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COUPLING OF 3-D CORE COMPUTATIONAL CODES AND A REACTOR SIMULATION SOFTWARE FOR THE COMPUTATION OF P.W.R REACTIVITY ACCIDENTS INDUCED BY THERMAL-HYDRAULIC TRANSIENTS

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ABSTRACT

Reactivity accidents due to thermalhydraulic transients as steam line break or boron dilution accidents in a Pressurised Water Reactor are characterised by large asymmetric reactivity effects in the reactor core. A best estimate calculation of the consequences of such accidents requires the coupling of three dimensional and time dependant neutronic and thermalhydraulic computations in the reactor core. Furthermore in order to compute the time variation of core inlet and outlet boundary conditions, the thermalhydraulic transient in the reactor coolant primary loops and in the reactor vessel has to be also computed.

This paper describes the possibilities of the SAPHYR system (Reactor PHYsics Analysis computer code System) to calculate these transients, by coupling neutron kinetics and thermal hydraulics in the reactor core and the primary system. The first part of the paper gives a description of the CRONOS-2, FLICA-4 and FLICA-S computer codes and coupling methods. The second part of the communication deals with a Steam Line Break calculation performed with the SAPHYR system.

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1. INTRODUCTION

Accidents in a Pressurised Water Reactor due to a reactivity insertion in the nuclear reactor core can be caused by a control rod ejection or thermal hydraulics effects as a boron dilution or a moderator specific mass increase. After the power increase feedback effects due to Doppler effect in fuel elements and moderator density variation. A best estimate calculation of the consequences of such accidents, with large asymmetric reactivity and feedback effects, on the fuel cooling generally requires three dimensional neutron kinetics and thermal hydraulics computations in the reactor core.

The French Atomic Energy Commission has recently developed a set of new computer codes for reactor physics computations called the SAPHYR system which includes the following computer codes :

- APOLLO-2 performs multigroup neutron transport assembly calculations.
- CRONOS-2 calculates three dimensional neutron transport, diffusion and kinetics in reactor cores.
- FLICA-4 is dedicated to three dimensional steady state and transient two phase flow calculations
- FLICA-S computes steady state and transients thermal hydraulics in reactor coolant system.

The SAPHYR system permits coupling of these various computer codes in order to improve accuracy of nuclear reactor analysis. This paper deals with the possibilities of coupling neutron kinetics with primary loops and core transient thermal hydraulics in the SAPHYR system in order to perform accurate computations of reactivity accidents.

The first part of this communication gives a brief summary of the possibilities of CRONOS-2, FLICA-4 and FLICA-S codes which have been coupled in order to compute reactivity accidents. The integration of these three computer codes in the SAPHYR system is then described. Finally, in the third part of the paper, validation of the coupling for Steamline Break Accident will be presented.

2 : THE SAPHYR COMPUTER CODES SYSTEM DESCRIPTION

The SAPHYR reactor physics computer codes system is made by the APOLLO-2 : 2-D neutron transport code, the CRONOS-2 : 3-D neutronic reactor core code, FLICA-4 : 3-D thermal-hydraulic reactor core code and the FLICA-S thermal-hydraulic transient system code.

2.1 : The CRONOS-2 computer code

CRONOS-2 [1] solves either de diffusion equation or the even parity transport equation using finite element or nodal methods. It is applicable to three dimensional Cartesian or hexagonal, time-dependent multi group simulation of reactor core.

The neutron cross sections are provided by two dimensional, 99 groups transport calculation with the APOLLO-2 code [2]. The library is obtained after a transport-diffusion equivalence adjustment and cross section data are tabulated versus various parameters such as: burn-up, isotope concentrations, UO2 temperature or moderator temperature and density.

The reactor core is represented with homogeneous elements where the flux is expended over truncated polynomial (usually the Lagrangian). The numerical integration involved in the finite element method is performed using the Gauss Lobatto formula which leads to more regular matrices than the conventional Gauss formula does. The regular matrices enhance the vectorisation and speedup the numerical resolution.

2.2 : The FLICA-4 computer code

FLICA-4 [3] is dedicated to three dimensional steady state and transient single and two-phase flow. The physical modelling consists of three mixture equations of mass momentum and energy balance equations. a phase mass balance equation permits thermal disequilibrium computations and a drift velocity model is used for non-homogeneous two-phase flow. Furthermore. viscous and turbulent effects. both for single and two phase flow. are taken into account and a transport-diffusion equation is used to compute the 3-D boron concentration distribution. Fuel elements temperature fields are computed by a one-dimensional heat conduction model.

The numerical scheme is a fully implicit finite volume method which allows computations with unstructured meshing. The equations are integrated over control volume. and the mass momentum and energy fluxes are expressed in terms of volume averaged variables by using an extension for two-phase flow of the Roe's approximate Riemann's problem solver [4].

2.3 : The FLICA-S computer code

FLICA-S [5] is dedicated to the thermal-hydraulic transient computations of primary reactor coolant system. It a highly modular computer code which includes special modelling of pumps. pressurizer. steam generators.

FLICA-S can treat up to four loops reactors. The vessel is represented by a multi-volume model which allows computations of flow mixing in the down-comer. the lower and upper plenum and the reactor core. a specific a punctual model may take into account vaporisation and water level variation in the vessel's dome.

The FLICA-S model consists of two pressure nodes in primary side: one for coolant loops, the other for pressurizer. The state equations of pressurizer are combined implicitly with those of coolant loops to improve numerical stability.

For fast reactivity transient computations. FLICA-S has two neutronic models: a one-dimensional two-group, time-dependent diffusion model. and a point kinetics model. Fuel thermal dynamics are modelled with a doubly lumped fuel node. These core models are useful when a simplified analysis without coupling with three-dimensional codes as CRONOS-2 and/or FLICA-4 is performed.

2.4 : Coupling strategy

The three computer codes are integrated in an unique software. A modularity concept which separate data structures and computational modules. is associated with an external user language GIBIANE. This. allows both the individual use of each one of the computer codes but also the construction of computational schemes which correspond to needs of the user.

The computer codes coupling has two aspects :

- geometric coupling with regards to the spatial domains of interest and mesh systems.

- parametric coupling for the physical data sharing.

The spatial domain of CRONOS-2 includes normally reflector while the FLICA-4's one is limited to fuel region. Moreover, the independent mesh systems of the codes to be coupled may be preferable to maximise the efficiency of numerical resolution of coupled codes. Thus, geometric coupling for the domains and the mesh systems will precede parametric coupling.

The structure for physical data sharing is defined by the objectives of the analysis and the computational procedure. For example, boron concentration may be not transferred (or may be transferred with some modifications) from the thermal hydraulic code to the neutronic one for the sake of conservative safety analysis. The data to be generated and transferred in a simplified combination of codes are also different from those in a complete combination, as is demonstrated in the following section. The data sharing should be defined at the beginning of the algorithm but can be redefined during the course of calculation.

Data which can be transferred between the codes are:

- fission power, gamma-ray heating are transmitted from CRONOS-2 to FLICA-4;
- effective fuel temperature, moderator density, moderator temperature, moderator pressure, solute concentration in the core comes from the FLICA-4 to CRONOS-2;
- thermal power, pressure drop in the core, flow rate, enthalpy, solute concentration at the core exit calculated by FLICA-4 are necessary to primary loops thermal hydraulics computations with FLICA-S;
- fission power, gamma-ray heating[1], core outlet pressure, core inlet flow rate, enthalpy and solute concentration are FLICA-4 boundary conditions computed by FLICA-S.

3. : APPLICATION TO A STEAM LINE BREAK ACCIDENT CALCULATION

Various reactivity transient calculations are possible using the coupling of these codes. In this section an application to a Steam Line Break accident computation will be detailed.

3.1 : Steam Line Break Accident description

A steam line break accident in a reactivity accident due to a sudden rupture of the steam generator outlet pipe [6]. The rapid increase of the inlet steam generator mass flow rate due to the break and the secondary temperature decrease due to the secondary pressure transient, lead to an increase of the heat transfer from the primary side to the secondary side. Consequently, the reactor coolant temperature decreases and, with a negative moderator temperature coefficient, the core reactivity increases.

A pessimistic scenario can be obtained if we suppose the reactor at zero power at hot stand-by status with the least initial negative reactivity and the most negative moderator temperature coefficient. All control rods are inserted except the most effective one. The stuck of the control rod is assumed in the sector of the broken loop. As the

[1] For simplified calculations, fission power and gamma ray heating may be calculated by ponctual or 1-D neutron kinetics models of FLICA-S.

primary system temperature decrease. the reactor can return to criticality and the fission power may grow asymmetrically as a consequence of space-dependent thermal-hydraulics feedback's. Because all rods are inserted. the power excursion may only be controlled by soluble boron which will be injected when primary pressure falls below the setpoint of safety injection. Time when occurs safety injection will strongly depend on the water temperature in the vessel dome. For a hot temperature vessel dome. when the pressurizer will be empty. the primary system pressure decrease may be slowed down by vaporisation of the dome water. this will delay the start up of solute boron injection.

3.2 : Steam Line Break Accident computational hypothesis

The accident was studied with a four loops reactor for which. asymmetric effects will be more important than for a three loops reactor.

The initial power level is at 10 MW. A -1500 pcm initial anti-reactivity margin is assumed for the reactor core. and the core is at the end of a cycle without initial solute boron concentration.

The break critical mass flow rate is computed by the Moody correlation. The pressurizer is located on the same loop than the affected steam generator. Thermal inertia of structures and materials is neglected except for the fuel rods.

No external neutron sources are taken into account and the three dimensional neutron kinetic calculations are only necessary when the core returns to criticality [7].

3.3 : Reactor meshing and computational procedure

Figures 1 and 2 presents the reactor core radial meshing which are used for CRONOS-2 and FLICA-4. The radial meshing for CRONOS-2 includes the reflector and there is one radial mesh by assembly. The radial nodalisation for FLICA-4 is made of 35 meshes. the meshing is refined in the stuck control rod region. There is 13 axial level for both codes

Figure 3 presents the primary nodalisation for FLICA-S. only one intact loop out of three is presented on the figure. The reactor vessel is divided onto 4 azimutal regions for the down-comer. the core and the upper plenum. the vessel dome is schematised by one volume.

The computational time step is 0.02 s.

The figure 4 presents a flow chart of the computational procedure. To determine the initial power condition. iterations between FLICA-4 and CRONOS-2 are performed until convergence on the power distribution (loop 1). The initial steady state thermal-hydraulic conditions in the primary loops has been previously computed by using FLICA-S.

At the beginning of the transient. only FLICA-4 and FLICA-S are coupled (loop 2). CRONOS-2 steady state computations will start only when the simplified FLICA-S core model detects that the core returns near to criticality. (loop 3). After the core has became critical. the transient computation is performed by coupling the 3-D neutron kinetic modules of CRONOS-2. FLICA-S and FLICA-4.

All this computational scheme is defined by the user by using the GIBIANE external language

3.4 : Results and discussion

Figure 5 displays the coolant temperature entering the reactor vessel. Asymmetric cooling appears after the steam lines of the intact SG are closed to terminate blowdown. The RCS pressure decreases monotonically (Figure 6) and reaches the setpoint of high pressure safety injection system. Figure 7 shows the concentration of soluble boron at the core inlet due to safety injection. Figure 8 shows the variation of mass flow rate of primary loops. The progressive cool down brings the reactor to criticality at 21 s and the fission power begins to increase (Figure 9).

Figure 11 and 12 display assembly-wise coolant temperature distribution in the core inlet after 20 s and 40 s. The space-dependent thermal hydraulic feedback accompanied by asymmetric cooling will increase the peaking of fission power. The local heating of coolant in the affected region. decreases the peaking of fission power and reduces the reactivity. Another important local effect is found in fuel temperature distribution as expected (Figure 10). The more the fission power increases. the more the Doppler feedback will act strongly in power peaking region than elsewhere. It contributes to reduce both [illegible] power peaking and the reactivity. Figure 13 and 14 shows fission power distribution after 20 s and 40 s. The global form remains almost unchanged during transient because the asymmetric effect of the stuck control rod dominates power distribution. Figure 15 shows axial fission power distribution. As the core is heated. the distribution becomes skew to the bottom of the core.

4. : CONCLUSIONS

In the frame in the SAPHYR software system for reactor physics computations: CRONOS-2. a three-dimensional neutronic code. FLICA-4.a three-dimensional core thermal hydraulic code. and FLICA-S a primary loops thermal-hydraulic transient computation code are coupled and applied to analyse a severe reactivity accident induced by a thermal hydraulic transient: the Steam Line break accident for a pressurised water reactor until soluble boron begins to accumulate in the core. The coupling of these codes has proved to be numerically stable. Furthermore it has yielded clear local feedback effects in the SLB. such as:

1. The fission power distribution is dominated by the stuck control rod and varies little during transient.

2. The peaking of fission power increases until the core is heated and then decreases due to local thermal hydraulic feedback effects.

3. Local feedback effects play a important role in decelerating the excursion of fission power.

4. Flow mixing in the core is significant.

5. Core power is transferred mainly to faulted region and their mixing in the core by cross flow is not important.

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fig. 1 : CRONOS 2 radial meshing

fig. 2 : FLICA-4 radial meshing

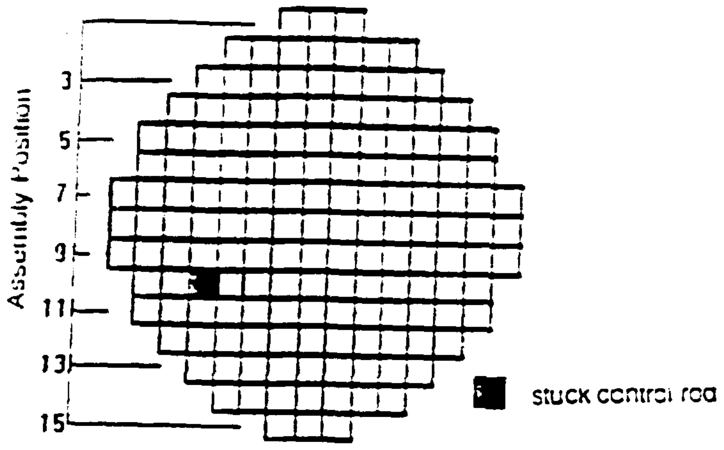


fig. 1 : CRONOS 2 radial meshing

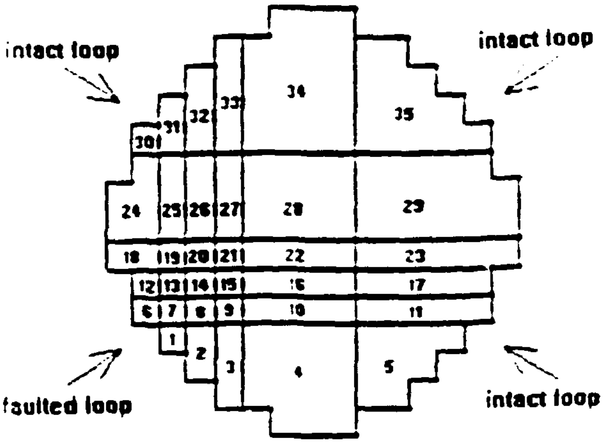


fig. 2 : FLICA-4 radial meshing

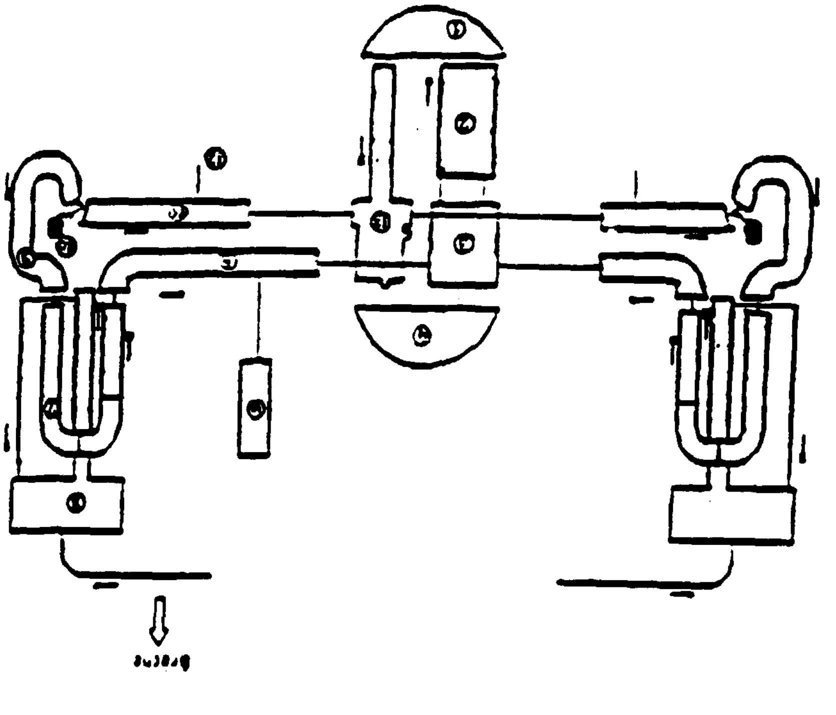


fig. 3 : FLICA-S Reactor nodalisation

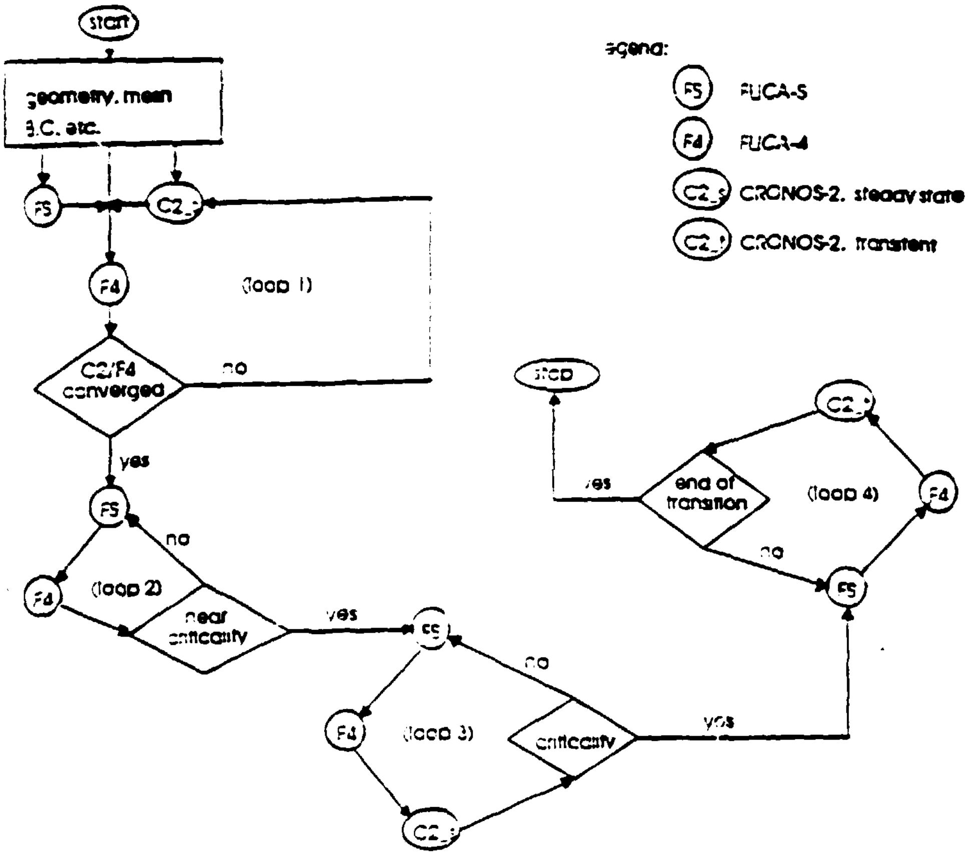


fig. 4 : Flow chart for the SLB analysis

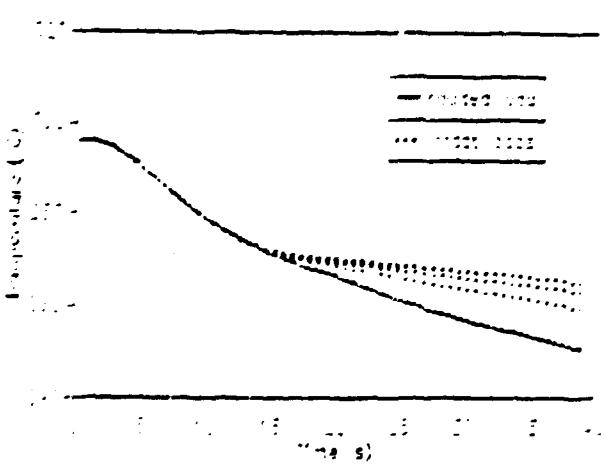


fig. 5 : Reactor vessel inlet temperatures evolution

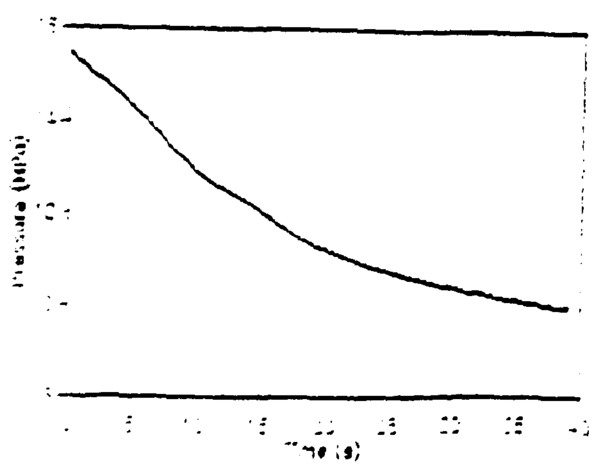


fig. 6 : RCS pressure evolution

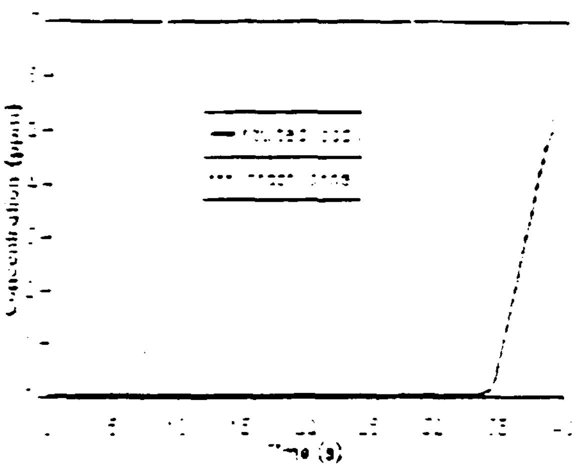


fig. 7 : Dilute Boron concentration evolution at core inlet

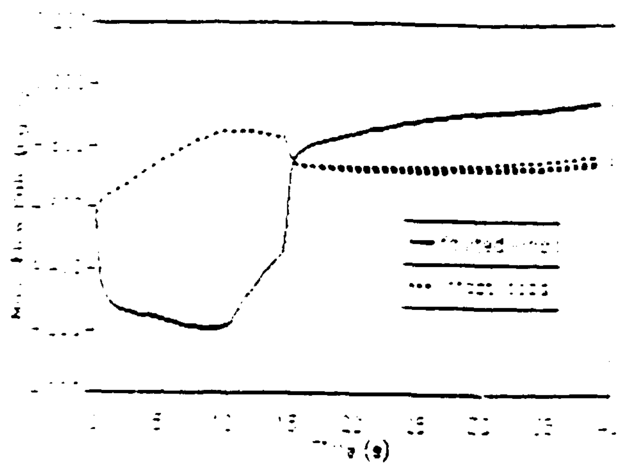


fig. 8 : RCS mass flow rate evolution

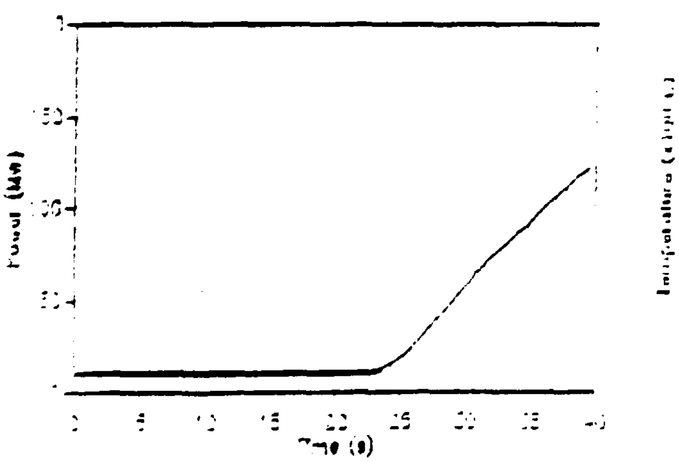


fig. 9 : Core fission power during the SLB accident

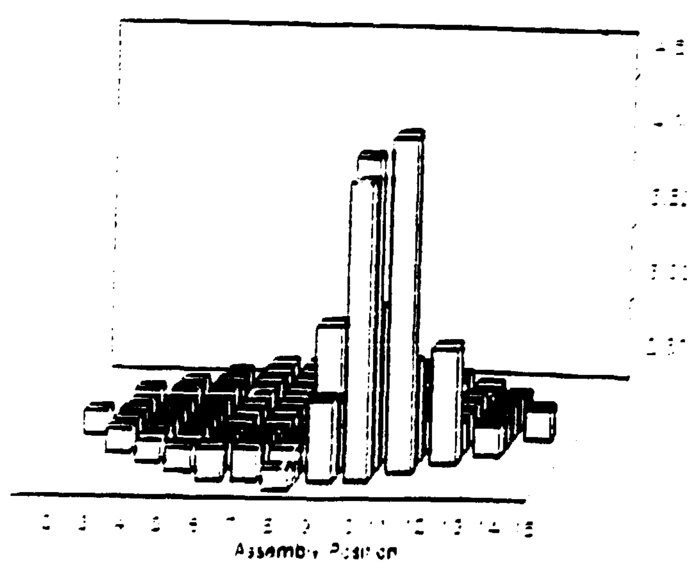
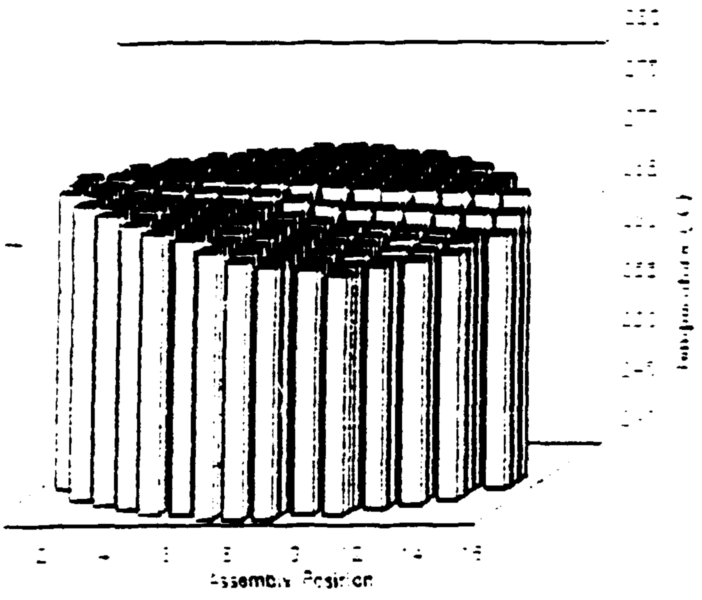


fig. 10 : radial effective fuel temperature distribution at t=40s



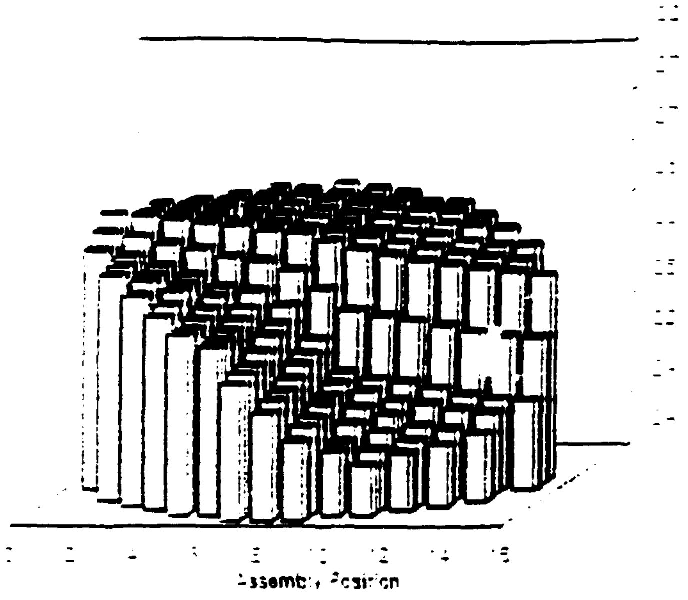
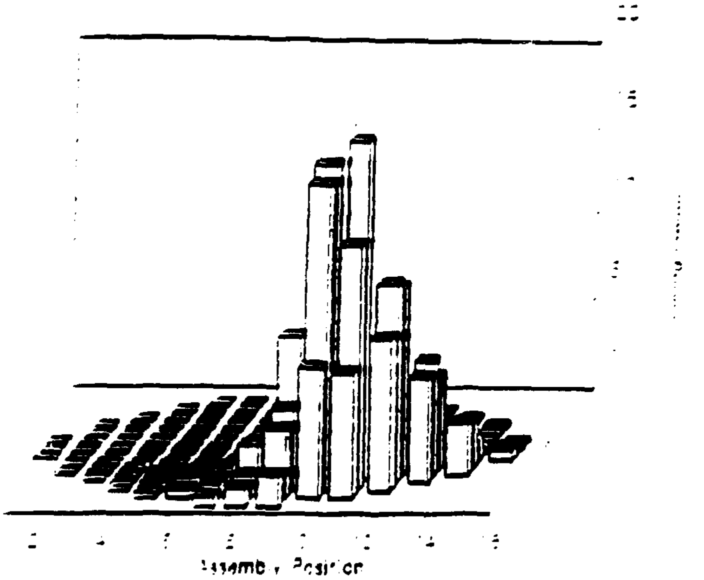


fig. 11 : Inlet core temperature distribution at t=20s

fig.12 : Inlet core temperature distribution at t=40s



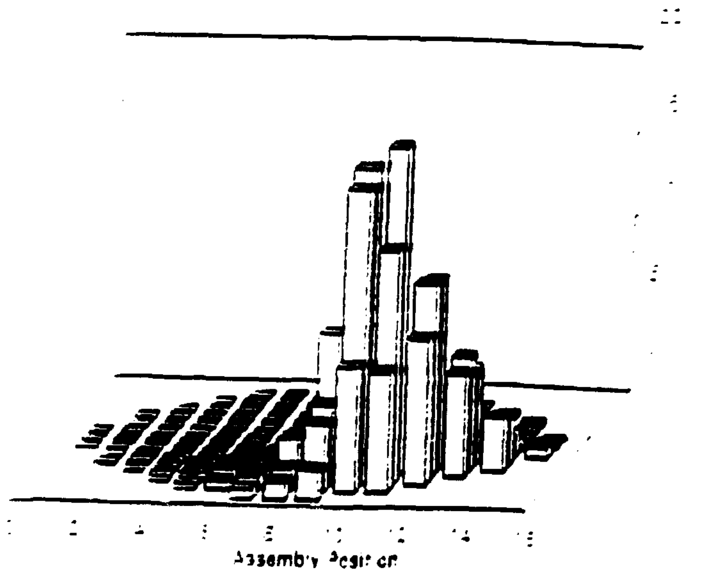


fig. 13 : Core power distribution at t=20s

fig. 14 : Core power distribution at t=40s

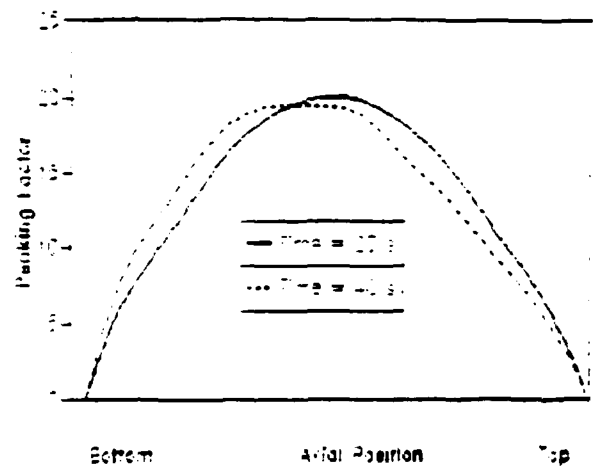


fig. 15 : Axial power distribution evolution during SLB accident