Burn-up Credit Applications for UO_2 and MOX Fuel Assemblies in AREVA/COGEMA

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For the last seven years, AREVA/COGEMA has been implementing the second phase of its burn-up credit program (the incorporation of fission products). Since the early nineties, major actinides have been taken into account in criticality analyses first for reprocessing applications, then for transport and storage of fuel assemblies Next year (2004) COGEMA will take into account the six main fission products (Rh103, Cs133, Nd143, Sm149, Sm152 and Gd155) that make up 50% of the anti-reactivity of all fission products. The experimental program will soon be finished. The new burn-up credit methodology is in progress.
After a brief overview of BUC R&D program and COGEMA's application of the BUC, this paper will focus on the new burn-up measurement for UO2 and MOX fuel assemblies. It details the measurement instrumentation and the measurement experiments on MOX fuels performed at La Hague in January 2003.

Keywords: R&D Program for BUC applications, Criticality experiment, code qualification, burn-up measurement, spent fuel, UOx, MOx.

1. Context of Burn-Up Credit (BUC)

For over ten years, burn-up credit (actinides only) has been used at the COGEMA plant in La Hague for spent fuel transport, interim pool storage and reprocessing applications. More than five years ago, an intensive research program was begun jointly with the French Atomic Energy Commission (CEA) and the French Institute for Radiological Protection and Nuclear Safety (IRSN). This research program will make it possible to take the main Fission products (neutron absorbers) into account in criticality studies. These Fission products are listed in Figure 1 below.

BU	**20 GWd/tU**	**40 GWd/tU**
Sm149	**980**	**1030**
Rh103	**790**	**1360**
Nd143	**530**	**900**
Cs133	**420**	**750**
Gd155	**390**	**1550**
Sm152	**250**	**490**
Sm151	**350**	**500**
Tc99	**240**	**440**
Nd145	**230**	**410**
Eu153	**150**	**390**
Mo95	**150**	**290**
Sm147	**150**	**230**
Sm150	**120**	**270**
Ag109	**100**	**250**
Ru101	**100**	**220**
Total for 200 fission products	**6120**	**11500**
Total for 6 fission products	***3460 (56%)***	***6090 (53%)***
Total for 15 fission products	***4950 (81%)***	***9080 (79%)***

Figure 1

After a brief history of BUC implementation at La Hague (chapter 2), the first part (chapter 3) of this article will describe the collaborative research program between COGEMA and CEA or IRSN, which is nearing completion. The second part of this article (chapter 4) gives details of the new method used to interpret burn-up measurements which will be applied to both UO_2 and MOX fuel assemblies.

2. A brief history of allowance for BUC in COGEMA La Hague

2.1. 1980 - 1990

At this time, all criticality studies were done at the optimum of moderation without BUC. No difficulties were encountered in the design of the first plants UP1 (at Marcoule) or UP2-400 (at La Hague) because the fuel assemblies were less enriched and the facilities were smaller.

At the same time (in the mid-eighties) COGEMA began its first R&D program for qualification of U and Pu (actinides only BUC). This IRSN R&D program was called the "HTC program" and it was conducted at VALDUC.

2.2. 1990 - 2000

1990 was the start-up of UP3. Since then, the BUC has been systematically applied for reprocessing because, without the BUC, the dissolver wheel is critical for enrichments higher than 2.5%. Likewise we started UP2-800 in 1995. Consequently a burn-up measurement is systematically performed to ensure the reprocessing times of both plants at La Hague

During this period, the increase of initial enrichment of the fuels led to the necessity of BUC not only for reprocessing but also for transportation and pool storage at La Hague. The actinide-only BUC was then used in Europe for:

- transport of the fuels from Germany, Holland, Switzerland, and Belgium to France

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- storage of all fuels (French and foreign) at La Hague pools.

In conjunction with the use of UC we adopt less penalizing criticality hypotheses. For example in the case of fuel storage, one less penalizing hypothesis consists mainly of taking into account a fixed number of rods (the criticality studies are no longer done at the optimum of moderation). Obviously this number of rods has to be guarantied.

In parallel with this generalization of use of actinides-only BUC, COGEMA decided to begin a second extensive R&D program to take fission products into account. This program is detailed in chapter 3.

2.3. 2000 - 2010

The complementary needs of BUC concern the extension of the range of acceptable fuels

- MOX
- UO_2 up to 5% enriched
- MTR

The need to incorporate fission products in BUC can be explained by the two levels by which the BUC can be applied:

- the first level results in little gain (a single and minimum fuel cycle in the core) but with a little constraint which consists only of a simple irradiation check
- the second BUC level can give the burn-up in the 50 least irradiated centimeters of the fuel. Nevertheless, this higher gain requires the implementation of a quantitative burn-up measurement.

The incorporation of fission products is consequently particularly interesting when the burn-up measurement is not easy to implement. The first level of BUC (with incorporation of fission products) can sometimes be enough. This is the case for EDF fuel storage at La Hague, for which today no BU measurement has been implemented, and also for the reprocessing of MOX. The paper in reference 1 gives a good example of the contribution of fission products for MOX. In this particular case with only a first level of BUC (with fission products) no gadolinium has to be used for reprocessing.

Another application of BUC (with fission products) is the ability to guaranty some liquid tank transfers - particularly liquids with high Pu concentrations.

3. The finalising criticality R&D program

This R&D which began in 1995 consists of two parts: an extensive experimental program at VALDUC and CADARACHE sites, and a code development and qualification program. The French criticality package CRISTAL is the main result of this R&D.

3.1 Minerve experimental and qualification program

The aim of this program is to:

1. Measure the effective capture cross-sections of the 15 major fission products in the Minerve reactor operated by the French Atomic Energy Commission at Cadarache. These measurements are made by oscillations of individual samples of the fission products in the reactor. Oscillations of spent UOX and MOX fuel samples is also under way. These measurements are being carried out by the SPEX Department to qualify the APOLLO2 Sn option of the CRISTAL package.
2. Measure and analyse samples of spent fuel from French and German pressurized water reactors which will allow the SPRC Department of the Atomic Energy Commission at Cadarache to qualify the Darwin fuel cycle package and the Cesar code (see respectively References [2] and [3]).

Some results have been published (see Reference [4]).

3.2. VALDUC experiments and interpretation

The program involves experiments and qualification of the French criticality package CRISTAL.

The experiments are carried out by IRSN at Valduc. These experiments represent a sub-critical approach involving the raising of the water level. Various configurations are studied, ranging from individual fission products to more general experiments on mixtures of products (see Reference [5]).

The experiments are interpreted by the Criticality Study Department of IRSN at Fontenay-aux-Roses with a goal of qualifying the industrial version (Apollo-Moret) of CRISTAL (see Reference [5]).

3.3. CRISTAL, the French criticality package

CRISTAL comprises three calculation routes:

1. The Monte Carlo route with the Tripoli code developed by the CEA's Department for the Study of Reactors and Advanced Modeling (SERMA) in Saclay.
2. The APOLLO 2 SN route used to make 1D and 2D calculations (calculations of standards); it is qualified by the CEA's Department of Reactor Physics in Cadarache.
3. The industrial route with coupling between the APOLLO 2 code used to assess multi-group cross-sections in infinite surface areas. And the MORET 4 code used to calculate complete 3D geometries using the Monte Carlo method but with cross-sections calculated by APOLLO, which requires a considerably shorter calculation time than the direct Monte Carlo method. This route has been qualified by the Criticality Study department of the CEA in Fontenay-aux-Roses.

Figure 3 below shows the various codes and sequences making up the CRISTAL package.

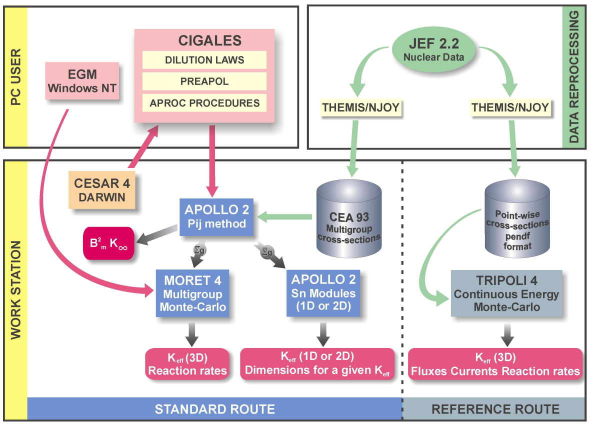

Figure 3

4. A new BU measurement for UO_2 and MOX fuels

4.1. Industrial and development objectives

COGEMA currently uses two methods for measuring burn-up, one based on gamma spectrometry and the other on passive neutron measurement. Both measurement systems are located in the shearing and dissolution facilities of both UP2 and UP3 reprocessing plant at La Hague.

There is also a portable system known as Python which uses a single neutron measurement and a total gamma measurement to obtain a burn-up profile along the axis of a fuel assembly. This system has been installed in various reactor spent fuel pools in Europe, making it possible to use burn-up credit for the transport of spent fuel casks and their interim storage in the COGEMA pools at La Hague. It does not involve gamma spectrometry since the technique currently used at La Hague requires a germanium detector with a liquid nitrogen cooling system that cannot easily be installed underwater in a reactor pool.

These two systems can be used to measure the burn-up of UO_2 fuel assemblies and are qualified by the French Safety Authorities for such fuels but not for MOX fuel assemblies.

The studies currently under way are therefore aimed at developing a system that could be used to measure the burn-up of UO_2 and MOX fuel assemblies as simply as possible, without the need for a liquid nitrogen cooling system.

Furthermore, the combination of neutron measurements and gamma spectrometry would simply involve modifying the software of the existing systems at La Hague, without any effect on the hardware.

4.2. Measurement system

In view of the industrial objectives mentioned previously, preference was given to a system that combined gamma spectrometry and neutron measurements and yet remained portable. There was no alternative but to use Cd-Zn-Te detectors which operate at room temperature. A considerable amount of development work was required nonetheless to optimize these detectors and the associated electronics for this type of application, since they are currently used almost exclusively in the field of nuclear medicine to measure low energy rays at low count rates.

The detectors were extended, making it possible to measure all the gamma rays of interest (661 keV for ^{137}Cs, 796 keV for ^{134}Cs and most importantly 1274 keV for ^{154}Eu). The electronics were optimized to handle count rates of 50,000 counts/second. Dual parameter analysis (energy and pulse shape) results in improved resolution and makes it possible to separate the ^{154}Eu peak from those of ^{60}Co at 1173 keV and 1332 keV. A resolution of the order of 1.5% has been achieved , i.e. 10 keV at 661 keV and 20 keV at 1.4 MeV.

4.3. Interpretation of the measurements

A considerable amount of work has also been carried out as regards analysis of the measurement results. A combined measurement analysis algorithm has been developed. It comprises four major stages that occur after the fuel assemblies pass in front of the detector.

1. Blind interpretation to be able to know with no data on the fuel assembly measured if the fuel is a MOX or an UO_2 fuel.
2. Validation of the average burn-up of the fuel assembly by analysis of neutron measurement (use of certain data such as irradiation history, burn-up of fuel cycles, initial isotopic composition).
3. Determination of burn-up at one end of fuel assembly due to acquisition of the axial profile of ^{137}Cs.
4. Validation of data provided by reactor operator (use of gamma spectrometry ray ratio to detect any possible inconsistencies in the data provided).

The first stage ensures that MOX fuel assemblies are not mistaken for UO_2 assemblies. Indeed, for a given burn-up, a MOX fuel assembly can have a far higher reactivity than a UO_2 fuel assembly. Likewise, the data validation module (Stage 4) guarantees that any significant error regarding the initial enrichment (or Pu content in the case of MOX fuel assemblies) will be detected. In addition to the average burn-up validation stage (Stage 2), Stage 3 makes it possible to set a minimum burn-up value which must be higher than the burn-up used for the criticality-safety study.

All of these stages use laws of correlation between a physical quantity (gamma peak ratio or neutron measurement) and burn-up. An evolution code is therefore required and is used on-line. This code is known as CESAR (see Reference [2]). A large amount of research and development work has been carried out (especially for MOX fuel assemblies) to create neutron effective cross-section libraries that are used to accurately characterize entire fuel assemblies.

This new measurement system can be used to measure both UO_2 and MOX fuel assembly burn-up and is far more accurate than existing burn-up measurement systems thanks to its data validation module.

The system is developed by CANBERRA under CEA specification.

4.4. Qualification Program

Since the R&D work is nearing completion, a program has been devised for qualifying the system.

Hot testing on 20 MOX fuel assemblies has been performed at COGEMA La Hague in January 2003. The burn-up of these 20 MOX fuels have measured.

Figure 4 below shows the positions of the detectors on the burn-up measurement stations in the shearing and dissolution facilities in COGEMA's UP2 and UP3 plants at La Hague.

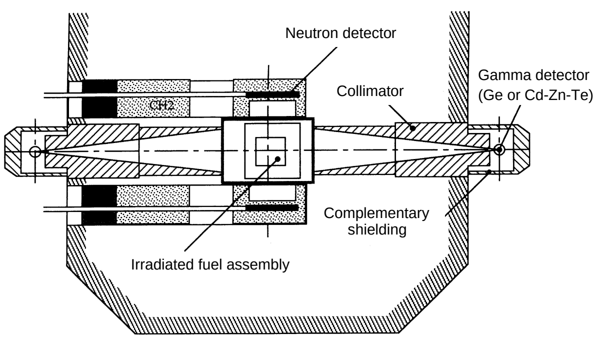

Figure 4

The qualification note is under progress, but we can already give the following results:

- The system will be able to validate the burn-up within +/-10%.
- A mistake in one declared irradiation cycle will be detected even if it does not change the cooling time of the fuel (ex. for a given fuel a declared irradiation history during reactor cycles 2, 3 and 5, and real irradiation history during reactor cycles 2, 4 and 5 can be detected)
- The initial ^{235}U enrichment or Pu content can be validated within a variation of about 1% (ex: a declared value of 5%Pu content can be distinguished with a 6%Pu content value)

This should mean that, in 2004, it will be possible to submit a report on the new UO_2 and MOX fuel assembly measurement system to the French safety authorities for approval.

5. Conclusion

The R&D work being carried out by COGEMA, CEA and IRSN on the subject of burn-up credit is now entering its second phase (with allowance being made for the major fission products as well as major actinides). It is also intended to make allowance for the burn-up credit of MOX fuel assemblies (actinides only to begin with). To this end, a new device for measuring the burn-up of UO_2 and MOX fuel assemblies has been successfully developed. The system is now at the qualification stage after real measurement on MOX fuels at La Hague.
All this R&D (in the fields of criticality and nuclear measurements) should ensure that COGEMA will be in a position to handle, in its La Hague facilities, the different types of fuel currently envisaged by electricity companies and fuel designers without any major problems.

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