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HORUS3D code package development and validation for the JHR modeling

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The definition and the validation of the HORUS3D package are described in this paper. HORUS3D (Horowitz Reactor simulation Unified System) is dedicated to the design studies of the future European MTR, called Jules Horowitz Reactor.

HORUS3D/N, based on APOLLO2 and CRONOS2 codes, allows the neutronics modeling of the Horowitz reactor with great confidence: compared to time-expensive reference calculations, the error on the assembly power distribution reaches only 1.4% and reactivity discrepancies do not exceed 150 pcm during the fuel cycle. An experimental program, organized in two phases, is planned. The first phase, called VALMONT, will allow the qualification of the calculation of the JHR fuel reactivity. The second phase, named AMMON, much more extensive than the first one, will allow the qualification of the assembly power map, the control rod efficiency, the reactivity coefficients, the Beryllium reflector, with the actual characteristics of the JHR assemblies.

HORUS3D/P is devoted to photonics calculations. An original scheme has been developed. It is based on a deterministic neutronics resolution, which prepares photon sources on the actual assembly geometry. Thereafter, the probabilistic code TRIPOLI4 is used to determine critical design parameters such as nuclear heating. An experimental program, ADAPh, is planned in order to decrease the important nuclear data uncertainties.

HORUS3D/Cy, based on the DARWIN package, allows to determine cycle design parameters such as residual heat with a great confidence thanks to a complete nuclear data and evolution chains. The experimental program: IRIS-γ will allow to extend the present qualification domain to the UMoAl fuel depletion.

HORUS3D/Th, the core thermal-hydraulics part, is based on the FLICA4 code and HORUS3D/Sys is devoted to System thermal-hydraulics problems. The specific characteristics of the JHR element led to a specific experimental program on the test facility called SULTAN-JHR.

KEYWORDS: MTR, JHR, modeling, neutronics, photonics, thermal-hydraulics

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1 Introduction

The complexity of the JHR fuel assembly geometry and the small dimensions of the core lead to neutronics and thermal-hydraulics modeling problems : strong flux gradients, neutron streaming effects, "flow redistribution", etc. Therefore, classical schemes cannot be used directly and a specific code package, named HORUS3D (HOrowitz Reactor simulation Unified System), is being developed.
This package has fulfill two main objectives : first, it is going to be used for the JHR Definition studies, which require a great accuracy in order to minimize the design margins. Secondly it has to prepare the future code platform to model the test devices with accuracy, from the design to the experimental interpretation. This point is critical to pass from a "cook & look" approach to a really predictive simulation.
The achievement of these objectives is hampered by the lack of appropriate integral experimental data. Hence, a rigorous methodical approach is required and a specific experimental program has been planned.
This paper presents the methodology, the development of the neutronics route, as well as the elementary validation against reference Monte Carlo results and the thermal-hydraulics developments.

2 General background

2.1 JHR description

The Jules Horowitz Reactor is the future European Material Testing Reactor [1,2]. The design studies are split in different phases. The preliminary design studies allowed to define the main options of the core and its elements, and to characterize a reference configuration. The Definition Studies will optimize the JHR's performances by testing modifications from the reference configuration.
The JHR assembly would be composed of 3x6 cylindrical fuel plates, maintained by 3 stiffeners (Figure 1). The external diameter of the assembly is close to 8 cm with an active height of 60 cm.
The fuel is composed of a dispersion of UMo powder in an aluminum matrix. The fuel plates are obtained in a rolling mill process. The Uranium-235 enrichment, less than 20%, respects the non-proliferation agreement. The central cavity can host either an aluminum filler, a hafnium control rod or an irradiation test device.
The preliminary core, about 100MW, consists of 46 assemblies, arranged in a triangular lattice inside a rectangular aluminum matrix (Figure 2). It is boarded on two sides by a beryllium reflector. The other two sides are left free in order to introduce mobile irradiation devices. These characteristics would be confirmed by the detailed studies.
The first kind of irradiation device consider is dedicated to material irradiation in the centre hole of an assembly and is called “isolated chouca” (Figure 1). Three “single chouca” can also be combined together in place of an assembly in a so called “clustered chouca”. The accidental ejection of such an experimental device adds a large quantity of water in the centre of the core and leads to strong effects on the safety parameters. Thus the goal of HORUS3D is also to determine with accuracy the neutronic parameters in accidental situations.
The second kind of irradiation device taken into account can be located either in the beryllium reflector or in a mobile mechanism in the water reflector, allowing to vary the neutron flux during the irradiation. This device, which is called “griffon” (Figure 2), accommodates a fuel pin in its centre under conditions similar to those in a PWR.

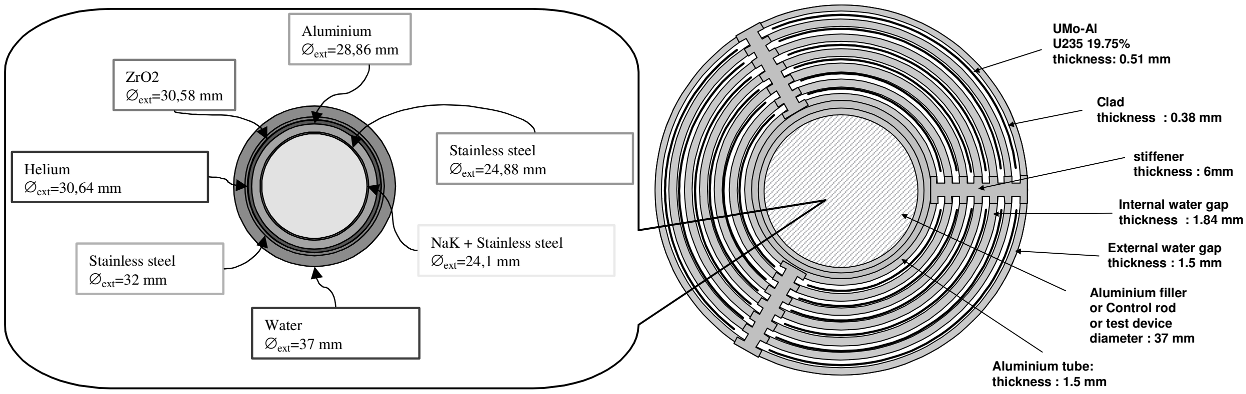


Figure 1 : Cross section of an experimental device "single chouca" inside the JHR assembly

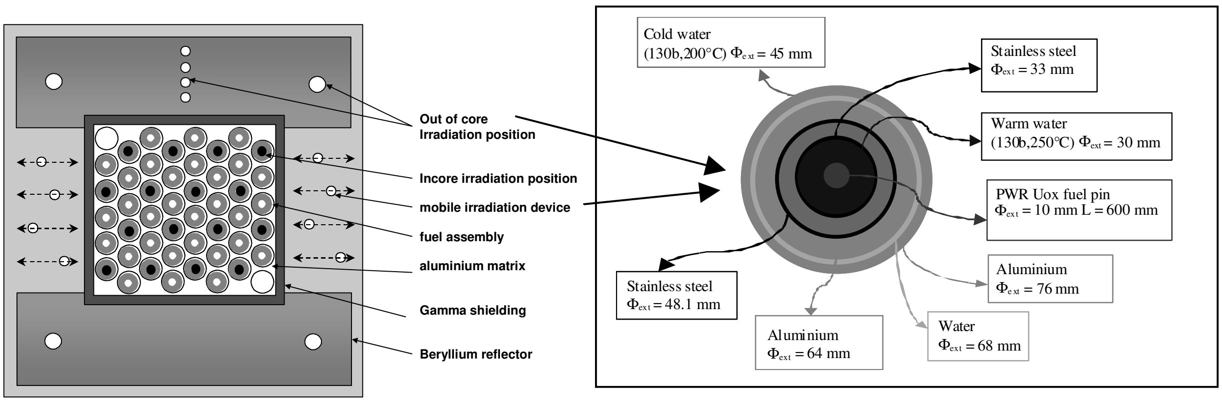


Figure 2 : Cross section of an experimental device "griffon" outside the JHR core

2.2 Strategy and Methodology

The JHR assembly and core design does obviously not allow the direct application of validated neutronics routes such as PWR schemes. Nevertheless, present codes are built of modules performing specific tasks (geometry, self-shielding, flux solver, etc.) and a user language is used to link the operators at run time. This allows a very flexible and powerful use of the codes, but leads to a more complex validation. Moreover, not only do the present JHR characteristics reduce the representative experimental data set, but the future design studies will also lead to important modifications. Therefore, the HORUS3D development strategy follows 3 axes :

1. Process splitting : an anticipating analysis of the modeling requests and the planning is essential to define the three different processes : Code developments, Scheme definition and validation, and the realization of a dedicated Experimental program.
2. The use of existing experimental validation studies, elaborated for other reactor types, is essential due to the lack of directly representative experiments. Therefore, the same options as in conventional schemes must be used in the JHR scheme as far as possible.
3. Incremental approach : The qualification process is based on well-identified steps (Figure 3). The Verification step guarantees the correct numerical resolution of the implemented models according to unitary tests. The Elementary Validation allows verifying that the scheme uses the suitable physical model and the best options. Successive benchmarks and the comparison of tested scheme results to reference results obtained by an exact resolution (like a Monte Carlo simulation code which uses the same Nuclear Data) are then performed. The benchmarks are defined in order to emphasise elementary physical difficulties, allowing to determine numerical and model biases and to avoid error compensation. The final step, called the Global Validation, consists in testing the scheme

results against representative experimental data. The numerical procedures and the nuclear database are evaluated. In the case of lack of experimental results, an error propagation method can be used to give an order of magnitude of the bias due to the potential nuclear data errors [5].

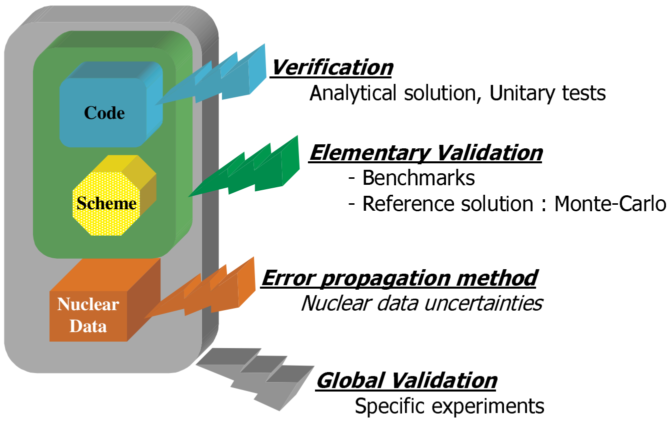


Figure 3 : Qualification process

3 Neutronics route

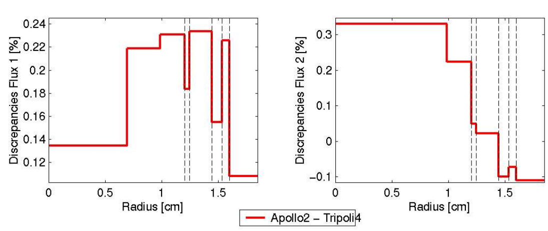
3.1 Assembly neutronics scheme

The JHR assembly neutronics route is performed with the APOLLO2 [3] code and its 172 group library. The complex geometry (Figure 1) necessitates the use of the exact 2D collision probability method. The optimized self-shielding model, developed for PWR UOX fuels, can be used. Nevertheless, many options have to be re-defined for the flux determination.
On terms of validation, we can identify three different physical problems : the self-shielding calculation, the flux resolution, the burn-up calculation, and that for the assembly with or without absorber and for experimental devices. Hence, 15 benchmarks have been defined and a reference solution for each has been determined. In order to avoid compensation effects, the calculations have to be compared on relevant parameters : the multiplicative factor k_∞, the well-known 6 factor formula ($k_\infty=\chi_{n,2n}\times\varepsilon_{even}\times\varepsilon_{odd}\times p\times f\times\eta$), the absorption rate and the fission rates, determined on an adapted 13 group energy mesh, and the flux on a 4-group mesh.
We have described in [6] the validation for the standard and control assembly route. One can observe (Table 1, Figure 4) that HORUS3D also models with accuracy test devices such as the Chouca irradiation device (Flux1 : e>0.907MeV Flux2 : e>5keV).

Table 1 : APOLLO2 discrepancies with TRIPOLI4 results (1D benchmark)

(pcm)	TRIPOLI (1σ)	AP2-TP4 (STD)	AP2-TP4 (Chouca)
χ	±120	45	5
ε_p	± 25	2	5
ε_I	± 28	-19	2
p	±161	-159	-242
f	±109	69	106
η	± 30	2	2
K_∞	± 9	-61	-140

Figure 4 : APOLLO2 Flux discrepancies (in %) with TRIPOLI4 results (CHOUCA)



3.2 Core Calculation

The 3D core neutronics route uses the new finite triangular elements developed in the CRONOS2 code [4, 6].
The reference Monte Carlo results were obtained with the TRIPOLI4 code, without any simplification; all geometry details have been represented and pointwise cross sections were

used. 50 Millions of particles were generated. The multiplicative factor is so determined with a precision of about 15 pcm (1σ) and the radial fuel plate power uncertainty arises to only 0.15%.
The comparison of the HORUS3D/N-V1 results with the TRIPOLI4 reference is presented in Table 2. One can observe that the different neutronics parameters are particularly well determined.

Table 2 : Comparison of HORUS3D/N-V1 and TRIPOLI4

Neutronics parameters Discrepancies (±1σ)	STD	12 CNT	1 Chouca I	8 Chouca I	3 Chouca G
Reactivity (pcm)	+ 195 ± 11	+ 58 ± 18	+ 3 ± 25	+ 65 ± 25	+ 493 ± 25
Assembly power : average standard deviation Max.	 - 0.2 % 0.7 % 1.4 %	 - 0.3 % 2 % 4.6 %	 - 0.1 % 0.7 % 2.0 %	 - 0.1 % 0.8 % 1.9 %	 0.0 % 0.9 % 2.1 %
Peak power	±10%	±15%	±10%	±10%	±10%
Fast flux	±5%	±5%	±10%	±10%	±10%
Thermal flux in the reflector Beryllium Water	 +3% +5%	 +3% +5%			
Reactivity worth		±1%			

3.3 Experimental Validation Program

This Validation will be followed by the comparison of HORUS3D/N results with experimental results. In order to complete the experimental database, an ambitious experimental program is in preparation. The first phase, foreseen during 2004, will focus on the fuel reactivity qualification by analyzing the effect of an UMoAl rod in an adapted critical UOX lattice : the Valmont program. The second phase, planned for 2007, consists of criticality experiments with a small lattice of fuel assemblies in reference and off-reference conditions (Figure 5). This phase, called AMMON and much more extensive than the first, will allow the qualification of the assembly power map, control rod efficiency, reactivity coefficients, Beryllium reflector effect with the actual characteristics of the JHR assemblies in order to fulfill the safety requirements of the reactor licensing procedure.

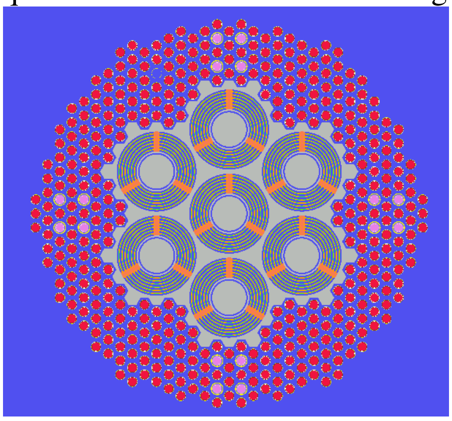


Figure 5 : AMMON reference configuration in EOLE facility

4 Nuclear Heating

The simulation of materials damage under irradiation is a key point in the definition of the next reactor generation. The experimental control of the sample's irradiation conditions requires knowledge and understanding of nuclear heating, due to both neutron and photon flux. The heating gradient has to remain lower than 5°C across the sample.
The nuclear heating of samples has two sources:

- It can be due to photons, coming from fission, capture and inelastic scattering which transfer their energy to electrons by means of pair creation, photo-electric and compton interaction. Inside the reactor core, the largest part of photonic heating is coming from

fission photons with an average energy of about 1 MeV and consequently it is proportional to the local power distribution. This contribution can be split into 2/3 due to prompt photons and 1/3 due to delayed photon.
- It can also be due to neutrons which transfer their energy after collisions by nucleus recoil or by slowing down of charged particles emitted by the nucleus (α,p). However, this contribution represents only about 10% of the total heating.

For that reason, the main part of the work on nuclear heating deals with the evaluation of photon heating.

In order to take into account the actual assembly geometry without a prohibitive running time, an original scheme has been developed. The photon calculation scheme is divided in several steps. Its point of departure is the calculation of the neutron flux in the assembly with the help of the deterministic code APOLLO2 following the standard calculation route. From this point on, we carry out successively :

1. The calculation of the photon spatial source distribution and its energy spectrum on the actual assembly geometry. A set of standard procedures allow to obtain the gamma source from the distribution of the neutron flux and a gamma yield matrix based on the JEF2 library for every kind of assembly.
2. Insertion of the photon sources in an input deck of TRIPOLI4 3D Monte Carlo code, configured in photon transport mode. The source is spatially weighted by the 2D power distribution coming from a core calculation using the CRONOS2 deterministic code. The Monte Carlo transport of photon of about 1 MeV inside the reactor is a fast process in comparison to neutron transport.

An automated procedure allows to interface the data generated by the APOLLO2 routines for all types of assembly and the TRIPOLI4 input deck. It takes into account the local power weighting obtained from deterministic core calculation and volume renormalization. It uses a set of libraries describing gamma sources, assembly geometry, assembly homogenization choices and dose response functions, which can be combined in order to deal easily with various core configurations.

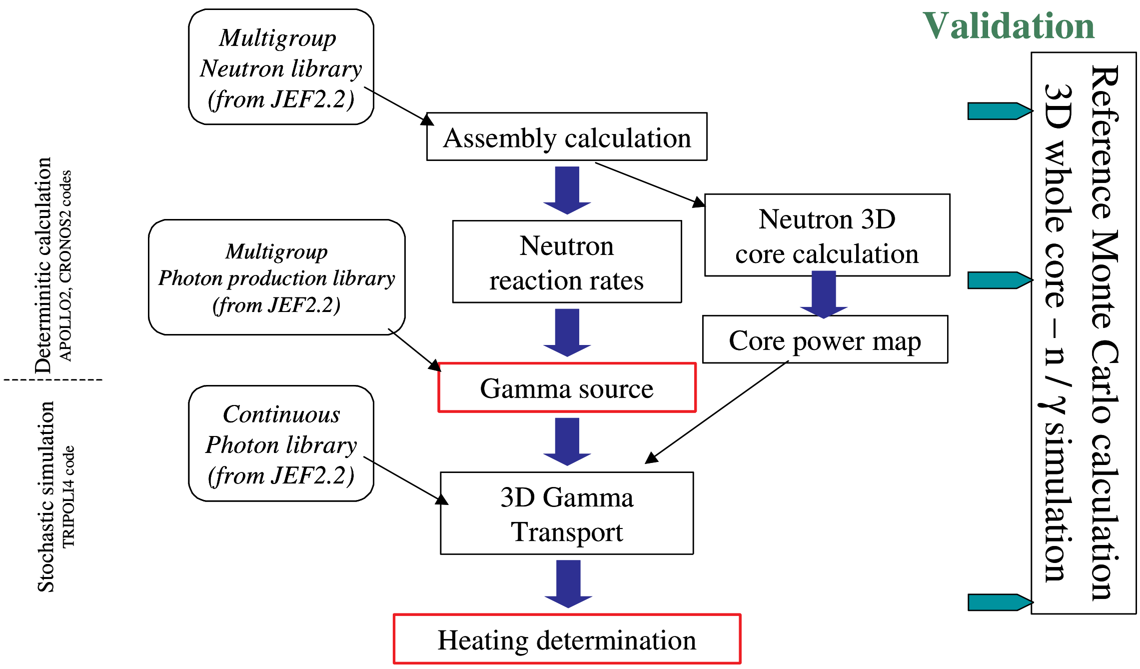


Figure 6 : Nuclear heating scheme

Validation
This calculation scheme has been validated thanks to a direct calculation using a Monte Carlo code in coupled neutron-gamma mode. This calculation allows to obtain heating reference values without any hypotheses about spatial homogenization, energy groups division or self shielding but to the detriment of computer time which is far much longer.
Inside the core, the gamma heating has been calculated to a maximum value of about 20 W/g, decreasing to 2 W/g in the reflector.

Qualification
However, we estimate that there are large uncertainties on gamma heating of approximately 30% (2σ), mainly due to the lack of knowledge about gamma spectra while the precision target for sample irradiation is less than 10%. In order to reduce these errors and to validate experimentally this calculation scheme, a set of experiments called ADAPh, are going to be carried out in the core of the research reactor EOLE in Cadarache.
EOLE is an experimental reactor of very low power (100W) dedicated to neutronics studies of moderated lattices, in particular those of industrial pressurized water reactor (PWR). The measurements of integrated gamma doses into various places more or less off center of the core are in progress by means of thermo luminescent dosimeters (TLD). The use of TLD is made necessary because of the importance of the involved dose rate. Various types of TLD are going to be tested (CaF_2:Mn, ^{7}LiF:Mg, Ti, Al^2O^3:C) to look for the best adapted. Any activation or saturation phenomena has to be avoid in order to permit measurement interpretation. The neutron dose sensitivity has also to be evaluated and minimized.
These data are going to be compared with those given by Monte Carlo code in coupled neutron-gamma mode as well as with result of the chain APOLLO2 – TRIPOLI4 gamma mode calculation.

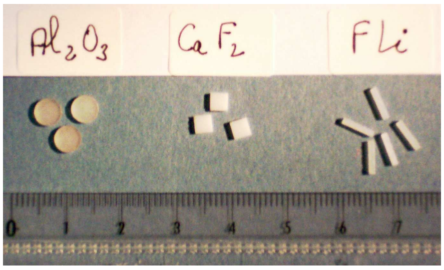


Figure 7 : Thermo Luminescent Dosimeters

5 Cycle problems

The main tool for cycle design parameters such as heating decay at CEA is DARWIN [13]. DARWIN is already validated for current fuels (UOx, MOx) or innovative fuels (MIX, APA, PuTh) and for every nuclear road (Pressured Water Reactor, Fast Breeder Reactor, Boiling Water Reactor, Advanced Reactors). DARWIN is also used in the back-end cycle for actinide incineration (SPIN) or long term interim storage studies. Its qualification domain has to be extend for the JHR reference fuel, UMoAl. An experimental program will take place in the OSIRIS reactor, based on the measurement of the gamma emission of irradiated UMoAl plates in the IRIS device. A sequence of irradiation and cool down periods will allow to distinguish several Fission Products.

6 Thermal-hydraulics

The thermal-hydraulic calculations are carried out with the FLICA4 code [14] and the CATHARE2 code [15]. These two codes are able to calculate both steady state and transient simulations. The 3D core thermal-hydraulic is solved by the two-phase flow computer code FLICA4. Based on fully unstructured meshes, the finite volume method implemented in FLICA4 provides an accurate description of the JHR assembly. The CATHARE2 code, which is a best estimate code for thermal-hydraulics nuclear reactor safety studies models, the JHR reactor with its all components (primary and secondary coolant systems, the pool with its own safety cooling heat exchanger) and uses a simplified approach for the core. CATHARE2 can be used to provide realistic boundary conditions for FLICA4 for calculations of the core thermal-hydraulics. The core power boundary conditions are given by neutronics computation.

6.1 Experimental Validation program

Due to the specificity of the JHR core geometry and the operating conditions, it appeared necessary to start an experimental program to validate FLICA4 and CATHARE2 codes. To achieve that, a test facility called SULTAN-JHR is being realized.

The SULTAN-JHR facility (Figure 8) is designed and built to provide known thermal-hydraulics conditions for a simulated full-length coolant channel of the JHR reactor, allowing the experimental determination of the thermal limits (both flow excursion (FE) leading to a flow redistribution and critical heat flux (CHF)) under expected JHR conditions. The facility is also designed to examine other thermal-hydraulics phenomena including onset of incipient boiling, single-phase heat transfer coefficients and friction factors, and pressure drop characteristics

The test section (Figure 9) simulates a single sub-channel in the JHR core with a cross section corresponding of a mean span (~50mm) that has a full reactor length (600mm), the nominal flow channel gap (1.5mm) and the plates in Inconel thick of 1mm. The tests with light water flowing vertically upward will be cover a heat flux range of 0-7 MW/m^2, a velocity range of 0.6-18 m/s, an exit pressure range of 0.2-1.0 MPa and an inlet temperature range of 25-180 °C.

The test channel is instrumented on the back of the channel wall (on the surface of the insulator) with 40 thermocouples positioned on the center of each side of the test section. The spacing is staggered to provide improved definition in the region close to the channel exit: every two centimeters at the bottom and every centimeter at the top. Pressure and temperature are measured at the test section inlet and outlet as well as the mass flow rate. Differential pressure measurements are located axially along the channel. The total differential pressure is measured too. A particular attention is made to calculate the plate temperature on the fluid side, by taking into account the thermal conductivity of the Inconel.

Tests will be conducted by decreasing the mass flow rate step by step at a constant power. The exit pressure and the inlet temperature are automatically controlled at a desired set point. The system is allowed to stabilize at each of the selected velocity settings. Once the minimum in pressure drop has been clearly determined (by observation of increasing pressure drop as velocity is further decreased), the test is stopped. Data are recorded continuously during the processes by the personal computer-based data acquisition system, which calculates mean values of the different measurements during the different steady states. From these tests the potential of the flow excursion is determined by detecting the minimum pressure drop in a plot of pressure drop versus mass flow rate.

Some tests will be conducted up to the Critical Heat Flux conditions.

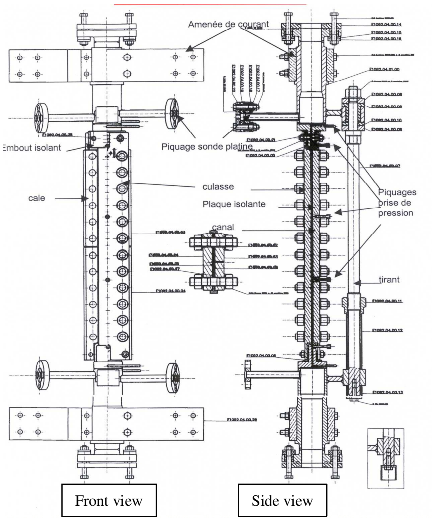


Figure 8 : SULTAN-JHR device description

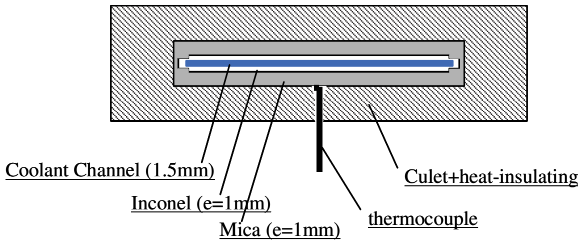


Figure 9 : Test section geometry

6.2 FLICA4 code

6.2.1 FLICA4 main features

FLICA4 is a three dimensional thermal-hydraulic code, dedicated to complex two-phase flow calculations required for the design and the safety analysis of nuclear reactor cores.
The FLICA4 two-phase flow model consists in a mixture mass conservation equation, a mixture momentum balance equation, a mixture energy balance equation and a vapor concentration equation to calculate thermal disequilibrium or subcooled boiling flows. The FLICA4 numerical method is based on a finite volume discretization. Numerical fluxes at cell interfaces are calculated using Roe's approximate Riemann solver. This method does not need a staggered grid and it makes use of characteristic information within the framework of a conservative method. This method enables conservation properties.
FLICA4 allows calculations with various kinds of elements : rectangular, triangular and hexagonal elements. Irregular quadrangular elements are also available. Different levels of refinement are also available (multigrid approach).

6.2.2 FLICA4 assessment strategy

A detailed validation of the code has been performed in order to ensure that the thermal-hydraulic models reproduce well experimental trends.
To achieve that in the scope of interest for the JHR, the SULTAN-JHR facility experimental data are used. Older experimental data, obtained during 70's have already been exploited in order to get preliminary results and a first adjustment of the models.
In order to minimize differences between computed and experimental data, an optimization of the closure relationships has been carried out. The mesh results from a compromise between the accuracy of the geometrical description of the reactor's core and calculation cost. All calculations are well spatially converged.

6.2.3 Uncertainty analysis

Validation consists in determining the best set of physical correlations compared with experimental results. To do that, a statistical approach with neural models and response surfaces is used for FLICA4. Uncertainties of the validation are deduced.
The last stage will be devoted to the determination of the global scheme uncertainties, using propagation methods or sensibility studies.

6.2.4 FLICA4 core calculations

The FLICA4 core calculations are built of two successive levels of refinement. First, the whole core is treated with a one cell per assembly discretization. After that, the hottest assembly is zoomed and calculated alone. Boundary conditions are deduced from the coarse calculation and supplied to the zoomed simulations. The hot assembly mesh uses irregular quadrangular elements. Each physical channel is cut into four cells. An example of results obtained for a steady-state calculation is given below. Transient calculation will follow the same scheme. During steady state or transient simulations, Onset of Nucleate Boiling margins, Onset of flow instability margins, void fraction and critical heat flux are key parameters evaluated.

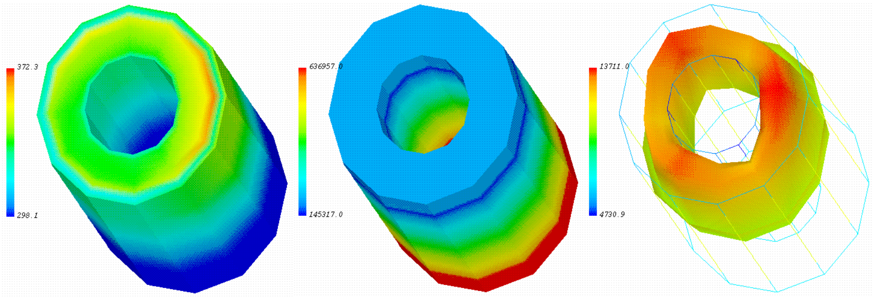


Figure 10 : FLICA4 hot element results (from left to right, temperatures, pressure, mass flow rate)

For a complete representation of the JHR system, FLICA4 is coupled with the CRONOS2 neutronics code and the CATHARE2 system code through the ISAS supervisor. This whole code system has been tested on a MSLB simulation for a PWR and on BWR systems. It will permit simulations of complex transients and will allow dimensioning of the JHR safety system.

6.3 CATHARE2 code

6.3.1 CATHARE2 main features

The CATHARE2 thermal hydraulic code has been developed jointly by CEA, IRSN (Nuclear Safety Institute), EDF (the French Utility) and FRAMATOME (the French Vendor). The objectives of the code are:

- perform safety analyses with best estimate calculations of thermal hydraulic transients in Pressurized Water Reactors for postulated accidents or other incidents,
- quantify the conservative margin,
- investigate plant Operating Procedures and Action management,
- be used as a plant analyzer in a full scope training simulator providing real time calculations.

The code is based on a 2-fluids - 6-equations model. The presence of non-condensable gases such as nitrogen, hydrogen, air, can be modeled by one to four additive transport equations. The code is able to model any kind of experimental facility or PWR (Western type or VVR), and is usable for other reactors (Fusion Reactor, RBMK, BWR, research Reactor)

The code CATHARE2 has a modular structure. Several basic modules may be assembled to model the primary and the secondary circuits of any Reactor or of any separated test or integral effect test facility. 0D, 1D and 3D modules are available. All modules can be connected to walls or heat exchangers with a 1D conduction calculation. Other submodules are used to compute point kinetic neutronics, pump speeds, accumulators, sources, sinks, …

The range of physical parameters in CATHARE2 is rather large and devoted to the study of many Nuclear Power Plants accidents : pressure from 0.1 to 16 MPa, liquid temperature from 20°C to 350°C, gas temperature from 20°C to 2000°C, fluid velocities up to supersonic

conditions, duct hydraulic diameters from 0.01 to 0.75m. An important and rather extensive experimental program has been carried out to support the validation of the code.
The numerical method in the CATHARE2 code uses a first order finite volume –finite difference scheme with a staggered mesh and the donor cell principle. The time discretization varies from the fully implicit discretization used in the 0-D and 1-D modules to the semi-implicit scheme used in the 3-D modules. These methods are known for their robustness in a wide range of flow configurations. A hyperbolic equation system is used to ensure the well posedness of the problem. Finally, at each time step and for every type of CATHARE2 module, one has to find the solution of a set of non linear equations : a full Newton-Raphson iterative method is then used.

6.3.2 CATHARE2 assessment strategy

At the onset of the CATHARE2 project, a very rigorous methodology for the development and the assessment was defined to achieve the "best estimate" objective. As a first step of the development, mass, momentum and energy equations are established for any module. They are derived from local instantaneous equations using some simplifying assumptions and averaging procedures. Many closures relationships must be developed to express the mass, momentum and energy transfers between each phase and the walls and at the interface.
The constitutive relationships are developed and assessed following a general methodology:

- Derivation of correlations by interpretation of separate effect tests or from the literature, by an appropriate fitting of certain coefficients
- Systematic assessment of the complete set of physical closure laws (revision) against a large matrix of separate effect tests
- Extensive assessment on integral tests in order to validate the general consistency of the revision

The qualification process covers the widest possible range of parameters. For the current revision , more than 1000 tests from 45 facilities will be calculated. The aim of this extensive testing is to verify the constancy and adequacy of the whole package of closure relationships, to estimate the accuracy of the predictions, to point out possible shortcomings in the models, which need physical developments.

6.3.3 Uncertainty analysis

Best-estimate codes such as CATHARE2 provide a good insight into the complex thermal-hydraulic behavior of reactors during accident transients and give a more realistic view of flow phenomena than previous models based on conservative approaches. However, to be used in safety studies, the uncertainty of predictions should be estimated. Methodologies have already been developed in FRANCE by EDF, FRAMATOME or IRSN. The purpose of all these methodologies is to propose tools to take into account the propagation of the uncertainties from the basic parameter uncertainties (physical models, numerical methods, plant data…) to the relevant responses of the code
Whatever the methodology, tools or methods are required for the identification of sensitive parameters, for quantification of their sensitivity, and for the determination of the uncertainties of the code correlations (basic uncertainties). Two main contributions to these problems have been developed. On one hand, the Discrete Adjoint Sensitivity Method (DASM) has been implemented in CATHARE. This mathematical tool provides in a single run the derivatives of selected output variables with respect to a set of code parameters. It is a powerful tool for sensitivity analysis. On the other hand a statistical method, CIRCE, has been developed which makes possible to calculate the uncertainties of the constitutive relationships based on qualification calculation results and DASM sensitivity values. Circe quantifies the uncertainties in the form of a covariance matrix of the parameters associated to the correlations.

6.3.4 CATHARE2 validation for JHR

Because CATHARE2 will be used for the safety and design purposes in the future JHR reactor, the JHR core geometry is specific (narrow channel) and the core flow conditions are different from PWR core conditions(low pressure, high heat flux and high velocity, it is necessary to improve the different CATHARE2 correlations, specially the wall friction and heat transfer correlations in single phase flow to nucleate boiling in both forced and natural convection which is one of the objectives of SULTAN-JHR facility. From the set of new correlations, which it will be developed, a determination of the uncertainties of these correlations with Circé will be performed with regard to Sultan-JHR data.

In parallel with the CATHARE2 qualification action, the modeling of the different JHR circuits has been developed. It includes the main circuit with its different elements (core, 3 pumps in parallel, 2 series of 2 heat exchangers in series and the pressurization system) and the auxiliary circuit with its heat exchanger immersed in the pool . The auxiliary circuit will be allowed to evacuate the residual power in case of pump loss by natural circulation.

The whole circuits are modeled with 45 elements corresponding to 420 meshes (0-D, 1-D and Boundary conditions elements) In particular, the core is simulated by a 1-D element which represents a mean core channel with its fuel plate and another 1-D element corresponding to the core bypass. A weight corresponding to the total core channel number is applied to the core channel element. The pool is simulated by two 1-D elements and two 0-D elements to take into account the evolution of the heat transfer coefficients between the heat exchanger and the pool to improve the sub-cooling margin in the core. The temperature of the pool is controlled at a constant temperature during the whole transient.

7 Conclusion

The preliminary design studies led to the definition of the main options of the future European MTR : the Jules Horowitz Reactor, and a reference configuration has been designed. The Definition Studies, in progress, will optimize the JHR performances by testing modifications from the reference configuration.

The computational tools, required for these studies, have to face up three difficulties : the specific characteristics of the JHR lead to modeling problems, the running time of the developed schemes has to be short enough for the design studies and the results have to be guaranteed by Validation studies. These difficulties, combined with the lack of specific integral experimental data, require a rigorous methodical approach.

An adapted and consistent neutronics/thermal-hydraulics code package: HORUS3D (HOrowitz Reactor simulation Unified System) is developed. The neutronics route has been presented with the Validation methodology based on successive benchmark definitions. The thermal-hydraulics part has focused on propagation methodologies and the experimental program, called SULTAN-JHR. The FLICA4 and CATHARE2 codes have been described, emphasizing the specific developments for the JHR modeling (core and system).

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