STUDY OF A SYSTEM FOR PWR FUEL ASSEMBLY CHARACTERIZATION DURING REFUELLING OPERATIONS

ABSTRACT

This final thesis project aims to define the main characteristics of an innovative out-core system to apply to a real PWR nuclear power plant during outages operations.

Thanks to the combination of the video system with commercial detection equipment, the device will strongly limit the possibility of fuel assembly misloading and so the reactivity incidents caused by a wrong placement of a fuel assembly inside the core. Besides, it will support the in-core burn-up measurement, concerning profile and average values, without important influences on the loading-unloading refuelling schedule. Furthermore, the possibility to recognize the typology of fuel, irradiated or fresh, combined with the theoretical automatic capability of the system could also decrease the effective dose to the workers. In addition the possibility to determine the amount of Plutonium isotopes and the total fissile material will be a very important feature in order to avoid nuclear proliferations.

The functions introduced in the device require the previous burn-up calculations, the determination of the passive neutron emission and a gamma ray spectrometry. For this reason, the project of this special passive device has required the utilization of a depletion code, ORIGEN-S, in which the data of a confidential spent fuel assembly database have been introduced, and of a Monte Carlo code, MCNP. The analysis of the huge number of ORIGEN-S simulations that has been conducted, has allowed understanding and analysing the sensitiveness of the burn-up and the main neutron emitters to some parameters. Thanks to a user made MatLab programme, the best fit has been implemented in the main correlations and a database has been automatically created with all ORGEN-S useful information, as for instance the fuel composition during the irradiation time and the gamma and neutron energy discretize spectrums for each simulation. The comparison with several technical publications has given good results.

Furthermore, taking into account the characteristics of a refuelling operation and the results of the depletion code, it has been decided to place the device into the transfer channel. For this reason several components, as for example the detectors and the pre-amplifier, will require the project of a special waterproof box. Once the placement and the whole equipment has been selected, the components have been modelled in the Monte Carlo software also taking into consideration to realize them in the more detailed way considering also the computational time. Difficulties have been encountered due to the slow processing speed of the old hardware used (Pentium III) and the
limitation of MCNP in very complex problems: gamma and neutron spectrum sources, very low detection efficiency and so big amount of particles simulated. However, the possibility of measuring the burn-up by the neutron flux has been tested and verified. No significant increment of refuelling operation time and the possibility of an automatic core loading verification allows increasing the availability of the nuclear power plant. In conclusion, despite the necessary future improvements that the project shows to require, it can be considered viable under all points under all aspects. As a matter of fact, with low necessary investments requested, the system allows environmental benefits, as the slightly decrease of nuclear waste amount.
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1. PREFACE

Due to Fukushima Dai-ichi NPP accident, happened in March 2011, the worldwide opinion about nuclear energy is getting worse. The combination of the stress test with the application of innovative devices to the NPP can be one of the first steps to win back the people confidence. The two fundamental problems, concerning this advanced source of energy, are the nuclear wastes and the dramatic consequences in cause of serious accidents, at least level 5 of INES scale.

For this reason, an innovative out-core device, which allows decreasing both factors in the fuel management area, has been designed.

This final thesis project does not aim to the fully development of the system but only to design its main functions and characteristics.

It is worth reminding that this project has obtained a scholarship for its development from Cátedra Argos in nuclear security of ETSEIB. Furthermore, it will allow to the student to obtain the European Master Of Science In Nuclear Engineering thanks to the collaboration of ETSEIB, Polytechnic of Turin and European Union inside the European Nuclear Engineering Network project.
2. ACRONYMS

BF  best fit
BU  burnup
BWR  Boiling water reactor
CEA  Commissariat à l’Energie Atomique
CT  cooling time
DA  destructive assay
FA  fuel assembly
FCL  fuel cycle length
FPD  fuel pellet density
FPP  fuel pellet porosity
FR  fuel rod
HM  heavy metal
IE  initial enrichment
IH  irradiation history
$k_{\text{eff}}$  effective multiplication factor
MCNP  Monte Carlo N-Particle
NDA  no-destructive assay
NDT  no-destructive testing
NE  neutron emission
NPP  nuclear power plant
PWR  pressurized light water reactor
SNP  spent nuclear pool
TGE  total gamma emission
WB  waterproof box
3. INTRODUCTION

Fuel assembly characterization is essential during the refuelling operation as well as in spent fuel manages operations like reprocessing plant or transportation. The combination of several different techniques allows determining numerous parameters such as for example burnup, fissile content, initial enrichment and amount of fission products with good accuracy.

During the last 20 years several methodologies and techniques have been developed by researches and utilities and nowadays fuel assembly characterization is also required by safety national agency (i.e. NRC) in order to confirm core-follow data, to implement BU credit in design of spent FA cask and finally to avoid misloading that may cause reactivity incidents.

Furthermore, it is worth reminding that most of the word’s inventory of Plutonium is stored in the irradiated fuel assembly of power reactors. The control and inventory of this potential source of weapons materials mainly by non-destructive method is an essential part of nuclear non-proliferation policy of the word community and so of IAEA as well.

This final thesis project has been specifically developed with respect to irradiated PWR 17x17 fuel assembly.

*The first part of this thesis* intends to resume currently fuel assembly characterization assay whereas *the second* one aims to design an on-line device which, without any influence on the loading/unloading schedule, allows determining: nuclear fuel burnup, axial burnup profile, fissile content, assembly typology (irradiated or fresh fuel) and finally the fuel assembly identification number. The purpose of the device implementation in fuel-management-core operation would be to validate automatically the final core loading, to avoid fuel assembly misloading, to improve the fuel management itself and the safety / availability of the power plant.
4. Irradiated Fuel Assembly Characterization

Due to fission in nuclear power plants a large amount of energy is produced. The composition of fuel pellet changes continuously during irradiation as well as in refuelling operation as a consequence of BU. FA assays allow calculating, monitoring and determinate irradiated FA parameters as for instance BU, fissile content and possibly cladding damages.

Irradiated FA characteristics depend on various fuel factors, which are governed by the physical conditions during reactor operation. Fuel parameters which are specifically considered during FA characterizations are:

- Burn-up or BU: it is the total amount of energy produced for amount of Uranium in nuclear fuel. It is measured in MWd/tU
- Cooling time or CT: it is the time passed since the fuel assembly is taken out of the reactor core
- Initial enrichment or IE: it is the content of $^{235}$U present in fresh fuel assembly. It can be uniform or not but in this master thesis it will be considered as uniform.
- Irradiation history: it is the information on how the total BU is distributed in time
- Fuel pellet porosity: it is the related density of fuel pellet in respect to UO$_2$ solid density
- Chemical shim: it is the concentration of Boron in moderator
- Burnable poisons: it is the amount of Gd$_2$O$_3$ which regulates the long term reactivity.
- Neutron flux and reactor power levels
- Fuel assembly design

You have subdivided the FA characterization methods as follow:

- No destructive testing or NDT
  - In-core measurement
  - Out-core measurement
    - Active techniques
    - Passive techniques
- Destructive assay (Annex A)
4.1 Non-destructive techniques

An NDA during a refuelling operation is necessary to characterize FA without neither destroying nor damaging them.

Neither the FA’s BU nor the fissile content can be determined directly but they can be calculated indirectly. In general, they are obtained by the combination of in-core measurements (i.e. neutron flux obtained by flux mapping) and corrected prediction thanks to out-core measurements (i.e. BU check and thermal power).

In-core BU measurement are more precisely than out-core ones. Several studies\textsuperscript{5,6,7} had shown that at maximum reactor records deviation from in-core design prediction is within the 4.2\%\textsuperscript{1} whereas the out-core ones are within the 5\% and 10\%\textsuperscript{11,12} in respect to reactor records.

However NRC\textsuperscript{9} recommends an out-of-core BU measurement in addition to the in-core one in order to:

- Confirm:
  - The reactor record and so the benchmarking of the physical code
  - The extrapolation of neutron flux done in flux mapping
- Prevent unauthorized loading (i.e. misloading) of assemblies due to inaccuracies in reactor
- Get BU records to verify that the appropriate subcritical margin is maintained
- Avoid an improper assembly identification
- Check the unloading criteria: a FA, with a determined IE, has not to be downloaded from the core and placed in the normal racket of SFP if its BU is less than BU specified in the final safety report of the NPP in order to maintain the $k_{eff}$ less than 0.95. In \textit{annex E}, it is shown an example of minimum BU label download of ASCO’-I & II NPP. Those limits are in function of IE and the spent fuel pool Boron concentration.

4.1.1 \textit{In-core measurement}

Reactor operation records or “core-follow data” are generated, achieved and controlled in compliance with regulation such as, for instance, 10CFR50 Appendix B of NRC in American NPP. They contain a lot of parameters like reactor power levels, reactor coolant flow, reactor coolant Boron concentration, fissile content and BU for each FA. A part of these data are obtained from plant instruments (i.e. Boron concentration and reactor coolant flow), but the minority cannot be directly
measured (i.e. BU and fissile content) therefore they are calculated by a nodal simulator neutronics code or are inferred from reactor testing parameters and reactor start-up.

Using the next equation, an American electrical utility calculates accurately the BU [MWd] cumulated in a generic FA:

\[
BU_i = \frac{RTP \times \Delta t \times CF}{m_f} \times P_i \tag{Eq. 5.1}
\]

Where:
- \( m_f \) [tons] is the mass of heavy fuel in metric tons of Uranium exactly measured in fuel fabrication facility
- \( RTP \) [MW] is the reactor rated thermal power; it is a constant for a given core
- \( \Delta t \) [days] is the irradiation time
- \( CF \) is the capacity factor: the measured power produced over the maximum amount of power which can be produced in the same \( \Delta t \). It is continuously defined by redundant methods.
- \( P_i \) is the relative power in FA, compared with to the reactor thermal power. It is determined by a lattice physics code (i.e. CASMO) and nodal simulation reactor code (i.e. SIMULATE) introducing the time integral of net thermal reactor power or \( E_{\Delta t} \).

The unique parameter in Eq. 5.1 which is not measured directly is \( P_i \), therefore an accurate BU determination needs a correct value of core distribution power and so an excellent work of flux map process, core instrumentation and finally of lattice and a nodal code.

In order to obtain \( P_i \) the reactor thermal power (\( P_{\text{CORE}} \)) must be known. It is determined by temperature and flow measurement of cooling water circulating through the reactor core assuming steady-state conditions, as follow:

\[
P_{\text{CORE,G}} = \sum_{i=1}^{n} \dot{m}_{w,i} \left[ h_{\text{out},i} (T_{\text{out},i} - P_{\text{out},i}) - h_{\text{in},i} (T_{\text{in},i} - P_{\text{in},i}) \right] \tag{Eq. 5.2}
\]

When:
- \( P_{\text{CORE,G}} \) [MW] is the total gross thermal power of the core
- $h_{out,i} \left( \frac{k_l}{k_{BG} + k} \right)$ is the outlet coolant’s enthalpy from the reactor in loop “i” in function of pressure and temperature
- $h_{in,i} \left( \frac{k_l}{k_{BG} + k} \right)$ is the inlet coolant’s enthalpy from the reactor in loop “i” in function of pressure and temperature
- $\dot{m}_w \left( \frac{kg}{s} \right)$ is the mass flow rate of water in loop i
- n is the number of loops of the reactor

To obtain the net thermal power of the core, $P_{CORE,N}$, it is necessary to subtract the thermal losses (such as for instance blow-down, seal water, component cooling water, heaters) which ANSI and Technical specification evaluate about 1% of $P_{CORE,G}$. The BU uncertainty contribution from the measurement uncertainty associated with total thermal core power is less than 0.1%.

Furthermore, to start the nodal code you need, $E_{st}$, the time integral of $P_{CORE,N}$ which can be determined exactly as follows:

$$E_{st}[MWd] = \int_0^{2t} P_{CORE,N}(t) \, dt \quad \text{Eq. 5.3}$$

By the way, Eq.5.3 is calculated in an approximate way discretising the integral in a series over a large number of time steps, n, as follows:

$$E_{st}[MWd] = \sum_{i=1}^{n} P_{CORE,N_i} \Delta t \quad \text{Eq. 5.4}$$

The BU assignment based on a core irradiation analysis that matches the actual energy output is often referred to the “core-follow” assembly BU.

4.1.1.1. Reactor analysis code

The methodology approved by NRC to determine core-follow BU is composed by two typologies of code:

1. The lattice physics code is a multi-group two-dimensional transport theory code used to conduct BU calculations on an assembly or on a single fuel pin and handles a geometry
consisting of cylindrical fuel rods with varied composition in a square-pitch array. This type of code needs the in-core neutron flux in input.

2. *The nodal simulation code* is a two-group three-dimensional program that solves the neutron diffusion equation for the homogenized nodal neutron flux in which FA are divided (normal 24 axial and 4 radial nodes). This type of code require cross section in input and neutron transport constantly computed by the lattice physics code and $E_{\Delta t}$. The nodal simulation code generates for each BU's step of 1 000MWd/tU:

   a. The isotopic inventory for each node radial and both axial
   b. Atom densities for each node radial and both axial to determine reaction rate
   c. Measured nodal power distribution
   d. Nodal Power
   e. Axial BU profile
   f. Xeon and Samarium reactivity worth
   g. Critical Boron concentration
   h. Core $k_{\text{eff}}$
   i. Expected detector rates as measured by a detector inside the FA during flux mapping to validate FA axial distribution prediction.
   j. Safety picking factor as:
      i. $F_{\Delta H}$ or peaking enthalpy factor: $\Delta H$ maximum over $\Delta H$ average of the same channel
      ii. $F_Q$ or total power peaking factor equal: nuclear peaking factor multiplied by the engineering uncertainly peaking factor

In general both codes are run with 100% reactor power and control rods withdrawn. The core follow calculation involves depleting the calculation of the nodal simulator of the current day and time in the cycle with the actual core state parameters as for instance control rods positions and chemical shim concentrations.

Due to the accurately reactivity balance NRC requires (i.e. predicted with in 1000pcm $\frac{\Delta k}{k}$) the level of accuracy of FA’s BU is elevated. Lots of studies$^{5,6,7}$ have shown that the reactor records deviate from core design prediction within the 4.2%$^7$ at most. It is worth reminding that a good extrapolation of core flux is fundamental for good results.
4.1.1.2. In-core BU measurement accuracy

The BU records do not generally differentiate between the data for instrumented and non-instrumented FA. At each flux mapping normally performed monthly a large database of core follow parameter is generated and the comparison of measurement and prediction at the instrumented FA leads to verify the accuracy of physical models code and so BU. A physical model code needs to be approved by NRC before using it in a NPP. As discussed before the exactness of these codes are, which have been widely validate $^{42}$, are good: the maximum reactor records a deviation from core design predicted within the $4.2\%^{3,5,6,7}$.

4.1.1.3. Flux Map process

Typically only two third of the core, and so the same amount of FA, are not instrumented and in this zone the flux needs to be determined by a flux map process.

The number of detectors that are used varies and depends on the size of the reactor and whether the detectors are fixed inside the core (BWRs and some PWRs) or are movable (most PWRs) in order to obtain a flux map. Roughly one-third of the fuel assemblies in a PWR core are instrumented. For example, a three-loop Westinghouse PWR (157 assemblies in the core) has 50 instrumented core locations, and a four-loop PWR (193 assemblies in the core) has 58 such locations $^{43}$. A typical PWR movable in-core instrumentation configuration is illustrated in Fig. 5-1. In a movable detection system, five or six fission chamber detectors are inserted through the instrumented core locations using retractable thimbles. The thimbles are closed to the leading reactor ends; thus they are dried inside and serve as a pressure barrier between the reactor coolant pressure and the atmosphere. Axial flux traces are recorded as the detectors are driven through the core. These traces typically contain 61 axial data points and provide a detailed axial neutron flux variation along the fuel assembly. During the flux map measurement, which typically lasts several hours, it is imperative that the reactor conditions (power, temperature, control rod position, etc.) are held as constant as possible by plant operators: the goal is to obtain a core power distribution that is instantaneous (i.e., a snapshot of the reactor state, although several hours would have elapsed between the first and last trace measurements). The pertinent reactor conditions are recorded along with the detector signals for post-processing. For data normalization purposes, cross calibration of the detectors is performed by inserting all of the fission chambers through a common core location.
Moreover, considering that in general a FA:

- Resides in the core for three fuel cycles
- Its probability to be instrumented is equal in each irradiation period and it is equal to 1/3 because in general only one third of the core is instrumented

Using the Binomial distribution and knowing that the events are independent you can obtained:

- The probability to be not instrumented in all fuel cycles is:
  \[ p_0 = b(0,3,0.33) = 0.3008 \]
- The probability to be instrumented in one fuel cycle is:
  \[ p_1 = b(1,3,0.33) = 0.4444 \]
- The probability to be instrumented in two fuel cycle is:
  \[ p_2 = b(2,3,0.33) = 0.2188 \]
- The probability to be instrumented in all fuel cycles is:
  \[ p_3 = b(3,3,0.33) = 0.03600 \]
- The probability to be instrumented at least once all over the fuel cycle is:
  \[ p_0 = B(1,3,0.33) = 0.69920 \]

As shown above the majority of FA is measured at least once during the irradiation time as their locations are changed from one cycle to next in order to improve in-core management; anyway a not negligible amount of FA could not be instrumented during their whole life, near 31 over 100. For this reason the flux map process has a big importance.

Thanks to the great value of p1 all the LWR’s generation up to now decided not to instrument all the core but in the new one (i.e. 3+ generation, EPR) all core will be instrumented to increase safety and reactor’s parameters control.

Despite a complete instrumentation, also in the 3+ generation the measure flux traces have to be processed: validation check, corrections for background, reactor operation fluctuation and normalization to a single detector.

Nowadays the majority of utilities uses a core analysis system (ARMP) developed by EPRI which interprets and use in-core measurements to provide a 3D map flux and so the 3D power distribution reflects changes in fuel and absorber composition as a consequence of BU.

Briefly, ARMP is based on the use of predictions form the nodal simulator code (i.e. SIMULATE) and coupling factors relating non-instrumented locations to the nearest instrumented FA.
The major flux map process steps\textsuperscript{1} are resumed as follows:

1) The “raw” measured traces are checked for validity, corrected for background interference, statistically evaluated for any available duplicate and symmetric traces, and normalized to a single detector response.

2) The assembly flux and/or power predictions are expanded from the nodal simulator number of axial nodes (~24) to that of the measured axial locations (~61).

3) Measured reaction rates at the instrumented locations are compared to their predictions from the nodal simulator code.

4) Correction factors are generated from the ratios of the measured to predict reaction rates at the instrumented locations and are used to adjust the predicted assembly power to infer the power in the uninstrumented assemblies.

5) Coupling or weighing coefficients for each non-instrumented location are derived using data from the nearest instrumented assemblies.

6) The power distributions at the non-instrumented locations are derived using the nodal simulator assembly powers along with the coupling coefficients.

7) Safety factors and margins to the Technical Specification thermal limits are computed using the inferred core power distribution.

Figure 5-2 illustrates the comparison between the predicted and the measured axial power in a typical instrumented FA. You can note seven discontinuities in the flux due to the presence of the same amount of spacing grid (SG). Limiting the analysis only to the relative error, marked by a yellow line, the outside spacing grid zone between the predicted and the measured axial power is always below 5% and so it is a good agreement.

Figure 4-1: Comparison of measured vs. predicted axial power distribution at an instrument core location (Ref 1)
The “measured core power distribution” is referred to the inferred core power distribution obtained by the combination of in-core measurement and correct prediction of nodal codes (i.e. SIMULATE-3).

To conclude, the flux maps are also used to calibrate the ex-core detectors and obtain the measure of the relationship between the amount of power generated in the top and in the bottom halves of the core.

4.1.2. Out-core measurement

As said before, out-core measurements are necessary in a NPP as well as in-core ones whereas they are less accurate \(^1,^2\).

Firstly, the most important functions of an out-core devices are:

- FA’s BU measurement
- Determining fissile and fission product amount
- Permitting core-follow data
- Avoiding FA core misloading and reactivity accidents
- Defining axial BU profile and so end-effect

Secondly, unlike in-core techniques, out-core ones can be basically performed in two places.

In the first one, the untypical but the modern ones, are applied inside refuelling operation without interfere or slowing loading/unloading schedule. For this reason they are called on-line device. A usual position can be the fuel tunnel transfer or in the transfer channel in fuel building. Up to the last years this employment was avoided and unusual because it executed an examination of a FA recently downloaded from the core submitted to a detector to a huge radiation field composed by gamma-rays (i.e. \(10^{19}\) gamma/s/MeV \(^{16}\)) and also by neutrons (i.e. \(10^8\) neutrons/s/MeV \(^{16}\)) which could cause damage and impossibility to measure. Furthermore it is worth reminding that the residual power generated by the decay of fission products (mostly beta and gamma) is not negligible, however it decreases in an exponential way. Considering a nominal linear power of 15.4 kW/m and applying the next relation of ANSI \(^{29}\):

\[
\frac{P(t)}{P_0} = 6.48 \times 10^{-3} [t^{-0.2} - (t + T_0)^{-0.2}] \\
Equ. 5.5
\]

Where:
- T [days] is the operational period
- t [days] is the time after shut down

The following values have been calculated to assume only an irradiation period equal to 372 days.

<table>
<thead>
<tr>
<th>Cooling time [days]</th>
<th>Time step</th>
<th>Thermal residual Power percentage</th>
<th>Thermal residual Power FA [W]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Shutdown reactor core</td>
<td>0.0609</td>
<td>4336</td>
</tr>
<tr>
<td>1.5</td>
<td>Out-core measurement</td>
<td>0.003985</td>
<td>283.732</td>
</tr>
<tr>
<td>37</td>
<td>Reloading FA inside core</td>
<td>0.01941</td>
<td>85</td>
</tr>
</tbody>
</table>

Table 4-1: Thermal residual power of a PWR

In order to apply those mentioned hard conditions out-core devices require “special” conditions like:
- Strong shielding
- Detection which can support to strong radiation field
- Cooling but only in some application

CEA in collaboration with EDF has recently designed, produced and patented one of these modern devices (i.e. NAJA device or PHYTON™ device, annex C). They use video techniques, active and passive nondestructive assays which allow to:
- Perform accurate burn-up measurements with passive neutron counting
- Determine FA identification number
- Verify the oxidation and some possibilities of cladding fuel rods
- Distinguish between fresh fuel/irradiated fuel or between UO2/MOX
- Define initial enrichment by interrogation neutron techniques
- Detect the presence and the kind of neutron absorber;
- Read the identification number by an Optical Character Recognition
- Perform the burn-up profile by gamma spectrometry
- Evaluate the activities of different fission and activation products thanks to gamma spectrometry
- Actuate an automatic core loading conformity control for PWR

Another type of these devices is PHYTON™ also developed by the CEA-EDF alliance; it has been described inside annexe C.

The second one is inside the spent fuel pool. This was the typical position and the first one used because measuring the exposure of a FA after a determined cooling time (i.e. 3 months) enables to eliminate a large part of gamma radiation due to the decay of short fission products. This allows performing analysis, such as neutron passive emission counts, in an easier way with fewer problems in terms of residual heat and radiation damages with allow to use more typologies of detectors like a HSGR.

In 1988 Los Alamos Laboratory built the first out core device composed by an HPGe (high-purity Germanium detector) and a fission chamber detector which allowed measuring BU: it was the Fork device¹² largely used and verified in American NPP. Nowadays recent out-core devices of this typology allows to:
- Determine an accurate burn-up measurement with passive neutron counting
- Distinguish between fresh fuel/ irradiated fuel or between UO₂/ MOX
- Define initial enrichment by interrogation neutron techniques
- Elaborate a burn-up profile by gamma spectrometry
- Determine the activities of different fission and activation products thanks to gamma spectrometry
- Operate a fuel cask loading conformity control for PWR

A description of modern industrial out-core device is reported in annexe C.

Lastly, the costs and risks associated with performing out-of-core BU measurements should be balanced with the risks and potential costs of not performing the measurements. Out of-core measurement campaigns require utility resources for planning and execution, revision of Probability Safety Assessment (PSA or APS) if an active method is used, increase the dose to personnel, increase the risk of damage to assemblies and potential fuel mishandling events due to the increased assembly movements, and have associated financial cost to the utility. All these parameters has to be taken under account when an out-core measurement device is designed.
4.1.2.1. No destructive active assay

A direct measurement of the fissile content of irradiated fuel, which is around 1% by weight of irradiated is possible using a large neutron source to induce fissions.

The source applied can be an accelerator, 14 MeV neutron generator, or an isotopic source with a high yield of spontaneous fission like $^{242}\text{Cm}$. The source, which has to be pulsed (i.e. on/off), is positioned near the irradiated fuel, where it produces an induced fission signal (i.e. neutrons) proportional to the amount of fissile material measured by a fission chamber with a Cd liner and a $\text{CH}_2$ moderator, as shown in figure 4-2.

Various different approach can be used in order to measure only induced delayed fission neutrons or only prompt fission neutrons shortly after or both two shortly after each interrogation source.

Specifically, Uranium, $^{235}\text{U}$, and Plutonium fissile content fuel, $^{239}\text{Pu}$ and $^{241}\text{Pu}$, can be determined. $^{235}\text{U}$ can be easy discriminated from $^{239}\text{Pu}$ because it emits more delayed neutrons per fission resulting in a discrimination ratio of 2.6, although for $^{241}\text{Pu}$ the discrimination ratio to $^{235}\text{U}$ is not applicable because it is near to one. This problem may be circumvent through iterative techniques, using calculated isotopic correlation generated by a depletion code (i.e. ORIGEN-S) and a BF toolbox (i.e. MatLab’s Curve Fitting Toolbox).

Nowadays this technique and other similar are used in several devices in SFP, reprocessing plant for MOX and fuel pellet industry and a lot of technical publication can be found.
4.1.2.2. No destructive passive assay:

Irradiation time over FA in reactor core allows to produce a great amount of energy by fission (i.e. for each FA an average of 60 kW), mostly thermal but also fast one, and a lot of fission products (i.e. around 800\(^{15}\)) are produced as well; therefore the composition of fuel pellet changes continuously in no irradiation time as well as in refuelling operation due to the decay of elements. The next equation resumes all the events of formation, destruction and decay which can occur.

\[
\frac{dN_i}{dt} = \sum_j \gamma_{j,i} \sigma_{f,j} N_j \Phi + \sigma_{c,i-1} N_{i-1} \Phi + \gamma' N'_i - \sigma_{f,i} N_i \Phi - \sigma_{c,i} N_i \Phi - \gamma N_i \quad \text{Eq. 5-6}
\]

where:
- \(\sum_j \gamma_{j,i} \sigma_{f,j} N_j \Phi\) is the yield rate of \(N_i\) due to fission of al nuclides \(N_j\)
- \(\sigma_{c,i-1} N_{i-1} \Phi\) is the rate of transmutation into \(N_i\) due to radioactive neutron capture by nuclide \(N_{i-1}\)
- \(\gamma' N'_i\) is the rate of formation \(N_i\) due to the radioactive decay of the nuclides \(N'_i\)
- \(\sigma_{f,i} N_i \Phi\) is the destruction rate of \(N_i\) due to fission
- \(\sigma_{c,i} N_i \Phi\) is the destruction rate of \(N_i\) due to all forms of neutron absorption other than fission \((n,\gamma;n,\alpha;n,p;n,2n;n,3n)\)
- \(\gamma N_i\) is the radioactive decay rate of \(N_i\)

Eq.5-6 is written for a homogeneous medium containing a space-energy-averaged neutron flux, \(\Phi\), with flux-weighted average cross sections, \(\sigma_r\) and \(\sigma_o\), representing the reaction probabilities. The flux is a function of space, energy, and time is dependent upon the nuclide concentrations.

Moreover, when a FA is discharged from the core, a lot of different particles and photons are produced and emitted spontaneously as for instance electrons, neutrons, positrons, alphas, gamma and x-rays. Unfortunately not all “particles” as useful to be used due to their own characteristics (i.e. range), their intensity and environmental characteristics (i.e. when a FA is downloaded, it remains above six meters of water all the time \(^{10}\), annex E). Thanks to the strong passive radiation field generated by irradiated FA passive NDA can be applied and commonly they are separated in two major categories:
- Gamma-ray methods
- Neutron methods
Both signatures of irradiated fuel elements are time-dependent variables. Each measurement must be corrected due to radioactive decay to allow the data to be correctly interpreted.

Each of these NDA provides some information about irradiated FA, but by combining these techniques, the level of confidence of the FA characterization is raised to the highest practical level\textsuperscript{15}.

\textit{Modus operandi:}

To obtain FA’s BU or fissile amount inside irradiated fuel, it is necessary to accomplish the following steps for both gamma-ray and neutron methods.

Firstly, depletion code which can reproduce irradiation time and decay time as ORIGEN-S has to be run for each BU to get the composition of the irradiated fuel, neutron and gamma-ray spectrum discretized in energy. Running the physical code changing IE, CT, fuel density and irradiation history a database can be build and also correlations between several amounts of fission products, NE or BU can be obtained.

Secondly, according to of the required characteristics of the out-core device, it is necessary to choose the detector equipment system and its placements. Then a Monte Carlo code (i.e. MCNPX) is obligatory to reproduce the amount of neutron and gamma-ray flux generated by FA which reaches the detector and produce counts. This operation allows taking into account geometrical factors, auto-absorptions, scattering and a lot of other parameters. Moreover also a gamma spectrometry is present to allow correlation, such as for instance between element amounts and BU.

In this way a new correlation between detector’s counts (i.e. $\text{BF}_3$ detector) generated by NE or gamma-ray emitted by FA and FA’s BU can be generated with a mathematical software (i.e. MatLab). It is indispensable to build a correlation for each IE, CT, different irradiation history and the concentration of Boron in water to use it in correspondent case. Considering more factors the relations will be more precious but it becomes very complicated.

As just shown for the correlation between BU/NE also for Pu mass/NE or fissile content/NE can be easily done.

The selection of detectors and placement of the device is fundamental because it radically changes the efficiency, the associated cost of materials and the detectors.
4.1.2.2.1 Neutron passive methods

The amount of neutrons generated by a 17x17 PWR FA with an IE of 3.5% when it is downloaded after the third irradiation period can be around $10^8$ neutrons/s/MeV.

There are three primary sources of neutron in irradiated fuels:

- Spontaneous fission of transuranic isotopes as Cm$^{242}$
- Alfa-neutron ($\alpha$, n) reaction with light material as O$^{18}$
- Photo-neutron reactions ($\gamma$,n)

The produced neutrons can undergo multiplication within the fuel material (i.e. due to fission), thereby producing additional neutrons. However, due to both to the concentration of Boron in water during refuelling operation, maintained around at 2600 ppm, and to the absorber material the $k_{eff}$ is kept below 0.95 in any situations also in case of accident.

The contribution of these sources to neutron passive flux depend on various parameters but the most important are: IE, CT, BU, irradiation history and fuel density. The first two become predominant when the CT reaches several months because photo-neutron source are produced by short fission products (i.e. $^{140}$Ba-La with $T_{1/2}$=12.8 days), which produce high-gamma rays (2.23 MeV). In the next table (table 5-2) elaborated by Los Alamos laboratory the principal sources of neutron emitted per second and per grams in irradiated UO$_2$ materials are resumed.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Half Lives $^{40}$ (years)</th>
<th>($\alpha$,n) Reaction $^{54-55}$</th>
<th>Spontaneous Fission $^{56}$</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{235}$U</td>
<td>7.038 ± 0.005 x 10$^5$</td>
<td>7.21 ± 0.72 x 10$^{-4}$</td>
<td>3.86 ± 0.99 x 10$^{-4}$</td>
<td>1.17 ± 0.12 x 10$^{-3}$</td>
</tr>
<tr>
<td>$^{238}$U</td>
<td>4.463 ± 0.0024 x 10$^9$</td>
<td>8.43 ± 0.84 x 10$^{-5}$</td>
<td>1.36 ± 0.02 x 10$^{-2}$</td>
<td>1.36 ± 0.02 x 10$^{-2}$</td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>87.71 ± 0.03</td>
<td>1.56 ± 0.16 x 10$^4$</td>
<td>2.60 ± 0.11 x 10$^3$</td>
<td>1.82 ± 0.16 x 10$^4$</td>
</tr>
<tr>
<td>$^{240}$Pu</td>
<td>2.413 ± 0.0016 x 10$^4$</td>
<td>4.25 ± 0.43 x 10$^3$</td>
<td>4.25 ± 0.43 x 10$^3$</td>
<td>4.25 ± 0.43 x 10$^3$</td>
</tr>
<tr>
<td>$^{242}$Pu</td>
<td>6.570 ± 0.006 x 10$^3$</td>
<td>1.56 ± 0.16 x 10$^2$</td>
<td>0.65 ± 0.10 x 10$^2$</td>
<td>1.04 ± 0.19 x 10$^3$</td>
</tr>
<tr>
<td>$^{242}$Pu</td>
<td>3.763 ± 0.009 x 10$^5$</td>
<td>2.27 ± 0.23</td>
<td>1.743 ± 0.015 x 10$^3$</td>
<td>1.743 ± 0.015 x 10$^3$</td>
</tr>
<tr>
<td>$^{241}$Am</td>
<td>432.0 ± 0.2</td>
<td>3.17 ± 0.32 x 10$^3$</td>
<td>3.17 ± 0.32 x 10$^3$</td>
<td>3.17 ± 0.32 x 10$^3$</td>
</tr>
<tr>
<td>$^{242}$Cm</td>
<td>0.4456 ± 0.0001</td>
<td>4.48 ± 0.45 x 10$^6$</td>
<td>2.25 ± 0.08 x 10$^5$</td>
<td>2.70 ± 0.09 x 10$^5$</td>
</tr>
<tr>
<td>$^{244}$Cm</td>
<td>18.099 ± 0.015</td>
<td>8.82 ± 0.88 x 10$^4$</td>
<td>1.081 ± 0.007 x 10$^2$</td>
<td>1.090 ± 0.007 x 10$^2$</td>
</tr>
</tbody>
</table>

Table 4-2: Principal sources of neutrons in irradiated UO$_2$ fuel (Ref 15)
The result that the \((\alpha, n)\) primarily sources are \(^{242}\text{Cm}\) and \(^{241}\text{Am}\) whereas the spontaneous sources ones are \(^{242}\text{Cm}, ^{244}\text{Cm}, ^{238}\text{Pu}, ^{240}\text{Pu}\).

![Figure 4-3: Neutron source rates for the five most prominent neutron-emitting isotopes and an exposure of 19.717 GWd/tU (ref 16).](image)

The total neutron rate integrated all over the energies has been plotted in the *figure 5.5*. The sum of the contribution of \(^{242}\text{Cm}, ^{244}\text{Cm}, ^{238}\text{Pu}, ^{240}\text{Pu}, ^{241}\text{Am}\) is very close to the 100% of the neutron rate. The role of Cm isotopes is fundamental: for short CT \(^{242}\text{Cm}\) gives the major contribution whereas CT is greater than eight months \(^{244}\text{Cm}\) is predominant and produce almost the total neutron emission. In general it is not possible to distinguish the individual isotopic sources of the neutrons emitted from a fuel assembly, therefore only the gross neutron emission rate has been measured and correlated with operator-declared values of BU and fission product. During the past 40 years the relation of BU and NE has been studied and improved thanks to research. The first one was linear \(^a\) but a better one has been elaborated. It has been published by IAEA inside IAEA-SM-352/33, and it has been reported following:

\[
BU = a \ NE^b
\]

*Equ. 5.7*

Where:

- \(a\) is a constant slightly depending on IE
- b is a constant depending closely on irradiation history close to 0.02 for normal ones.

In addition, the constants a and b can be calculated using an evolution code (i.e. ORIGEN-S) taking closely into account IE, CT and the irradiation history of the FA examined. The accuracy of this correlation is good: in fact the relative error of NE is close to 2%.

Furthermore, other correspondences can be theoretically produced between BU and any fission product as shown before inside the *modus operandi* subchapter.

To conclude, one advantage of the neutron techniques over the gamma-ray technique is the high penetrability of neutrons. The high penetrability of neutrons reduces the problem of measuring the interior portions of the assembly: the neutrons produced in the interior rods of a PWR assembly contribute as much to the measured signal as the neutrons originating in the exterior rods.

4.1.2.2.2 Gamma-ray passive methods

The amount of gamma-rays generated by a 17x17 PWR FA with an IE of 3.5% when it is downloaded after the third irradiation period can be up to $10^{18}$ photons/s/MeV.

Gamma-ray signature of irradiated fuel assembly originate products of the fuel from fission and the activation products of the cladding and associated structural material. The gamma intensity field decreases very fast with increasing of CT because a great amount of gamma is produced by short-live fission products.

Gross gamma activity or the gamma spectra can be measured and correlated to BU, CT and amount of total fissile material inside an irradiated fuel assembly. It can be used as integrating functions or for detailed measurement at a single axial location. However, due to the slope of correlation laws for gamma emitters which is always smaller than the neutron emission, BU determination using passive neutron counting that leads to more accurate results than gamma measurement. Anyway the combination of both techniques is performed to verify the results and avoid of errors, mostly human.

The application of gamma-ray techniques to the measurement of irradiated fuel assemblies has several limitations based upon the physical properties of gamma photons. One important limitation is the problem of self-attenuation within the fuel assembly. Therefore, measurements using gamma-ray
techniques are primarily used for the exterior rods of LWR fuel assemblies. Radial BU distribution may also influence the measured gamma-ray intensities. 

Each of the measurable isotopes has limitations in correlation with cooling times or BU values. The selection of the "best" correlation for either quantity may depend upon the CT and the scope.

In table 5-3, you have reported the main accurate and useful correlation for gamma spectrometric BU determination for a PWR's FA 17x17 with an IE of 3% and a CT equal to 3 years. The activity is given in Curie/g, Bu in GWd/tU and CT in days.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Correlation law</th>
<th>Cooling time</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>0 to 90</td>
</tr>
<tr>
<td>$^{134}$Cs</td>
<td>$a$ BU^2 ($a = 0$)</td>
<td>![Green]</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>$a$ BU ($a = 3000$)</td>
<td>![Green]</td>
</tr>
<tr>
<td>$^{154}$Eu</td>
<td>$a$ BU^b ($a = 5$, $b = 2$)</td>
<td>![Green]</td>
</tr>
<tr>
<td>$^{134}$Cs/$^{137}$Cs</td>
<td>$a$ BU^b ($a = 10^{-2}$, $b = 1$)</td>
<td>![Green]</td>
</tr>
<tr>
<td>$^{154}$Eu/$^{137}$Cs</td>
<td>$a$ BU^b ($a = 10^{-3}$, $b = 1$)</td>
<td>![Green]</td>
</tr>
</tbody>
</table>

Table 4-3: Overview of gamma spectrometric BU determination (Ref 14)

Legend:

<table>
<thead>
<tr>
<th>Means</th>
</tr>
</thead>
<tbody>
<tr>
<td>Really not appropriate</td>
</tr>
<tr>
<td>Not appropriate</td>
</tr>
<tr>
<td>Can be successfully used</td>
</tr>
<tr>
<td>Recommended</td>
</tr>
</tbody>
</table>

Table 4-4: Legend Table 4-3
In table 5-5, you have listed some correlation for gamma spectrometric CT determination for the PWR’s FA 17x17 with an IE of 3%. The same legend has been applied.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Correlation law</th>
<th>Cooling time</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>0 to 90</td>
</tr>
<tr>
<td>$^{144}$Ce/$^{137}$Cs</td>
<td>$a \exp^{b CT}$ (a = 10, b = -0.002)</td>
<td>Yellow</td>
</tr>
<tr>
<td>$^{106}$Ru/$^{137}$Cs</td>
<td>$CT^{0.5} a \exp^{b CT}$ (a = 1, b = -0.02)</td>
<td>Green</td>
</tr>
</tbody>
</table>

Table 4-5: overview of gamma spectrometric CT determination (Ref 14)

In general it is better to choose correlation composed by two elements because it is more stable and more precocious.

As discussed before, measuring total gamma emission by a simple proportional detector as the Geiger-Muller counter and knowing FA’s BU, the CT can be defined by the following relation 14:

$$CT = a (BU) \left( \frac{TGE}{BU} \right)^{b(BU)}$$  \hspace{1cm} Equ. 5.8

When a (BU) and b(BU) are quadratic functions of BU, determined by fitting parameters calculation results. Equ.5.8 is valid only if CT is bigger than several months. Furthermore, obtaining a collimated gamma emission in a proper way, it is possibly to define also axial BU profile in order to estimate “end-effects”. It is necessary a scanning of total gamma emission or TGE in function of the height of FA, z. Extremity BU or EBU is defined by the relation 4:

$$EBU(z) = \frac{z}{z_0} \frac{\int_0^z TGE(z)dz}{\int_0^z TGE(z)dz} F(CT)$$  \hspace{1cm} Equ. 5.9
Equ.\,4 assumes that there is a linear proportionality between BU and TGE$^{\text{14,19}}$ correct for short CT by a factor $F$ computed as follow for PWR reactors:

$$F(CT) = 0.5 \cdot CT^{0.1} \quad E\text{qu. \,5.10}$$

5. **Innovative out-core nondestructive device**

5.1. **Introduction**

In the first section of this project the major FA characterization techniques have been analysed and described, whereas the second part intends to resume the study done about the design of an innovative out-core nondestructive FA characterization system, taking into account the characteristics and the refuelling operations of a real Spanish PWR nuclear power plant: ASCÔ-I. This project proposes to develop the major features and aspects of this system and not the full development of it.

In this part you will be shown:
- The required functions of this device
- The methodology of work
- The choice of the correlation applied to the device
- The selection of the FA to simulate

In the next subchapters, it will be shown:
- The characterization of nuclear fuel burnup with ORIGEN-S
- Selection of the device placement
- The choice of special detector equipment
- The test of the innovative out-core device with a Monte Carlo techniques, MCNP
- The application of the innovative out-core device inside a refuelling operation
- An economical and environmental evaluation
- The future developments
- Conclusions

The design of this innovative and special system requires:
A large application of a depletion code: ORIGEN-S
An intensive use of a Monte Carlo code: MCNP
The analysis of the refuelling operation
The project of special detection system whose several parts must be positioned inside water

5.2. Features and objective of the system

A nondestructive on-line out-core system which allows to determinate:

- Fuel assembly burn-up by the detection of neutron passive flux and by the analysis of specific gamma-ray emitted by fission products (i.e. $^{137}$Cs) with a gamma spectrometry
- The axial burn-up profile by the integration over energy of prior gamma spectrometry
- The amount of fissile materials: $^{235}$U, $^{239}$Pu, $^{241}$Pu
- The typology of fuel assembly: irradiated or fresh
- The fuel assembly identification number

To perform these functions the system is basically composed by three parts:

- Neutron detection equipment
- Gamma-ray detection equipment
- Video control device

Furthermore, the device has been specifically projected to avoid:

- Damages on the FA
- A great modification of transfer channel
- Any influence on the loading/unloading schedule.
- Modification of the $k_{eff}$
- An additional fuel displacements in order to avoid extra FA misloading as specified by NRC
- A big revision of Probability Safety Assessment (PSA or APS)
- An additional effective dose to the workers
- Increase time of the FA displacement from the core in the spent fuel pool and so for the refuelling operation
As specified above to avoid the modification of the keff and an additional dose to the workers the system is composed only by passive techniques. Moreover non-destructive techniques have been chosen to obtain an on-line device which can operate inside a refuelling operation.

Knowing that, the out-core measurements are in general less accurate than in-core one \(^1\), this device does not intends to substitute FA characteristics measured by in-core computational code but aims to give an extra instruments to verify them and their assumptions as for instance the extrapolation of neutron flux in no instrumented zones. If for instance a part of in-core instrumentations do not work well, this innovative nondestructive on-line out-core system will be essential for the measure of BU and so for the new FA disposition in the next core.

The implementation of this device inside a refueling operation allows to:

- Avoid FA mishandling
- Avoid FA misloading core loading decreasing the possibility of the reactivity accidents
- Perform an automatic validation of final core loading
- Improve the global availability of the plant
- Compare in-core and out-core burn-up value
- Generate a more exact inventory of fissile materials in SFP and of reactor records
- Facilitate the inspection of IAEA

At the end, the more accurate value of BU would allow to implement end-effect in cask design which consents to improve the capacity of SPF and FA transport.

### 5.3. Methodology of system design

Once defined the targets of the work several steps, have been done to design the system. These are listed here and shown in the next figure:

1. Choice of the fuel assemblies to simulate along the project
2. Choice of useful correlations applied in the device
3. Characterization of nuclear fuel burnup with ORIGEN-S
4. Elaboration of ORIGEN-S results, curve fitting and creation of a database by MatLab
5. Choice of the device placement and definition of the environmental characteristics
6. Choice of detection equipment
7. Modelling of the detectors and project of special shielding which allow them to work inside water

8. Test of the innovative out-core device with a Monte Carlo techniques, MCNP

9. Elaboration of MCNP results

10. Economical evaluation

11. Environmental evaluation

12. Possible future developments

13. Conclusion

5.4. Choice of correlation applied in the device

As commented before the first step of the project has been the choice of the correlation to use in the device.

5.4.1. Fuel Assembly Burn-up

For the determination of BU in function of the NE, it has been decided to apply equ.5.7² because it has a good accuracy and can be easily determined by the combination of a depletion code and a curve fitting toolbox. It has been reported here:
\[ BU = a \cdot N E^b \quad \text{Equ. 6.1} \]

Where:
- "a" is a constant slightly depended on IE
- "b" is a constant dependent to irradiation history

Constants "a" and "b" will been calculated by ORIGEN-S in at step n.4.

In addition, the relation between \( ^{134}\text{Cs}/^{137}\text{Cs} \), mass ratio, will be used to determine BU by a gamma spectrometry in order to compare this value with the previous one even if it is less accurate \(^3\). It is given by:

\[ \frac{\text{Cs}^{134}}{\text{Cs}^{137}} = a \cdot \text{BU}^b \quad \text{Equ. 6.2} \]

This correlation has been chosen because it has been supposed to make the out-core measurement with a CT equal to 1.5 day and so inside the range from zero to nighty days where its accuracy is good (table 4-3). Moreover \(^{134}\text{Cs}/^{137}\text{Cs} \) correlation has to be preferred to \(^{137}\text{Cs} \) because it is more stable \(^1\).

The Cesium peaks chosen for the correlation are the following.

<table>
<thead>
<tr>
<th>Decay yield</th>
<th>Energy</th>
<th>Intensity</th>
</tr>
</thead>
<tbody>
<tr>
<td>[%]</td>
<td>[MeV]</td>
<td>[g/s]</td>
</tr>
<tr>
<td>(^{137}\text{Cs})</td>
<td>99.9</td>
<td>0.6617</td>
</tr>
<tr>
<td>(^{134}\text{Cs})</td>
<td>85.5</td>
<td>0.795</td>
</tr>
</tbody>
</table>

Table 5-1: Cesium peaks selected

The peak of \(^{134}\text{Cs}\) has been selected from the most of the element because it has:
- A great decay yield (85.5 %)
- A good ORIGEN-S output intensity (9.708*10^{16} g/s for simulation E501 at recharge one)
- A bigger energy than that the 604 keV radioactive decay photons which has the highest decay yield
5.4.2. **The axial burn-up profile**

To determine Axial BU profile eq.5.9 will be applied. It has been reported as follow:

EBU is defined by the relation\(^4\):

\[
EBU = \frac{\gamma}{\gamma_0} \int_0^{y_0} TGE(y) dy \int_0^y TGE(y) dy \ 0.5 \ CT^{0.1}
\]

*Equ. 6.3*

Where:
- EBU is the extremity BU in MWd/tU
- TGE is total gamma emissions in photons per second
- CT is cooling time in days
- \(y\) is the coordinate along the FA active length

5.4.3. **The amount of fissile materials**

The amount of fissile material and of Plutonium isotope for each FA will be determined by own correlation derived by the best fitting of ORIGEN-S simulation thanks to the curve fitting MatLab toolbox (*annex F*).

5.5. **Choice of the fuel assembly to simulate**

The first big effort during the process of design of this innovative out-core device has been the ORIGEN simulations. Anyway before running them the FA’s BU has to be chosen. Thanks to a Spanish NPP, a confidential database of discharged FA during all plant life has be obtained and deeply analysed by statistical software, Minitab-16 (*annex F*). For reason of military and national security the names of the NPP must not be revealed. The database contains a lot of information about ninety hundred and four PWR 17x17 FA; the fuel is sintered UO\(_2\), 95% of porosity, only enriched in \(^{235}\)U (*Annex G*). Specifically in this database you can find:
- Burn-up [MWd/tU]
- Discharge day [day]
- Initial enrichment of the FA [%]
As reported in the annex G, the database FA has been divided in several groups depending on the IE but only five have been chosen. The following table shows the selection operated.

<table>
<thead>
<tr>
<th>IE</th>
<th>From</th>
<th>Up to</th>
<th>Group</th>
<th>Number of FA</th>
</tr>
</thead>
<tbody>
<tr>
<td>2,1149</td>
<td>2,1151</td>
<td>A</td>
<td>53</td>
<td></td>
</tr>
<tr>
<td>2,5829</td>
<td>3,1031</td>
<td>B</td>
<td>52</td>
<td></td>
</tr>
<tr>
<td>3,1029</td>
<td>3,1031</td>
<td>C</td>
<td>52</td>
<td></td>
</tr>
<tr>
<td>3,1484</td>
<td>3,1486</td>
<td>D</td>
<td>44</td>
<td></td>
</tr>
<tr>
<td>3,2287</td>
<td>3,2691</td>
<td>E</td>
<td>100</td>
<td></td>
</tr>
<tr>
<td>3,3500</td>
<td>3,3665</td>
<td>F</td>
<td>73</td>
<td></td>
</tr>
<tr>
<td>3,3965</td>
<td>3,4085</td>
<td>G</td>
<td>39</td>
<td></td>
</tr>
<tr>
<td>3,5881</td>
<td>3,5999</td>
<td>H</td>
<td>38</td>
<td></td>
</tr>
<tr>
<td>3,6000</td>
<td>3,6266</td>
<td>I</td>
<td>134</td>
<td></td>
</tr>
<tr>
<td>3,8476</td>
<td>3,8567</td>
<td>L</td>
<td>23</td>
<td></td>
</tr>
<tr>
<td>4,0667</td>
<td>4,1088</td>
<td>M</td>
<td>28</td>
<td></td>
</tr>
<tr>
<td>4,1213</td>
<td>4,1588</td>
<td>N</td>
<td>117</td>
<td></td>
</tr>
<tr>
<td>4,1603</td>
<td>4,2120</td>
<td>K</td>
<td>47</td>
<td></td>
</tr>
<tr>
<td>4,4381</td>
<td>4,4695</td>
<td>P</td>
<td>41</td>
<td></td>
</tr>
<tr>
<td>4,4699</td>
<td>4,4966</td>
<td>Q</td>
<td>39</td>
<td></td>
</tr>
<tr>
<td>4,5159</td>
<td>4,5486</td>
<td>Q</td>
<td>24</td>
<td></td>
</tr>
</tbody>
</table>

**Total** 904

Table 5-2: Fuel assembly database selection

Besides the next table shows the BU characteristics of each group.

<table>
<thead>
<tr>
<th>IE</th>
<th>From</th>
<th>Up to</th>
<th>Group</th>
<th>Burnup characteristics of the selected IE</th>
</tr>
</thead>
</table>

**Table 5-3**
Once selected the groups, a further choice has been done: ten FA have been chosen from each group trying to cover the whole range of BU in the best way. In fact as a result of analysis of the FA database (annex D), the burnup inside a group, therefore with a close IE, is distributed as a Gaussian as a result of the Limit Central Theorem as a consequence of random fission process.

In annex F, all the FA selected have been reported classified for IE code and listed with increasing burnup.

### 6. Characterization of nuclear fuel with ORIGEN-S

Once the FA have been selected and so the IE and the total BU obtained, four main typology of ORIGEN-S simulation have been done to investigate the variations in neutron and gamma-spectra at each out-core measurement.

To investigate parameter sensitivities, one of the base-case parameter has been varied, whereas the other parameters were fixed. Parameter variation included:

- a) Initial enrichment
- b) Fuel cycle length
- c) Irradiation history with various combinations of power cycles (I) and shut-down periods (D)
- d) Fuel pellet density

A total of three hundreds simulations have been run.

For each simulation done, the following item have been obtained in three different text files:

- The composition of fuel in function of time during irradiation time and refuelling time
- The gamma spectra at each out-core measurement
- The neutron spectra at each out-core measurement
After that, all data have been loaded, elaborated and exported to Excel database by MatLab scripts and programmes (annex F).

To conclude, a deep analysis of this depletion code is done in annex G.

6.1. Simulation input

In order to obtain the composition of the fuel and the neutron and gamma-ray spectra for each simulation, ORIGEN-S requires:

- Selection of the FA typology
- The initial and isotopic composition of the fuel
- The discretization in energy of gamma and neutron spectra output and so the selection of an ORIGEN-S library
- The scheme and the characteristics of FA life inside the fuel cycle: irradiation period and refuelling period
- The required output

6.2. Selection of fuel assembly typology

As discussed before the FA typology selected is a PWR 17x17 with UO$_2$ pellet. A description of this FA is reported in annex G.

6.2.1. Initial composition of the fuel

All over the simulations, except in group simulation “d”, the initial composition of the applied fuel is:

- Total mass of Uranium Isotope: 491289.2 g
- Total mass of Oxygen Isotope: 66006.4 g
- Total mass of UO$_2$: 557295.6 g

Specifically the Uranium isotopic composition, which depends on the IE, is:

- $^{234}$U: 0.0119
- $^{235}$U: IE
- $^{236}$U: 0
- $^{238}$U: 99.74-IE
whereas for the Oxygen:

- $^{16}\text{O}: 0.99759$
- $^{17}\text{O}: 0.00037$
- $^{18}\text{O}: 0.00204$.

6.2.2. **Library selection**

See Annex G.

6.2.3. **Fuel cycle scheme**

ORIGEN-S allows altering irradiation and refuelling periods.

The user can define the power level and the length of each power period. Imposing a BU and the fuel cycle duration the code automatically computes a neutron flux and thanks to an exponential matrix method computes the new fuel composition at each time step. Instead during a decay period, the neutron and the gamma fluxes are not external computed but they result from the radioactive decay of the fission products. As well as in the power period in the decay period the user can define the length and the time steps.

In this final thesis project the total BU has been divided in three fuel cycles, with the same lengths and the same power basis alternated by a decay period which represents a refuelling operation. This has been assumed equal to 37 days, as for the reported study NRC 43 for all the IE groups whereas the fuel cycle length depends on the group as shown in table 5-3. The last one period of decay has been imposed equal to 1.5 days because at this time the NDA out-core measurement is performed and afterward information for this project are not useful. Figure 6-1 reports an example of a complete fuel cycle scheme for IE-1, where:

- I-1, I-2 and I-3 are the subsequent irradiation periods
- R-1, R-2 and R-3 are the refuelling periods.

![IE-1 complete fuel cycle](image_url)
This basic scheme of fuel cycle is followed by each group except simulation typology c, where the subsequent scheme of irradiation and decay periods has been changed. However, an incomplete power cycle due to a FA damage has not been taking into account.

Finally, the power basis for each simulation is computed as follow:

\[
\text{Power Basis} = \frac{BU \ [MWd/tU] \times Fuel \ mass[tU]}{Fuel \ cycle \ Length \ [day] \times Number \ of \ irradiation \ cycle} \quad \text{Eqn. 7.1}
\]

6.2.4. **Output**

At the end of each ORIGEN-S a very large output, more or less 25 pages, is written by the code. Thanks to OPUS software, which is included in the scale package, you can select and export the information you want to analyse automatically in a text files.

Not all the file has been saved, but only the parts of interest for this current project; specifically:

- Composition of the fuel at each time steps of the whole period, both irradiation and decay period
- Gamma-ray spectra at each device measurement: CT equal to 1.5 days
- Neutron spectra at each device measurement: CT equal to 1.5 days

The spectra have been calculated and saved only when the device is supposed to operate and so 1.5 days after the finishing of each power period. The value assumed is the same of minimum time which of the EPR reactor\(^{44}\) has to be subcritical before the FA can be extracted.

The whole composition of the irradiated fuel assembly, which contain an enormous number of fission products\(^ {46}\), has not been exported from the output because it is not useful and significant. For this reason only the main and the useful fission products in line with an ORNL study\(^ {45}\) have been exported. Of course a special attention is reserved to the major neutron and gamma emitters. Anyway the total mass of the elements selected corresponds approximately to the 87% of the entire fuel pellets mass. To conclude, the elements chosen in the Excel database are always in the same order and classified for typology as follows:

1) **Actinides:**

   a) \(^{234}\)U, \(^{235}\)U, \(^{236}\)U, \(^{238}\)U
   
   b) \(^{238}\)Pu, \(^{239}\)Pu, \(^{240}\)Pu, \(^{241}\)Pu, \(^{242}\)Pu
   
   c) \(^{242}\)Cm, \(^{244}\)Cm, \(^{245}\)Cm, \(^{246}\)Cm, \(^{247}\)Cm
   
   d) \(^{241}\)Am, \(^{242m}\)Am, \(^{243}\)Am, \(^{237}\)Np
2) Lanthanides:
   a) $^{143}$Nd, $^{144}$Nd, $^{146}$Nd, $^{148}$Nd, $^{150}$Nd
   b) $^{144}$Ce, $^{144}$Pm, $^{147}$Pm
   c) $^{147}$Sm, $^{148}$Sm, $^{149}$Sm, $^{150}$Sm, $^{151}$Sm, $^{152}$Sm, $^{154}$Sm
   d) $^{154}$Eu

3) Volatile fission products
   a) $^{133}$Cs, $^{134}$Cs, $^{135}$Cs, $^{137}$Cs, $^{138}$Cs
   b) $^{85}$Kr, $^{131}$I, $^{133}$Xe

4) Metallic and other fission products
   a) $^{126}$Sb, $^{129}$Sb, $^{90}$Sr, $^{95}$Mo
   b) $^{95}$Zr, $^{95}$Nb, $^{99}$Tc, $^{101}$Ru, $^{106}$Ru, $^{103}$Rh, $^{109}$Ag,
   c) $^{79}$Se, $^{108}$Pd, $^{126}$Sn

All the simulations output files have to be modified to be loaded and elaborated by MatLab software because it can read text files in matrix setup. The exact operation step by step is described in annex F. Moreover a big effort has been done to write a program which is able to load and reshuffle the matrix row in a predefined user order and create a correspondent Excel database. Its setup has been thought in order to allow future developments.

6.2.5. Simulation typology description

As said before, four main types of ORIGEN-S simulation have been done. They are listed as follow:
   a) Initial enrichment
   b) Fuel cycle length
   c) Irradiation history with various combination of power cycles (I) and shut-down periods (D)
   d) Fuel pellet density

Following you they will be described whereas the correspondent code assigned to each simulation output file is detailed in annex F.

6.2.5.1. Initial Enrichment

This simulation typology has been done for all IE and so for fifty FA. An additional discretization in energy for the gamma spectra, one hundred and forty four groups, has been imposed to obtain more realistic spectra to simulate in MCNPX. Because of the ORGINE-ARP interface limitation in very detailed energy group discretization it has not been possible to make only one simulation, as usual,
for each FA. For this reason it has been decided to split the gamma energy spectra in three parts and simulate each separately, joining them by the next elaboration in MatLab. In this way each FA has required three simulations for a whole amount of one hundred and fifty runs.

The FA characteristics imposed in input are those derived from the confidential database analysis. For this reason, the results obtained from these simulations will be assigned the adjective “default”.

6.2.5.2. Fuel cycle length

This simulation typology has been done only for IE-1 and IE-5 to limit the number of simulations maintaining the complete IE range. This study wants to analyse probable differences respect to the default results which can be found changing the fuel cycle length. The variation of the irradiation period has been imposed equal to more or less two months instead of the nominal duration. In the subsequent table the variation cases are shown with the correspondent simulation code:

<table>
<thead>
<tr>
<th>IE group</th>
<th>Default cycle length</th>
<th>Simulation code FCL variation</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>[months]</td>
<td>F1 [months]</td>
</tr>
<tr>
<td>IE-1</td>
<td>12</td>
<td>10</td>
</tr>
<tr>
<td>IE-5</td>
<td>18</td>
<td>16</td>
</tr>
</tbody>
</table>

Table 6-1: Fuel cycle length simulation

For this typology of simulation, forty runs have been done.

6.2.5.3. Irradiation history

As for the last simulation typology, only FA belongs to IE-1 and IE-5 has been used only for to limit the number of simulations maintaining the complete IE range. This study intends to analyse the changes of neutron and gamma spectra compared with the default case modifying the normal sequence of irradiation and refuelling operation. In fact during a typically sequence, a FA after a power cycle is always reinserted in the core whereas in reality that cannot be true. Nowadays the last techniques of in-core fuel management may require not using a FA in two neighbour power cycles leaving it inside
the spent fuel for a whole irradiation period. For this reason all possible variations compared with the default case have been tested and listed in the next table and figure, where:

- \( I \) is an irradiation period in days (green cylinder)
- \( D \) is an extra decay period in days (red cylinder)
- \( R \) is the refuelling operation in days (yellow cylinder)

This typology of simulation has required a big effort; in fact eighty runs have been necessary.

### 6.2.5.4. Fuel pellet density

This typology has been done only for IE-1 and IE-5 to limit the number of simulations maintaining the complete IE range. This study wants to analyse possible differences compared with the default results which can be found changing the fuel pellet density which is normally equal to 95% of the theoretical one (10.95 g/cm\(^3\)). For both IE groups one case has been imposed with FPD equal to 93% and
another equal to 97%, maintaining the default value of the fuel pellet volume. The input data have been described in the next table.

<table>
<thead>
<tr>
<th>Fuel pellet density</th>
<th>Simulation code</th>
<th>UO2 theoretical density</th>
<th>UO2 real density</th>
<th>Mass UO2</th>
<th>Mass U</th>
<th>Mass O</th>
</tr>
</thead>
<tbody>
<tr>
<td>[%]</td>
<td>[g/cm³]</td>
<td>[g/cm³]</td>
<td>[g]</td>
<td>[g]</td>
<td>[g]</td>
<td></td>
</tr>
<tr>
<td>93</td>
<td>P1</td>
<td>10.5</td>
<td>9.765</td>
<td>545563.1</td>
<td>480946</td>
<td>64617</td>
</tr>
<tr>
<td>95</td>
<td>Default</td>
<td>10.5</td>
<td>9.975</td>
<td>557295.6</td>
<td>491289</td>
<td>66006</td>
</tr>
<tr>
<td>97</td>
<td>P2</td>
<td>10.5</td>
<td>10.185</td>
<td>569028.1</td>
<td>501632</td>
<td>67396</td>
</tr>
</tbody>
</table>

Table 6-4: Fuel pellet density simulation scheme

For this typology of simulation forty runs have been done.

6.3. Fuel assembly characterization results

Once ORIGEN-S simulations have been done, all data has been elaborated by MatLab implementing a Best Fit function thanks to the Curve Fitting MatLab Control Box (Annex F). In this sub-chapter only the main results will be showed, whereas a specific treatment for each main neutron emitters and Plutonium isotopes have been reported in annex G.

In the annex F, the methodology and the parameters used during the Best Fit process have been described. In the same annex you can find all the obtained BF parameters values divided for typology of study.

Specifically in this subchapter will be showed:

- The analysis of ORIGEN-S spectrum simulations
- The neutron emissions in function of BU
- The $^{134}$Cs/$^{137}$Cs ratio in function of the BU
- Total fissile material in function of the NE
- Plutonium isotopes amount in function of the NE

6.3.1. Spectrum analysis

The neutron and the gamma fields emitted by the irradiated FA after 1.5 days of CT for each ORGEN-S simulations have been calculated by the depletion code. Once the all the runs have been finished
the data have been loaded and elaborated by the MatLab software by a user-made programs (annex F). Following a gamma and neutron discretized energy spectrums analysis will be done.

The next legend will be applied during this sub-charter:

- R1: spectrum after the first power cycle at a CT equal to 1.5 days
- R2: spectrum after the second power cycle at a CT equal to 1.5 days
- R3: spectrum after the third power cycle at a CT equal to 1.5 days

The following figure examines the neutron spectrum.

![Figure 6-2: Neutron flux generated by irradiated fuel assembly](image)

The first graph on the left indicates the three neutron spectrums of simulation E5O10 at the successive discharge at a CT equal to 1.5 days and so when the out-core measurement is supposed to be made. You can appreciate that:

- The energy discretization applied is not constant but it is finer for energy smaller than 3 MeV.
- The spectrum shape at the different moments is the same. It only varying in magnitude due to the presence of the same radioactive elements in a different amount. It is worth reminding that a neutron element emitters generate always the neutron at the same energy which is related to the Q-value of the reaction.
The gap between R1/R2 spectrum and R2/R3 one is not the same due to the exponential behaviour of total NE (Equ.2.1).

All the successive three graphs show the comparison between spectrals relative to different simulations (E5O1, E5O4, E5O7, E5O10) at different down-load. You can affirm that:

- For more elevated BU, a bigger NE in each energy bins corresponds.
- At R1 the differences between the NE spectrums are very little because the BU difference among simulations is very small. However dividing this gap between all the forty four energy groups, the difference in each energy bin cannot be appreciable in MCNP simulations.
- The situation described for R1 is the same at R2 and R3.
- If the number of energy bins is smaller the NE difference inside an energy bin will be more evident.

The next figure shows the behaviour of the gamma spectrum.

![Gamma Spectrum Figures](image)

**Figure 6-3: Gamma-ray flux generated by irradiated fuel assembly**

The first graph on the left indicates the three gamma spectrums of simulation E5O10 at the successive discharge at a CT equal to 1.5 days and so when the out-core measurement is supposed to be made. You can appreciate that:
- The energy discretization applied is not constant but it is finer for energy smaller than 1 MeV in order to distinguish the $^{134}$Cs and $^{137}$Cs peaks of interest applied in equation 2.2.

- The spectrum shape at different moments is the same but it only changes in magnitude for the same reason discussed for the neutrons.

- A higher BU, and so a successive fuel cycle, corresponds to a higher GE emission. In fact, as the Eq.2.3 explains, the total gamma emission is proportional to the BU.

- The gap between R1/R2 spectrum and R2/R3 one is not the same due to the exponential behaviour of total NE (equ.6.1).

- The difference between the gamma spectrum relevant to different out-core measurements is more appreciable in a bigger energy than in a smaller one as well as discussed for the neutron spectra.

All the successive three graphs show the comparison between spectrum relative to different simulations (E5O1, E5O4, E5O7,E5O10) at different downloads. You can affirm that:

- For more elevated BU, a bigger GE and a higher GE in each energy bins correspond.

- At R1 the differences between the GE spectrals are very little because the BU difference among simulations is very small. However dividing this gap between all the forty four energy groups, the difference in each energy bin cannot be appreciable in MCNP simulations.

- The situation described for R1 is the same at R2 and R3.

- If the number of energy bins is smaller the GE difference inside an energy bin will be more evident.

In order to visually observe the difference between the peaks of $^{134}$Cs and $^{137}$Cs of different BU, we have zoomed the precedent gamma spectrum in two different ways: the first allows noting the 0.6617MeV $^{137}$Cs peak whereas the second the 0.795MeV $^{134}$Cs one. The two peaks are highlighted by a light blue arrow.
Examining the two precedent graphs we can affirm that:

- The Cesium peaks differences between successive power cycle are clear and appreciable.
- The Cesium peaks differences between FA inside the same power cycle are close to be not appreciable.
- The peaks increase with the BU.
These considerations are very important for the IES MNCP simulations, and for similarity to others groups, because they allow to limit the number of runs only to some simulation (i.e. E5O1, E5O5, E5O8 and E5O10). The values of the not mentioned simulations will be obtained for extrapolations.

Due to the very fine gamma energy group discretization to obtain a good MCNP it is necessary to simulate huge number of particles which require a very elevated run time considering the elevated number of cells inserted and their volumes.

6.3.2. Neutron emissions in function of Burn-up

One of the more important device’s objectives is to determine BU in function of the neutron emissions. Following one graph for each main simulation category will be showed.

This first graph shows the relationship between BU&NE varying the IE. All the IE follow in very accurate way a power relationship, as follow:

\[ y = a \times x^b \]

where:
- \( y \) is the NE [number of neutrons emitted/second]
- \( x \) is the BU [MWd/tU]
- \( a \) is a constant very similar for each IE apart from very low IE
- \( b \) is a constant which is proportional to the IE and depends also from the FCL

The goodness of the BF, which is measured by the Adjusted R-Square (annex F), increase with the IE but it is always very high and greater than 0.999 where 1 represents the perfect correlations. These results confirm the quality of Equ.2-6 suggested by IAEA.

Finally, the different marker colour indicates the diverse IE, whereas a typology of marker highlights the different fuel cycle as follows:
- * \( \rightarrow \) First cycle
- * \( \rightarrow \) Second cycle
- 0 \( \rightarrow \) Third cycle
Moreover in the previous figure, you can note:

- Each FA follows the BF corresponding correlation
- The three different fuel cycles in each IE
- The NE increase with IE and BU
- The mayor differences of calculated values from the BF are placed in the connection between the different fuel cycle
- Difference in the initial $^{235}\text{U}$ enrichment affect significantly the NE

All the further parameter studies in this sub-charter will be referred to the BF correlation just encountered indicated by a light blue line.

This next graph shows the relationship between BU&NE varying the FCL differencing into IE1 and IE5. In both two IE not a very huge differences has been encountered even if the IE1 shows a bigger absolute divergence compared with the IE5.
Moreover you can note that the E1F1 simulations produce a less NE compared with the E1F2 ones due to the presence of a mayor amount of $^{242}$Cm with increases during the irradiations. This behaviour is more evident in IE5 due to the higher BU and the greater amount of $^{242}$Cm with cause a smaller relative difference compared with BF correlation.

In the following graph the results of fuel pellet porosity parameter sensitiveness are showed. The variation of FPP does not influence significantly the NE even if a greater divergence can be encountered in IE1 due to the smaller BU. Moreover, the NE related to a P2 simulation shows a slight increment compared with the P1 simulation thanks to a more elevated fuel pellet density.
The following figure shows the behaviour of NE in function of difference irradiation history.

You can appreciate that a different irradiation history is one of the most important parameters affecting neutron source terms, particularly for short cooling time $^{14}$. The NE is dramatically altered for both IE by IH even if the BU is identical. The principal cause of these perturbations is the build-up...
of the $^{242}\text{Cm}$. Plutonium-241 is precursor of $^{241}\text{Am}$, which capture a neutron and produce $^{242}\text{Am}$, which decays with 16.01 hours to $^{242}\text{Cm}$. For a significant downtime, the $^{241}\text{Am}$ inventory increases, thereby a production of $^{242}\text{Cm}$. The resulting influence upon the relative neutron rates becomes less important as the cooling time increases. For this reason the biggest difference have been shown for the double downtimes applied after the second irradiation cycle.

6.3.3. $^{134}\text{Cs}/^{137}\text{Cs}$

Another method utilized to measure the BU in the project innovative out-core device is the ratio between the amount of $^{134}\text{Cs}$ and $^{137}\text{Cs}$ which can be determined by the gamma spectrometry. In the following figure this rate is shown for each IE.

![Figure 6-10: Cesium-134/Cesium 137 ratio vs. burnup](image)

All IE follow the next BF correlation:

$$y = a \times x^b + c$$  \hspace{1cm} \text{Equ. 7.3}$$

where:

- $y$ is the ratio between the mass of $^{134}\text{Cs}$ and $^{137}\text{Cs}$
- $x$ is the BU
- $a$, $b$ and $c$ are constants; $a$ and $b$ are always positive whereas $c$ is always negative. It is showed that the absolute value of $a$ is enormous compared with $b$ one.
The parameter sensitiveness has shown that the ratio between the mass of $^{134}$Cs and $^{137}$Cs:

- Mostly increases with the BU but not with the IE
- It is very dependent on the IH due to the $^{134}$Cs characteristic which is very sensitive to shutdown periods because of its very short half-life (2.6 years)
- It is significantly dependent for the IE
- It is not influenced from the fuel pellet density
- It is slightly affected by the FCL due to the very short half-time life of $^{134}$Cs (2.6 years) compared with $^{137}$Cs ones which is 30.1 years
- Adjusted R-Square (annex F) is around 0.997 for all IE and so it is not very good. This fact can be highlighted noting the difficult of the BF to represent in a correct way ratios belong to different fuel cycle.

To conclude, you can affirm that knowing the IE, it is possible to determine the BU by the measure of ratio of gamma detector counts relative to selected peaks of $^{134}$Cs and $^{137}$Cs. However, how NRC highlights $^{1}$, the BU determined is less accurate that one calculated by NE/BU correlation but it will be useful for a further BU verify.

6.3.4. **Total fissile material**

In the next figure the NE in function of the total fissile materials for all IE has been plotted.

![Figure 6-11: Total fissile material vs. burnup](image)

All IE follow the next BF correlation:

$$y = a * x^b + c$$  \hspace{1cm} Eq. 7.4
where:
- \( y \) is the NE
- \( x \) is the per cent of total amount of fissile material, composed by \(^{239}\text{Pu}, \, ^{241}\text{Pu} \) and \(^{235}\text{U} \), respect to the initial heavy material mass
- \( a, b \) and \( c \) are constants which depends strongly from the IE; \( a \) is always positive whereas \( b \) and \( c \) are always negative. It is showed that the absolute value of \( c \) is enormous compared with \( b \) one.

The parameter sensitiveness has shown that the total fissile material:
- Decrease due to BU increcent as normal
- It is not altered by variation of IH and FCL
- It is barely affected by the FPP in a mayor way for greater IE
- Adjusted R-Square (annex F) is always bigger than 0.999, therefor the accuracy of BF is very good. The IE5 BF has the best accuracy.

To conclude, you can affirm that knowing the IE, it is possible to determine the total fissile amount by the measure of neutron detector counts.

6.3.5. \(^{137}\text{Cs}\)

Cesium-137 is a very interesting isotope useful to measure the BU due to its linear behaviour. In fact as you can appreciate in the next figure a unique BF can be applied for all the IE. Moreover thanks to its quite long half-live time, 30.1 years, it is slightly sensitive to the IH, the FPD and the FCL.

![Figure 6-12: Cesium-137 vs. burnup](image)
To conclude measuring the amount of $^{137}$Cs by a gamma spectroscopy the typology of FA, fresh or irradiated, can be determined.

6.3.6. **Plutonium isotopes**

In the following two figures are shown the relation of each Plutonium isotope with the NE.

It is worth remembering that if the BF goodness of the fit is more or less the same, one single correlation related to all IE has been preferred to one for each IE, in order to limit the number of equations needed for the analysis.

For this reason only for $^{240}$Pu a unique correlation which is related to all IE has been made. All the enormous work done is reported in *annex G* whereas in the next figure the BF IE5 correlation is reported.

![Plutonium isotope vs. burnup](image)

**Figure 6-13: Plutonium isotope vs. burnup**

The parameter sensitiveness has shown that:

- $^{238}$Pu, $^{241}$Pu and $^{242}$Pu have a power behaviour in function of the BU
- $^{239}$Pu and $^{240}$Pu have an exponential behaviour in function of the BU
- All the Plutonium isotopes increase with the BU
- All the Plutonium isotopes are sensitive to the IE with the exception of the $^{240}$Pu
- You cannot define a behaviour of constants involved in the BF
The accuracy of the BF correlation are good, exceptional for the $^{238}\text{Pu}$ where the adjusted $R$-value is equal to one, but very poor for the $^{239}\text{Pu}$.

The amount of Plutonium isotopes are not affected by the IH, the PFD and the FCL.

The follow figure shows the $^{240}\text{Pu}$ trend in function of BU. As discussed before a unique correlation has been used for all IE thanks to the good accuracy (Adjusted $R$-value equal to 0.998). Furthermore you have preferred to apply a BF correlation which approximates in a very good way the $^{240}\text{Pu}$ amount at high value of BU to allow a very precious inventory for spent FA in the SFP.

![Figure 6-14: Plutonium-240 vs. neutron emission](image)

To conclude, you can affirm that knowing the IE, it is possible to determine directly in accurate way all the Plutonium isotope with the exception of $^{239}\text{Pu}$. In order to determinate the amount of $^{239}\text{Pu}$ correctly you suggest to apply the following equation:

$$m_{239\text{Pu}} = m_{\text{total fissile materials}} - m_{241\text{Pu}} - m_{235\text{U}}$$  \hspace{1cm} \text{Equ. 7.5}$$

where $m$ is the mass.

### 6.4. Database

As previously discussed, thanks to a MatLab program, called *Program_database (Annex F)*, 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-S 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-S information in the same place and not in different text-files as the OPUS 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 6-15: database scheme](image)

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 OPUS gives us. This operation is done by a sort of complex row reshuffling. In fact the default isotopes hold in OPUS 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-S 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 annex F, all the Excel file which compose the Excel
database are listed whereas the whole Excel database is reported only in digital format for reason of space.

7. Selection of the device placement

Once the ORIGEN-S simulations have been done and elaborated, the device placement can has been chosen. In order to avoid any influence on the loading/unloading schedule and any additional fuel displacements as specified by NRC\textsuperscript{1}, the nondestructive measurements must be accurately located. For this reason a refuelling ASCO-I operations has been deeply analysed (annex E).

Moreover the device placement taken into account:

- Easy access to the place for periodically maintenance
- A constant FA position during the measurements in order to obtain a fixed geometry factor of the detectors
- The presence of electrical connection
- The placement of NAJA device\textsuperscript{13}
- The presence of an elevate gamma and neutron radiation field which could affect the measurements
- The magnitude of gamma and neutron spectrum

Besides, the selection must be not cause:

- An additional effective dose to the workers
- A great modification of the building structure
- Fuel assembly damages

Thanks to the refuelling analysis, three possibly placements have been selected:

- Fuel tunnel channel which connect reactor building to fuel building
- The spent fuel pool
- Transfer channel

The summary of advantages and disadvantages of those placements is resumed in the next table.
<table>
<thead>
<tr>
<th>Place</th>
<th>Advantages</th>
<th>Disadvantages</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel tunnel channel</td>
<td>– Dry zone possibility for detector equipment</td>
<td>– Hard access</td>
</tr>
<tr>
<td></td>
<td>– No additional dose to the workers</td>
<td>– Very strong radiation due to the lack of shielding</td>
</tr>
<tr>
<td></td>
<td></td>
<td>– Lack of electrical connections</td>
</tr>
<tr>
<td>The spent fuel pool</td>
<td>– Possibility to increase the time of the measurements</td>
<td>– Presence of an elevated gamma and neutron field due to the presence of irradiated and spent FA</td>
</tr>
<tr>
<td></td>
<td>– Dry zone for detector equipment</td>
<td></td>
</tr>
<tr>
<td></td>
<td>– Fixed position of FA</td>
<td></td>
</tr>
<tr>
<td>Transfer channel</td>
<td>– Fixed position of FA: horizontal at the exit of fuel tunnel transfer</td>
<td>– Wet zone constriction for most of neutron and gamma detector equipment</td>
</tr>
<tr>
<td></td>
<td>– Possibility to modify the magnitude of gamma and neutron field, increasing the detector distance from the FA</td>
<td></td>
</tr>
<tr>
<td></td>
<td>– Presence of electrical connections</td>
<td></td>
</tr>
<tr>
<td></td>
<td>– No additional dose to the workers</td>
<td></td>
</tr>
<tr>
<td></td>
<td>– Enormous freedom in the detector placement</td>
<td></td>
</tr>
<tr>
<td></td>
<td>– Easy access during the power cycles</td>
<td></td>
</tr>
</tbody>
</table>

Table 7-1: Possible device placements

The transfer channel has been preferred due to the great amount of advantages. However, due to the presence of water inside the transfer channel during the outage operation, two special “water proof boxes”, one for each detector equipment, have been projected.

In order to obtain a fixed position of the FA during the nondestructive measurements, it has been decided to make the detection measurements when the FA is arrived to the end of the runway tracks of the fuel transfer facility trolley. At this point, the FA is located horizontally inside the stainless steel container.
7.1. Transfer channel

During the refueling operations the transfer channel is filled by borated water (2600 ppm, 70°C, pH 7). The walls are made by concrete whereas the FA container fuel transfer facility by SS 316.

Thanks to the “Informe final de Seguridad” of ASCO-I NPP, the real dimensions of the transfer channel have been rebuilt. To optimize the duration of a MCNP simulations, the modeled geometry has been limited using the neutron mean free path. In fact where necessary, the concrete wall thickness has been supposed equal to five times the neutron mean free path, whereas for the water twelve times from the outer border of the FA.

The determination of the concrete and borated water m.f.p. is reported in the annex H.

8. Choice of the special detector equipment

Once the ORIGEN-S simulation has been done and elaborated, the equipment has been chosen. The magnitude of gamma and neutron spectral conditions the selection of the detector equipment and his placement inside the transfer channel.

As said before, the innovative out-core device is composed by the main parts:

- Gamma detector equipment
- Neutron detector equipment
- Video control system

8.1. Gamma detector equipment

The gamma detector equipment, a difference of the neutron one, has to operate also a spectrometry.

Nowadays you can obtain a spectrometry with an HPGe, with a NaI(Tl) detector or other scintillator. Unfortunately the HPGe, which provides the best resolution performance and the fastest decay time, requires operating at cryogenic conditions. Due to this necessity, the HPGe detector option has be refused because it is not reasonable applying it in this device.
Discarded the HPGe hypothesis, the LaBr$_3$(Ce) has been preferred to the NaI(Tl) because provides a better resolution performance by approximately a factor of 2 $^{60}$. Note that neither the NaI(Tl) detectors nor the lanthanum bromide detectors can approach the resolution of an HPGe detector. The efficiency for LaBr$_3$ is about 1.3 times that of NaI(Tl) for the same volume and the decay time constant is slightly more than 10% of the NaI detector decay time. On the basis of photoelectron yield, LaBr has higher efficiency and temperature stability than NaI(Tl).

Moreover, the smallest available crystal dimensions, 1”$\times$1”, has been selected to improve resolution and because the enormous gamma field present allows using a small size detector without any problems. A description of scintillator detector and pulse processing is reported in annex I.

Basically, the gamma detector equipment has been selected taking into also account the placement (chapter 7). The commercial selected components are listed next:

- LaBr$_3$ detector 1”$\times$1” ORTEC
- Pre-amplifier: AS2612 SAINT-GOBAIN
- Amplifier: 572 A ORTEC
- Multichannel analyser: ASPEC-927A ORTEC
- High voltage supply: 660 ORTEC
- NIM case: 4002E ORTEC

The gamma equipment will allow determining for each FA:

- $^{134}$Cs/$^{137}$Cs ratio for equ.6.2
- The TGE of a delimitate energy range of the gamma spectrum which is proportional to the whole TGE equ.6.3

8.1.1. **Waterproof box**

All the gamma instrumentations except the LaBr$_3$ detector and its pre-amplifier will be placed outside the transfer. In this way, the waterproof box is need only for these two components. This choice allows to a simpler maintaining and limit the possibility of damages only to the detector if the waterproof box will be broken or perforated.

Considering the absence of hard environment conditions (water 50°C, pH 7) the Aluminium has been chosen for the WB. Thanks to the low atomic number of the Al, Z=13, a low BS production will be as results from the following equation $^{56}$:
where:

- $N_a$ is the Avogadro number
- $Z$ is the atomic number
- $H$ is the Mark Plank’s constant
- $C$ is the light velocity
- $m_e$ is the electron mass

The WB trace out the shape of the LaBr$_3$ detector and its pre-amplifier. It contains also a PVC support which allows maintaining the correct horizontal position.

The technical drawings have been reported in the annex whereas the modelled of the WB and the LaBr$_3$ detector is in the MCNP modelling sub-chapter.

8.1.2. **Placement**

As mentioned in the first chapter, one important characteristic of the gamma-ray is the self-attenuation within the fuel assembly. Therefore, measurements are primarily planned for the exterior rods of the FA. For this reason the position choice for the detector will allow to measure two faces of the fuel assembly and geometry factor equal to one. In this way a more representative measure of FA feature will be obtained. In this sense of view the fuel transfer channel side with the fuel elevator has been discarded.

However, the big problem encountered is the enormous gamma flux which for instance is equal to $5.04 \times 10^{19}$ photons emitted for seconds for simulation E501 at recharge one. It is worth reminding that nowadays the modern electronic can process a maximum of one signal per nanosecond. A possible position, with a compatible gamma flux, has been calculated thanks to an approximate iterative model which schematizes the source as linear, the detector as point and the borated water as a slab shielding.

The distance obtained between the FA axis and the detector is equal to 422 cm the whole iteration process has been described in **annex J**.
Once this distance has been obtained, you need to check if the photons produced by the radioactive decay of $^{137}$Cs and $^{134}$Cs could reach that distance and produce a useful signals.

The results obtained, for the selected peaks, have been listed in the following table:

<table>
<thead>
<tr>
<th></th>
<th>Intensity at the detector</th>
<th>Time required for 1000 pulse in the spectrum</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{134}$Cs</td>
<td>0.735</td>
<td>1360</td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>0.735</td>
<td>1360</td>
</tr>
</tbody>
</table>

Table 8-1: Cesium peaks intensity at the detector

Also assuming that each Cesium photon in the detector will produce a count, this detector position is not compatible with the feature of this typology of out-core device which require at maximum an overall time of maximum 5 minutes for the whole out-core measurements.

For this reason the detector can be placed nearer the FA but a special and very expensive electronics is needed. In particular you are talking about a multichannel which incorporate inside a single channel to discriminate signals before process them. In this way a better flux can be provided to the gamma detector and a nearer position can be applied.

The position choice allows obtaining the $\gamma$-rays for two edges of the FA due to the 45 degree position obtaining a meaningful signal. In the next figure a section of a part of the transfer channel is showed. In the middle you can denote the FA modelled and the container; the gamma detector is denominated GD.
It worth reminding that, a difference of the neutron equipment, the LaBr3(Ce) detector position will not be fix, but it will be changed several times along the FA active length to obtain the BU profile with equation 2-3. In stops the TGE(γ) will be measured whereas the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio and the TGE will be the overall sum of the values measured in each position.

The time request for each independent spectrometry is function of:

- The number of the gamma-detector position along the active length
- The ratio between the Cesium peaks and the background necessary to validate the operation decided during the test phase. This is a sort of critical limit using the Currie Criterion. For example, it can be assumed equal to five
- The overall availability of time for the overall out-core measurement

Unfortunately, the background of the placement will be unknown up to the calibration equipment phase. For this reason, at this stage of the project the single spectrometry time cannot be defined.

To conclude, the LaBr3(Ce) detector will be moved horizontally on runway tracks by a stainless steel rigid pusher chain driven by an electric motor located on the service floor.

However, from the theoretical point of view, the LaBr3 energy scale must be calibrated before the first out-core measurement. Calibration is performed by using the peaks of a known source, such as Europium-154 or cobalt-60. Because the channel number is proportional to energy, the channel scale can then be converted to an energy scale.

Because of the presence of radioactivity in the transfer channel (i.e., background radiation and presence of radioactive and active material), the spectrum should be analysed when the FA is not present. The background radiation must then be subtracted from the actual measurement.

### 8.2. Neutron detector equipment

The choice of the neutron detector equipment has considered several factors which are listed following:

- The presence of an intensive gamma field
- The necessity of a good detector efficiency in order to limit the time needed for the neutron detector measurement
- The spectrum of neutron field generated by the irradiated FA which is composed by thermal and fast neutrons
- The thermalization of neutrons due to the collisions in the borated water which cause that the average energy of the neutron flux decreases form to 2 MeV to around 30 KeV for the selected neutron detector position
- The necessity only of a number of counts proportional to the neutron flux and not of a spectrometry as for the LaBr$_3$ detector.

Thanks to the elevated detector efficiency at energy around 30 KeV, the low cost and the elevated ability of discriminate gamma rays from neutrons the BF$_3$ has be selected. In fact you should expect that the most gamma-ray interactions will result in low-amplitude pulses at energy smaller than the alpha produced one $^{65}$ (annex H). In this way a simple amplitude discrimination can then easily eliminate these gamma rays without sacrificing neutron detector efficiency.

It is worth reminding that the gamma-ray due to its characteristics does not interact directly with the proportional gas, but it primarily in the wall of the counter creating secondary electrons that may produce ionization in the gas. Because the stopping power for electrons in gases is quite low, a typical electron will deposit only a small fraction of its initial energy within the gas before reaching the other wall of the counter. For this reason, the amplitude of these pulses is low compared with the alpha generated by the neutron capture which produces a count at least an energy bigger than 0.84 MeV. The elevated energy of particles emitted produced in the neutron capture by BF$_3$ derive from the large Q-value that is around 2.3 MeV.

The neutron detector will work in the source range and for this reason a pulse mode has been choice.

Basically the neutron detector equipment components selected are listed next:

- BF$_3$ detector: 2021 LND
- Pre-amplifier: 142 IH ORTEC
- Amplifier: 590 A ORTEC
- Multichannel analyser: ASPEC-927A ORTEC
- High voltage supply: 660 ORTEC
- NIM case: 4002E ORTEC

Specifically you have selected the 2021 LND BF$_3$ detector because it is proportional gas is enriched at 96% of $^{10}$B which allow reducing the time of detector counting thanks to the higher reaction rate.
Finally, it has been selected the ASPEC-927A ORTEC multichannel analyser even if single channel analyser will be sufficient because it is shared with the gamma equipment. In this way using a double enter multichannel you have saved money comporting a cheaper investment. For the same reason, the 660 ORTEC double high voltage supply and the NIM case have been selected.

8.2.1. Waterproof box

The waterproof box, which allows keeping dry the BF$_3$ Neutron Detector, is composed basically of two parts:

- The external Aluminium box
- A PVC which keep in a fix position the detector

The Aluminium has been preferred to other materials because:

- It produces a low amount of Bremsstrahlung $equ.8.1$
- Not special mechanical and anti-corrosive characteristics are required for this environment (borated water at 50°C, pH 7).

The application of PVC in the detector support allows to low rate of photon production form BS, a low cost and a small load stress induction.

Furthermore, in addition to the detector water box an ulterior Aluminium box, internal shielded by Lead, has been positioned for the allocation of the pre-amplifier 142IH.

Locating the pre-amplifier closer to the detector and so inside the transfer channel is you obtain to reduce the rebound of signal inside coaxial cables and the signal noise.

The pre-amplifier WB has been placed below and centred to the neutron detector Thanks to this arrangement you have maximized the neutron flux at the detector and minimized the possible perturbations in the pre-amplifier. Reminding that the electronics is perturbed by the neutron flux $^{23}$ the presence of a Lead shield and of the detector between the 192IH and the FA is very useful.

The technical drawings have been reported in the annex K whereas the modelled of the WB and the BF$_3$ detector are placed inside the MCNP modelling sub-chapter.
8.2.2. Placement

To decide the neutron detector position you have considered:

- The neutrons can travel short distance (maximum 2 m) inside borated water due to the neutron capture high rate of absorption of $^{10}\text{B}$
- A presence of source range neutron flux
- The very important axial importance of BU (annex C, ref. 16)
- Absence of strong upper limitation in detector counts thanks to the detector selected
- A major detector count rates allows obtaining a BU measure in smaller time decreasing the cost of the operation.

In order to facilitate maintaining and limit the damage probability due to WB perforations, only the BF$_3$ detector and the WB has been placed inside the transfer channel, whereas all the other neutron instrumentation are placed outside the transfer channel in a dry zone.

Considering the factor listed before, the neutron detector has been placed at the opposite wall of FA elevator at the same high of the FA in the central position. Thanks to this arrangement the geometrical detector yield is equal to one. Next, the 2D disposition is showed whereas the 3D drawing is in the subchapter 9.5.

---

Figure 8-2: BF$_3$ placement
The described position of the neutron detector, ND, will be fix during all the measurements.

8.3. Video control system

The video system has to read the FA number in a very hard condition:
- Very high gamma ray flux
- High neutron flu
- Sumerge in borated water
- Borated water temperature: 50ºC
- Low visibility

The equipment selected belongs to the Ahlberg Electronics and it is basically composed by an underwater camera for high radiation area, a control unit for the camera and its remote control and an underwater light. Thanks to the repetitive but fix position of the FA during the measurements no movements of the camera are needed. In this way, the equipment and the given software can read automatically the FA ID number even if the presence of an operator during the implement of the system is suggested.

Specifically the equipment is composed:
- N71Z underwater camera for high radiation areas
- S40-35 Underwater light with bracket
- R150 remote control camera (left side), P150R Camera control unit (right side) and a monitor which can be stored in the camera rack

Figure 8-1: Video control unit
To attenuate the strong condition over the N71Z camera a far placement has been choice. The N71Z camera and the light will be placed at 2 m from the up edge of the FA in the floor of the transfer channel whereas all the monitor and the hardware will be placed in the fuel building in a dry zone. For this reason, a cable of 20m will be provided.

9. Overall out-core device scheme

The overall out-core device scheme is schematized is the following figure. The yellow boxes indicate that the component is placed in the transfer channel inside water, whereas the light blue ones indicate that the module is positioned in a dry zone.

As you have discussed before some components are shared by the gamma and the neutron equipment. All components selected are listed next:

- 1 → Pre-amplifier: 142 IH ORTEC
- 2 → Amplifier: 590 A ORTEC
- 3 → Pre-amplifier: AS2612 SAINT-GOBAIN
- 4 → Amplifier: 572 A ORTEC
- 5 → Multichannel analyser: ASPEC-927A ORTEC
- 6 → 20m video camera cable (it has not been plotted)
- 7 → BF₃ detector: 2021 LND
- 8 → LaBr₃ detector 1”x1” ORTEC
- 9 → S40-35 Underwater light
- 10→ N71Z underwater camera
- 11→ R150 remote control camera (left side), P150R Camera control unit
- 12→ High voltage supply: 660 ORTEC
- 13→ PC
- 14→ NIM case: 4002E ORTEC

The datasheet of each component, as well as THEIR technical drawing, has been reported inside annex K.

10. MNCP components modelling

Due to the normal computer available for the MCNP-X calculation and the very low detection efficiency, a very large amount of particles (i.e. 10⁸) has to be shot. Moreover the presence of gamma and neutron fields requires the simulation, at least, of both two particles and the “mode p n”. In this way MCNP will follow both particles for the energy deposition but an additional time is required. Considering also the very big volume of the transfer channel, 900 m³, and the FA very complex geometry, the time required for a simulation could reach more days.

In order to limit the simulation time in a reasonable way, maximum twenty four hours, some assumption and techniques have been tried.

An advanced MCNP geometry has been applied to the FA but without success, due to the limitation of MCNP in very complex geometry. For this reason you have been constrained to homogenized the FA because for only reproducing the FA with basic geometry at least 30 pages of codes will be required and specifically:

- 905 superficies
- 1520 cells

An input file like this, it is impossible to be run by a normal computer because several days can be required. You would like to remember if a more advanced computer way available, 30 pages MCNP
could be loaded and run without any problems. It is worth reminding that the MCNP license is property of the UPC physical and nuclear department.

Besides, also for limit the run time not all the transfer channel has been simulated but only the portion of interest, the Bremsstrahlung has been switch off by physics card. Due to the presence of low atomic number materials near the gamma detector this physic assumption is not poor. Moreover, the MCNP model which generates this peculiar radiation is not the true one but only an approximation by a simple formula.

The assumptions done have a drastic impact on the runtime of the problem, cutting up to 50% of the simulation time. In this way simulating $10^8$ particles the simulation time is more or less 15 hours.

In the next subchapters the modelling of all the components will be described.

### 10.1. Fuel Assembly

The modelling of the FA has required an enormous effort due to its complexity and the limitations of MCNP about describing very complex geometry. The spacing grids, the low and top nozzles have not been taken into account because they can be neglected due to their low importance in MNCP calculations. These have been done in similarity to a lot of criticality studies.

In the following scheme is illustrated all the process actuated.

---

**Figure 10-1: Fuel assembly modelling process**
Firstly, you have tried to insert the FA as heterogeneous composing it only by the fuel pellets, the pressurized Helium inside the fuel pins, the cladding and of course the moderator. As commented before the way E.1 cannot be ridden because of the computer features whereas the repeated-structures or advanced geometry has been tested.

It is worth remanding that the primary goal of the repeated-structures capability is to make possible to describe only once, the cells and surfaces of any structure that appears several times in geometry. Using an advanced geometry, the amount of computer memory needed will be extremely reduced and in this way the MCNP program could be run.

MCNP basically contains two commands for compose advanced geometry.

The lattice command allows reproducing lattice of a fundamental cell in a hexagonal or square way but one limitation of this modelling method is that no geometry can exist outside the lattice elements. Because lattice elements that are not defined do not exist, and because particles are killed when they exit elements that are defined, no geometry can be described other than the universe 0 lattices. For example, a concrete wall cannot be described next to the FA. The lattice itself, filled with the appropriate cells, must be the only geometry in the problem. For this reason the lattice command cannot be used in this project whereas it could be useful for determine critically of a core.

Instead the fill command capability extends the concept of an MCNP cell. The user can specify that a cell is to be filled with something called a universe. A universe is either a lattice or an arbitrary collection of cells. A single universe, described only once, can be designated to fill each of any number of cells in the geometry. Some or all of the cells in a universe may themselves be filled with universes. One a universe is created it can be copy and translated by the like but command. The fill command has five very important limitations:

- A universe cannot be translated more than nine hundreds ninety nine times
- The user can create nine universes at most
- Two different universes must not have a surface or more in common
- The surfaces of a filled cell and surfaces of the filling universe must never coincide
- The source can be uniquely defined as part of the universe cell like a sampled source in a complex cell

The capacity of fill card has been tested successfully in program FA2x2_E.i which contains a 2x2 PWR FA (annex H).
Once all the characteristics of fill card have been understood and assimilated, you have tried to apply the more universe concept to the whole FA in program FA17x17.E.i (Annex H). Basically, two different universes have been defined and moved to each FA positions. One is the fundamental FA cell composed by moderator, fuel pellet, Helium and cladding whereas the second universe cell is the tube guide cell. It is composed by the cladding and the moderator. A section and an axonometric view of the FA are showed in the next figure. In the right part, thanks to the no representation of the water and the FA container, you can note the special disposition of the 25 unfuelled pin.

Unfortunately this program failed when runs due to the too low sampling efficiency in whole FA cell\textsuperscript{3}. The program FA2x2.E.i works only because the sampled zone respect to the FA17x17.E.i is smaller and so the sampling efficiency is sufficient. The unique solution to this problem will be define a point central source but it would be a poor assumption. For this reason the E.2.2.1 option has been discarded.

The last technique possible to improve the sampling efficiency, E.2.2.2, tries to disassembly the two universes before built, creating three smaller ones: one for the fuel pellet, one for the cladding and one for the moderator. The Helium has been neglected to avoid the maximum number of translated cells. Unfortunately this program cannot work again due to the overlapping of the border of two different universes.

The whole process described, which has required months of hard work, has comported to leave the heterogeneous FA modelling in favour of the homogenized one due to the MCNP limitation to describe repetitive and complex geometry. For this reason, you have to take in consideration to utilize another MCNP program more suited for this specific application for future developments of this project.
The FA homogenization has been done in a deterministic way maintaining the external dimensions of the fuel, height and side, and the total mass of each element. Specifically it has been divided in two zones:

- The fuel zone which is composed by the a homogenized mixture of the cladding, the fuel pellets and the helium
- The moderator zone which is constituted only by borated water

The following equations have been applied to calculate the fuel zone weighted density and diameter of the homogenized FA:

\[
\rho_{H,\text{Fuel zone}} \text{ [grams cm}^{-3}\text{]} = \frac{\sum_{i=1}^{n} \rho_i * V_i}{\sum_{i=1}^{n} V_i} \quad \text{Equ. 10.1}
\]

\[
d_H = \frac{\sqrt{\sum_{i=1}^{n} V_i}}{H * \pi} \quad \text{Equ. 10.2}
\]

where:

- \(\rho_i\) is the density of the element \(i\) which composes the fuel zone [grams cm\(^{-3}\)]
- \(V_i\) is the volume of the element \(i\) which constitutes the fuel zone [cm\(^3\)]
- \(H\) is the height of the active FA length [cm]

A weighted density equal to 9.41 g/cm\(^3\) and an equivalent diameter of 15.45 cm have been obtained.

The source has been set up as:

- Cylindrical and correspondent to the fuel zone cell
- Homogenized as possible due to the utilization of a zero dimensional depletion code. No radial or axial profile has been imposed.
- Multi-particles: gamma and neutrons are emitted in the same probability
- Multi-energy: you have imposed two different spectrum sources one for the gammas and another for the neutrons. The energy discretization is that imposed for ORIGEN-S simulations whereas the probability of each particle to be emitted in a single energy bin has been obtained thanks to the relative source intensity
- No isotropic: it emits only \(2\pi\) in the direction of the detection equipments. Thanks to this assumption, the results will have lesser residual errors due to the bigger amount of useful particle shot.
No problem of sampling efficiency has been encountered.

Following some words about the homogenised source profile.

The assumption done is not correct, however it is the only possibility in this project. In fact, as discussed in the annex G, ORIGEN-S is a zero-dimensional depletion code and it does not calculate the emission profile in function of the radium and the height but only an average value. To implement the BU profile in the MCNP maintaining as depletion code ORGEN-S, an enormous numbers simulations will be needed.

Specifically, it would be necessary to divide the FA active length in some parts, i.e. twenty, applying to each a correspondent neutron flux and compute the relative BU. However, in this way the BU will be the output and not the input as in this project. For this reason the axial irradiated flux profile cannot be implemented in MCNP, therefore the BU profile determination capacity of this innovative device cannot be tested. However, it will be equally implemented and discussed.

For these reasons the gamma detector equipment will be simulated only in the middle of the active length of the FA to maximize the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio. Choice another point is possible but it only decreases the $^{134}\text{Cs}/^{137}\text{Cs}$ ratio, not the meaning of the simulation.

Future developments of this project have to into account the possibility to change the depletion code or the process in order to simulate also the axial gamma spectrum profile. To implement this feature, the following softwares can be used:

- KENO VI (SCALE) which can compute the spectrum source in function of the active length of the FA
- NEWT 2D (SCALE) which can determinate the radial variation of flux emitted, both gamma and neutron, thanks to the application of discrete ordinates techniques
- TRITON (SCALE) which utilizing several softwares in a iterative way, it can compute the axial flux uploading the FA composition and so the microscopic cross sections
- ATTILA TRANSPIRE 3D which use the FEM method to calculate the axial irradiated FA profile.

In conclusion, it is worth reminding that the modelled FA is contained by a SS316 box which is implemented in the fuel transfer facility (annex E). This component allows to the FA to be transported horizontally inside the fuel transfer tube and to be swung to a vertical position once the out-core measurements are finished.
10.2. Gamma detection equipment

The part of gamma detection equipment which enters in the transfer channel modelled zone is constituted by the LaBr$_3$ detector and its pre-amplifier.

The follow component has been modelled:

- Photomultiplier photocathode of Bialkali (1)
- The photomultiplier Borosilicate glass cover (2)
- The Aluminium detector house (3)
- The Aluminium box (4)
- The PVC support (5)
- The pre-amplifier (6)
- The Lanthanum Bromide cristal (7)

In the left side of next figure a 2D axial section of the modelled done is shown whereas in the right one a VISED 3D plot made is reported.

![Diagram of LaBr$_3$ detector MNCP modelling](image)

Figure 10-3: LaBr$_3$ detector MNCP modelling

The detector WB box has been plotted in the wireframe mode in order to show the internals.

The LaBr$_3$ detector spectrometry is got imposing a tally F8, pulse height tally detector, in the crystal. In order to obtain a more real gamma spectrum a geb command elaborated experimentally by NERG research team has been implemented in the MCNP input.
10.3. Video control system

The MCNP modelling is not necessary because the camera is positioned outside the simulated transfer channel volume.

10.4. Neutron detection equipment

The BF3 detector, the amplifier and the amplifier WB has been modelled because enter in the transfer channel modelled zone.

Basically the modelling is composed by:

- The Aluminium box (1)
- The Aluminium detector house (2)
- The BF3 (3)
- The PVC support (4)

The effective neutron detector count is obtained from the number of collision is the Boron trifluoride. The detailed process is described in annex H. In the left side of next figure a 2D axial section of the modelled done is shown whereas in the right one a VISED 3D plot, which includes also the pre-amplifier water prove box, is reported.

The detector WB box has been plotted in the wireframe mode in order to show the detector house.
10.5. Transfer channel

The modelled transfer channel has been essentially composed by concrete walls, the spent fuel gate and the borated water. As discussed in the precedent subchapter, the whole transfer channel has not been modelled but only the part defined of interest.

Moreover, no other systems have been modelled neither the railway of the transfer fuel tube facility because they have been considered negligible. It is worth reminding that, if a material and a component are not strictly necessary for the simulation and it is negligible, it is duty not represents it, in order to not increase unnecessarily the computational time.

The technical drawings of the modelled transfer channel have been reported in the annex K. In the next figure a 3D representation of the transfer channel with all the components modelled for the MCNP simulations is reported. It has been made by the MCNP visual editor.

In the three different views, the borated water has not been plotted to denote the others modelled components. The different colours represent different cells and not different materials. In the axonometric view you can note by a different colour (violet), the presence of a part of the transfer spent pool gate in the top of the transfer channel wall. This gate is open during outage operation and for this reason it has been filled by water. In addition, you can see the presence of a well which allows the swing of FA from the measurement position (horizontal) to a vertical one. In the front view you can observe the position of the two detectors both two placed in the opposite side of the FA elevator:

- The neutron detector in correspondence of the central line of the homogenized FA
Finally in the top view, the same detectors positions along the active length are marked as the geometry of the FA elevator. As commented before, this particular neutron detector placement allows improving the number of counts thanks to the radial function maximum.

11. Test of the innovative out-core device with MCNP

As discussed before, to test the innovative out-core device a Monte Carlo technique has been applied, specifically MCNP of Los Alamos National Laboratory. The input file of this programme, Test_device.is, has reported in MCNP annex. In order to prove the goodness of all assumptions done, the two correlations used in the device has been verified:

- The relation between the fuel assembly burnup and the neutron detector counts, caused by the neutron flux
- $^{134}$Cs/$^{137}$Cs ratio in function of the BU

Specifically, the simulations have concerning the fuel assembly with initial enrichment of $^{235}$U equal to 4.53%. Moreover, thanks to the analysis of neutron and gamma spectral, flux generated by the irradiated fuel assembly determined by the ORGEN-S code, some FA has been simulated, precisely: E501, E505, E508 and E510. For each of this case, three different runs have been done, one for outage operation. It worth reminding that a MCNP run, with the program previously described, will require at least 15 hours for shooting $10^8$ particles. In the annex H the whole MCNP input is reported.

In the following table, the result results for the BU vs. $BF_3$ detector counts has been reported.

<table>
<thead>
<tr>
<th>Simulation</th>
<th>Refuelling operation</th>
<th>F4 (MeV/cm2/h)</th>
<th>F4 (1/cm2/h)</th>
<th>Average energy neutron flux (MeV)</th>
<th>$\sigma(n,\alpha)/\sigma_{TOT}$</th>
<th>Number of collisions per history</th>
<th>BF3 count-1 (cps)</th>
<th>BF3 count-2 (cps)</th>
<th>BF3 detector counts (cps)</th>
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<td>E501 R1</td>
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<td>5.9E+01</td>
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</tr>
</tbody>
</table>

Figure 11-1: MCNP results
The BF$_3$ detector count have been determined considering the sensibility value present in the detector data sheet and a user made correlation which takes into account the characteristics of products of the radiative capture of $^{10}$B and the detector characteristic, volumes and materials. The specific calculations have been reported inside annex H whereas the final formula obtained is reported next:

$$C = 0.5 \times \left( 0.7326 \times C_{MCNP} \times \int \Phi_O(E) \, dE \times \frac{\sigma_{n,a}(E)}{\sigma_{tot}(E)} + S_E \times F_4 \, tally \times \int \Phi_O(E) \, dE \right)$$  
Equ. 11.1

Where:

- $C$ is the effective number of BF$_3$ counts per second
- $C_{MCNP}$ is the total number of collisions in the BF$_3$ per history done. This value can be obtained from the table 126 of MNCP output
- $F_4$ tally$^{18}$ is the surface current tally [ neutrons/cm$^2$ per neutron history]
- $\frac{\sigma_{n,a}(E)}{\sigma_{tot}(E)}$ is the probability to have a radiative capture per collision happened in the BF$_3$. The microscopic cross sections have been obtained by the JANIS-3 OECD database.
- $\Phi_O(E)$ is the neutron flux per unit of energy obtain from ORIGEN-S simulation
- $S_E$ is the detector sensitivity present in the product datasheet [number of neutron detector counts per unit of flux]

In the following graph, the relation obtained between the fuel assembly burnup and the neutron detector counts has been reported.

![Figure 11-2: MCNP results BF$_3$ counts vs. BU](image-url)
This correlation follow can be best fitted with a good accuracy by the following power function:

\[ y = a \times x^b \quad \text{Equ. 11.2} \]

where:
- \( y \) is the effective number of BF\(_3\) counts per second
- \( x \) is the burn up [MWd/tU]
- \( a \) and \( b \) are constants which are equal to 1.438e-13 and 3.491

The goodness of the results is clear. In fact the Adjusted R-square value is equal to 0.9988 where 1 represents the perfect function (annex F). However, comparing this BF value with that obtained by the ORIGEN-S simulation you can note that the accuracy of the fit is decreased about 0.11% due to the intrinsic probabilistic nature introduced by the Monte Carlo technique.

Thanks to the elaboration of MCNP outputs, you can affirm that the evaluation of burnup by the measure of a neutron flux in the BF\(_3\) is possible. However, because of the low slope of the computed correlation at low burnup, relative to the irradiated fuel assemblies which have been only irradiated for a power cycle, it will be problematic determine those value by the innovative out-core devices.

The next figure shows the relation between the nuclear fuel burnup and the collision rate per history.

![Figure 11-3: MCNP results: collision per history vs. BU](image)
The collision rate seems to increase with burnup, however more experimental points are needed to confirm this tendency.

The average neutron spectral flux energy at the BF$_3$ detector is around 30 keV for all the simulations; any particular correlation can be determined. Nevertheless, it is worth highlighting the enormous power of thermalization of water. In fact, in more or less 40 cm, the neutrons generated by the fuel assembly at an average energy of 2 MeV reach the BF$_3$ detector with 30 keV causing a decrement of energy around 98.5%.

Unfortunately, the correlation between $^{134}$Cs/$^{137}$Cs ratio in function of the BU cannot be evaluated in this project due to the very low detection efficiency. For this reason, very bad results have been obtained. In fact to achieve a useful peaks of Cesium, an enormous amount of particles have to be shot. An approximate hand-made calculate has shown the necessity to shot at least $10^{11}$ particles. Considering that using a Pentium-III, a neutron and gamma MCNP simulation with $10^8$ particles requires around 15 hours, $10^{10}$ particles will need 625 days and so this run is not accessible. However, using a cluster the time can be reduced exponentially up to several days. It is worth reminding that this is only a computational problem not a theoretical one, neither a wrong placement of the LaBr$_3$ detector because the position has been verified by deterministic calculates.

Considering that in the MCNP simulations:

- The presence of gamma and neutron spectral source equally probable
- The probability to shot from the whole spectrum:
  - A $^{134}$Cs selected photon is around 0.192%
  - A $^{137}$Cs photon is around 0.687%
- The probability to be captured by the LaBr$_3$ per photon which reaches the crystal is:
  - $2.47 \times 10^{-7}$ for a $^{134}$Cs selected photon
  - $1.42 \times 10^{-7}$ for a $^{137}$Cs photon

the probability to have a detector counts for photon created is around to:

- $4.88 \times 10^{-10}$ for a $^{137}$Cs photon
- $2.3712 \times 10^{-10}$ for a $^{134}$Cs 0.795 MeV photon

Remembering that $10^8$ particles have been shot for each MCNP simulation, no-one detector count for these two peaks has been obtained.
However, in the real situation an enormous total gamma emission around $10^{19}$ photon per second is present, therefore a great number of counts related to the Cesium peaks of interest can be obtained and processed in a small time.

This final consideration points out the computational problem encountered.

In conclusion, the necessity of an enormous number of particles has been highlighted also during the energy MNCP calibration of LaBr$_3$. In fact, during the $^{137}$Cs calibration one billion of gamma photons, with an energy equal to 0.6617 MeV, has been launched but only 182 particles has hit the LaBr$_3$ crystal resulting a capture probability for history done equal to $1.42 \times 10^{-7}$. For too smaller amount of points, the energy calibration cannot be realized.

12. Application of the innovative out-core device inside a refuelling operation

The application of this innovative out-core devise in refuelling operations will cause the revision of technical procedures and, on the other hand, of the APS even if in a negligible way. Thanks to the device placement selected, no additional fuel assembly movements are necessary as required by NRC. Substantially, the way covered by the fuel assembly from the reactor core to the spent fuel pool will be the same, however an additional time will be required. Considering that the requested time to download a complete PWR core is 15 days and supposing an added time of 5 minutes for each fuel assembly, the duration of a refuelling operation will be incremented at maximum about 4%. As widely discussed, this out-core device does not pretend to substitute the in-core FA characterization but it aims to support it when necessary. For this reason, the additional measurements of BU, total fissile material and Plutonium isotopes will be applied only when they are requested by the control room. In this way, the increment of the outage operation is meaningless respect to the improvement done.

For example, if the control room encounters an error in an area of the flux map processing, which causes an wrong extrapolation of the neutron flux and so of the BU, the out-core measurement can be only applied to the FA correspondents. On the other hand, the Plutonium isotopes determination
can be applied only to the FA which has been irradiated for the last power cycle and so they will be stored definitely in the spent fuel pool. In this way a more accurate inventory will be obtained.

The application of the passive neutron burnup characterization, equ. 6.1, does not require the spectrometry, therefore it can be applied independently.

All the detection out-core measurements will be applied only when the FA is downloaded from the core, whereas the fuel assembly verification number also in the upload phase.

In the next figure, the PWR scheme is shown. Presently, the video controls are present only on the spent fuel mast bridge (3) and in the refuelling machine (1) in the reactor building, annex E. This device provides an additional fuel identification number in the transfer channel which allows decreasing the fuel misloading in reactor core and in the spent fuel building. Combining this system with the capacity of the device to determinate the typology of the FA, fresh or irradiated, measuring the amount of Cesium-137, the probability of reactivity incidents in the core and in the spent fuel pool strongly decreases. It is worth reminding that most of the misloading events are caused by human errors.1 Thanks to the combination of detectors with the video control system, the reactor core damage can slightly decrease. To determine this amount an APS is needed.

---

*Figure 12-1: PWR video control system (ref 65)*
13. Economical evaluation

Thanks to the support of ORTEC, SAINT-GOBAIN, LND and Ahlbert Electronics vendors a first approximate economical evaluation only regarding the cost of the components needed in this innovative out-device studied is done in the next table. All the prices are in euro and VAT included.

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**Total amount 50.277 € VAT included**

Figure 13-1: Economical Evaluation

Of course this evaluation does not take into account several factors, such as for instance the needed of a constant presence of one operator, the periodically maintenance, the calibration process, the cost of operational process reviews, the assembling cost and so on. However it can be used for a first approximate economical evaluation. An initial investment of more or less fifty thousand euro for an innovative out-core device which can increase the availability of the plant, the efficiency of the in-core nuclear management and aims to decrease the reactivity accidents is absolutely negligible for a NPP. For this reason, from the economical point of view this project is highly viable.
14. Environmental evaluation

Due to Fukushima Dai-ichi NPP accident, happened in March 2011, the worldwide opinion about nuclear energy is getting worse. The application of innovative devices to the NPP can be one of the first steps to win back the people confidence. The two fundamental problems, concerning this advanced source of energy, are the nuclear wastes and the dramatic consequences in cause of serious accidents, at least level 5 of INES scale.62

The application of the projected innovative out-core device allows to decrease both factors in the fuel management area. Thanks to the combination of the video system with the detection equipment, the device will strongly limit the possibility of FA misloading and so of reactivity incidents caused by a wrong placement of a FA inside the core. Furthermore, the possibility to recognize the typology of fuel, irradiated or fresh, combined with the automatic capability of the system, could also decrease the effective dose to the workers.

As widely discussed along the thesis, this innovative device will allow determining a more accurate BU, at least for several numbers of FA, achieving better core configurations and an improved use of the fuel. In this way the same core will produce a greater amount of energy saving fissile materials or in the other side the same amount of energy will cause a smaller amount of nuclear waste. Moreover, a higher fissile material consumption, mainly $^{239}$Pu, allows producing nuclear wastes which are less interesting for the nuclear weapon productions.

To conclude the more exact inventory of Plutonium isotopes stored inside the spent fuel pool allows facilitating the IAEA periodical controls and borders the nuclear proliferation activity.

15. Problems encountered during the device design

As widely discussed in the previous chapters, two main problems have been encountered.

The first one is the MCNP limitations for very complex geometry described with the fill and lattice functions.63,54 The constrains involved with these commands, such for instance the limitation on the number of universes and the impossibility to utilize a lattice function if additional objects are required, have negated the possibility to utilize the advanced geometry concept53,54 which allows to reduce substantially the computational time required for a run.
The second problem encountered is the computation time. In fact due to the very detector efficiency and the overall volumes of cells, the number of particles needed in a MCNP simulation to obtain a useful $^{134}\text{Cs}$/$^{137}\text{Cs}$ peak ration is enormous even if several assumption have allowed to reduce the computational time up to 15 hours for $10^8$ history utilizing a Pentium-III. Unfortunately, an approximate deterministic calculation has shown the necessity to increase the history at least to $10^{11}$ in order to test also the burnup determination by a gamma spectrometry. This amount of particles with a neutron and gamma source requires using a Pentium-III around 625 days. For this reason, this function could not be tested. A future development of this device has to take into account the use of a cluster.

16. Future possible developments

This final thesis project does not pretend to project the whole innovative out-core device but it only aims to define its main features. For this reason, in this project phase, several possible developments are possible and required in order to obtain the best final configuration of the system.

As discussed in the thesis, due to the characteristic of ORIGEN-S, which is a zero dimensional depletion code, it is not possible to calculate the BU profile which constrains to impose no-profile source in MCNP. Replacing ORIGEN-S with more sophisticated software, i.e. TRITON of SCALE package, the flux profile can be calculated and inserted in the Monte Carlo input file.

Surely, the FA has to be modelled as heterogeneous: each fuel pin has to be composed, at least, by the fuel pellets, the pressurized Helium and the Zircaloy cladding. This important modification can be implemented utilizing a different Monte Carlo code, as KENO 3D, but it will substantially increase the computational time due to the major number of described cells. For this reason the application of a Variance Reduction Tool has to be considered. However thanks to this modification, the capacity of the detector to detect the BU profile could be tested.

Moreover, in order to improve the accuracy of the detection yield by MCNP, you may:

- Introduce the real irradiated fuel pellet element composition as done in some interesting research programs
- Improve the detailing of some elements yet inserted as the FA container and the detectors and insert new components
- Determine the number of alpha produced inside the BF$_3$ detector due to neutron captures directly by MCNP.
Furthermore, in order to obtain a more exact correlation between the neutron detector counts and the BU, a parameter sensitiveness analysis has to be done. For example, the following factors may be studied:

- Amount of Boron in the water
- Background in the transfer channel
- Wrong placement of the FA during the measurements
- Density of the borated water which can influence the neutron thermalization

Thanks to the immense capacity of MCNP, the material and the shape of the detector water prove boxes can be improved to optimize the detector responses, as done in some technical publications.

Furthermore, it is worth reminding that the BF$_3$ and LaBr$_3$ detectors and the microscopic cross sections of the elements which compose the FA are influenced by the temperature. For this reason you suggest to perform a:

- CFD analysis imposing the heat generation term due to irradiation as volumetric source in the detectors
- Calculate an average value of the fuel pellets temperature afterward apply the correspondent library in MCNP

Besides, very important device improvements will be obtained if quicker instrumentations are applied to obtain an increment of the number of counts per seconds. In this way, the time required for the out-core measurement will be reduced consequently also the possible impact in the refuelling operation. The instrumentation can be bought, if it exists or in the other case we have to take into consideration the possibility to produce it.

An active neutron technique to determine the initial enrichment of the fuel assembly can be implemented as done in NAJA$^{13}$. However, this modification will cause the necessity to calculate the $k_{\text{eff}}$ which has to remain always below 0.95$^{13}$ during the refuelling operation.

In conclusion, if the device will be displaced to the spent fuel pool a big revision of the project of the main feature is not necessary. In this way, several more function, as the burnup credit applied to fuel transport and storage casks, can be easily applied thanks to the exponential decrease of gamma field caused by the radioactive decay of short-live fission products. For this new placement, the analysis of several current research programs about out-cores devices, such as for instance the PHYTON$^{\text{TM}}$ and the NAJA device$^{10}$ can be useful.
17. Conclusions

Characterization of nuclear fuel burnup with ORIGEN-S: irradiated fuel isotopic composition analyses and gamma-neutron spectral study

- The impossibility to generate both radial and axial profile of gamma and neutron field caused by the irradiated fuel assembly is a big limitation. For this reason, a change of the depletion code has to be taken into consideration.

- The deep analysis of parameter sensitiveness about the main neutron emitters has shown the variation of the initial $^{235}$U enrichment. This, in addition to the irradiation history is the most delicate factors. In fact, as in the case of the most elevated neutron producer $^{242}$Cm, an additional shut-down time will comport a huge modification in the production due to the short half-life compared with the outage operation. The comparison with the technical publication has highlighted at the same time the validity of the results but also the necessity of more experimental points.

- In order to give a correct interpretation of the passive neutron measurements of the fuel assembly, some information of the initial fuel material is necessary, and as far as short cooling times, additional information about the irradiation history are required.

Test of the innovative out-core device with a Monte Carlo techniques, MCNP

- Thanks to the elaboration of MCNP outputs, it is possible to affirm that the evaluation of burnup by the measurement of a neutron flux in the BF$_3$ is possible. However, because of the low slope of the computed correlation at low burnup, relative to the irradiated fuel assemblies which have been only irradiated for a power cycle, it will be problematic to determine those value through the innovative out-core devices.

- When making a comparison between the best fit accuracy regarding the burnup and the neutron emission of the ORIGEN-S and MCNP output, this value has shown a decrease due to the uncertainty introduced by the MCNP random features. In fact, the R-adjusted value for an initial enrichment equal to 4.22% has been reduced about 0.11%.

- The analysis of the energy average neutron spectrum at the BF$_3$ detector seems to increase with the burnup. More experimental points are required for a more accurate best fit.

- Because of the very low detection efficiency of Cesium peaks measured by detector spectrometry tally, the correlation between $^{134}$Cs/$^{137}$Cs ratio cannot be easily tested in MCNP. A number of particles shot equal to $10^9$, which requires around 15 hours with a Pentium-III, is not sufficient. At least $10^{10}$ particles are desirable which necessitates more or less 1500 hours. This computer time, without the availability of a cluster computer, is not viable.
Implementation of the device in the outage operation

- This device does not aim to substitute the in-core burn-up determination, but it will provide a support when required. In fact, the measurement will not be applied to all fuel assembly whereas to a small amount which the utility defines. For this reason the additional time needed is a negligible amount. Surely, the main modification caused by the implementation of this device inside a refuelling operation will be the revision of the refuelling procedures.

Environmental evaluation

- The combined implementation of a combination of detector equipment with video controls will allow decreasing both the reaction incidents related to the fuel assembly misloading and the nuclear waste amount due to better burnup characterizations therefore allowing a more sophisticated in-core configuration. As a matter of fact, the more accurate Plutonium isotopes inventory in the spent fuel pool will consent to limit the nuclear proliferations and facilitate the IAEA inspections.

Future developments

- Due to the initial phase of this project several developments have to be applied. In fact the implementations of a more detailed geometry, the application of the real composition of the irradiated fuel and the assumption of a cosine flux source profile in MCNP modelling are necessary. Furthermore, very important device improvements will be obtained if quicker instrumentations are applied. If this equipment exists, then it can be bought, or if it is not the case, then the possibility to produce it should be taken into consideration. A big revision of the project will not be necessary if the device is displaced to the spent fuel pool. In this way, several additional functions, as the burnup credit applied to fuel transport and storage casks, can be easily applied thanks to the exponential decrease of gamma field caused by the radioactive decay of short-live fission products.

Out-core devices research

- Measurement of passive gamma rays and neutrons emitted by the decay of radioactive isotopes is a widely used and fundamental out-core technique to verify and determine characteristics of nuclear materials. Several initiatives are currently under way in order to investigate the use of advanced nondestructive assay methods to support the in-core evaluation measuring the fuel assembly burnup, the fissile material and Plutonium content in irradiated nuclear fuel.

Conclusive evaluation

- Taking into account the economic viability, the utilization of commercial detector equipment and the limited increase of the overall refuelling operation this project is considered viable.
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