07	3,356	1,03E+15	0,9999	0,9999	6,08E+06
		EXP-1	2,50E+07	8,81E-05	7,96E+16	9,93E-01	9,93E-01	5,33E+07
	E5	POWER-1	2,27E-07	3,378	1,59E+15	0,99999	0,99999	7,54E+06
		EXP-1	2,51E+07	8,44E-05	8,38E+06	9,93E-01	9,93E-01	5,47E+07

Table F-6: BU vs. IE

F.4.5.1.2 ^{238}Pu-NE

X->Pu238 Y->NE	Model		a	b	SSE	R-square	Adjusted R-square	RMSE
	E1	POWER-1	2,4340E+06	1,4170E+00	3,5540E+14	9,9980E-01	9,9980E-01	3,5630E+06
	E2	POWER-1	1,1560E+06	1,5040E+00	2,4880E+15	9,9950E-01	9,9950E-01	9,4260E+06
	E3	POWER-1	9,4380E+05	1,5260E+00	3,6860E+15	9,9960E-01	9,9960E-01	1,1470E+07
	E4	POWER-1	1,1140E+06	1,4480E+00	6,4140E+15	9,9940E-01	9,9940E-01	1,5130E+07
	E5	POWER-1	1,0280E+06	1,4470E+00	4,4860E+15	9,9960E-01	9,9960E-01	1,2660E+07

Table F-5:Plutonium-238 vs. NE

F.4.5.1.3 ^{239}Pu-NE

X --> Pu 239 Y--> NE	Model		a	b	c	d	SSE	R-square	Adjusted R-square	RMSE
	E1	EXP-2	7,102E+03	4,540E-03	3,522E-11	1,771E-02	2,359E+14	9,999E-01	9,998E-01	3,012E+06
		POWER-2	2,997E-83	2,664E+01	0,000E+00		2,002E+17	9,615E-01	9,587E-01	8,611E+07
	E2	EXP-2	1,038E+04	3,725E-03	2,193E-07	1,309E-02	1,056E+07	9,797E-01	9,774E-01	6,372E+07
		POWER-2	2,997E-83	2,664E+01	0,000E+00		2,002E+17	9,615E-01	9,587E-01	8,611E+07
	E3	EXP-2	1,201E+02	5,211E-03	9,153E-07	1,224E-02	3,322E+17	9,643E-01	9,602E-01	1,130E+08
		POWER-2	1,558E-89	2,837E+01	0,000E+00		3,873E+17	9,584E-01	9,553E-01	1,198E+08
	E4	EXP-1	5,149E-02	8,899E-03			4,739E+17	9,577E-01	9,562E-01	1,301E+08
		POWER-2	1,217E-80	2,566E+01	0,000E+00		4,782E+17	9,574E-01	9,542E-01	1,331E+08
	E5	EXP-1	9,031E-03	9,232E-03			3,825E+17	9,692E-01	9,681E-01	1,173E+08
		EXP-2	3,750E-03	8,706E-03	1,896E+00	1,444E-02	3,798E+17	9,696E-01	9,661E-01	1,209E+08

Table F-6:Plutonium-239 vs. NE

F.4.5.1.4 ^{240}Pu-NE

X--> Pu 240 Y--> NE						
Model	a	b	SSE	R-square	Adjusted R-square	RMSE
POWER-1	4,641E-04	4,001E+00	1,340E+17	9,969E-01	9,969E-01	3,001E+07
EXP-1	1,270E+07	3,568E-03	8,513E+16	9,980E-01	9,980E-01	2,398E+07

Table F-7:Plutonium-240 vs. NE

F.4.5.1.5 ^{241}Pu-NE

X->Pu241 Y->NE	Model		a	b	SSE	R-square	Adjusted R-square	RMSE
	E1	POWER-1	6,7960E-01	3,2440E+00	3,4590E+15	9,9790E-01	9,9780E-01	1,1110E+07
	E2	POWER-1	1,0130E-02	3,8570E+00	1,9030E+16	9,9630E-01	9,9620E-01	2,6070E+07
	E3	POWER-1	8,4700E-04	4,2180E+00	3,2180E+16	9,9660E-01	9,9650E-01	3,3420E+07
	E4	POWER-1	2,2770E-03	4,0590E+00	2,2350E+16	9,9800E-01	9,9790E-01	2,8250E+07
	E5	POWER-1	2,2430E-03	4,0480E+00	1,9310E+16	9,9850E-01	9,9840E-01	2,6260E+07

Table F-8:Plutonium-241 vs. NE

F.4.5.1.6 ^{242}Pu-NE

X-->Pu242 Y --> NE	Model		a	b	SSE	R-square	Adjusted R-square	RMSE
	E1	POWER-1	2,2180E+05	1,4510E+00	9,3710E+14	9,9940E-01	9,9940E-01	5,7850E+06
	E2	POWER-1	1,4390E+05	1,5400E+00	3,4120E+05	9,9930E-01	9,9930E-01	1,1040E+07
	E3	POWER-1	1,1000E+05	1,5910E+00	4,5610E+15	9,9950E-01	9,9950E-01	1,2760E+07
	E4	POWER-1	1,7280E+05	1,5270E+00	2,0490E+05	9,9980E-01	9,9980E-01	8,5540E+06
	E5	POWER-1	1,7560E+05	1,5280E+00	2,0990E+15	9,9980E-01	9,9980E-01	5,6580E+06

Table F-9:Plutonium-242 vs. NE

F.4.5.1.7 Fissile material-NE

X--> Fissile material Y --> NE	Model		a	b	c	SSE	R-square	Adjusted R-square	RMSE
	E1	POWER-2	6,2760E-03	-5,6320E+00	-4,7740E+07	1,5610E+15	9,9900E-01	9,9900E-01	7,6040E+06
	E2	POWER-2	5,5600E+01	-3,9410E+00	-1,0060E+08	5,9740E+15	9,9890E-01	9,9880E-01	1,4870E+07
	E3	POWER-2	2,2360E+02	-3,6990E+00	-1,2000E+08	6,7960E+15	9,9930E-01	9,9920E-01	1,5870E+07
	E4	POWER-2	2,6410E+00	-3,2290E+00	-1,7290E+08	9,9530E+15	9,9910E-01	9,9900E-01	1,9200E+07
	E5	POWER-2	3,3342E+04	3,2390E+00	-1,6700E+08	3,6890E+15	9,9970E-01	9,9970E-01	1,1690E+07

Table F-10:Total fissile material vs. NE

F.4.5.1.8 $^{134}Cs/^{137}Cs$ - BU

X--> BU Y --> Cs 134/ Cs 137		Model	a	b	c	SSE	R-square	Adjusted R-square	RMSE
	E1	POLY-2	-2,3970E-11	3,7640E+06	2,5330E-03	3,6800E+05	9,9680E-01	9,9650E-01	1,1670E-03
		POWER-2	1,4320E-04	6,3640E-01	-1,3800E-02	3,2620E+05	9,9710E-01	9,9690E-01	1,0990E-03
	E2	POLY-1	2,3480E-06	1,0590E-02		1,1850E-04	9,9290E-01	9,9260E-01	2,0570E-03
		POWER-2	7,7310E-05	6,8860E-01	-1,1670E-02	5,9520E-05	9,9640E-01	9,9620E-01	1,4850E-03
	E3	POLY-1	2,2240E-06	1,2060E-02		1,4910E-04	9,9250E-01	9,9230E-01	2,3080E-03
		POWER-2	1,4280E-04	6,3510E-01	-1,8860E-02	6,0120E-05	9,9700E-01	9,9680E-01	1,4920E-03
	E4	POLY-1	1,7920E-06	1,6810E-02		1,4340E-04	9,9060E-01	9,9020E-01	2,2630E-03
		POWER-2	1,0680E-03	4,5380E-01	-4,1680E-02	3,0420E-05	9,9800E-01	9,9780E-01	1,0610E-03
	E5	POLY-1	1,7230E-06	1,7220E-02		1,3970E-05	9,9110E-01	9,9080E-01	2,3340E-03
		POWER-2	1,1530E-03	4,4710E-01	-4,4010E-02	2,1900E-05	9,9860E-01	9,9850E-01	9,0007E-04

Table F-11: 134Cs/137Cs vs. NE

F.5 References

[1] MatLab team, *MatLab Manual*, MatWorks,2010