A REVIEW OF FAST REACTOR ACTIVITIES IN SWITZERLAND – April 1985

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1. Introduction

On 23 September 1984 nuclear energy in Switzerland successfully circumnavigated the most dangerous political hazard since its introduction 15 years ago. On this day the Swiss people voted 55:45 against an initiative calling for a halt to all nuclear construction, and 54:46 against a companion initiative which aimed at the same end by taxation and rationing of all nonrenewable energies plus hydro-electric power. Another important event was a vote in the Federal Parliament in March 1985, in which a clear majority agreed that current projections of electricity demand warrant construction of another large power plant. From the legal point of view this was equivalent to granting a general permit for the Kaiseraugst nuclear power plant (According to the Swiss licensing procedure two additional permits concerning the safety of the plant are necessary, but these can be granted by the Government without consulting Parliament). However, the future of this much-debated project remains uncertain, since antinuclear politicians now exert even stronger pressure on the responsible parties to take up negotiations for cancelling the plant.

Concerning a national energy research policy, a number of different committees and interest groups have made somewhat conflicting recommendations. As a result, at effectively the same level of total R+D funding, the research programmes have been further diversified. In the nuclear fission field, for example, there are activities related to many different reactor concepts, including the Light Water Reactor, the Light Water High Converter Reactor, the High Temperature Reactor, the Liquid Metal Fast Breeder Reactor and the recently proposed new concept of a small heating reactor. In 1984 the total expenditure for fast reactor activities remained the same as that in the previous year, – but the budget for 1985 has declined. The 6.0 million Swiss Francs expended in 1984 have been allocated to an LMFBR safety programme (46 %) and a fuel development programme (54 %).

All activities reported below are carried out at the Federal Institute for Reactor Research (EIR). In the natural convection studies described in Section 5, the Nuclear Engineering Laboratory (LKT) of the Federal Institute of Technology at Zürich is actively participating. In the past twelve months collaboration with foreign research organizations in the Federal Republic of Germany, France, Italy (JRC Ispra) and the U.K. for the LMFBR safety programme, and the Federal Republic of Germany and the U.S.A. for the fuel development programme has proved to be very fruitful. In this context an attachment agreement with CEA-DERS at Cadarache is worth mentioning, since it enabled an EIR staff member to participate in the prediction and analysis of the SCARABEE-APL in-pile tests.

2. Hypothetical Core Disruptive Accidents

A study of the influence of burnup on the consequences of an unprotected LOF in a large power reactor (SNR-1300) is nearing completion. The work was stimulated by previous analyses which shed some doubt on the assumption that with respect to energy release an EOEC calculation is always conservative, an assumption frequently made in risk evaluation. A higher energetics potential of the BOEC core could be due to (1) early voiding of fresh fuel assemblies, which can drive a large number of sodium-filled irradiated assemblies into transient overpower conditions and hence increase the potential for an MFCI, and (2) delayed dispersal of fresh fuel with correspondingly delayed negative reactivity feedback.

Parallel calculations were carried out at KfK using SAS3D and at EIR using the CAPRI/KADIS code. This allowed interesting comparisons of the performance of different computational models to be made. It was found that the work energy potential of the BOEC core is only about one-tenth of that of the EOEC core. The calculations showed that, in the BOEC core, the transient overpower conditions arise in peripheral core regions with small sodium void worth and the power transient is rapidly terminated by fuel dispersal in the central region, where the fuel reactivity worth is high. It was concluded that for conservative bounding cases a CAPRI/KADIS calculation gives similar results to a more detailed SAS3D calculation. As an example the work energy calculated by the two codes for the BOEC core agreed within 10 %.

3. Fuel-Coolant Interaction

A series of freon-water experiments in a simple tube geometry, which began in 1982, has been terminated. Preliminary analyses have indicated that the observed supercritical pressure spikes resulted from a type of water hammer effect in which a pressure wave passing through a highly compressible fluid-vapour mixture impinged upon a less compressible pure fluid region. This led to the suggestion that, in contrast to the assumption made in current MFCI models, liquid-liquid contact in the interaction region is not a prerequisite for the generation of high pressure spikes.

In order to pursue these ideas analytically, a simple theoretical model of the water hammer process was devised. The model was compared with measurements in air-water mixtures performed in the same geometry as the vapour explosion experiments. The calculated parameters (velocities of the original and the reflected wave front, pressure magnification at the boundary of the air-water mixture) are in general agreement with the measurements. The results support the above idea that a water hammer effect may play an important role in fuel-coolant interactions - at least in certain geometries.

4. Response of Structures to HCDA Loading

The activities related to the analysis of HCDA induced fluid-structure interaction effects in the primary circuit of an LMFBR have been continued. They comprise (1) the further development of the containment code SEURBNUK/EURDYN, (2) the validation of this code and (3) its application, together with the general purpose structural mechanics code ADINA, for a generic reactor roof study. In the past twelve months emphasis was on the analysis of the Cadarache containment experiments MARA-4 and MARA-8, for which information was made available to EIR through an EIR/CEA collaboration agreement. This work led to a number of improvements to the EIR version of SEURBNUK/EURDYN. It has been decided that many of these improvements will be incorporated in a standard version of SEURBNUK/EURDYN which is jointly developed by EIR, Euratom and the UKAEA.

MARA-4 models an inner tank and a core support. Preliminary SEURBNUK/EURDYN calculations showed that for such a geometry the explicit fluid-structure coupling used in the early versions of the code becomes numerically unstable. The problem could be overcome by introducing a new semi-implicit coupling algorithm. The MARA-8 test geometry is similar to the well known MARA-1 model except for the roof which is flexible in the former and rigid in the latter case. This model allowed testing of a newly developed flexible roof option for SEURBNUK/EURDYN.

Based on the MARA-4 and MARA-8 containment experiments a comprehensive code comparison exercise involving the codes SEURBNUK/EURDYN, CASSIOPEE and SIRIUS was carried out in collaboration with the CEA. The comparison showed that for the important parameters the different codes predict similar values and that these values are in overall good agreement with the experiments. No discernible effects due to differences in the numerical approaches (e.g. Euler versus Lagrange

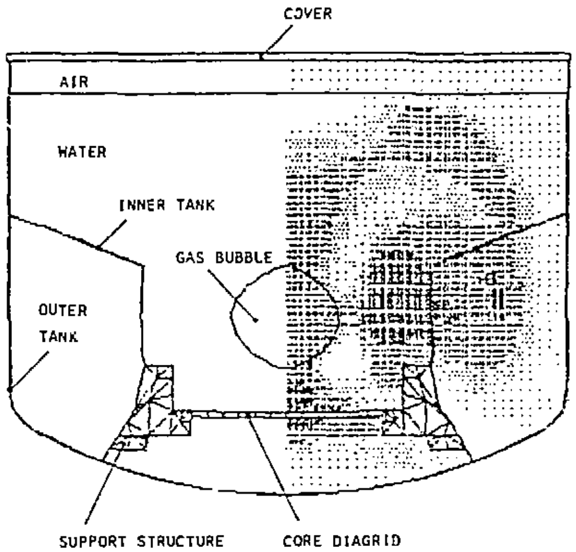

Fig. 1: MARA-4

Analytical model and calculated pressure distribution 0.2 ms after firing of the charge (Dark areas correspond to high pressures and vice versa). Expansion waves reflected by the inner tank and the water/air interface are clearly visible.

for the fluid mesh, finite element versus finite difference for the structure discretisation) could be identified. As far as roof impulses are concerned the use of dynamic (rather than static) material data during the charge calibration results in noticeably better agreement with the experiment. The timing of events, strains and displacements, however, appear not to be similarly affected. Some calculational uncertainty results from the fact that in the different codes certain structure boundary conditions have to be approximated in different ways.

The study of the dynamic behaviour of a three-dimensional reactor roof structure has progressed to a stage which allows a fluid-structure interaction calculation to be carried out. The objectives of the study are to determine whether (1) possible fluid-structure interaction effects between a sodium slug and a three-dimensional reactor roof can be simulated by a substitute 2-D model of the roof and (2) in a realistic reactor situation such interaction effects are important, i.e. lead to a redistribution of the energy released in a HCDA. The methods adopted in the study have been described in previous activities reports.

The study has shown that it is possible to establish a relatively simple 2-D/3-D equivalence for the roof not only in the case of linear but also in the case of non-linear material properties of this structure. The adopted model is homogeneous and has an equivalent density which preserves the total mass. The equivalence is achieved by adjusting the Young's modulus and the yield strength within physically realistic limits. The applicability of the model was confirmed, since (1) lateral motion of the roof, indicating true 3-D behaviour, was minimal, and (2) a single set of material properties could be found independently of the boundary conditions between the roof annulus and the rigid central plug.

5. Natural Convection in a Subassembly

The SONACO natural convection experiment in a 37-rod sodium-cooled bundle has been described in detail in previous activities reports. Particular features of the experiment are the two different cooling modes: The heat generated in the bundle is transferred either to cooler sodium in an annulus surrounding the hexagonal wrapper of the bundle (radial mode), or to a heat exchanger in a plenum above the bundle (axial mode). Depending on reactor design both cooling modes can play an important role as a mechanism for removing the decay heat from a subassembly with seriously disturbed flow conditions.

After sodium filling earlier in the year the SONACO test section was taken to its full power, which had meanwhile been increased from 10 to 30 kW, in Autumn 1984 (The maximum power of 30 kW corresponds to a rod surface heat flux of 6 W/cm^2, i.e. a heat flux which is typical for a large power reactor such as Super-Phenix when at 5 % of its full power). Up to the present a comprehensive series of tests has been carried out with the radial cooling configuration and the inlet of the bundle completely blocked. Emphasis has been on tests with a constant power distribution, but a few tests have also been performed with radially varying power distribution.

A representative selection of the tests was analyzed using two different "porous medium" computer codes, BACCHUS-T and INCA. The BACCHUS calculations were performed jointly by EIR and CEN Grenoble in the framework of a collaboration agreement, and the INCA calculations were contributed by NPDE Dounreay. Table I gives a summary of the conditions for these analyzed tests together with measured and

Table I: SONACO Tests with Radial Cooling

Test No	Annulus Inlet Temperature (Deg C)	Annulus Flowrate (l/sec)	Total Power (kW)	Maximum Temperature Difference (Deg C) BACCHUS-T	INCA	Experiment
536	233.4	0.988	8.06	50.3	52.0	50.0
1115	251.7	0.991	9.88	61.2	61.4	60.1
1120	248.8	0.487	10.07	68.5	67.8	66.4
1125	243.5	0.244	10.06	80.8	79.1	79.0
1130	237.2	0.123	10.09	108.1	105.6	105.0
1136	220.6	0.241	19.57	146.7	140.1	142.5
1159	221.7	0.247	29.42	207.9	196.1	202.2
1254*	235.1	0.238	10.00	75.7	74.8	74.0
1258*	234.9	0.243	9.87	86.0	83.8	84.4
1350	229.2	0.0612	10.18	172.0	168.0	160.8

*) Tests with radially varying power distribution

predicted maximum temperature differences (maximum temperature in the bundle minus annulus inlet temperature). It can be seen in the table that this important parameter is well predicted. Similar conclusions could be drawn for the detailed axial and radial temperature profiles.

Complementary information has been obtained from movable and fixed velocity probes. In this context measurements of subchannel velocities in the bundle are worth mentioning, since they have directly confirmed the existence of predicted internal circulation loops. The measurements showed that, in a purely natural convection regime, the local velocity fluctuations are relatively large and instabilities with very low frequencies occur. For the range of test conditions studied so far these velocity fluctuations, however, do not appear to induce (or be the result of) temperature fluctuations of any significant size in the peak temperature region.

In the next phase of the SONACO experiment efforts will concentrate on the axial cooling configuration. Preliminary tests have indicated that with this configuration local temperature fluctuations have already become significant at a relatively low power level. This raises interesting questions concerning the limits of the range over which a stable natural convection regime predominates and the applicability of computer codes beyond this range, i.e. in a regime which is characterized by single-phase non-stationary conditions. It is hoped that the next phase of SONACO will provide the data necessary to answer these questions, and to this end the experiment is currently being equipped with a fast data acquisition system.

6. (UPu)C Fuel Development

(R.W. Stratton)

Work in 1984/85 has concentrated entirely on completing the sphere-pac (UPu)C fuel pins for the FFTF test AC-3. In the year before an incident in a waste conditioning box had led to a contamination of the fuel lab. Following clean-up and recommissioning of equipment, the fuel and pin fabrication was restarted in Spring 1984. By the end of April 1985 about 27 pins will have been successfully fabricated.

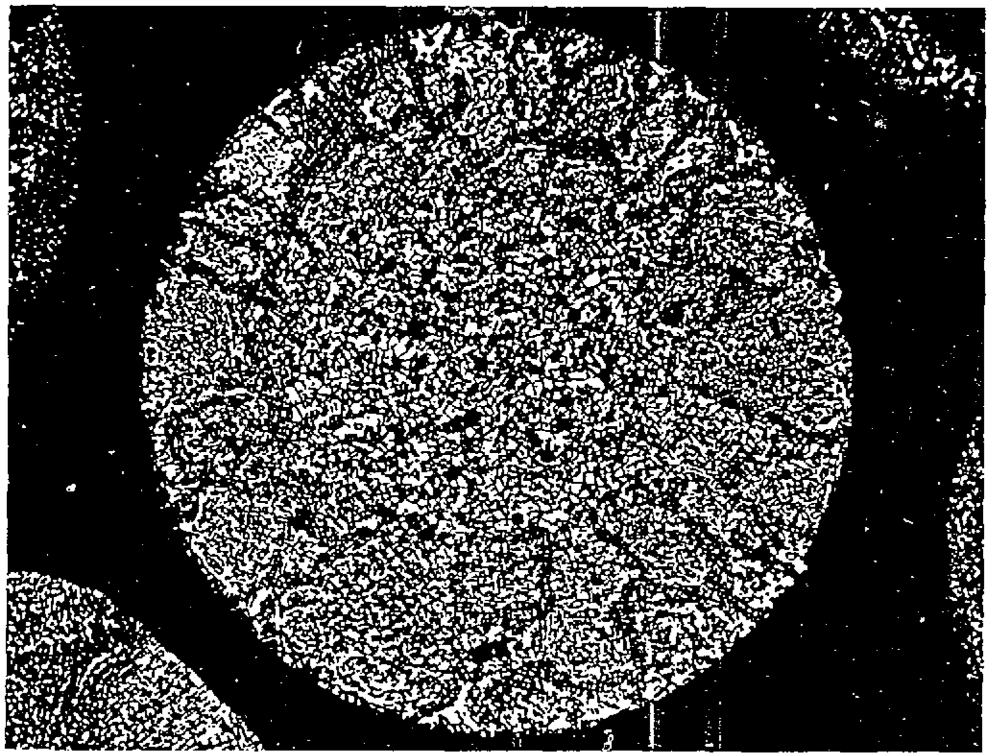

Fig. 2: Microsphere from the Fuel Production for the FFTF Irradiation Test

Diameter of the sphere:	0.8 mm
Matrix:	U-Pu monocarbide
White phase:	U-Pu sesquicarbide

Pores appear black

Due to strictly controlled procedures, consistent fuel quality was achieved with fuel to specification regarding stoichiometry, form, density, etc. The sinter oven and criticality limits on the fissile material were factors controlling fuel production rates. Nevertheless, towards the end of the campaign when enough fuel became available, as many as 5 fuel pins per week could be fabricated.

Production of the pins proceeded well with only minor deviations from the specifications. In all cases the smeared density reached 78 to 80 % of the theoretical density, corresponding to a value at the high end of the permitted range. The very flat axial density profiles in all pins are particularly worth mentioning. Welding produced very consistent results as seen in the dimensional inspection and radiography. A positive feature was also the low level of alpha contamination found after welding the closure end-cap, requiring only minor decontamination. This was due to the special pin-box docking device used and clean handling during vibro-filling.

For experiment approval in the U.S.A. extensive calculations of the fuel/pin behaviour during steady state operation and possible transients were carried out in collaboration with Los Alamos National Laboratory. In this context an extension for calculating the "cumulative damage fraction" was added to the EIR-developed sphere-pac code SPECKLE-IV.

Due to changes in FFTF operating plans, the experiment is now due to operate at a maximum rating of 80 kW/m over about 600 EFPD to a burnup of approximately 9 % fima.