CRITICALITY STUDIES: ONE OF THE TWO PILLARS OF CRITICALITY SAFETY AT THE BELGONUCLEAIRE MOX PLANT

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Abstract

The present paper focuses on the criticality studies performed by the Engineering division of Belgonucleaire. These are one of the two pillars of the criticality prevention implemented for the Belgonucleaire MOX producing plant.

1. INTRODUCTION

Belgonucleaire operates a mixed plutonium/uranium oxide (MOX) fuel producing plant located in Dessel, Belgium. Since 1986, the plant is working at an annual capacity of 35-40 tons of MOX, using a MIMAS fabrication process. A diagram of the fabrication process is given in Figure 1. MOX is manufactured by mixture of plutonium oxide and depleted or natural uranium oxide powders. The whole production line is installed in dry glove boxes.

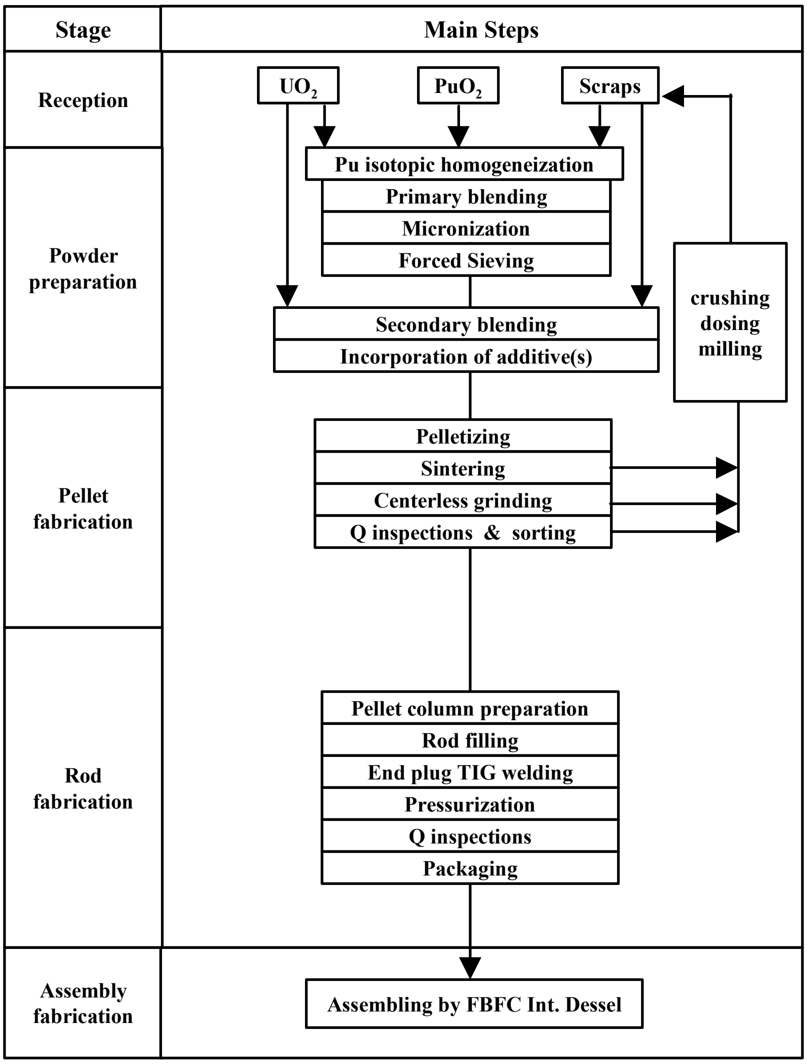

FIG. 1. MIMAS production process at Belgonucleaire.

Due to the important throughput of fissile material, criticality prevention is one of the major concerns for the safe operation of the plant. In that sense, it must address all the steps shown in the Figure 1. Criticality safety is relying on two pillars which are the criticality studies (the theory) and the reliable operational practice. As the human factor is the most difficult aspect to manage, a quality assurance system has been developed for both the definition of the specifications and the operational practice. The interested reader is referred to Ref. [1] for the latter aspect. The present paper focuses on the criticality studies, the first pillar of criticality safety.

2. GENERAL METHODS

The evaluation of the configurations of fissile material is based on criticality calculations, performed by the Engineering division of Belgonucleaire for the main part. The authorized amount of fissile material in parts of the production line is based on *safe mass* (1-Dimension) calculations. For more complex configurations, and for storage structures, *safe geometry* (3-Dimensions) calculations are performed.

The safe mass is determined from the computation of reflected spheres. Pessimistic assumptions are taken for the plutonium content, the isotopic composition and the density and humidity of the material. The applied uncertainties take into consideration the double batching of fissile material (50%), calculation errors and heterogeneity effects (15%) and an extra arbitrary margin according to the licence prescriptions of the plant (10%).

For the safe geometry calculations, besides the pessimistic assumptions on the fissile material, the characteristics of the infrastructure (absorbing and reflecting materials) are also taken in a conservative way in the model. The calculations are generally performed by means of Monte Carlo codes which provide the effective neutron multiplication factor k_{eff} with its standard deviation σ.

The criterion

$$k_{eff} + 3\sigma < 0.95$$

must be satisfied in both normal and accidental conditions. This is compliant with the ‘double contingency principle’: no single error, regardless of its occurrence probability, may lead to criticality. Variable density of water is always considered to pursue the moderation optimum.

Computations for new configurations or for modified installations are always initiated by a calibration calculation: the previous configuration or a similar installation is modelled in order to verify the coherence between present calculations and the older ones.

The validation of the computer chain and its application to the MOX plant was audited by the independent inspectorate Association Vinçotte Nucléaire (AVN).

3. COMPUTING TOOLS

To achieve the calculations, several codes are used in routine.

The 1-D problems (safe mass determination) are calculated with the transport code (deterministic) ANISN [2], coupled with the cross sections library (16 energy groups) developed by Hansen & Roach [3].

For the 3-D problems (safe geometry problems), Monte Carlo calculations are performed with the code KENOVa [4]. The cross sections for non fissile materials (CH_2, Cd, Pb) are here again taken from the H&R 16 groups library. For the fissile material, the cross sections are prepared through a calculation performed by the multi-purpose code WIMS-8a [5]. This modular code uses its own library WIMS'97 (69 or 172 energy groups) based on JEF2.2. For instance, for a storage of rods, the typical route is a pin cell calculation terminated by a smearing of the fuel, cladding and moderator (air between rods with a certain amount of water). Then a condensation is performed in order to obtain the fissile material cross sections within 16 groups, at the order P_1. These cross sections are read and written in a ANISN format by an interface program.

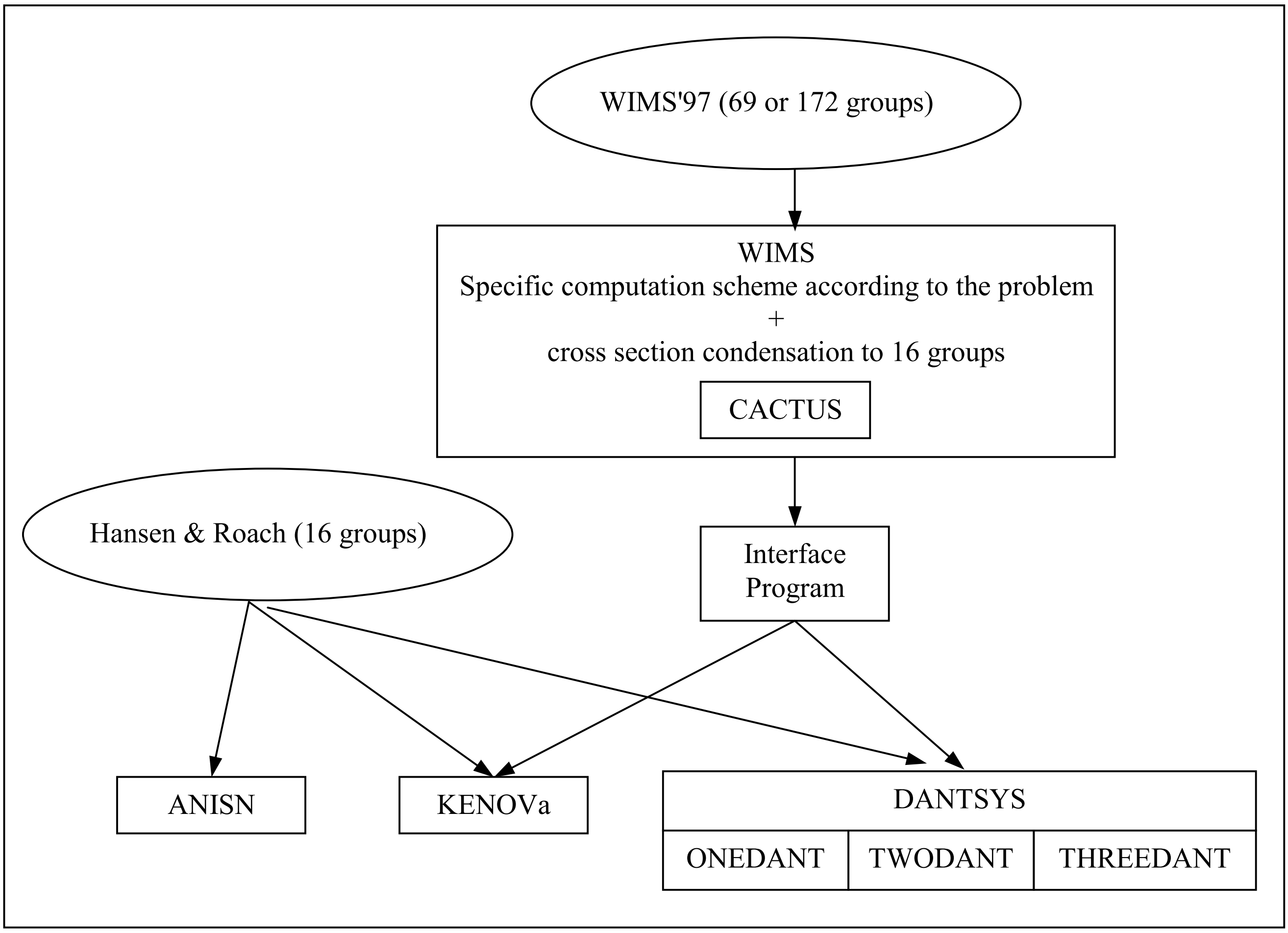

FIG. 2. Structure of the computing tools used for the criticality studies at Belgonucleaire.

Figure 2 shows the computing tools used in Belgonucleaire for the criticality studies performed for plants of the fuel cycle. It is worth noting that

- 1-D problems may also be performed following the WIMS-KENOVa, or H&R-KENOVa, or WIMS-ONEDANT, or H&R-ONEDANT route,
- 3-D problems may be performed either with the WIMS-THREEDANT chain or using KENOVa with H&R cross sections.

This affords cross checks made by the same or by other engineers, increasing the confidence in the calculated results.

DANTSYS [6] is a package of deterministic transport Sn computer codes allowing to describe a wide range of geometries. Only the three most employed codes of the package DANTSYS are listed in Fig. 2., namely ONEDANT, TWODANT and THREEDANT, respectively for 1, 2 and 3-D problems. ONEDANT employs the same calculation method as ANISN but is quicker. In that sense, the separation between ANISN and ONEDANT, appearing in the Figure 2, is mainly due to historical reasons and because ONEDANT is included in a much larger package. Two dimensional Sn transport calculation is also implemented in the WIMS package as the TWOTRAN module. THREEDANT is the most interesting tool for describing X-Y-Z geometries. Other options such as hexagonal pitch exist.

When the geometry of the problem may be simplified in a conservative way, the CACTUS module of WIMS reveals to be very efficient and flexible. This is a 2D transport calculation, using the ‘characteristics method’ [7]. It affords to describe very general and complex geometries. In reactor physics, for instance, CACTUS is used to calculate PWR, BWR and WWER fuel assemblies.

The computer codes and their use are validated against various experiments :

- critical experiments from open literature [8];

- proprietary programmes and international programmes with the Belgian VENUS reactor : VENUS PRP experimental programme (> 200 experiments), VIP experiments (VIP-PWR, VIP-BWR, VIPEX, VIPO) in which the reactor is loaded with high Pu content MOX rods;
- international programmes in progress: REBUS (validation of burn up credit through experiments carried out with spent fuel) and KEOPS (sub-critical measurement of the k_{eff} on powders and pellets).

Moreover, Belgonucleaire is taking part in various international benchmarks coordinated by the OECD.

4. TYPICAL EXAMPLES

The criticality studies are performed in order to get various parameters, for instance:

- the minimum mass and/or minimum volume (1-D problems) of a fissile material leading to criticality, from which the safe mass and/or safe volume can be deduced. For such problems, the density of the moderator in the mixture is calculated by $d_{mod} = \left(1-\left(d_{OX}/D_{OX}\right)\right) D_{mod}$, where D_{mod} and D_{OX} are respectively the theoretical densities (no mixture) of the moderator and fissile oxide and where d_{OX} is varied from zero to D_{OX} in order to get parameterised curves as shown in the Figure 3 ;
- the sub-criticality level of a storage. In Figure 4, for example, the k_{eff} level in accidental conditions is obtained in two steps: first the individual boxes are gradually flooded with water, the space between the boxes staying dry. Then the water content in the ambient air filling this space is modified in order to catch the maximum interaction between the boxes ;
- the maximum amount of fissile material (rods in a storage room, powder boxes, cans) ;
- geometry reactivity effects. Figure 5 shows an example of a studied configuration for cans in a glove box ;
- the material reactivity effect, indicating whether a neutron absorbing material must be included in the conception of a storage structure ;
- the reactivity effect of the reflector, accounting for the presence of operators near the fissile material ;
- etc.

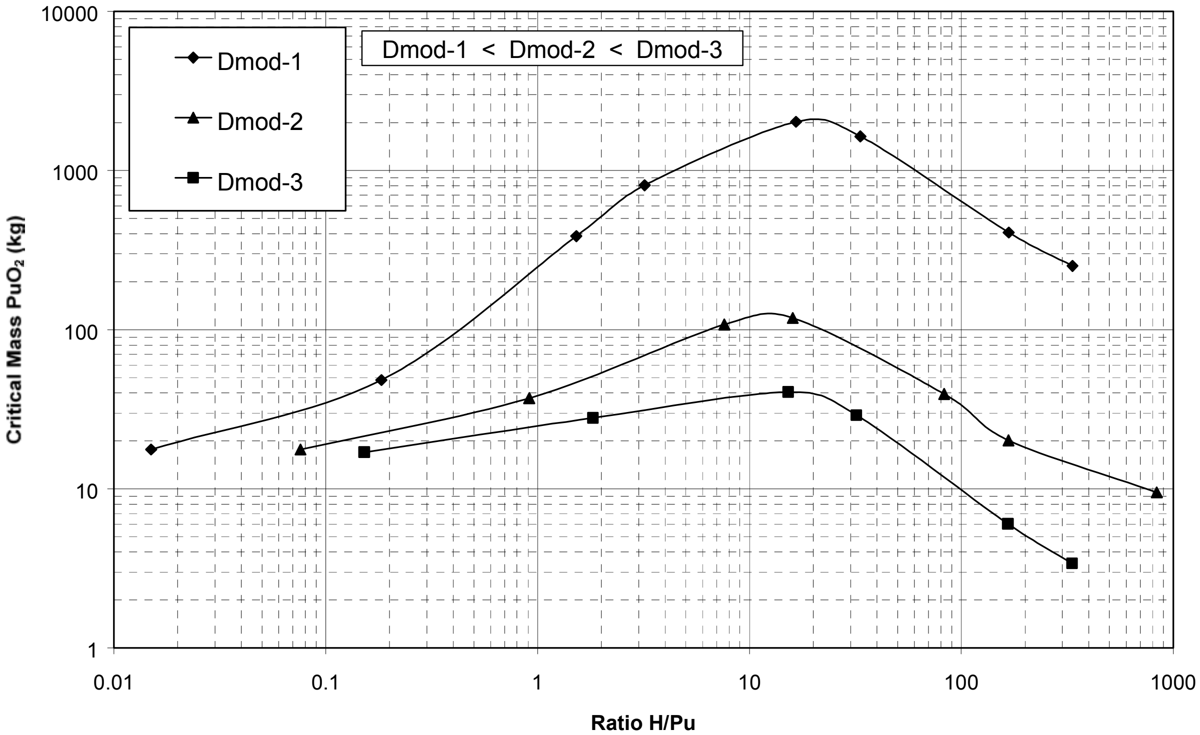

FIG. 3. Example of critical mass versus ratio H/Pu

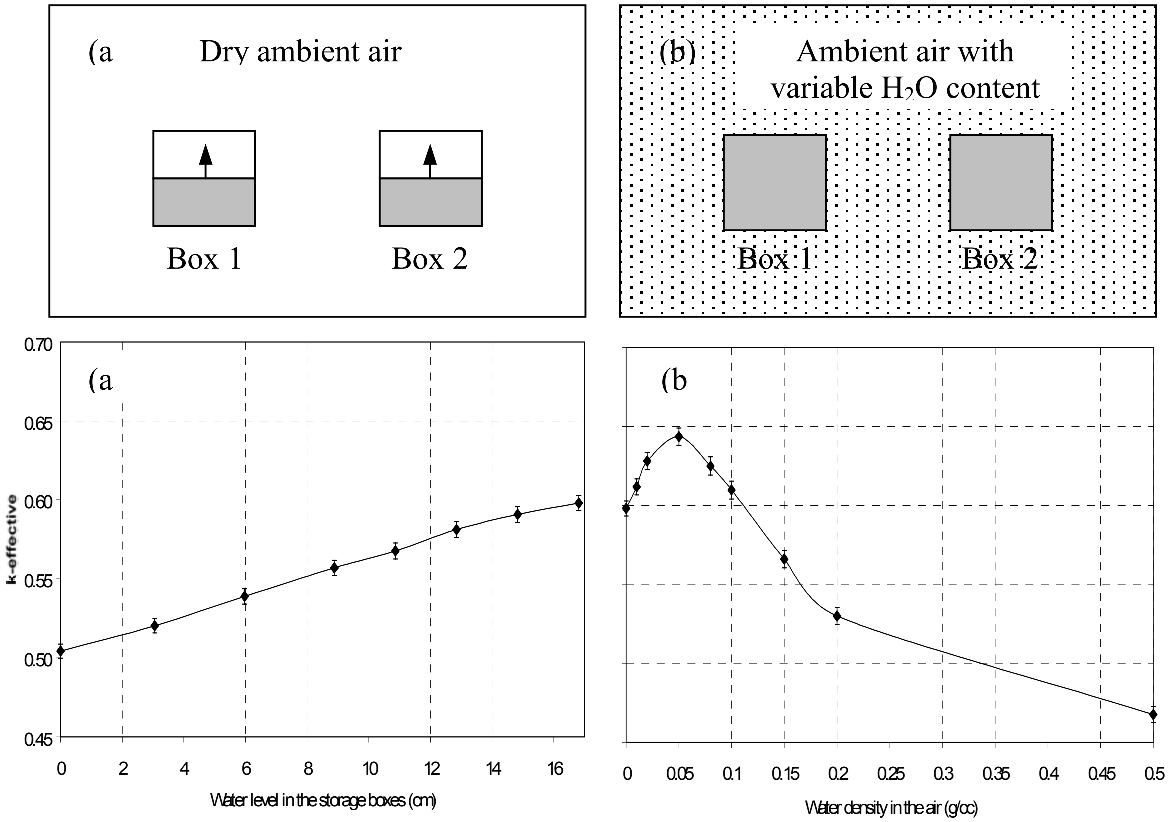

FIG. 4. Accidental k_{eff} levels in a storage room. (a) The water level is first raised into individual storage boxes. (b) Then the water density of the air is varied to find the maximum interaction between the boxes, leading to the maximum k_{eff}.. The bars indicate 3σ.

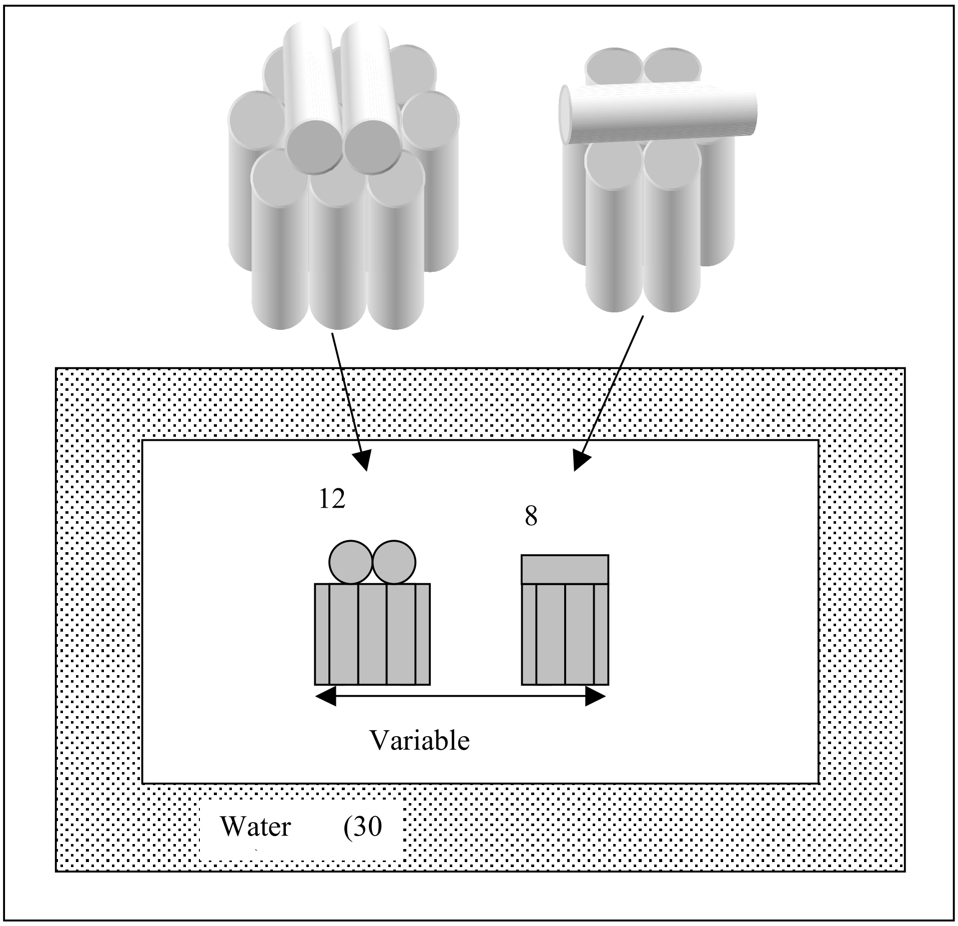

FIG. 5. Model for reactivity effect evaluation due to the geometry of cans in a glove box

The fissile materials frequently considered at Belgonucleaire are PuO_2, PuO_2-UO_2 (MOX) and Pu. These may be mixed with other compounds containing mostly O, H and C. The reference isotopic Pu (239/240/241/242) vector is usually taken as 70/18/10/2 but other possibilities can be studied. The physical form of the fissile material may be solid (rod, pellet), liquid (samples for chemical analysis), powders (master and secondary blends) and scraps. The powder density may be varied from 2 to 5 g/cm^3 and contains as much as 5 w/o water.

5. CONCLUSIONS

For the criticality prevention of its MOX plant, Belgonucleaire considers the two aspects, ‘good reliable operational practices’ [1], and the ‘technical specifications’ as two pillars of criticality safety. The technical specifications are defined on the basis of the criticality studies performed by the Engineering Division for the most part. Various codes and libraries are therefore used in routine and, due to the great flexibility of the computing tools, additional cross checks are performed by other computing routes. The same tools are also used for other criticality purposes: spent fuel storage pool, shipping casks, etc.

Besides these studies, mainly focused towards the criticality level, kinetic aspects should be considered in the near future.

References

[1] KOCKEROLS, P., ‘Criticality prevention in a MOX fuel plant’, WONUC, Moscow, Erchevo (2000).

[2] ENGLE, W. W., A Users Manual For ANISN - A One Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering, K-1693, Oak Ridge, Tennessee (1973).

[3] HANSEN, G. E. and ROACH W. H., Six And Sixteen Group Cross Sections For Fast And Intermediate Critical Assemblies, LAMS-2543 (1961).

[4] PETRIE, L. M. and LANDER, N. F., KENO-Va: An Improved Monte Carlo Criticality Program With Supergrouping Computing Telecommunication Division At ORNL (1990).

[5] WIMS8a, The ANSWERS Software Package, a General Purpose Neutronics Code, AEA Technology (1999).

[6] ALCOUFFE, R. E., BAKER, R. S., BRINKLEY, F. W., MARR, D. R., O'DELL, R. D., WALTERS, W. F., DANTSYS: A Diffusion Accelerated Neutral Particle Transport Code System, LA-12969 (1995).

[7] HALSALL, M. J., "A Users Guide to the WIMS-E Module WCACTUS", AEEW-R 1710 (1983).

[8] INTERNATIONAL HANDBOOK OF EVALUATED CRITICALITY SAFETY BENCHMARK EXPERIMENTS, NEA/NSC/DOC(95)03 (2000).