Chapter 11
Overview of simulation software used in design studies and safety analyses for French research reactors

The design or modification of research reactors and their experimental devices (including the modification of cores), just like the associated safety demonstration – including during safety reviews – is based on studies, generally performed using simulation software[308], in various areas: neutronics or criticality (for cores and areas dedicated to fuel storage), thermohydraulics (for cores and cooling systems), structural mechanics (for metal structures and civil engineering works), etc. It is chiefly the operators (CEA, Institut Laue-Langevin) that perform these studies, but IRSN may also perform them as part of its expert assessment of the files submitted by these operators to the ASN.

It is, of course, important to validate simulation software before it is used for studies. The ability of each simulation software tool to accurately or conservatively represent the physical phenomena in question must also be established as part of a safety demonstration, or the expert assessment of such a demonstration.

In this respect, section 8.3 on the so-called "BORAX" reference accident presents examples of general demonstration tests performed in reactors or on models to reinforce evaluations performed by calculation. Indeed, this type of verification can be desirable, or even necessary, in cases where the evaluations performed by calculation are subject to or result in too high a level of uncertainty (including due to simplification of the modeling) or when the software has only been validated separately to its various physical models.

308. The expression "computer code" is also used.

It is also important to recall here the special importance, in the case of a new reactor (or a reactor that has undergone substantial modification), of startup tests (or restart tests) performed by the operator on various items of equipment or systems to ensure, as far as possible[309], that they are able to perform the functions for which they were designed, with the performance expected from the design studies largely based on the use of simulation software.

Some of the software used[310], in gradually improved versions over time, and their key applications[311] in French research reactors are described briefly below. This description is generally limited to the possibilities offered by these software tools, the context and aims of the studies for which they are used, as well as some elements concerning the modeling, and precautions to be taken to achieve a satisfactory level of confidence in the results obtained – adjustments based on operating experience or the comparison of different software.

A number of these software packages were initially developed for power reactors (for example FLICA, CATHARE and SIMMER). They have been adapted for use with research reactors (e.g. for reactors using uranium- and aluminum-based fuel in the form of plates, with heavy water as a coolant or moderator, etc.). In addition, neutronics and thermohydraulics software for cores and reactor systems can be combined[312], for example CRONOS-FLICA, CRONOS-CATHARE or even CRONOS-FLICA-CATHARE — this last combination forms the HEMERA (Highly Evolutionary Methods for Extensive Reactor Analysis) software chain.

Lastly, attention may be drawn to the fact that the complexity of research reactor cores, which combine standard fuel elements, fuel elements that may partially contain neutron absorbers, control absorbers (rods or plates) in or close to the core and highly diverse experimental devices in various locations in or close to the core (for example, loops that may be cooled with coolants other than those used to cool the core in which they are installed, such as liquid sodium), naturally requires the use of relatively sophisticated simulation software, especially in relation to neutronics. The 2004 discovery of melted rods in the driver core of the CABRI reactor (section 10.1.2), which was caused by underestimating the temperatures reached in the rods in question, confirms this complexity.

▶ Neutronics

- **APOLLO**: this software[313] for two-dimensional (2D) simulation in the field of neutronics, based on neutron transport theory (the Boltzmann equation) in a

309. Indeed, it is not possible to envisage creating accident situations in order to check that the equipment designed to control these situations works correctly.
310. In particular, see "Neutronics", a monograph by the Nuclear Energy Division of the CEA, Le Moniteur, 2013.
311. The applications detailed in this chapter also form the subject of publications.
312. Although they were produced for pressurized water reactors, some of these coupled software packages have been, or can be, used for certain research reactors, while others need to be adapted.
313. In the field of neutronics, a distinction is drawn between the expressions "software" and "calculation scheme": "calculation scheme" refers to the sequence of physical models linked to a clearly defined "library" of cross sections.

stable state (stationary), but able to simulate fuel burnup[314] (so-called "evolution" code) and take into account a large number of neutron energy groups (300 for normal calculations), is principally used to determine the "libraries" of cross sections[315] than can then be used with the CRONOS software presented below. These are multi-parameter "libraries" of cross sections (the parameters may be temperature, water density, etc.) "condensed" into several energy groups and homogenized in "cells" selected for the representation of the system being studied (an assembly, a rod or a plate, a pellet, etc.). In principle[316], APOLLO (2) can also be used to determine the core neutron balance (production of neutrons *via* fission, absorption and leakage) with the relevant neutron parameters (neutron balances such as the effective neutron multiplication factor k_{eff}; kinetic parameters such as neutron life or delayed neutron production; neutron feedback; the efficiency of absorbers, etc.).

– **CRONOS**: This software for the three-dimensional simulation of reactor core neutronics solves either the transport equation or the diffusion equation by using the finite element method for several neutron energy groups (two groups are sufficient for the current calculations). It can be used to determine the distribution, in three dimensions, of the core power as well as the change in this power over time during incident or accident transients, the efficiency of neutron absorbers, etc. CRONOS can also simulate fuel burnup (so-called "evolution" code). The cross sections needed for the code come from calculations performed using the APOLLO software and are entered as input data. CRONOS is a multi-reactor type code: there is nothing in its organization or structure that makes an assumption about the type of reactor for which the calculations are being performed. This means that calculation schemas using CRONOS (2) have been assembled (notably in terms of meshing) for a large number of reactors, including research reactors (figure 11.1).

– **MCNP:** This software for three-dimensional geometrical simulation, developed by the Los Alamos National Laboratory is historically the first simulation software based on the Monte Carlo method (Monte Carlo N-Particle transport code). MCNP can be used for a variety of particles (neutrons, electrons, photons, etc.). It is used in a range of fields, including, in addition to reactor physics, radiological protection, dosimetry, criticality and medical physics.

  For a reactor core, the principle behind the software involves following the history of each neutron in the system being studied, from birth (external source, from fission, etc.) to death (capture by a nucleus or leakage outside the system). With the MCNP software, a continuous neutron energy spectrum is usually used, but a discretized spectrum may also be used. Although the MCNP software can simulate fuel burnup (so-called "evolution" code), it is not suitable (similarly to the other Monte Carlo software described below, at the current stage of its development) for simulating transients in a reactor, as the neutron feedback is not correlated to temperature.

314. Fuel consumption through irradiation.
315. Cross sections constitute indicators of the likelihood of interaction between neutrons and the material; this likelihood depends on the neutron energy.
316. Highly complex calculation with APOLLO version 2; it will be simpler with version 3.

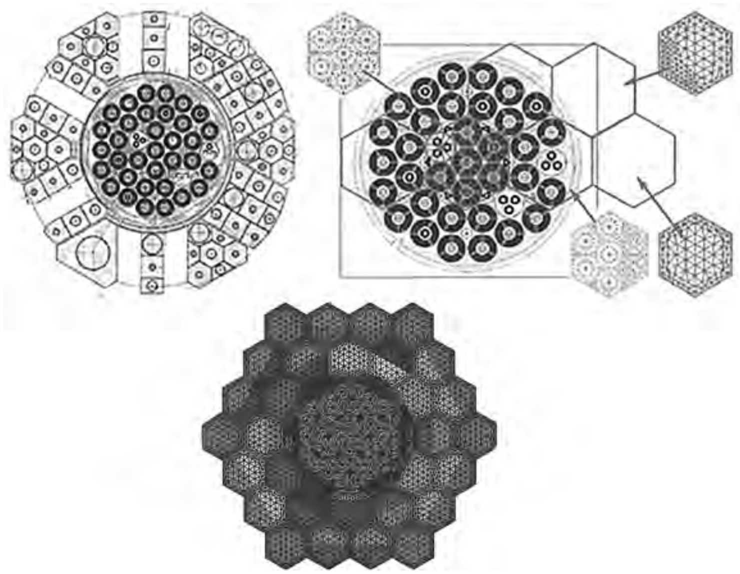

Figure 11.1. From the actual core geometry of the Jules Horowitz reactor (top left) to its subdivision into hexagonal macro-elements (top right) and meshing in finite isoparametric elements (bottom), performed by the CEA in order to calculate, using the CRONOS (2) software, the core power distribution (source: CEA monograph). © DR.

The history of each neutron depends on its interactions with the material. The distance traveled by the neutron between two collisions, the nuclei involved and the interaction types are parameters that are randomly sampled by using experiment results grouped into "libraries" of nuclear data. In this way, by increasing the monitoring of many neutrons, it is possible to simulate the natural behavior of the system and calculate numerical values close to certain core neutron parameters (balances such as k_{eff} and kinetics coefficients, but not temperature-dependent feedback). As this type of calculation is based on probabilities, it is necessary to perform extensive random sorting to reduce the statistical uncertainty[317]. Some calculations can last several months, hence the importance of using powerful computers. The geometric representation of the system studied is based on a precise, geometric description of the surface of objects, defined according to the problem to be solved and which may vary a great deal in size (from a core area to a fuel pellet, for example). This is known as "surface representation". The MCNP software package can therefore be used for precise neutronics calculations.

– **TRIPOLI** (*TRIdimensionnel POLYcinétique*): This three-dimensional simulation software tool, under development by the CEA since the 1960s, uses the Monte Carlo method to solve the transport equation for neutrons and photons, the latter

317. The statistical uncertainty regarding the result of a calculation is given by the central limit theorem: the standard deviation of the result is proportional to the inverse of the square root of the number of neutrons monitored.

resulting from the nuclear reactions generated by the neutrons (fission or capture – the photons take the form of γ radiation). Similarly to the MCNP software, a continuous neutron energy spectrum is usually used for TRIPOLI, but a discretized spectrum may also be used. The TRIPOLI software can simulate fuel burnup (so-called "evolution" code), but, for the same reason as in the case of the MCNP software, it cannot simulate transients in a reactor. With TRIPOLI, the system being studied can be processed by surface definition (as for MCNP) or using a combinatorial volume method (in which the user specifies the volume types and the link between the volumes). It is mainly used for reactor core physics, criticality and radiological protection. The TRIPOLI software is frequently used in France for precise neutron calculations (so-called "reference" calculations).

Examples of use:

The CEA uses the TRIPOLI software for neutronics studies at its research reactors (CABRI, the Jules Horowitz reactor, etc.). It also used TRIPOLI (4) in parallel with APOLLO (2)[318] to examine the impact of UMo fuel on the cycle duration and performance of the Institut Laue-Langevin high flux reactor.

- **MORET:** This simulation software, under development by IRSN since the 1970s, calculates neutron transport using the Monte Carlo method. It is generally used with a discretized neutron energy spectrum. The geometric representation is less detailed than that possible using the meshing tools associated with MCNP and TRIPOLI. The MORET software makes it possible, for complex three-dimensional systems containing fissile materials, to determine the following main values (excluding feedback correlated to temperature): the effective neutron multiplication factor (k_{eff}), the neutron flux, the reaction rates (fission, absorption and diffusion) in the different volumes, neutron leakage outside the system and the kinetic parameters of the system (proportion of delayed neutrons and their generation times, neutron life, etc.). The geometric model of the system being studied is processed using the combinatorial volume method. More specifically, the software is used to study the criticality risks in nuclear facilities (i.e. the appearance of an uncontrolled chain reaction outside the reactor cores in operation), in its "environment" known as CRISTAL[319], which offers different datasets (and other software such as APOLLO (2) and TRIPOLI (4)).

  Examples of use:

  - The MORET software is principally used by IRSN for its expert assessments of the criticality risks in fuel cycle facilities. However, for the past ten years or so, IRSN has also been using it for reactors, as in the case for a study designed to learn lessons from a fuel loading error that occurred in reactor 4 of the Dampierre nuclear power plant (Loiret department) in 2001. At the start of the 2010s, IRSN also used MORET (5) to simulate tests

318. CEA publication cited in footnote 300.
319. The CRISTAL formula is being developed and qualified as part of a collaboration between IRSN, the CEA, AREVA-NC (Orano) and AREVA-NP (Framatome). This ensemble includes "libraries" of nuclear data, calculation procedures, simulation software and interface tools. Its purpose is to evaluate the criticality conditions of nuclear facilities and fissile material transport packaging.

performed in the American reactor SPERT in the 1960s, which aimed to study the response of a reactor core to stepwise reactivity insertions. This simulation was carried out as part of an exercise to benchmark simulation software packages organized by the IAEA (on innovative methods for research reactors[320]), which aimed to evaluate the ability of different simulation software tools (used in reactor design, safety demonstrations or the expert assessment for these demonstrations) to reproduce some measurements performed directly on various research reactor cores, in the fields of both neutronics and thermohydraulics. The MORET (5) simulation software notably made it possible[321] to reproduce the radial distribution of power in SPERT-IV-D 12/25 core assemblies (see figure 11.2), which was then used for assembly heating calculations using the CESAR software from the ASTEC code (see below).

- In 2008, in the context of the review of the BORAX-type accident for the ORPHÉE reactor, the CEA decided that the envelope reactivity insertions considered likely for the reactor would not lead to an explosive interaction between melted fuel and water (steam explosion). In order to evaluate the basis for this conclusion, IRSN performed a study in 2010 using the MORET (5) software to independently determine the reactivity insertions for the scenarios identified by the CEA, including the simultaneous rupture of the two “cold sources”, the “hot source” and the nine horizontal thimbles (figure 11.3). Physically, these items of equipment, most of which contain a gas, create leakage spaces for neutrons, which do not therefore participate in the chain reaction. If heavy water gets into these spaces, the effect of neutron reflection by the heavy water is heightened, which increases core reactivity. The reactor model used was refined in order to re-establish some neutron parameters based on the calculations performed when the reactor was designed (before 1980) – using the TRIPOLI and TRIDENT software – or measured, such as the multiplication factor k_{eff}, for various control absorber positions, critical absorber position, efficiency in terms of reactivity, etc. The IRSN study indicated a maximum reactivity insertion significantly higher than that derived from the CEA calculations, leading the CEA to update its own studies by using a more recent version of the TRIPOLI software, which confirmed IRSN’s results. It was then important for IRSN to make sure that the simultaneous failure of all horizontal thimbles, in particular, could be ruled out, by ensuring that the thimble material (AG3NET alloy) was sufficiently ductile at the end of its service life. Due to having underestimated the fluence[322] received by these thimbles, the operator reviewed its thimble replacement schedule. More specifically, the ASN ordered (on the advice of the GPR) the operator to check the schedule

320. IAEA Coordinated Research Project 1496 (2008-2013): Innovative Methods in Research Reactor Analysis.
321. See IRSN communication at the TOPSAFE 2012 conference: *Interpretation of reactivity accident transient on research reactors on example of SPERT-IV-D 12/25 Benchmark*, Ivanov E., Maas L., Écrabet F.
322. Value used to establish the maximum service life for thimbles (see footnote 231).

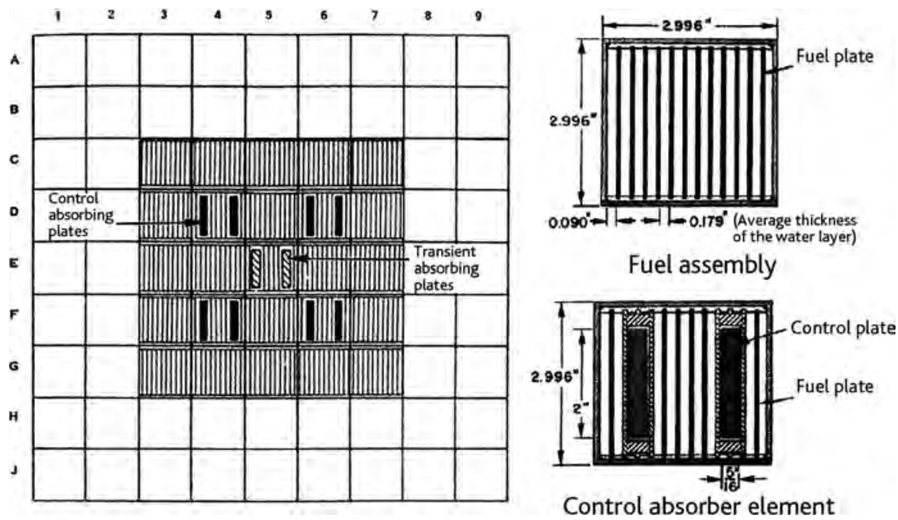

Figure 11.2. On the left, general diagram of the SPERT-IV-D 12/25 core; on the right, a fuel element and control element containing control plates. © Phillips Petroleum Company-Atomic Energy Division.

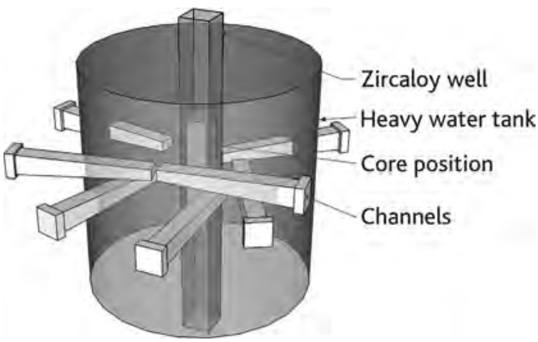

Figure 11.3. 3D model of the ORPHÉE reactor, notably showing the neutron channels, prior to use of MORET simulation software. © IRSN.

for replacing the horizontal thimbles and "cold source" thimbles to ensure that the reactivity "weight" of the devices with very low ductility was limited (with the operator giving a precise definition of the ductility and reactivity "weight" criteria used).

- In 2011, a similar study[323] was performed by IRSN using MORET (5) software to evaluate the envelope nature of the reactivity insertion

323. For this study and the previous, see the IRSN publication presented at an IAEA conference in Rabat, Morocco, in 2011, entitled "Safety approach of BORAX type accidents in French research reactors", Chegrani Y., Gupta F., Tiberi V., Heulers L.

identified by the CEA in its study on the BORAX-type accident for the Jules Horowitz reactor. The reactivity insertion corresponds to the ejection of a control rod containing hafnium, which is the material that absorbs neutrons. An envelope reactivity insertion value had been determined by the CEA on the basis of calculations performed using the APOLLO (2), CRONOS (2) and TRIPOLI (4) software. The aim of the IRSN study was to verify this envelope nature, notably by calculations to establish sensitivity to various parameters. The MORET (5) software can be used to determine the reactivity insertion by taking the difference in two effective multiplication factors (k_{eff}) calculated for two core states: control rod inserted and control rod ejected (leaving a "water hole").

The modeling used with the MORET (5) software had already been verified[324] for a core configuration by comparing certain parameters, such as k_{eff}, to those from the CEA calculations (APOLLO (2) and TRIPOLI (4)).

The parameters studied in the sensitivity calculations were core fuel burnup, the initial configuration of the control rods and the reactivity of experimental devices (figure 11.4).

The results obtained using the MORET (5) software confirmed the results of the CEA study into the reactivity insertion in the event of control rod ejection, notably: the higher reactivity "weight" of the control rods in the first core ring and the conservative nature of the new core in comparison to an irradiated core. They also demonstrated the weak influence of experimental device reactivity on the reactivity insertion in the event of an accident.

▶ Thermohydraulics

– **CATHARE** (*Code Avancé de THermohydraulique pour l'Etude des accidents de Réacteurs à Eau*): This two-phase thermohydraulics "system code[325]" has mainly been developed and used for safety analyses on pressurized water reactors (a study on the thermohydraulic behavior of reactors during incident or accident transients, updates of the associated processes), and research and development work. It has also been incorporated into IRSN's SOFIA simulator[326].

The CATHARE software has been under joint development by the CEA, EDF, AREVA-NP and the IRSN since 1979. The core and systems selected for a study

324. The match was found using the same library of cross sections as that used by the CEA.
325. A "system code" allows a system and its components (fuel, exchangers, pumps, structures, etc.) to be modeled in their entirety.
326. SOFIA (*Simulateur d'Observation du Fonctionnement Incidentel et Accidentel* – simulator for observing operation during incidents and accidents) is an information system used by IRSN for studies and training. It can calculate and monitor changes in the physical parameters of a pressurized water reactor in real time. It can also be used to simulate equipment failure and operator actions. The calculation can be stopped at a given moment to examine the state of the facility, and it is possible to go back to modify the scenario being studied. The reactors modeled in SOFIA are those in the French nuclear power fleet (900 MWe, 1,300 MWe, 1,450 MWe and EPR reactors).

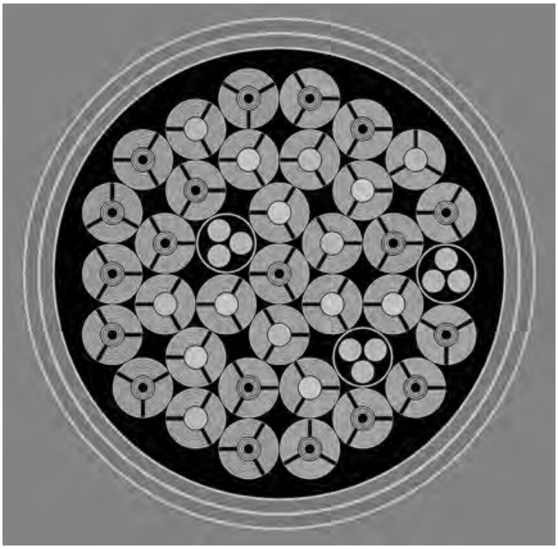

Figure 11.4. One of the core configurations of the Jules Horowitz reactor studied by IRSN using the MORET (5) software (with the inserted rods in red, the withdrawn rods in blue, and the experimental devices in green). © IRSN.

can be modeled in one dimension (1D), with the core represented by a "standard" channel or assembly, but the CATHARE software also has a 3D module that can provide a three-dimensional representation of the vessel and core.

Example of use:

In the early 2010s, the CEA, in order to draw up the preliminary safety analysis for the Jules Horowitz reactor, and IRSN, to perform the expert assessment of this report, used the CATHARE (2) software to study the "guillotine break of a particular element" (RGEP) accident for this reactor (the single core water supply collector – see figure 5.11). The aim was to ensure that this type of break could not initiate core melt in the reactor. The criteria selected for this purpose were a zero void fraction in the core (no boiling) and a maximum fuel plate cladding temperature of 400°C (to avoid failure due to creep).

The two cases studied (figure 11.5) were a double-ended guillotine break in the pool and a guillotine break with limited displacement in a room (a bunker, where the piping in this bunker has an anti-whip restraint). Several conditions were selected for the initial state of the reactor just before the break, namely those that initially seemed the most conservative (maximum reactor power, minimum core cooling flow rate, minimum water pressure at the core outlet, minimum water temperature at the core inlet and minimum water level in the reactor pool). The simulations performed notably resulted in:

- the reactor scram being triggered almost immediately, due to the low-pressure threshold at the core outlet being exceeded;

- gravitational flow in the emergency suction lines, from the pool, compensating for the outflow via the break and maintaining a satisfactory water inventory in the reactor coolant system (figure 11.5);
- in the case of the break in the bunker, the outflow through the break decreasing as the bunker filled with water and the broken section of piping was submerged.

The minimum margins in relation to the criteria selected were generally achieved shortly after the reactor scram.

The study performed by IRSN notably assessed the sensitivity of the results obtained by the CEA – indicating compliance with the criteria given above – to some scenarios, including, for example, the behavior of the reactor coolant pumps (risk of cavitation) just after a break has occurred, or the time it takes for a break to open. This study highlighted a risk of failing to comply with the criteria and, consequently, showed that the CEA needed to provide evidence to justify that there would be a sufficient water flow rate through the pumps to cool the core, even if the reactor coolant pumps were (temporarily) operating in degraded mode.

In addition, the CATHARE (2) software allowed the CEA to determine the forces experienced, notably by the anti-whip restraint in the case of a break in the bunker; these forces had to be calculated for the mechanical design of this device. The CEA calculation showed the full importance of this device, as a double-ended break in the bunker could lead to core melt.

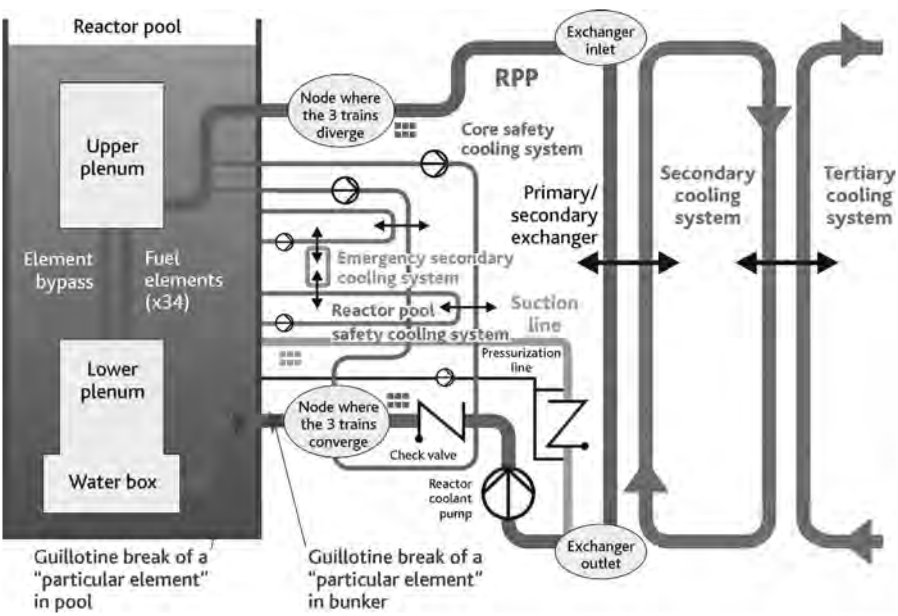

Figure 11.5. Diagram of Jules Horowitz reactor circuits and position of guillotine breaks (RGEP) studied. © Georges Goué/IRSN.

– **FLICA, DULCINEE:** These software tools are used to simulate the thermohydraulics in a reactor core and the thermal behavior of the fuel. They have been used at French research reactors for several decades. The DULCINEE software has a neutron model known as a "point kinetics" (or "0D") model, which can perform calculations combining simplified neutronics and thermohydraulics.

  The FLICA (4) software can be used to create a three-dimensional representation of a reactor core and process the two phases of the cooling fluid (liquid and steam). For thermal transfers in the fuel, the modeling is one-dimensional (1D).

  Together with the CRONOS software, the FLICA software can be used to obtain a more detailed representation (3D) of the core for transient studies performed using the CATHARE "system code". Figure 11.6 shows how they can be combined in the HEMERA software chain.

  Example of use:

  In order to determine the thermal energy deposited in the Jules Horowitz reactor fuel in the event of the accident reactivity insertion selected for the BORAX-type accident (control rod ejection) study, the CEA used[327] the combined CRONOS (2) and FLICA (4) software (without modeling the expansion of the fuel plates, a phenomenon that reduces the thickness of the water channels between the plates, thereby generating negative reactivity). The "point kinetics" DULCINEE code was also used for sensitivity studies, as this software ("0D") is suitable for small cores such as that at the Jules Horowitz reactor.

– **CFD (Computational Fluid Dynamics) codes**: The use of this type of simulation software is becoming more common, including at research reactors, to determine fluid flow on a local scale by solving Navier-Stokes equations averaged over time and space, in a domain discretized using meshes ranging from a millimeter to a centimeter in size.

  Examples of use:

  - In 2010, the Institut Laue-Langevin performed, in collaboration with the Argonne National Laboratory (ANL, Illinois, USA), studies[328] on the feasibility of "converting" RHF to UMo fuel with low uranium-235 enrichment. Two CFD-type software tools were used: the STAR-CD software (used by the ANL) and the CFX software developed by ANSYS[329] (used by the ILL). The validity of the models was checked through comparison with in-reactor measurements and benchmarking against the results of various models. These studies showed that changing the fuel without modifying the fuel plates would result in a significant degradation

327. In particular, see the CEA presentation at the TOPSAFE 2008 conference: "*The BORAX accident in the JHR*", Maugard B., Elie J.-P., Trémodeux P., Iracane D., Lemoine P., Ratel G., Berthoud G. *et al.*
328. In particular, see the ANL–ILL–communication at the RERTR 2010 conference on reduced fuel enrichment for research and test reactors: *"Thermal-hydraulic safety analyses for conversion of the Laue Langevin Institute (ILL) High Flux Reactor (RHF) from HEU to LEU fuel"*, Tentner A., Thomas F., Bergeron A., Stevens J. (https://www.rertr.anl.gov/RERTR32/pdf/S10-P4_Tentner.pdf).
329. ANSYS Inc. is an American company.

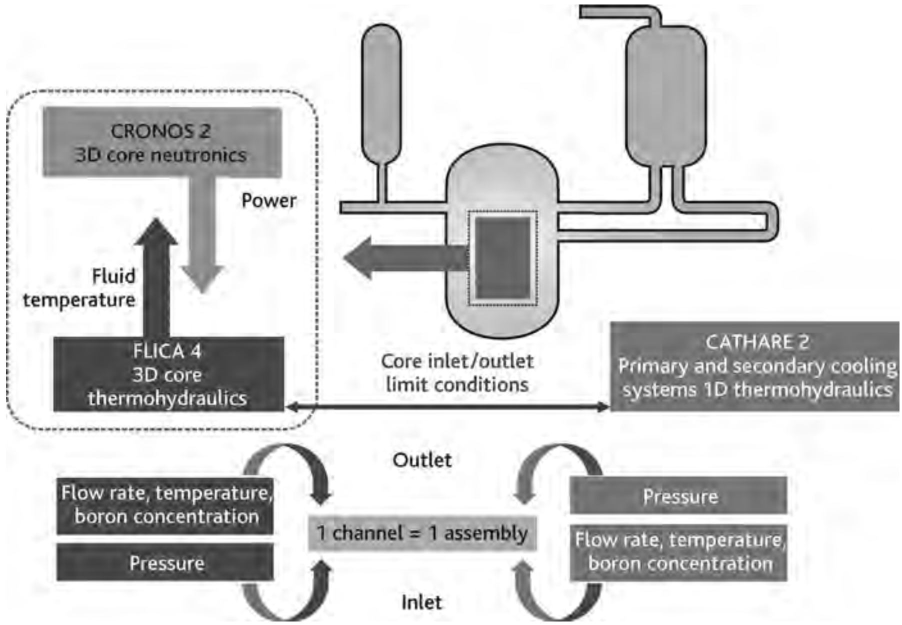

Figure 11.6. The combined CRONOS (2), FLICA (4) and CATHARE (2) software packages in the HEMERA chain: interface parameters between these three software packages. © IRSN.

in reactor performance, notably in terms of neutron flux. Other designs were then studied for the fuel element. One of these, which increased the quantity of fuel without changing the external dimensions of the plates, would maintain good reactor performance while securing safety margins in relation to the risk of boiling in the water channels between the plates. "Converting" the RHF core, however, is still dependent on developing and qualifying a new type of fuel with higher density than UAl.

- In 2010, the Institut Laue–Langevin also used a CFD code (CFX) to study the behavior of RHF thimbles, to demonstrate that the core fuel element would not melt in the event of a reactivity insertion resulting from the rupture of one or several thimbles.

▶ Thermomechanics

– **SCANAIR**: This software, which has been under development by IRSN since 1990, can be used to simulate the thermomechanical behavior of fuel rods in pressurized water reactors during power transients, and evaluate the risks associated with a loss of leaktightness or cladding failure. It is notably used to define, prepare and interpret tests to assess the capacity of fuel rods to withstand such transients, such as those that have been, or will be, performed as part of the CIP program at the CABRI reactor. The SCANAIR software can simulate rapid reactivity insertions (Reactivity Injection Accidents [RIA]) or slow power ramps such as those that could result from the rupture of steam piping or even the uncontrolled withdrawal of a rod cluster control assembly in the core of a pressurized water reactor. In particular, the SCANAIR software models the thermomechanical interactions between the fuel pellets (UO_2, $UPuO_2$) and rod cladding, the boiling of the coolant (water) and the various deformation mechanisms of the cladding material.

Example of use:

In its research to explain the melting of the fuel rods in the driver core of the CABRI reactor, discovered in 2004 (section 10.1.2), the CEA, the operator of this reactor, used several simulation software tools, including APOLLO (2), TRIPOLI (4), DULCINEE and SCANAIR. As indicated in section 10.1.2, the CEA concluded that the effects of the transients at CABRI on the rods of the driver core were insufficiently estimated in the safety analyses performed before these transients were carried out. The CEA then decided to develop a new calculation tool for the studies to be performed before future tests in the pressurized water loop as part of the CIP program. This tool links the SCANAIR[330] software to appropriate datasets. As part of the expert assessment of the file sent by the CEA aiming to demonstrate that the driver core could undergo future tests as part of the CIP experimental program without any damage, IRSN, as the developer of the SCANAIR software, had the expert assessment of the new CEA tool completed by AVN (Belgium), which did not issue any contraindications regarding the use of this tool. In addition IRSN used the SCANAIR software for a study[331] aiming to assess the validity of new strength criteria for the cladding of fuel rods in the CABRI core, which were proposed by the CEA. The aim of this IRSN study was to assess whether these new criteria were consistent with the results of tests performed in the SPERT and NSRR reactors in the USA and Japan respectively. These tests had made it possible to determine a failure threshold for stainless steel cladding expressed in terms of the energy deposited in the fuel (approximately 240 cal/g). In order to perform this study, it was vital to strictly use the same version of the SCANAIR software and datasets as those developed by the CEA for its own calculations of the impact of the future CIP tests on the driver core. The CEA made these elements available to IRSN. The IRSN study showed that the new criteria, expressed[332] in terms of maximum cladding temperature (1,300°C) and maximum equivalent cladding deformation (3.65%), were consistent with a failure threshold of 240 cal/g.

▶ Core melt accidents

– **SIMMER:** This software, which combines neutronics and fluid mechanics, can simulate a core melt accident in a fast neutron reactor. It was initially developed

330. The CEA had initially envisaged linking the CATHARE and SCANAIR simulation software, but in the end it decided to use just the SCANAIR software and undertake major work to calibrate and validate the thermohydraulic module to adapt it to the configuration of the CABRI driver core.

331. See the communication by IRSN at the IGORR 12 conference in 2009: "*Analysis of CABRI driver core new safety demonstration for fuel rods integrity during fast power transients*", Écrabet F., Pelissou C., Moal A.

332. Excluding the absence of fuel melt (the melting point of UO_2 is around 2,840°C).

at Los Alamos, from 1974 onwards. PNC (Power reactor and Nuclear fuel development Corporation, Japan), FzK (Forschungszentrum Karlsruhe, Germany, later the Karlsruher Instituts für Technologie [KIT]) and the CEA have continued its development for studies on fourth-generation fast neutron reactors. In France, it was used in the 1980s and 1990s to study theoretical core melt accidents in fast reactors (chiefly SUPERPHENIX).

Example of use:

In the 2000s, IRSN, together with FzK, adapted the SIMMER III software for the Jules Horowitz reactor, for exploratory studies into a BORAX-type accident. Numerous adaptations were necessary, notably to be able to correctly simulate the neutronic behavior of the core and the fuel used in this type of reactor, in the form of curved plates[333]. These studies have notably shown that the energy deposited in a core like that of the Jules Horowitz reactor could potentially exceed the inclusive value of 135 MJ and that the sequences involved (the simultaneous ejection of several control rods, for example) should be highly unlikely (through robust design, manufacture and in-service monitoring measures).

– **MC3D:** MC3D is a multi-phase thermohydraulics software developed by the CEA then IRSN. It can be used to simulate the steam explosion that would result from a thermodynamic interaction between the fuel (notably when it is in a liquid state) and the coolant in a reactor; such a phenomenon could occur during a reactor core melt accident. In particular, this software can be used to determine the dynamic pressures on structures (such as the walls of the reactor pool, for example). To begin with, it simulates the first phase of the thermodynamic interaction, called premixing, which is the rough mixing of the two fluids accompanied by varying degrees of vaporization. In some conditions, premixing can be destabilized, which may lead to a violent explosion similar to a detonation (second phase).

Example of use:

The CEA used the MC3D software to study the interactions between melted fuel and water during a BORAX-type accident for the Jules Horowitz reactor[334], notably to determine the loads that could be experienced by, firstly, the reactor block vessel and piping connections and, secondly, the reactor pool walls and floor; these loads would be created by the shock waves and their multiple reflections, as well as the expansion of the steam bubble.

333. IRSN communications: International Conference on the Physics of Reactors, PHYSOR 2008: Upgrading of the coupled neutronics-fluid dynamics code SIMMER to simulate the research reactors core disruptive RIA, Biaut G. *et al.*; TOPSAFE 2008 conference, "*Reevaluation of BDBA consequences of research reactors",* Biaut G. *et al*. See also the joint IRSN-CEA communication at the 18th International conference on Nuclear Engineering [ICONE]) in 2010: "*Validation of SIMMER III neutronics module for the simulation of reactivity injection accident in material testing reactors*", Chegrani Y., Ivanov E., Di Salvo J., d'Aletto C.

334. CEA communication quoted in footnote 317.

– **ASTEC**: The ASTEC (Accident Source Term Evaluation Code) simulation software system aims to simulate all the phenomena that would occur during a core melt accident in a water reactor, from the initiating event to any releases of radioactive products outside the reactor containment, with the exception of the steam explosion, which can be processed using the MC3D software, and the loads on structures, which can be processed using a software package like Cast3M (see below). The ASTEC software system (see figure 11.7) was developed jointly over a number of years by IPSN then IRSN, together with its German equivalent GRS; since then it has been developed further by IRSN alone. The applications of the ASTEC system chiefly concern the safety analysis of pressurized water reactors, with the evaluation of the radioactive releases that could result from core melt in such a reactor, and the examination of the procedures to implement in the event of such an accident. The ASTEC software system is also used by IRSN in its level-2 probabilistic safety analyses for nuclear power reactors. Lastly, it has also been used in the preparation and interpretation of experimental programs, in particular the Phebus FP global test program and in the tests carried out as part of the ISTP (International Source Term Program).

Examples of use:

- The CEA used the IODE software from the ASTEC system to study iodine transfer in the reactor building of the Jules Horowitz reactor during a BORAX-type accident – it used the CERES and GAZAXI software to evaluate the contribution made by the principal radionuclides to the (effective) doses during their migration into the environment[335].
- As part of the previously mentioned exercise to benchmark simulation software organized by the IAEA (the coordinated research project on "Innovative Methods in Research Reactor Analysis" — 2008–2013), IRSN performed calculations using the CESAR thermohydraulics software in the ASTEC system to interpret the reactivity insertion tests performed in the SPERT reactor. The CESAR software had to be adapted for fuel in the form of plates. This software made it possible to find the same plate cladding temperatures as those measured in the SPERT-IV-D 12/25 core assemblies (see above for the neutronics simulation using the MORET (5) software).
- IRSN also used the CPA software from the ASTEC system, which is dedicated to reactor containment thermohydraulics, to assess the effectiveness of a new confinement management system (dynamic rather than static) proposed by the operator of the RHF for accident situations[336]. The objective was to evaluate the conclusions drawn from the operator calculations that aimed to demonstrate the possibility of maintaining the reactor building at negative pressure (in relation to the annulus between the two containment walls) in such situations, taking into account the

335. CEA communication quoted in footnote 317.
336. See the communication by IRSN at the RRFM 2010 conference: "*Development of a numerical tool for safety assessment and emergency management of experimental reactors*", Maas L., Beuter A., Seropian C.

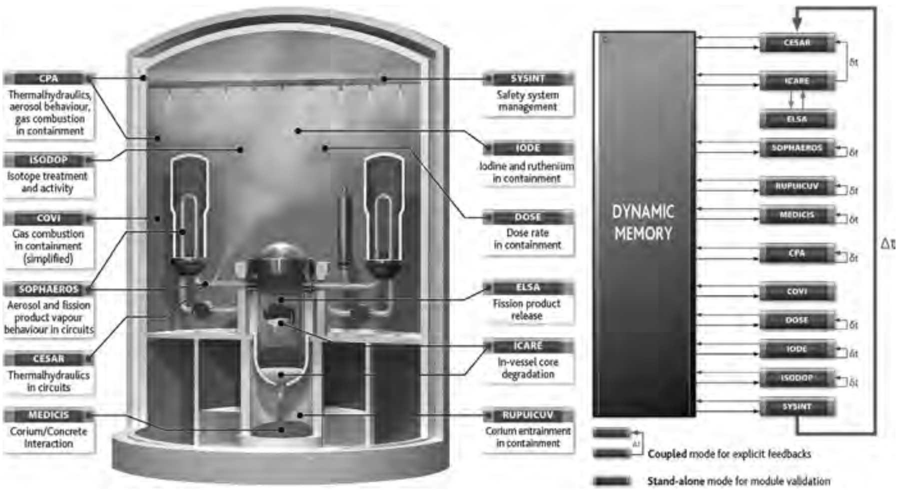

Figure 11.7. The various phenomena involved in a core melt accident (pressurized water reactor) and the modules that simulate them in the ASTEC software system. © IRSN.

increase in air temperature (due to the fission products released in the reactor building and the heating of the pool water in the event of core melt), the possibility of direct air leaks into the environment and the "swelling" operation of the annulus between the two containments (the internal concrete containment and the external metal containment). Three accident scenarios were studied: a BORAX-type accident, the fuel melting under water and the fuel melting in air.

▶ Mechanics

– **Cast3M, ASTER, ANSYS software:** Cast3M is a finite element simulation software developed by the CEA for structural and fluid mechanics. ASTER (*Analyses des Structures et Thermomécanique pour des Etudes et des Recherches* [simulation code for thermomechanical and structural analysis for research]) is a similar software developed by EDF. ANSYS Inc. is an American company that creates and distributes various structural mechanics software packages (including for loads leading to major deformation).

Examples of use:

- Cast3M is generally used by the designers and operators of French nuclear facilities for applications relating to metal structures or civil engineering works (reactor pools and buildings, etc.), notably at research reactors. It is widely used by IRSN, which also occasionally collaborates with the CEA for special development work. For example, in the field of civil engineering, developments[337] include improving the laws simulating the delayed or

337. These developments are produced as part of theses, linking IRSN and other partners, including the CEA.

dynamic behavior of concrete works in the event of loading during an accident (such as an earthquake, for example). These developments are then integrated into Cast3M, thereby being made available to all Cast3M users.

- The Institut Laue-Langevin used the ASTER software for the design studies for the PCS 3 building at RHF (which forms part of the post-Fukushima "hardened safety core").

– **EUROPLEXUS, LS-DYNA, RADIOSS**: EUROPLEXUS is a finite element simulation software for fast dynamics phenomena taking into account structures and fluids, originally developed by the CEA (PLEXUS code) and the Joint Research Centre (CCR) in Ispra in Italy (PLEXUS-3C), then reworked by a user group including EDF and ONERA. LS-DYNA is a computer code of the same type, developed in the USA by the Livermore Software Technology Corporation (LSTC), as is RADIOSS, developed by Altair Engineering. These software can be used to study the behavior of structures subjected to shocks, for example.

Example of use:

For the Jules Horowitz reactor, the CEA used[338] the EUROPLEXUS and RADIOSS software to study the behavior of the reactor pool structures in the event of a BORAX-type accident. It did this by modeling a steam bubble with characteristics such that it leads to the cases of overpressure determined previously using the MC3D software.

▶ Evaluations in emergency situations

During emergency situations[339] or emergency exercises, the operators of French research reactors and IRSN would base, or do base, their assessments on evaluations performed using simulation software simplified to a greater or lesser degree. More specifically, IRSN has a simulation software used for facilities other than pressurized water reactors in the nuclear power fleet, which can be used to determine the transfer of radioactive products within a facility and the releases into the environment (quantity and kinetics of releases for each radionuclide). This software models, in a simplified way, radionuclide leakage between rooms, transfers via ventilation systems and releases outside the facility. The deposition rate (for aerosols) and the efficiency of filtration devices are entered in the software as data. The software is used to produce and update "standard accidents" sheets (see section 7.7). In addition, it is used by IRSN experts to define the scenarios used in emergency exercises, and may also be used in the expert assessments performed by IRSN. This type of simplified software lends itself well to pre-calibration with the relevant data for the various research reactors, which makes it possible to create models available for rapid use in emergency situations or during exercises.

338. CEA communication cited in footnote 317.
339. In practice, as soon as an on-site emergency plan is triggered.

Science and Technology Series

Elements of nuclear safety
Research reactors

Jean Couturier, Hassan Abou Yéhia et Emmanuel Grolleau

This publication gives a global overview of the diversity and complementarity of research reactors, some of which have been or are still being used to conduct experiments that are essential for the development and operation of nuclear power reactors, including in relation to safety issues. This work highlights the many uses of these reactors, which have very different designs, use highly varied quantities of radioactive substances with varying levels of risk for safety and radiation protection, and which — in many cases because they are old or have been shut down — require appropriate measures to control the ageing or obsolescence of some of their equipment, as well as, on an organisational and human level, to ensure that they continue to be operated safely. For some research reactors, safety and radiation protection aspects must be considered, taking into account that two types of operators are present at the same time within these reactors: reactor operating personnel and operators in charge of experimental devices using neutrons from the reactor for fundamental or applied research purposes. There are two specific chapters on the safety standards established under the aegis of the IAEA for research reactors and on serious accidents, notably those involving criticality and reactivity, in research reactors. The second part of the work focuses on French research reactors, including the regulations and official documents applicable to these reactors, on lessons learned in France from significant events and accidents — as well as abroad, such as the Fukushima Daiichi nuclear power plant accident in 2011 — on the consideration of reactivity accidents in the design of French research reactors, and on the ten-yearly safety reviews carried out in France.

The Institute for Radiological Protection and Nuclear Safety (IRSN) is a public body undertaking research and consultancy activities in the field of nuclear safety and radiation protection. It provides public authorities with technical support. It also carries out various public service missions entrusted to it under national regulations. In particular, these include radiological monitoring of the environment and workers in France, managing emergency situations, and keeping the public informed. IRSN makes its expertise available to partners and customers both in France and worldwide.

IRSN
INSTITUT
DE RADIOPROTECTION
ET DE SÛRETÉ NUCLÉAIRE

Head office
31, avenue de la Division Leclerc
92260 Fontenay-aux-Roses
RCS Nanterre B 440 546 018
Telephone +33 (0)1 58 35 88 88
Postal Address
B.P. 17 - 92262 Fontenay-aux-Roses Cedex
France
Website www.irsn.fr

ISBN : 978-2-7598-2356-7
9 782759 823567