Framatome ANP Capabilities in Nuclear Criticality Safety Studies for Transport Packages and Storage Installations

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ABSTRACT

FRAMATOME ANP, the world leader in nuclear fuel supply must master the full scope of activities including design, fabrication and fuel assemblies delivery.

One of the main concern is to ensure that the nuclear criticality safety analyses are performed in compliance with all the regulations in force. Increased ^{235}U enrichment for PWR assemblies and ^{235}U content for MTR elements imposed to make a complete review of our safety transport analysis files.

Moreover, since 1996 Framatome ANP is performing the criticality studies for the different stages of the fuel cycle including spent fuel storage facilities which is the backend fuel cycle strategy for many countries.

The paper deals firstly with a description of the criticality calculation codes and methodology, developed by the IPSN and in use for more than twenty years by the French nuclear industry. These tools have been qualified against a wide experimental database including fissile media, shapes and configurations met for these specific studies. The paper follows with the experience feedback that Framatome ANP has acquired in the past five years in the different fields concerning particularly :

- Improvements of existing transport fresh PWR fuel cask,
- Design of a new transport fresh MTR fuel cask,
- Justification of interim spent fuel storage.

through real examples developed showing the different safety considerations used to fulfill the regulations and the safety authorities requirements.

The above examples demonstrate the expertise Framatome ANP has acquired to be able to support specific customer needs.

1. INTRODUCTION

The calculation codes and methodology used in France for the assessment of criticality safety have been developed and validated for more than two decades. The Institute for Nuclear Protection and Safety (IPSN) together with the French industry (from enrichment to reprocessing processes) have supported a complete experimental program to qualify a calculation scheme for the different configurations encountered in the nuclear fuel cycle where criticality safety is of concern.

Computer codes package APOLLO1-MORET III for now and CRISTAL in the near future allow to perform criticality safety studies for any kind of fissile material, shapes and configurations.

As the worldwide leader of fuel vendor, Framatome ANP has big concerns with criticality safety aspects of fuel manufacturing plants as well as of fuel transportation to nuclear power plants. Moreover as Framatome ANP offers a wide range for fuel services such as backend solutions, interim and final fuel storage criticality safety approach can be performed.

After a brief description of the calculation methodology and codes the paper will mainly deal with transport casks and backend cycle aspects.

2. COMPUTER CODES AND STANDARDS

In France, criticality studies are mainly carried out using the "Commissariat à l'Energie Atomique" APOLLO-MORET system of codes. This system is being adopted by the industry. It includes tools to describe the codes inputs and two main codes: APOLLO1 and MORET III. A brief description of the system is presented in this section and the Figure 1 shows the links between the different codes.

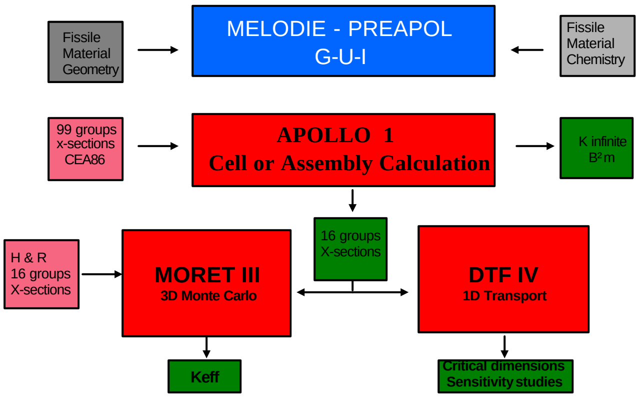


Figure 1

APOLLO1 is an assembly code widely used in the framework of reactor physics.
The code is linked with the 99 groups CEA86 microscopic cross sections library (JEF1 level).
The main features used in criticality calculations are :
Self-shielding: the code makes use of the Livolant-Jeanpierre theory. It is based on the flux factorisation in two components : a fine structure with strong variation within the resonance and a macroscopic flux presenting a soften variation. For each mixture of resonant nuclide and a scatterer (characterised by the background cross section), a very fine resolution of the slowing down equation is carried out and the effective reaction rates are calculated in each APOLLO energy group depending on the temperature (Doppler effect) and the background cross section. A double equivalence (homogeneous-heterogeneous and continue-multigroup) is made which enables the calculation of the self-shielded cross sections models which take into account the presence of resonant nuclides in different physical zones and the mixture of resonant nuclides are also available.
The flux calculation : the collision probability method is used in one or two dimensional geometry. Leakage calculation in the fundamental mode theory allows the calculation of the diffusion coefficient and so the search of the material buckling for $k_{eff} = 1$ or the determination of the K_{eff} for a given buckling. The homogenisation is made so that the reaction rates are preserved.
The spectral calculation is performed with 99 energy groups followed by a cell homogenisation and a 16 groups condensation according to the well known Hansen & Roach energy group boundaries. The 16 groups cross section characterising the fuel neutronic behaviour are then input into the MORET III Monte Carlo code.
A pre-processing code called MELODIE treats all the basic data, mainly dimensions and chemistry, checks them and finally prepares the APOLLO1 code input data.
MORET III is a three dimensional multigroup Monte Carlo Code using a $P0_C$ (transport correction) or P1 representation of the scattering anisotropy.
Its very powerful geometry capabilities allow an easy representation of complicated configurations without any approximation. The equipment's are divided into simple elementary

volumes (sphere, boxes, finite cylinders, hexagons, ...).
Combinatory operators as "reunion, intersection, inclusion, ..." and array options are then used to combine these elementary volumes. A graphical tool system is available to make a deep verification of the geometry using two dimensional coloured view as well as the convergence curve resulting of the sampling of the neutrons batches.
The nuclear safety criticality limits are the following ones :

- $K_{eff} \leq 0.95$ for an individual package in isolation,
- $K_{eff} \leq 0.98$ for an array of packages.

The above values are agreed by the French IPSN.

3. FRESH FUEL TRANSPORT CASKS

3.1. PWR Fuel Transport Cask

Framatome ANP, foremost world supplier of PWR fuel, delivers fresh UO_2 fuel elements, directly or via Fragema, from the factories of its subsidiary FBFC (Franco Belge de Fabrication de Combustible) to the French and foreign nuclear power plant sites (Belgium, China, Germany, Korea, South Africa, Spain, Sweden, Switzerland). For the past 25 years, these deliveries have been made by road, rail and sea in RCC shipping containers.
Since 1994, the French Safety Authority has undertaken a process of reviewing all former-design packaging in order to check the consistency of their safety file with the regulations in force.
The RCC packages are classified: “fissile industrial package 3 ”, as per the 1985 IAEA guidelines amended in 1990. In this respect, the packaging designer must justify the sub-criticality of the package after the completion of accident tests as defined in the regulations by a series of 9 m drops, drops on bars and fire tests. Now the safety analyses on the RCC packaging have shown the limits of this design as soon as shell perforation is envisaged.
Framatome therefore decided in 1997 to review the design of its UO_2 shipping containers.
The general requirements of the RCC container Upgrading Project meets three main objectives :

- safety constraints with allowance for the new accident hypotheses,
- allowance for product changes : ^{235}U enrichment of 5% and UO_2 theoretical density of 100%,
- operational improvements whenever possible.

Regarding the safety issues, the analyses conducted on the RCC showed that there was a need to offer an upgraded concept with the following functions :

- mechanical protection of the assembly, maintaining the fissile array within a given volume at the end of mechanical testing like drop onto bar,
- confinement, in order to limit fissile material dispersion within the container.

The need to fulfil this dual function led to a "door-based" protection system which offers a continuous barrier along the assemblies and at the ends. In addition to confinement, this system ensures the axial and radial restraint of the assemblies in case of drops and easily meets the requirement for preservation of the initial geometry – considered sub-critical – under regulatory accident fire and drop conditions. This new packaging, based on an adaptation of the RCC container, is named FCC.

Adding doors creates 2 neutronic chambers and thus calls for the consideration of accident hypotheses where these chambers could be flooded by water. The preliminary criticality studies demonstrated in this case the need for neutronic "isolation" between each assembly. This finding led to the concept of double wall doors into which a loaded neutron-poisoning component is inserted.
To ensure neutronic "isolation", the neutron-poisoning plates are taken out of the inter-assembly space and replaced by the central core of the frame, itself filled with neutron-poisoning material.

The move from the RCC to the FCC therefore means replacing the existing frame by a reverse T double wall stainless steel frame, filled with neutron-poisoning material. On this frame, two continuous double-wall stainless steel doors are mounted, forming two neutronic cavities each intended to accommodate an assembly or rod box. The radial clamping of the assemblies performed until now by clamping frames is provided by a set of pads inserted into the doors. Reinforced top and bottom plates guarantee axial restraint of the assembly in the cavity, in the event of axial drop.
The assembly made up of the frame, doors and top and bottom plates forms the double neutronic chamber which ensures the required confinement function. The detailed criticality studies below take the hypothesis that there is conservation of the neutronic cavity volume, keeping in mind that a reduction in the latter would be more favorable. It must therefore be guaranteed that the neutronic chamber will remain leak-tight following the mechanical tests.
The anti-perforation barrier is provided by the double-wall doors. These doors were designed to ensure that the door stops the bar, the aim being to keep the assembly on the inside and in the cavity volume.
Figure 2 hereafter shows the design of the FCC as well as the calculation models used in the criticality simulation for normal and accidental conditions.

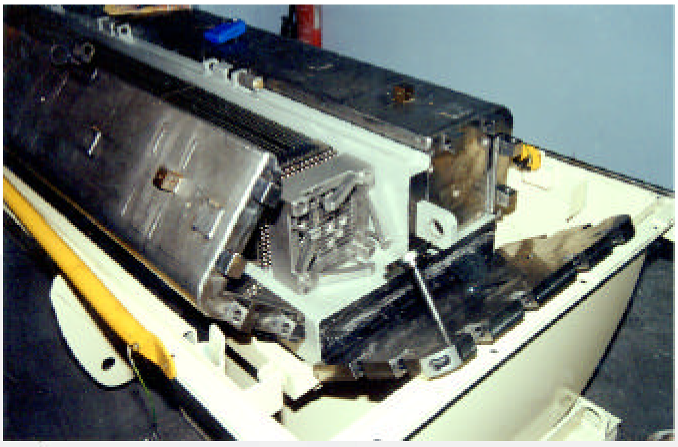
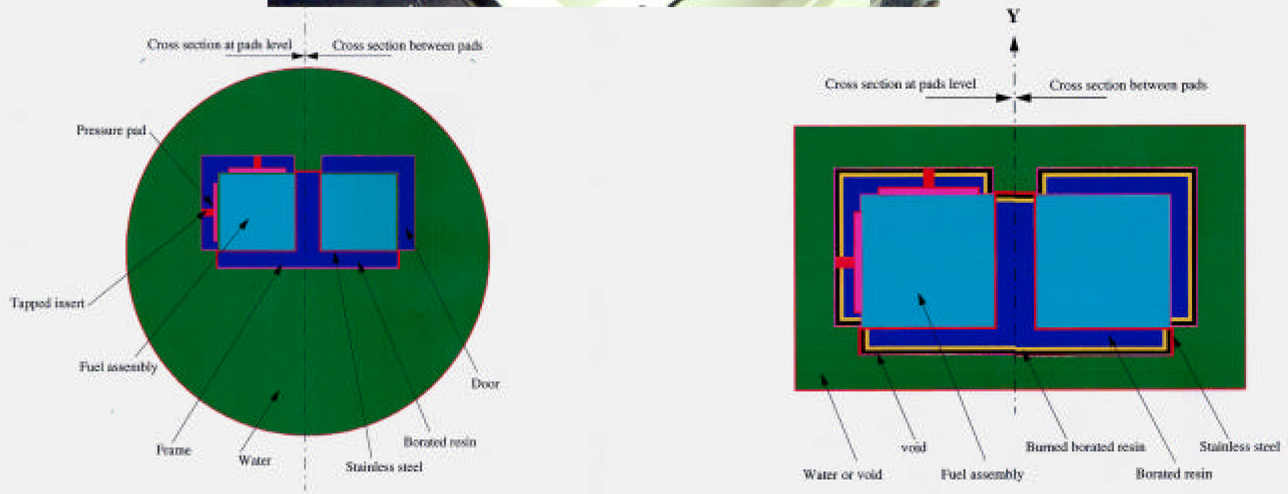


Figure 2



3.2. MTR fuel transport cask

CERCA manufactures fuel for research reactors all over the world. To comply with customer requirements, design and fabrication of Material Testing Reactors elements may be composed of various configurations such as uranium enrichment (LEU, MEU and HEU), uranium quantity, fuel alloy (Al, Si, Zr, …) or geometry (square, cylindrical, …). World-wide transportation of these elements requires a flexible cask which accommodates those different designs and meets the international transportation regulations. To be able to deliver most of fuel elements, and to cope with non-validation of casks used previously, CERCA decided to design and handle manufacturing of its own cask.

During all the phases of the cask design, emphasis has been put on the criticality safety aspects to comply with international requirements. Keeping in mind the large range of geometry of MTR fuel elements, three inner baskets have been designed to accommodate this aspect : two for large elements, and one for six "classical" MTR elements.

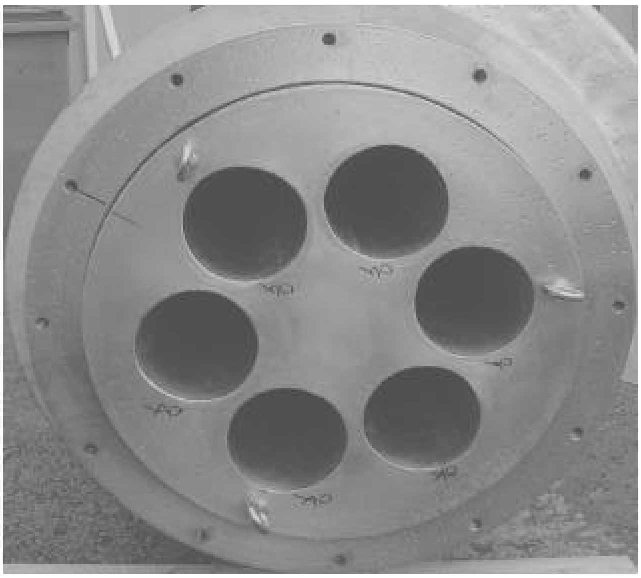
As fuel integrity is the main criticality safety viewpoint, emphasizes were put on the mechanical behavior of the fuel geometry during the regulatory tests of dropping and punching. To ensure the fuel integrity, a thick stainless steel inner shell is added between the shock absorbers and the inner fuel basket. The neutron absorber as well as its thickness were carefully chosen in order to keep a reactivity level as low as possible to meet the criticality safety criterion even after regulatory mechanical and thermal tests.

Criticality safety evaluations have been performed following two situations :

- One large fuel element with a robust skeleton is placed inside the inner basket ; thanks to the cask concept, no fuel damage can occur. This complies with RHF and FRM2 fuel elements,
- Six "classical" MTR elements, with a more simple structure around the fuel, are placed inside an inner basket ; even if some tests showed that their simple structure could stand IAEA mechanical tests, a penalizing bounding limit of an homogeneous U metal – Water mixture accounted for a complete fuel geometry loss. Therefore the content of the cask is defined as a maximum weight of uranium versus the enrichment.

Figure 3 show the entire package as well as the inner fittings with the multi-compartment basket.

Figure 3



4. INTERIM SPENT FUEL STORAGE

For many countries spent fuel storage facilities is the backend fuel cycle strategy. Among the existing solutions dry storage has a growing success.
For the six past years Framatome ANP had to face with such projects with ex-USSR countries.

In 1996, Framatome ANP had obtained its first contract for the VVER-type NPP at Medzamor in Armenia. The dry storage facility there was designed to receive the spent fuel assemblies previously stored in the plant's fuel pit. That was the first Framatome ANP's first order for a dry storage facility, for which the company holds a license to use the American Nuhoms® process, developed by the Vectra company which belongs to Cogema.

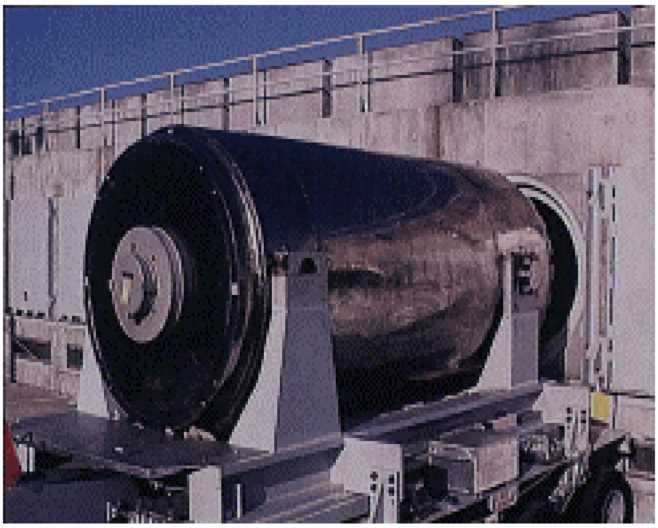
In 1999, Framatome ANP obtained a second contract for a dry storage facility. It concerns construction of the first part of the infrastructure necessary for the final shutdown of the Chernobyl NPP. It covers :

- Design and construction of a facility that will be used to package 25,000 spent fuel elements. This is a delicate operation that consists in separating the two fuel bundles before confining them inside capsules, which are themselves inserted into fuel storage canisters. The latter are stainless steel cylinders specially designed to house spent nuclear fuel.
- Design and construction of reinforced concrete structures for dry storage of the 256 canisters.

As far as criticality safety is concerned two major items has been pointed out to perform the calculations :

- even if "dry" is the master word of the design, abnormal and accidental water leakage inside the fuel canister has to be accounted for,
- to by pass the reactor fuel history, no credit is taken for the fuel depletion. For reactivity viewpoint fuel is considered a its maximum fresh enrichment.

Figure 4 shows a canister entering the storage module as well as one of the computational model used for the criticality 3D computations.



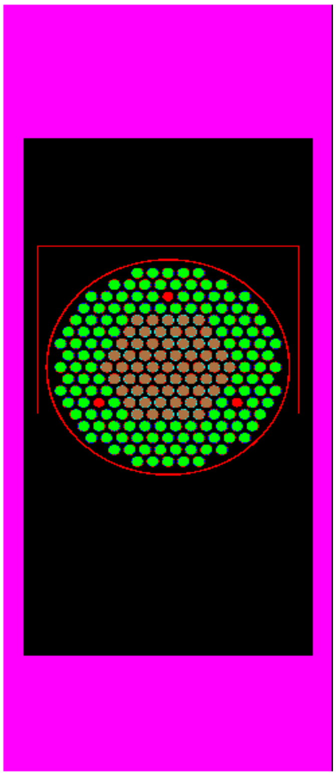


Figure 4

5. CONCLUSION

The paper showed through some real projects the experience feedback Framatome ANP has acquired in the past five years in the field of criticality safety approaches. The expertise Framatome ANP has gained to fulfill the regulations and the safety authorities requirements enables to support any specific customer needs.