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Nuclear Reaction Data and Nuclear Reactors: Physics, Design and Safety

13 March - 14 April 2000

Miramare - Trieste, Italy

Core Physics Calculations

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WORKSHOP ON NUCLEAR REACTION DATA AND NUCLEAR REACTORS: PHYSICS, DESIGN AND SAFETY

23 February - 27 March 1998

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TRIESTE, International Centre of Theoretical Physics,

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1. Objective

1.1 General

In order to satisfy the needs of utilities, to be accepted by the public and to meet safety requirements, a nuclear power plant must produce electrical energy as safely and as cheaply as possible.

To achieve these two goals, the reactor system must be designed and operated so as to:

- ensure the safety of the plant with respect to radioactive release under all normal, degraded, incident or accident conditions;
- maximize the energy extracted from the fuel in accordance with the operating strategies adopted.

Accident severity is generally classified on the basis of the potential radioactivity likely to be released. This classification comprises several levels. Each of them is specified according to a set of very general criteria. The purpose of these criteria is to limit the potential risk associated with an event to values acceptable to the public in terms of probability and of the population exposed. The nuclear power plant designer must, for his part, translate these criteria into constraints that can be applied to the project. These constraints are determined on the basis of:

- the adoption of a global safety approach;

- the analysis of the plant behaviour and the resulting response of materials forming the various barriers under normal, degraded or extreme conditions;

- the capacity to control the overall system operating mode under all credible conditions.

In core physics practice, the main constraints are imposed at the design stage on the local values of linear power density, effective heat transfert between the fuel cladding and the reactor coolant and mechanical strength of the fuel. These constraints define a plant unit operating range in terms of alarm setpoints and controls preventing the occurrence of potentially hazardous situations. In addition, they allow fuel management schemes to be defined that meet the safety criteria.

The design limits are defined in conjunction with a methodology for assessing uncertainties and margins which translate the need to ensure that the results of safety analysis are conservative in view of an imperfect knowledge of the parameters. This lack of precision mainly arises from:

- the basic data (cross sections, physical and chemical properties of the materials, etc.) and approximations and simplifications in the numerical calculations;
- the anticipatory nature of the design calculations.

In fact, although the uncertainty ranges associated with basic data and methods are quantified in the qualification process and are supplied with the design code packages, the manufacturing parameter variation laws, the characteristics of the monitoring devices and the loading contingencies are only known once the design has been completed. It is therefore impossible to explicitly incorporate into the design the range of variation of the associated parameters. And, even it were possible in theory, it would turn out to be quite impracticable at the project stage owing to the enormous amount of data involved.

1.2 Project Parameters: Reactivity and Power Distribution

Under the "containment barriers" safety approach, the neutronic design of power reactor cores must comply with the essential goal of guaranteeing fuel integrity in the face of any form of attack. Fuel integrity may be compromised by several factors which can be interconnected:

- an increase in the heat source due to an upset nuclear system neutron balance;
- a reduction in the heat exchange capacities between the fuel material and the coolant, caused by a reduction in the quantity of the latter in the primary system;
- a local degradation of the heat exchange quality between the fuel rod cladding and the surrounding coolant, either due to the presence of deposits on the cladding surface or due to the formation of a vapour film or to partial dryout of the channel by plugging;
- a local overpressure, due to pellet - cladding interaction and/or the build-up of fission gases;
- mechanical and thermal-mechanical effects: distortion, contact shock or coolant jet produced by the particular shape of components.

Apart from effects of a mechanical nature, all the causes of degradation of the "first barrier" listed above can be traced to a nearby or remote common origin, i.e. a localized and/or generalized rupture of the equilibrium between the nuclear generation of thermal energy and the capacities of the system to remove it under physically acceptable conditions.

The condition of a nuclear reactor core is governed by two main parameters:

1) the reactivity, which measures, under transient conditions, the deviation of the system neutron population from equilibrium conditions and, under steady-state conditions, the deficit in the neutron balance (as an integral value, this parameter is unique and independent of position at all times for a given core configuration);

2) the power, which represents in macroscopic terms the spatial distribution of fission events in the core. This parameter depends on the properties of the medium and on the space and energy distribution of the neutrons within the multiplying system. Consequently, it may be extremely sensitive to local conditions.

At the design stage, the purpose of the core physics calculations is to evaluate these two parameters with the maximum accuracy compatible with the size and complexity of the problems to be treated, the application of numerical methods and the limits imposed on their qualification by experimental uncertainties. The values thus calculated, with the attendant uncertainties and margins, are then compared to the design limit values established to ensure that the safety criteria are respected.

The reactivity and the power distribution, which possess equal importance in the design studies, are not homogeneous quantities, however, and are different in kind.

While the power is the image of the instantaneous distribution of fissions in the multiplying system at each instant (it therefore represents an objective reality), the reactivity, under steady-state conditions, does not have any real physical meaning. As a mathematical object, it translates the requirement ever present in the studies to describe by static and homogeneous equations the conditions in which the neutron population is not in equilibrium. In this case, instead of explicitly inserting in the balance an inhomogeneous term expressing this disequilibrium:

$$A_0\Phi_S + P_0\Phi_S = S \qquad (1\text{-}1)$$

where A_0 and P_0 represent respectively the absorption plus leakage and production operators of the multiplying system and S is a generic source term representing either the neutron surplus or the deficit, the following approximation is made implicitly:

$$S = \rho_0 P_0 \Phi_0 \qquad (1\text{-}2)$$

where:

$$\Phi_S = \Phi_0 + \delta\Phi \qquad (1\text{-}3)$$

and ρ_0 is a proportionality constant called the **reactivity**, Φ_S is the solution to equation (1-1) and $\delta\Phi$ is the difference between Φ_S and Φ_0, the latter being the solution to the eigenvalue equation derived from equation (1-1) :

$$\left[A_0 + (1-\rho_0)P_0\right]\Phi_0 = 0 \qquad (1\text{-}4)$$

This approximation is equivalent to taking into account only the Φ_S component parallel to the fundamental mode, since the function $\delta\Phi$ in Eq. (1-3), defined as the difference between Φ_S, the solution of the inhomogeneous equation, and the fundamental mode, must strictly satisfy the equation:

$$(A_0 + P_0)(\Phi_0 + \delta\Phi) = \rho P_0 \Phi_0 \qquad (1\text{-}5)$$

which, in the light of Eq. (1-4) implies the relationship :

$$(A_0 + P_0)\delta\Phi = 0 \qquad (1\text{-}6)$$

and which, finally, in accordance with the "Fredholm alternative" (REFs 1 and 2), has one unique solution : $\delta\Phi = 0$.

Thus, an equation that is inhomogeneous by nature is converted into its homogeneous equivalent with a considerable gain in the simplicity and, above all, in the speed of the calculations.

This approach requires some comment :

- By laying down the condition of proportionality between the inhomogeneous term and the neutron flux [Eq. (1-2)], the latter is arbitrarily emptied of all modes apart from the fundamental: in other terms, the system is projected onto the fundamental mode artificially generating, between the actual and calculated flux values, differences that may be considerable in a few particular cases (shut down sub critical reactor, fast transients, etc.).

- On the other hand, the reactivity is, by definition, quite insensitive to higher modes: being linked to the fundamental mode, its value is identical whether it is calculated in the actual state of the system or in its projection onto the fundamental mode. We thus dispose of an absolute parameter that only depends on the intrinsic properties of the multiplying system and remains unaffected by its state.

Eq. (1-4) can also be written in other forms. By defining :

$$k_0 = \frac{1}{1-\rho_0}$$
$$\lambda_0 = \frac{1}{k_0} \qquad (1\text{-}7)$$

where the parameter k_0 is called the **effective multiplication factor**, the canonical forms are obtained :

$$A_0 + \frac{P_0}{k_0}\Phi_0 = 0 \qquad (1\text{-}8)$$

or :

$$A_0\Phi_0 + \lambda_0 P_0\Phi_0 = 0 \qquad (1\text{-}9)$$

Finally, Eq. (1-9) has an adjoint, containing the operators A_0^t and P_0^t, transposes of A_0, P_0 :

$$A_0{}'\Phi_0^* + \lambda_0 P_0{}'\Phi_0^* = 0 \qquad (1\text{-}10)$$

the solution of which, Φ_0^*, is called the adjoint flux or "neutronic importance".

By multiplying the two sides of Eq. (1-2) by Φ_0^* and rearranging, we obtain the generic definition of reactivity :

$$\rho_0 = \frac{\langle \Phi_0^*, S \rangle}{\langle \Phi_0^*, P_0\Phi_0 \rangle} \qquad (1\text{-}11)$$

where, as usual, the symbol $< >$ indicates an integration over space, here the whole of the system, and over energy.

Since every nuclear system operates under criticality conditions ($\rho_0 = 0$), equation (1-11) also becomes, by extension, the definition of the reactivity change $\delta\rho$.

Another definition of the reactivity change can be obtained without introducing any approximations, from the state equation of the system after perturbation, which is written generically :

$$A'\Phi' + \lambda' P'\Phi' = 0 \qquad (1\text{-}12)$$

where

$$
\begin{aligned}
A' &= A_0 + \delta A \\
P' &= P_0 + \delta P \\
\Phi' &= \Phi_0 + \delta\Phi \\
\lambda' &= \frac{1}{k'} = (1-\rho') \\
\rho' &= \rho_0 + \delta\rho
\end{aligned}
\qquad (1\text{-}13)
$$

A_0, P_0, Φ_0, ρ_0 being, respectively, the operators, the eigenfunction and the reactivity of the reference state before perturbation.

By directly substituting the definitions of Eq. (1-13) in Eq. (1-12) we obtain :

$$
(A_0 + \delta A)\Phi' + [1-(\rho_0+\delta\rho)](P_0+\delta P)\Phi' = 0 \qquad (1\text{-}14)
$$

By multiplying the left side of Eq. (1-14) by Φ_0^*, we obtain :

$$
< \Phi_0^*, (A_0 + \delta A)\Phi' > + < \Phi_0^*, [1-(\rho_0+\delta\rho)](P_0+\delta P)\Phi' >= 0,
$$

and

$$
< \Phi_0^*, A_0\Phi' > + < \Phi_0^*, (1-\rho_0)P_0\Phi' > + < \Phi_0^*, \delta A\Phi' > + < \Phi_0^*, (1-\rho_0)\delta P\Phi' > - < \Phi_0^*, \delta\rho P'\Phi' >= 0
$$

By rearranging and applying the properties of the adjoint flux, we obtain again :

$$
< \Phi_0^*, [\delta A + (1-\rho_0)\delta P]\Phi' >=< \Phi_0^*, \delta\rho P'\Phi' > \qquad (1\text{-}15)
$$

which finally gives :

$$
\delta\rho = \frac{< \Phi_0^*, [\delta A + (1-\rho_0)\delta P]\Phi' >}{< \Phi_0^*, P'\Phi' >} \qquad (1\text{-}16)
$$

This expression for the reactivity change is strictly equivalent to the one classically obtained from Eq. (1-11).

All the other definitions, notably the well-known first order definition derived from Classical Perturbation Theory (REFs 3 and 4) :

$$
\delta\rho = \frac{< \Phi_0^*, [\delta A + (1-\rho_0)\delta P]\Phi >}{< \Phi_0^*, P\Phi >} \qquad (1\text{-}17)
$$

and that commonly used in design:

$$\delta\rho = \ln\frac{k'}{k_0} \qquad (1\text{-}18)$$

are approximations of Eq (1-16). Consequently, they are only valid in limited cases and certainly not generically.

2. BASIS OF CALCULATION TECHNIQUES

2.1 Boltzmann Equation and Diffusion Approximation

The solution of the Boltzmann Eq. (1-9) by numerical means involves its discretization in space, energy and probability of interaction and/or direction of neutrons depending on the approximation chosen for the transport operator.

However, the system obtained after discretization is generally too complex for the calculational capacities of industrial machines and for execution times compatible with design constraints. It cannot therefore be solved in its original form without calling on other simplifications. In design, therefore, the diffusion approximation is generally adopted.

The diffusion system state equation can be obtained from the transport equation, either on the basis of physical considerations or mathematically, by an asymptotic expansion (see REF. 5).

The generic diffusion equation is written, after energy condensation, space homogenization and discretization of the space and energy operators, as follows :

$$A_0\Phi_0 + \lambda_0 P_0\Phi_0 = 0 \qquad (2\text{-}1)$$

with : $\lambda_0 = \frac{1}{k_0}$

where

$$A_0 = \begin{pmatrix} A_1 & 0 & . & . & 0 \\ -\Sigma_{2\to1} & A_2 & 0 & . & 0 \\ -\Sigma_{3\to1} & -\Sigma_{3\to2} & A_i & . & . \\ . & . & . & . & . \\ -\Sigma_{G\to1} & -\Sigma_{G\to2} & . & -\Sigma_{G\to G-1} & A_G \end{pmatrix}$$

(G being the number of energy groups)

with : $A_i = -\mathrm{div}(D\mathrm{grad}) + D_i B^2 + S_{ai}, \qquad \leq i \leq G$

and : $P_0 = [\chi_{i,j}] * [\nu\Sigma_{f_{i,j}}], \qquad \leq i,j \leq G$

where

$[\chi_{i,j}]$ is the matrix giving the energy distribution of neutrons generated by fissions and

$[\nu\Sigma_{f_{i,j}}]$ is the system multigroup fission operator having the form

$$[\nu\Sigma_{f_{i,j}}] = \begin{bmatrix} \nu\Sigma_{f1} & 0 & . & 0 \\ 0 & \nu\Sigma_{f2} & . & 0 \\ 0 & 0 & . & 0 \\ 0 & 0 & . & \nu\Sigma_{fG} \end{bmatrix}$$

In all practical cases, the energy coupling between neutrons generating fissions and those generated by fissions is neglected, this being justified by a statistical analysis. Consequently, the term $[\chi_{i,j}]$ can be simplified by converting it into a vector with the following form (see REF. 6) :

$$[\chi_i] = \begin{bmatrix} \chi_1 \\ \chi_2 \\ . \\ \chi_G \end{bmatrix}$$

Operators A_0 and P_0 P_0do not explicitly depend on time, are homogeneous in space and energy subdomains and vary in dimension depending on the type of discretization chosen.

Thus, for the PWR calculations, only two energy groups are normally employed: the first so-called "fast" group, which covers all neutrons with an energy exceeding 0.625 eV, and the second so-called "thermal" group, which includes neutrons with an energy of less than 0.625

eV. This distribution is justified by the dependence on the energy of the fission cross sections of U235, the fuel that is conventionally used in PWRs. The exponential increase in U235 cross sections in the thermal range in this type of reactor enables neutrons belonging to the second group, despite their small number, to generate over 70 % of the fissions.

As far as space is concerned, it is usual to carry out the calculations on the fuel pin scale, since it is the power of the hottest pin under normal, degraded, incident and accident conditions that determines the conditions that might make it difficult to meet the safety criteria.

Eq. (2-1) is associated with boundary conditions which, depending on the geometry of the problem, may consist either in zero flux or in zero current at the interfaces. They take on the following general form:

$$\alpha\Phi_0 + \beta\frac{\partial\Phi_0}{\partial n} = 0 \qquad (2\text{-}2)$$

2.2 Stage Separation: "Cell" Computations and "Core" Computations

To find the solution to Eq. (2-1), along with its boundary conditions [Eq. (2-2)], it is essential to know the explicit values of all the space and energy components of the matrix operators A_0 and P_0 in each volume element of the hyperspace within which the problem is defined and bound. In the diffusion approximation, the number of dimensions is determined by the three space dimensions plus the number of groups chosen for the energy discretization.

Owing to the specific physical features of the interaction process between neutrons and matter, its energy dependence and the geometry of the media involved, the calculation of these components is generally very complex. That is why it is always carried out in modern design packages by codes that are generically called "cell" codes in opposition to those that are specially charged with solving the reactor neutron balance equation, which are classically called "core" codes.

In the design packages, the separation of functions between the "cell" and "core" codes expresses the need to apprehend and process independently the two heterogeneity levels of the calculation :

1) interaction of neutrons with matter, with energy and space characteristics conditioned by local effects involving the geometry of the media, the energy dependence of the cross sections and the energy distribution of the neutrons (neutron spectrum);

2) the system neutron balance, a more global and macroscopic parameter, which first of all translates the coupling between core regions. This coupling is kept going mainly by the fast neutrons, which, in a first approximation, are insensitive to local effects.

The first aspect includes very accurate and detailed modelling of the physical media with a high number of energy groups but limited to a restricted spatial region (of the order of the neutron mean free path distance). It is processed by "cell" codes. The second, on the other hand, requires a description of the overall system that may be highly simplified in terms of the number of energy groups and the description of these media. This description falls within the scope of the "core" codes.

The "cell" codes have features that vary with their age and design, the sophistication of their physical models and the type of data they must supply to the diffusion codes with which they are coupled, but all of them share several main features, i.e. :

- they use a very large amount of basic data (microscopic cross sections of all the isotopes composing the media, isotopic concentrations, detailed geometry, etc.);

- they call on a set of approximations and physical models, backed by a sophisticated numerical calculation process with which they are intimately interconnected;

- they handle big amounts of data, not only during execution of the actual calculation, but also in the output process when conditioning of the data to be transmitted to the downstream "core" codes is required.

2.3 Generation of Constants for the Core Computations

The microscopic cross sections of all the isotopes contained in the core and its immediately surrounding media are compiled in libraries issued by international organizations.

At present, in Europe, the JEF-2 (Jointly Evaluated File) library is used. This library, issued in 1992, is the fruit of international collaboration between European countries and Japan, extending recently to other partners, including the U.S.

The data contained in these libraries cannot be used directly. Preliminary processing integrating the resonance parameters, group formation and energy condensation must be performed before going on to format the data in the shape required by the "cell" codes.

In France, this task is carried out by the NJOY - THEMIS code pair which, in the PWR case, generates two types of library, one with 99 groups and another, more detailed one, with 172 energy groups.

2.4 Numerical Approximations

The discretization of a problem defined and bound in a limited domain always involves defining points, segments or subspaces belonging to the same domain, to which the explicit calculation of the numerical values is limited.

The relations to be established between the unknowns at the chosen positions (mesh points) depend on the mathematical formulation of the problems to be treated and their boundary conditions. In any case, they are never strictly accurate: the discretization process introduces various kinds of errors (truncations in series, rounding in numbers, etc.) which, particularly in finite-difference techniques, significantly impact the current solution.

Once convergence has been achieved, the difference between the solution obtained and the exact solution increases as the approximations involved in the discretization process become more demanding. The solution obtained by numerical means coincides with the exact solution only at the limit when the calculation mesh becomes infinitely fine (see REFs 7 and 8).

It is possible to mitigate, at least partially, the drawbacks due to meshing by coupling the discretization with a polynomial expansion of the required solutions between the points (nodes); this is a feature of the "finite-element" and "advanced nodal" methods (see REFs 9 and 10).

These general considerations all apply to the use of numerical means to solve the neutron balance equation. In this field, significant advances have been made in recent years as a result of the progress achieved in solving techniques. We now have a wide range of methods available that can be broken down into four main families:

- finite-difference;
- finite-element;
- mixed finite-element;
- nodal methods.

2.5 Mesh-Dependent Corrections to Operators

In addition to simplifying assumptions in the solutions, Eq. (2-1) comprises two additional approximations affecting the values of the operator components, i.e. :

1) the explicit time-independence of the system;

2) the generalized knowledge of operator components throughout the system.

The first approximation implies that system changes over time are considered as an infinite succession of steady states, each differing only slightly from its predecessor and being completely independent and uncoupled from it.

In the absence of external perturbations, the adiabatic approximation is perfectly justified owing to the difference existing between the time constants involved:

- the neutron life time, a few tenths of a second for those with the longest lives (less than one per cent of the total), an infinitesimal fraction of a second (from 10^{-3} to 10^{-7} second depending on the system under consideration) for all the others;
- the change in the nuclear properties of the media following the production of energy several hours, or even several days before the slightest significant effect can be detected.

During this time, the operators, and here is the second approximation, are assumed to be known at all points in the system.

However, even if this knowledge is achievable from a theoretical point of view, since there are no physical reasons against it, in practice, owing to cost, memory allocation and computational efficiency considerations, only the cross sections of a very limited number of points are directly accessible. All the others may be reconstructed from these data (which correspond to special physical conditions) by algorithms taking into account the local values of the parameters. The most modern and most sophisticated methods apply interpolation in parameterized tables, others employ analytical expansions, and yet others use a combination of both of these.

The approximations described above require the presence in the "core" computer codes of subroutines capable of:

a) updating the data on the isotopic compositions of the nuclear fuel so as to incorporate the effects of energy production. This generally involves the so-called "burn-up calculation subroutines that employ analytical formulations or, more often, matrix solution methods to solve the nuclide change equations node by node in simplified form,

b) for PWR calculations, supplying the local data needed to define the interpolation parameters. Owing to the close coupling existing between the production of heat and the local physical conditions of such reactors (temperature, moderator density, etc.) these data are produced by code functions simulating a thermal-hydraulic calculation in simplified form, or else they are transmitted directly by a thermal-hydraulic calculation module coupled with the "core" code.

The data thus calculated allow the local physical parameters to be evaluated, provided the parameterized tables prepared by the "cell" code are interpolated to update, at each time step and at each node, the components of the operators defining the state equation of the multiplying system.

3 COMPUTATION SCHEMES

3.1 Structure and Qualification of Computational Code Packages

The core physics design of a nuclear power plant involves performing a large number of calculations, the purpose of which is to apprehend values describing the behaviour of the system under all operating conditions. We have seen that, owing to the complexity of the equations and the number of parameters, these calculations cannot be carried out by analytical methods. Consequently, design requires codes capable of solving the problems by numerical methods. These codes need high-capacity computers, either large mainframes or work stations connected together in a network.

The codes reflect the approach applied in the industrial design of nuclear power plants and are designed with the express purpose of solving, independently from each other, a specific and limited portion of the problems to be handled and to output results that can be used as the input data for downstream codes. It follows that, on the whole, the various codes form macro modules with a modular structure made homogeneous and coherent by the links existing between the elements forming it. Often, this complex structure benefits from a user macro language shared by all the modules, dynamic memory allocation and standardized input - output and information management procedures. The whole system is called a "design code package."

In general, each code possesses a range of computational options corresponding to the various approximation levels adopted in describing the behaviour of the reactor. Selection of a set of homogeneous assumptions from those available is then conditioned by the type of physical system to be studied and the degree of the representation desired by the user. The options must be chosen to ensure strict consistency between the various computational stages and preclude a situation where the accuracy achieved at a particular stage in the modelling of any phenomenon is compromised by any rougher approximations introduced subsequently. This consistent and homogeneous set of approximations, choices and options forms what is generally called a "design computation procedure."

The process of qualifying a computational scheme thus designed and structured, with its procedures, involves performing a large number of comparisons between the calculated and measured values of a fairly limited number of integral parameters, representative of the behaviour of the reactor and reactor operating models, using a set of experimental data representative of the situations facing the plant operator (see REF. 11).

The information gained from these comparisons allows the two main goals of the qualification to be attained, i.e. :

a) First, the exact definition of the range of applicability of the computation scheme, i.e. determination of the systems it is capable of analyzing without modification (this domain is defined in terms of the possible variation ranges of fundamental parameters such as, for example in conventional PWRs, the moderating ratio, the fuel enrichment and/or the type of absorbers used in the control rods);

b) Second, quantification of the level of uncertainty involved in the evaluation of such parameters, assuming the planned procedure has been applied in accordance with the instructions and within the validity interval defined above.

The experimental data that can be used in the qualification process may have the following sources, depending on the type or parameters required and the amount of information available :

- specially targeted experiments, carried out on critical mock-ups in Research Centres or

- start-up tests and operating experience of operating nuclear plants, i.e. "feedback of operating experience."

The critical experiments supply very accurate data on reactivity values and on fine reaction rate distributions. They also record the interaction effects of absorbers and measure their relative worth. Operating experience essentially involves the flux maps made by scanning the system at precise moments in time, cycle lengths, differential worth of absorber banks and power and temperature effects. In addition, it supplies information allowing irradiated fuel analysis to be carried out under certain conditions.

Qualification is essential to obtain the requisite permits from the Safety Authorities. It is just as necessary to assess the possibility of handling project changes or requests from the utility that entail using the code package outside the validity range already qualified.

3.2 PWR Design Code Packages at FRAMATOME

The SCIENCE code package, commissioned in 1995 at FRAMATOME for PWR core physics design, is a good example of a modern core analysis tool combining a high-level physical model with a user-friendly environment (see REF. 12).

The quality and accuracy of the results obtained with this package are based in the first place on the quality of the physical models it applies:

- transport calculations on fuel assemblies, collision probabilities method, implemented using the APOLLO 2-F code (industrial version of the APOLLO-II code - see REFs 13 and 14).

- 3-D core calculations by nodal methods with expansion and fine power reconstruction, coupled to a feedback model using parameterized tables produced by the SMART code (see REF. 15).

As stated above, the APOLLO-II code has been developed by the French Atomic Energy Commission (CEA). It was adapted, industrialized and qualified by FRAMATOME for its own needs and incorporated into the SCIENCE code in this last configuration under the Name of APOLLO 2-F.

SMART code has been developed jointly by FRAMATOME and BWFC, on the ground of the NEM (Nodal Expansion, Method) methodology.

You will find further information on the characteristics, performance and qualification of the SCIENCE Code Package in REF. 16.

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