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Institute FOR Safety Technology

JOINT RESEARCH CENTRE

COMMISSION OF THE EUROPEAN COMMUNITIES

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EXECUTIVE SUMMARY

NUCLEAR FISSION

The Institute participates in the execution of the following nuclear fission specific programmes:

- Reactor Safety
- Waste
- Safeguards

In the present pluriannual programme a number of activities in the reactor safety programme were either eliminated or had to suffer strong reductions. Continuity could be assured to those projects which are highly relevant for the assessment of residual risk of nuclear power plants.

The LOBI facility, a 1:700 volume scaled model of a four loop 1300 MWel. PWR was recommissioned after the execution of modifications. Since then four tests were performed in which a steam generator tube rupture, a 5% cold leg break, a primary system cooldown with main coolant pumps off and the bypass behaviour in the upper plenum/upper downcomer were investigated respectively. This experimental programme provides the necessary data for an extensive code application and assessment effort. Extensions of this programme to solve accident management problems are being studied.

Six tests were performed in the FARO facility in which up to 125 Kg of molten UO_2 at about 3000°C were released from the furnace. In various test sections fuel freezing and plugging phenomena were investigated as well as the potential of a fuel jet to perforate a steel plate, respecting boundary and initial conditions of a sodium cooled reactor.
The study of these phenomena is highly relevant for the understanding of in-vessel and ex-vessel phenomena during a core melt down. Test results are used to verify models and codes which should predict very low probability events in a nuclear power station.
Because of its build in flexibility the FARO facility will most probably be used next to solve light water reactor severe accident problems.

Always in the frame of unlikely but severe events a large effort was started in cooperation with interested Member States on the potential release of radioactive products into the outer containment during a fuel melt down accident.
This project has its focal point in the execution of in pile tests in the Phebus PF facility at C.E.A. Cadarache which are planned to start in 1992. JRC provides a deputy project manager and a small team which is integrated in the project work on site. Shared cost activities are being carried out by Member States to help in the choice of instrumentation, definition of the test matrix, execution of out of pile tests e.g. on fission product chemistry and analysis of detailed lay out of the experimental rig. At Ispra existing calculation tools are widely used in this preparatory phase but a more general and wider ranging activity for a European Source Term Suite has been launched which is intended to integrate European models and codes.

Work on Fast Breeder reactors suffered severe cuts in the last few years. A still ongoing activity is the development of the European Accident Code (EAC). The

second version of the code is being released which compared to the first presents advanced modelling in the areas of fuel pin behaviour, molten fuel motion inside the pin and fuel motion in the coolant channels. Moreover the code now simulates properly the hexcan geometry and has a new detailed neutronics calculation.

Using its large hot cell facilities the Institute cooperates with the Materials Institute in analysing with non destructive inspection techniques radioactive components of reactors. This is part of the so called PISC III activities. In the reporting period X-ray and ultrasonic inspections were carried out as well as microstructural investigations.

The Institute's main task in the waste program is the construction, commissioning and operation of PETRA, a facility which is capable to perform typical high active processing steps in fuel reprocessing including verification, optimisation and demonstration of waste treatment and conditioning processes in function of confinement and disposal concepts. The construction, terminated in january, was followed by cold commissioning tests which are expected to terminate in early 1990.
Always in the frame of the waste programme the construction of a fully automised waste barrel monitor based on multiplets correlation analysis could almost be terminated. This work draws from experience gathered in measuring campaigns executed on behalf of external contractors.

Concerning Safeguards the Institute is involved in the development of non destructive assay methods, it performs a large series of contractual tasks in support of DG XVII and it runs the Performance Laboratory PERLA which constitutes a valid point of contact between instrument development and application. This laboratory is now functioning with Pu and U test samples of known isotopic composition and weight which are representative for in field campaigns. Ten training courses were organised in 1989 for EC and IAEA inspectors. The instrument development work concentrated mainly on improving the accuracy of existing measuring methods such as high resolution gamma spectroscopy, active neutron interrogation and others via the development of better data acquisition, elaboration and interpretation systems.

NUCLEAR FUSION

The work in the nuclear fusion area concentrated mainly on the construction of the European Tritium Handling Experimental Laboratory (ETHEL) the preparation of tests in that laboratory and the analysis of safety problems specific to a LiPb blanket.
The ETHEL project made significant progress e.g. the building is cast up to the roof, a 350 m^3 double walled steel container is ready for installation, experimental glove box suites were successfully tested. According to planning the construction will be terminated in 1990. Contemporarily tests and analyses are performed in the areas: assessment studies of tritiated waste, Tritium recovery from liquid LiPb eutectic, recycling of hydrogen and deuterium from first wall materials, hydrogen interaction with fusion relevant materials.
Interesting results were obtained in fusion blanket tests which simulate the interaction between liquid LiPb and water. Experiments simulating steam generator tube rupture showed that the chemical reaction is self limiting due to hydrogen and Lithium oxide formation and that steam explosions are unlikely to

occur. In collaboration with Westinghouse Hanford Company the transport of potentially radioactive species in the alloy has equally been measured.
More general safety analysis work was conducted for ITER and NET related to loss of flow and loss of coolant accidents. The work was performed with a new developed code called THERM which allows a 2-dimensional thermal and thermomechanical analysis of major reactor components including the proposed divertor plate.

NON NUCLEAR ACTIVITIES

Industrial Hazards

The Institute broadens the spectrum of those activities which are not solely applicable to the nuclear area and in which experience, gained in the past particularly in thermodynamics and structural analysis, find challenging new applications.
Research on chemical and fluid dynamic phenomena associated with runaway reactions continued as part of the Industrial Hazards programme. A facility called FIRES (Facility for Investigating Runaway Events Safely) is planned to be completed in 1990. Most of the 1989 resources were dedicated to this installation. In addition 1:1 scale model tests were performed in a glass reactor to determine the stirrer performance . A 2 litre reaction calorimeter was used to gain basic knowledge of the chemical processes prior to their investigation in FIRES.
Satisfactory agreement between experimental data and results of a mathematical simulator named FISIM (FIres SIMulator) were obtained. This indicates that interactive real time computer simulation of batch-type chemical reactors could become a promising control and operation tool also for industry.
Investigations on multiphase flow phenomena in reactor relief systems during venting were continued measuring the influence of viscosity and surface tension on vented mass fractions and related flow patterns.
Analytical studies performed in parallel make use of existing U.S. codes and concentrate on the development of a new computer programme called VESSEL which models one-dimensionally the transient behaviour of multicomponent mixtures of chemicals during a venting process.
The nuclear reactor safety code RELAP5/Mod 1 has been used to describe the behaviour of a petrochemical plant under off-normal conditions.

On request of the Institute for Remote Sensing Applications a small effort in the area of modelling marine transport processes is persued. Comparative analysis of available codes with differing models and numerical techniques was performed by applying them to particular situations e.g. circulation pattern in Northern Adriatic, or by evaluating their physical application limits.

Reference Methods for the Evaluation of Structure Reliability

The objective of this activity is to improve the modelling of non linear structural mechanics problems in both transient dynamic and quasi static regimes including the behaviour of materials under those conditions. The Institute has pluriannual experience in this area for steel structures and cooperates with a large number of organisations in the Member States which most recently have decided to combine their efforts in an association of laboratories. When enlarging the scope of this research including e.g. structures made of concrete, experience had to be

gained quickly by testing reinforced concrete elements and developing global physical models. Uniaxial cyclic bending tests were performed and biaxial bending tests in the presence of axial compressive loads are being prepared. Research also continued on steel specimens to determine dynamic material response under biaxial loading conditions showing that the von Mises criterion is really valid only for the initial yield surface.
The Institute already possesses small and large testing machines to determine material properties. To those a large reaction wall with a height of 16 m and a width of 21 m will be added, the construction starting end 1989. This facility is designed for large and full scale testing of sub-assemblies and full structural systems. It will be available to industry for testing innovative design concepts and prototypes.
In the area of computational mechanics the development of Plexis-3C, a finite element computer programme developed in collaboration with CEA for the analysis of nonlinear transient dynamic problems involving compressible fluids has been completed. Modelling of natural convection has been extended from 2D to 3D situations.

Exploratory Research

The Institute was in charge of an action entitled H/D extraction purification and separation in which gas-solid separation processes were analysed to resolve problems of purification and extraction of gases. Gas chromatographic techniques were employed to characterize Na-, Ca-, and Ni mordenites as adsorbent materials.
The final aim of these studies is to develop a model which allows to predict the adsorptive properties of materials and which thereby may give guidance in the production of adsorbants for given separation or purification processes.

The Institute also contributed to the Boron Neutron Capture Therapy Project. Calculations were performed in order to optimise the neutron energy spectrum of a neutron beam. It is intended to develop epithermal neutron beams which have optimal penetrations in the boron containing tissues.

Work for Third Parties

These activities can be subdivided into 3 groups:

- activities in progress since 1988
- activities which started in 1989
- activities which are being prepared

In the first group extensive use is made of existing nuclear installations and related competences. It includes a contract for increasing the capabilities of the LOBI installation, the delivery of special safeguards equipments, experiments for the incineration of low activity resins and decontamination studies of LWR components.

In the second group a contract was signed which foresees the use of a nuclear installation under construction. Smaller activities were agreed in the development of finite elements codes.

Contracts which are being prepared concern both nuclear and non nuclear activities. It is envisaged to launch larger projects with participation of several

contractors for one particular task in which the multidisciplinary nature of the Institute is valorized.

Associated Laboratories

A European Association of Structural Mechanics Laboratories has been set up whose partners have agreed to enhance the competitive position of the Community's construction industry, improve analysis methods, develop construction standards for structures subject to severe dynamic loading.
A detailed collaborative research program is under development which will make use of existing specialist manpower and high performance testing facilities. The setting-up of a European Association of laboratories, universities and industries for the analysis of chemical processes is being discussed.

Large installations

The Institute is responsible for the construction and operation of all large nuclear installations and has to provide the necessary support i.e. waste treatment and storage, transport of radioactive material, licensing of installations and nuclear cleaning.
Under this heading fall: PETRA, PERLA, ETHEL and FARO, described above.

In addition there are facilities non necessitating a nuclear infrastructure i.e. LOBI, LDTF and the Reaction Wall.

All of them are subject of a detailed planning effort in both the construction and operation phases and the nuclear ones have to respect a quality assurance program.

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1. MAIN ACHIEVEMENTS AND MILESTONES

1.1. SPECIFIC PROGRAMMES

1.1.1. REACTOR SAFETY

THE LOBI PROJECT

Introduction

The general aims of the project are:

- Experimental investigations of the thermohydraulic behaviour of simulated primary and secondary cooling systems of a pressurized water reactor for a range of loss-of-coolant accidents (LOCA) and Special Transients.
- Use of the experimental results for the development and/or improvement of analytical models and the independent assessment of large system codes used in light water reactor safety analysis.

In particular, at the JRC, the activities are focussed on:

- Operation of a large-scale two-loop high pressure, full-power integral system test facility (LOBI) for the loss-of-coolant and special transients experiments which is used for the experimental investigation of the influence of the thermohydraulic behaviour of individual components of a PWR cooling system on the course of LOCA or Special individual components of a PWR cooling system on the course of LOCA or special transient conditions with special emphasis on accident management aspects.
- Comparison of experimental results with pre-test and post-test prediction calculations for the independent assessment and improvement of large system codes used in LWR safety analysis.

Long term objectives have remained during the years essentially unchanged in philosophy, however test aims and sequence for test execution have changed some times in consequence of new acquainted knowledge and may change in future according to new research priorities.

The work programme comprises A-Tests (BMFT Programme) and B-Tests (Community Programme). For the BMFT Programme the definition of these experiments is performed in the "Working Group LOBI-A" with German experts and for the Community Programme in the "Working Group LOBI-B" with experts from all E.C. member countries.

The LOBI test facility is an approximately 1 : 700 scale model of a four-loop 1300 MWe PWR and has two primary loops, the intact loop representing three loops, and the broken loop representing one loop of the reference PWR. Both primary loops are containing a coolant circulation pump and a steam generator. The simulated core consists of an electrically heated 64 rod bundle housed in the pressure vessel model. Nominal heating power is 5.3 MW.

An extensive measurement system and a specially tailored data acquisition system are used to monitor and record the main thermohydraulic parameters prior and during each test. The facility was operated until June 1982 in the MOD1 configuration, providing experimental data for the large break LOCA scenario. The present MOD2 configuration incorporates several improvements with respect to both mechanical components and instrumentation as well as data acquisition system in order to satisfy the requirements of the small break and Special Transients experimental programmes.

The LOBI experimental programme is complemented by an extensive code application and assessment effort. Test results prediction and evaluation are made by the LOBI team with RELAP5/MOD1-EUR and RELAP5/MOD2 which have been converted by LOBI into IBM versions. LOBI calculations by other groups are performed with RELAP, CATHARE, DRUFAN/ATHLET, TRAC and RETRAN.

Achievements

During 1989:

- Recommissioning of LOBI test facility (plant and instrumentation) after modification:
  - new heater rod bundle with improved TC instrumentation in the upper plenum region to trace ECC water;
  - new upper power connecting plate with the possibility of improved bypass behaviour simulation;
- Commissioning of a volume control system;
- Execution of experiments of the LOBI-MOD2 Programme;
- Specifications, pre-test calculations and preparation of scheduled experiments;
- Test analysis, post test calculations, data qualification and documentation of experiments executed;
- Instrumentation service and improvement.

As far as the test results (status October 1989) are concerned:

- Experiment BC-04: Detailed investigation of by-pass behaviour in the upper plenum/upper downcomer region for forward and reverse flow. Data are used for characterizing the facility for code calculations.
- Experiment BL-30: 5 % Cold Leg Break. ECC-Water Injection: 2 of 4 HPIS and Accumulator in cold leg of intact loop. This experiment belongs to a test series on the influence of break size with the experiments BL-00 (0.4 %), A2-81 (1 %) and BL-02 (3 %). Observed phenomena include early decoupling between primary and secondary side, loop seal formation and clear out. No bundle dry-out was observed, heater rod temperatures were close to saturation temperature. (Comparable also to BL-01, same test, however with hot leg injection).

Experiment BL-22: Steam generator tube rupture with three phases: (1) Initial phase (0...420 s) without operator intervention, (2) Cooldown of secondary side intact loop with 100 K/h, maintaining sufficient subcooling in primary system, (3) Reduction of pressure and temperature in primary system by simulated pressurizer spray system.

Post-test analysis of experiment BL-21 (steam generator U-tube mixture) with RELAP5/MOD1-EUR and documentation of results.

Comparative analysis of pre- and post-test predictions for experiment BL-12 (1 % cold leg break, no HPIS), performed with different codes (RELAP5/MOD1-EUR, RELAP5/MOD2, TRAC-PF1, CATHARE and DRUFAN/ATHLET). Results were presented in October 89 at "NURETH-4" at Karlsruhe.

Parametric study with RELAP5/MOD1-EUR for intentional primary system depressurization (accident management), using experiment A1-93 (SBLOCA cold leg, no HPIS). Results were part of the contribution to the "CSNI Specialist Meeting on International Coolant System Depressurization", Garching, June 1989.

FARO

Introduction

FARO is an experimental facility where a number of phenomena related to severe accidents can be investigated using real reactor materials. The main feature is the possibility of melting quantities of 100-150 Kg of UO_2. The melt can then be released into test sections to study:

- fuel jet impingement on structures;
- fuel freezing and plugging in channels;
- coolant jet penetration, fragmentation and in general fuel-coolant interaction problems.

The experimental programme was firstly devoted to the studies of LMFBR severe accident problems in the three test sections BLOKKER I, BLOKKER II, TERMOS.

Since 1988, studies are under way to assess the possibility of using the facility for LWR severe accident studies. A programme is now being defined for the study of invessel melt quenching phenomena, while the LMFBR test programme is now being concluded.

A series of models and computational codes have been developed for the test precalculation and interpretation, in particular the CONDIF and SMURF codes, for predictions of the molten pool behaviour, the codes JET 3D, BOUNDY and MELT for the description of plate perforation tests. For plugging and freezing tests, the French CEA code BUCOGEL was used, while for jet penetration the JENA code has been developed.

Achievements

In 1989 six tests were performed, where the melt has been delivered from the furnace to various test sections. The melt quantities ranged between 105 and 125 Kg.

The main objectives of these experiments have been reached and concern:

1. In the BLOKKER I test section information about the response of a stainless steel plate 40 mm thick and 400°C preheated to a perpendicular impinging molten UO_2 jet (FARO test 45). 90 Kg of a coherent melt jet of initial diameter 22 mm played on the plate for 12 s. 19 Kg of solidified melt remained on the plate. A maximum plate temperature of 1100°C was measured. No ablation occurred thus confirming the result obtained in a similar test performed in 1987 (FARO test 37, [1]), in which 70 Kg of melt played upon the plate for 6 s. Fig. 1 shows a frame of the high speed film of the test which represents the melt jet impinging against the plate. Fig. 1.1 shows the plate after crust removal.
2. UO_2 melt temperature measurements by ultrasonic thermometry (FARO tests 46, 48, 49). Transient measurements were performed by tungsten sensors both in the furnace by introducing a sensor in the pool at the end of the melting and just before releasing the melt, and in the intermediate storage vessel of the BLOKKER series [2]. Melt temperatures ranging from 2950°C to 3100°C have been measured. Fig. 1.2 gives the response of a zone of a probe installed in the intermediate vessel for test 48. Time zero corresponds to the release start. The departure from zero of the trace indicates the arrival of the melt in the vessel. From the stationary portion of the curve it can be deduced a bulk

temperature of 2950°C which corresponds to the initial temperature of the melt for the freezing test reported below.

3. In the BLOKKER II test section the freezing and plugging of molten UO_2 in an array of 7 stainless steel circulator channels (FARO test 48). The tubes were 2 m long, 20 mm thick and their inner diameter ranged from 4 to 6 mm.

   Their initial temperature was 400°C. The melt was released from the intermediate BLOKKER vessel. A preliminary analysis indicates that plugging occurred before material passed through the tubes whatever the diameter. The penetration lengths range between 300 mm (∅ 4 mm) and 845 mm (∅ 6 mm).

4. TERMOS 2. First analysis of the experimental results confirm the general behaviour of 100 - kg scale molten UO_2 / sodium interaction as already observed in the TERMOS 1 test.

As far as the analysis is concerned, BUCOGEL [3] a 1-D computer code which models the freezing of an incompressible multi-component fluid flowing through the interior of a cold structure, developed by CEA has been used to analyse the flow of molten UO_2 falling through the FARO furnace release channel into the TERMOS test section. The length of this channel is 2.5 m and calculations concerning two internal diameters, 50 mm and 80 mm, have been performed. It has also been used to analyse the flow through a modified guide tube to be used in BLOKKER experiments. With particular interest to establish the validity of the code it was used to calculate the formation of blockages in small diameter tubes as realized in the BLOKKER II experiment. Fig. 1.3 shows the crust growth history along the inside of a tube, internal diameter 3.7 mm, whereas Fig. 1.4 shows more clearly how the flow of molten UO_2 will be blocked just after 0.9 s.

Some precalculations have been performed using the 2-D hydrodynamics code SEURBNUK-EURDYN [4] to examine the feasibility of carrying out FCI experiments involving the dropping of molten UO_2 into water contained in a stainless steel vessel. Due to code limitations it was necessary to assume that all the FCI's occur in a well-defined mix-region below the water/cover gas free surface, and that the mix-region is modelled as a homogeneous bubble or gas-bag. During the expansion of the bubble, see Fig. 1.5, pressure waves interact with the vessel and the water accelerates towards the roof, however, the magnitudes of the distortion of the vessel and the roof impact depend strongly on an accurate description of the enrgy source. Due to the lack of this knowledge two simpler energy sources were chosen, an ideal gas expansion and a low density explosive. Roof impact pressures were found to be rather high in both cases, however, it was felt that the final energy partition was unrealistic due to the fact that the bubble is not allowed to break through the surface of the water into the cover gas.

Fig. 1. - Meltjet impinging the plate (from a high speed film)

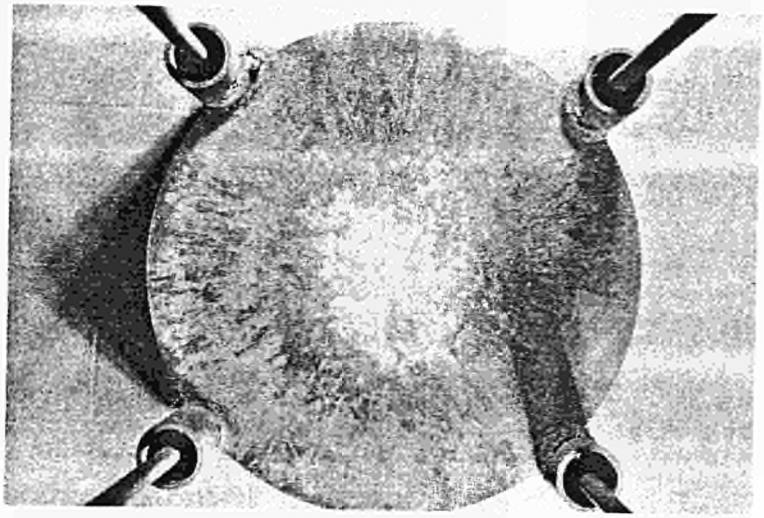


Fig. 1.1 - Plate after crust removal

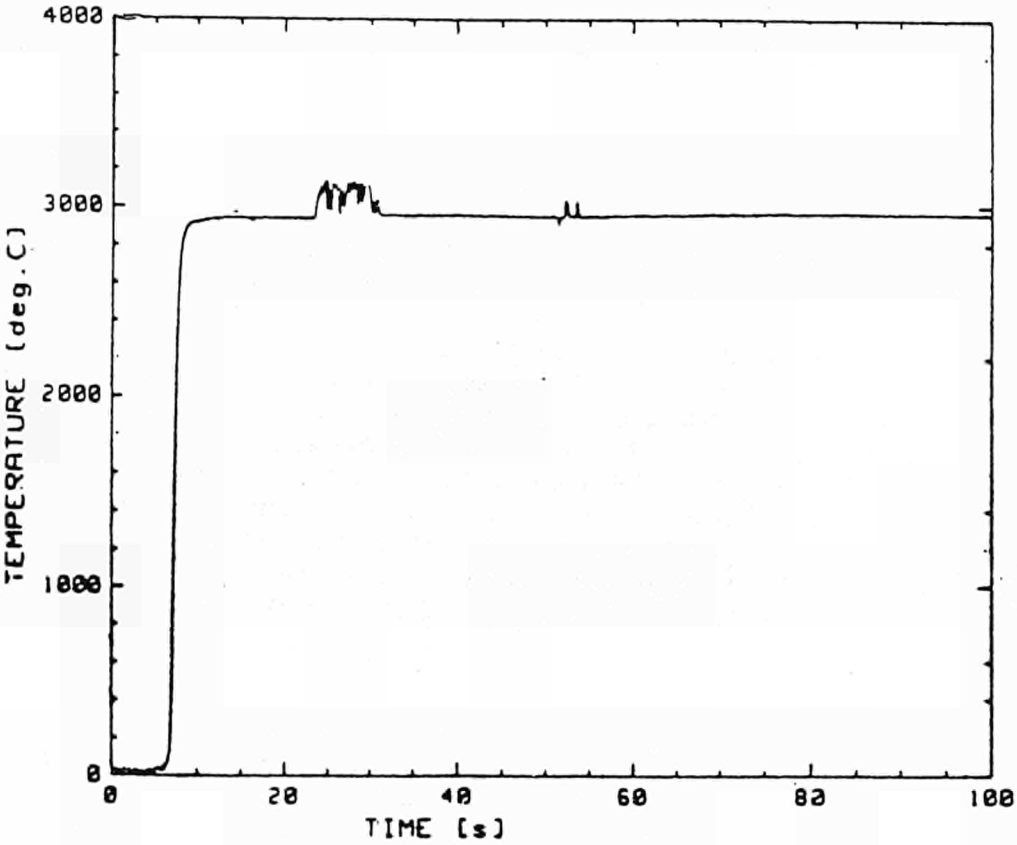


Fig. 1.2 - Example of an ultrasonic sensor reponse (test n. 48)

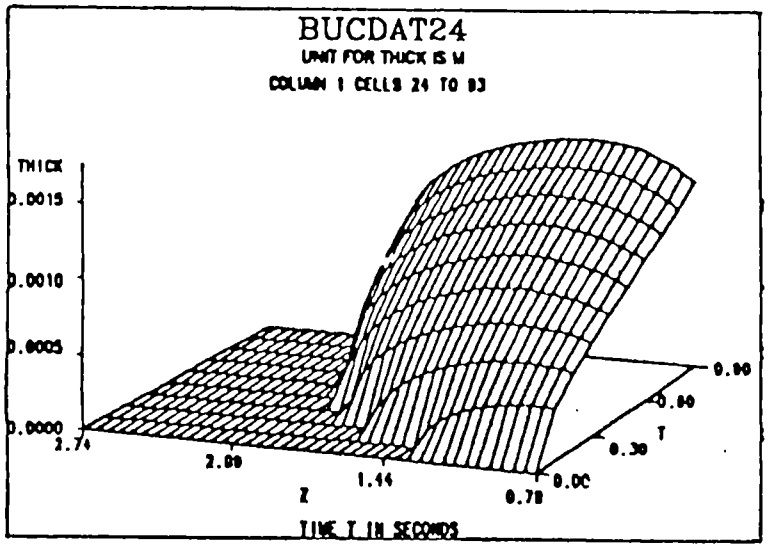


Fig.1.3 BUCOGEL calculation of FARO-BLOKKER II test; . UO2 thickness versus time (t) and axial length (z)

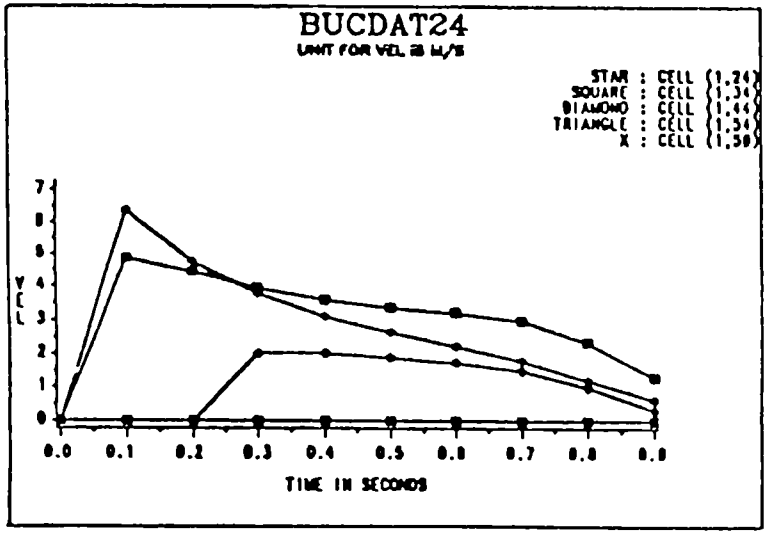


Fig.1.4 BUCOGEL calculation of FARO-BLOKKER II test; UO2 velocities versus time at di. ent positions

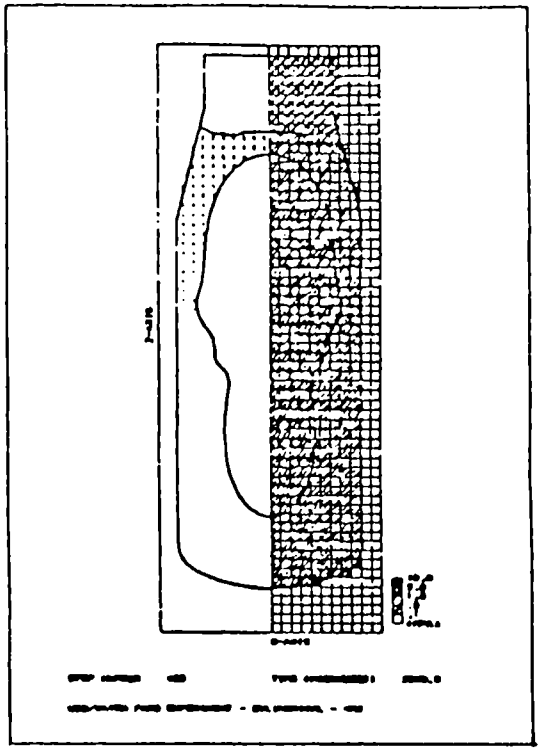


Fig. 1.5 SEURBNUK calculation of UO2/Water interaction: Configuration of the interaction zone at time = 4 ms

SOURCE TERM

Introduction

With the programme 1988-1991 this research area became one of the most important in the Commission Reactor Safety Programme. The objective was to set up a series of well coordinated actions contributing in a significant way to the estimation of the source term, that is the quantity and quality of radioactive release to the environment in case of hypothetical LWR severe accidents.

The most important of these initiatives was to join the french Phebus FP programme with a substantial funding as well as participation in the project and analysis work. A convention CEA-Commission was signed on July 12, 1988: the Phebus Fission Product Project was at that time in its pre-design phase.

In addition, other actions have been started: scoping calculations of Phebus FP tests, code development and validation, development of advanced instrumentation for inpile tests, study of physical and chemical properties of aerosols and FPs released during severe accidents. All these works were or are mainly performed by national laboratories through Shared Cost Actions (SCA).

The JRC contribution during 1989 was mainly concentrated on code implementation and validation and in general on the coordination and analysis of the activities made under contract. In relation to Phebus FP, a JRC team detached at Cadarache assured the participation in the design and development of the facility the analysis group located at Ispra gave substantial contribution in the definition of scientific objectives and test matrix of the programme and in scoping calculations.

Achievements

Phebus FP programme

In the Phebus FP facility, located at the CEA Cadarache Centre, the core, the primary circuit and the containment building of a pressurized water reactor (PWR) are represented by a test fuel bundle, the in-pile section, FP line, experimental system and containment tank. The present Phebus reactor will be modified, receiving a new core cooling system and a new (FP) building.

The components of the Phebus FP facility are designed against the following scenario:

- the test fuel is re-irradiated inside the in-pile section, using the existing LOCA pressurized water loop, during 2 weeks, in order to regenerate a significant inventory of fission products with half-lives between 10 h and 10 d;
- the loop is than blow down, under simultaneous reduction of the reactor power, and the in-pile section isolated from the LOCA LOOP (which is only used to cool the outer structures during the experimental phase);
- for the experimental phase (see fig. 1.6) the inner volume of the in-pile section is swept by a steam-hydrogen flow while reactor power and test fuel temperatures are increased. The resulting test scenario includes all stages of a severe core degradation in a pressurized water reactor accident during which fission products are released and trannsported though FP line and experimental system into the containment tank before being released into the atmosphere tank.

The main design parameters of the test facility were defined and documented.

Milestones of the project in 1989 have been

- beginning of detailed design of the experimental equipment,
- complete revision of civil engineering, due to new seismic criteria,
- breaking ground for the construction site,
- specification and ordering of large components,
- industrial architect selection and working plan.

Instruments and measurements have been examined both against experimental/analytical requirements and for feasibility. Several R & D activities had to be launched to qualify instruments for the anticipated Phebus FP environment (high temperatures, high radiation levels, transient conditions).

A first safety report was compiled in 1989 and discussed with the competent authorities.

Fig. 1.7 shows the overall project planning 1989-1992.

In parallel an important effort has been devoted to the preliminary definition of the Phebus FP test matrix and the basic experimental requirements. This work was based on the results of a number of preparatory analysis and calculations performed by several european teams (Phase A, scoping calculations SCA). The common objective was the identification, from the results of Source Term studies performed on national reactor designs of the main phenomena taking place in LWRs during severe accidents.

From the results of these analysis and calculations a second step (Phase B, scoping calculations SCA) was started by the same teams to assess the capability of the proposed Phebus FP lay-outs to reproduce, during the 5 planned tests, the main phenomena identified in phase A. The results of these works have been analyzed in close collaboration by the JRC-Ispra and CEA-Cadarache teams: a number of important indications came out both for the lay-out dimensioning and for the definition of a preliminary test matrix which was approved by the Phebus FP Committee in September 1989 (Fig.1.8).

Additional support to the Phebus FP project came from the results of the SCA contracts on instrumentation. Following general instrumentation review contracts passed in 1988, two researches started in 1989. The first, carried out at UKAEA-Harwell, deals with a system based on gamma-spectrometry, to differentiate between flowing FPs and deposits on pipe walls. The second one, performed by the Italian Lavoro e Ambiente , deals with the technological upgrading of an on-line detector of aerosol physical properties, providing high frequency of measurements.

Aerosol and FP properties

Two important researches (SCAs) are coming to a conclusion. The first one, at UKAEA-Winfrith, is an extensive research on FP chemistry related to LWR severe accident conditions, for instance on the effect of high temperature, on the chemical compounds which might be generated in the presence of central rod material vapours, and boric acid. The results of these tests are still being analyzed and their application for model development and validation is under consideration.

EUROPEAN BIBLIOTHEQUE

The second one, performed in the NAUA KfK-Karlsruhe facility, is an experimental investigation on the potential contribution to the overall Source Term due to resuspension in the containment atmosphere of aerosols deposited and dissolved in the sump water, in case of sump water boiling during the late stages of an accident. The results have shown that in this case the amount of radioactive material which became airborne might not be negligable.

A new research action on chemistry was started at Winfrith in the FALCON facility. With the objective of making the european studies on FP chemistry as complementary as possible to Phebus FP, a seminar with the participation of 40 european specialists was organized on June 27-28 by UKAEA Winfrith and JRC, where the programmes under way have been presented and discussed. Preliminary studies have also been carried out in collaboration with ENEL-Milan for a possible research programme to be performed at Ispra on aerosol resuspension in the primary circuit. The aim is to provide an extensive and qualified data base on the mechanisms which lead to a resuspension of deposited aerosols in pipes under different thermohydraulic conditons.

Code development and test analysis

In 1988 elements of the Source Term Code Package from the USNRC were implemented on the HRC's AMDAHL computer. Apart from some software problems, the modules available, designed for reactor analysis, required some adaptation to allow calculation of the PHEBUS-FP experiments. In 1989 a calculational sequence has been established for PHEBUS analysis, named CHAIN, combining elements of the STCP and modules developed by the JRC and contractors. CHAIN has been applied to the benchmark problem of the Phase B shared-cost actions and to the calculation of the first Phebus test FPTO.

The boundary conditions assumed in the thermohydraulic module MARCH3 are not appropriate to PHEBUS, and it was decided to replace this module by a combination of the JRC code BUTRAN and the fission product release models (CORSOR) of the STCP.
BUTRAN was modified by the introduction of a simple fuel melting model in which the melting temperature is independent of the oxidation state and molten materials do not relocate. The CORSOR release correlations were then incorporated and the outcome, BUTCOR, was applied to the Phase B benchmark, where it proved adequate in the stages before massive material relocation. Fig. 1 compares the peak fuel temperature calculated by CHAIN and results from other Phase B participants.

Because of its short running time BUTRAN was also applied to a preliminary investigation of the strategy of preoxidation suggested for the PHEBUS-FP experiments. The results indicate that it is difficult to achieve significant partial oxidation along the length of the bundle in a reasonable time without provoking severe oxidation at the hottest parts of the rods i.e. the region close to the core midplane (radial variations in rod temperature are small).

BUTRAN is convenient and fast-running, but cannot handle the material relocation expected in the later stages of a PHEBUS test. The JRC has consequently acquired and installed the CEA bundle degradation code ICARE-2. A comparison has been made between the predictions of ICARE-2 and BUTRAN for that portion of the benchmark FPTO for which they are expected to agree. Results for the peak clad temperature, which brings out the importance of the correlation selected for the kinematics of clad oxidation and of the modelling of convective heat transfer between rod and gas mixture.

In order to perform Phebus calculations, it was also necessary to modify the thermohydraulic module MERGE for the primary circuit. It was necessary to allow the imposition of a constant temperature on the outside of a pipe, of temperature as a function of time on the inside surface of a pipe and of heat flux to the pipe wall as a function of temperature. The modified MERGE was applied to the TMLB' sequence of Phase B and gave satisfactory results. The complete TRAPMELT3 has been operated with input from BUTCOR without problems, and, more recently, an interface has been constructed allowing the code to operate using input data from ICARE-2.

The thermohydraulics of TRAPMELT3 is strictly single-phase, all condensation and reevaporation of steam being neglected. CEA-Cadarache have developed a code coupling the deposition calculations of TRAPMELT with the two-phase thermohydraulics of CATHARE-2 and have made the resulting CATHARE-TRAPMELT available to the JRC. The JRC intends to perform some evaluation studies on the code working remotely on CRAY 2 machine.

MACE of the STCP, which calculates the thermohydraulics of the containment, is unsuitable for PHEBUS, and an alternative fast-running code CONT has been written which allows the imposition of structure surface temperatures and also "natural" heat removal by diffusion into slablike structures, with condensation heat transfer modelled by a modified Uchida correlation.

CONT has been tested in conjunction with the other elements of CHAIN in a preliminary calculation of FPT-0.

The CONTAIN 1.04 code is available to the JRC. It is more sophisticated than CONT (although as used at Ispra it remains zero-dimensional), particularly in its condensation modelling, and has been applied to a study of various design options for the PHEBUS containment. Such options include the type of condenser structure to be inserted in the containment vessel, and the strategy to be adopted for the temperature control of the vessel walls in order to achieve optimum representativity. Relative humidities calculated with CONT and with CONTAIN bring out the importance of condensation modelling for the overall thermohydraulics predictions.

Although most components of CHAIN are compact and fast-running, TRAPMELT is very time-consuming, and has other restrictions as well. Work is underway to improve the graphical output of TRAPMELT, add a restart facility and to speed the code up by the dynamical elimination of unnecessary equations.

As a more general and wider-ranging activity, the development is being launched, coordinated by JRC-Ispra and supported by shared-cost actions. of a European Source Term Suite, which is intended to integrate European models, codes and, it is hoped, expertise in the area of fission product release and transport. The basic structure of the suite attempts to take a modern approach to code integration. Consultation with national experts is taking place with a view to establishing groundrules for the construction of the suite, with due emphasis on the choice of datastructures, quality assurance procedures and the expected distributed computer hardware upon which the completed suite will run.

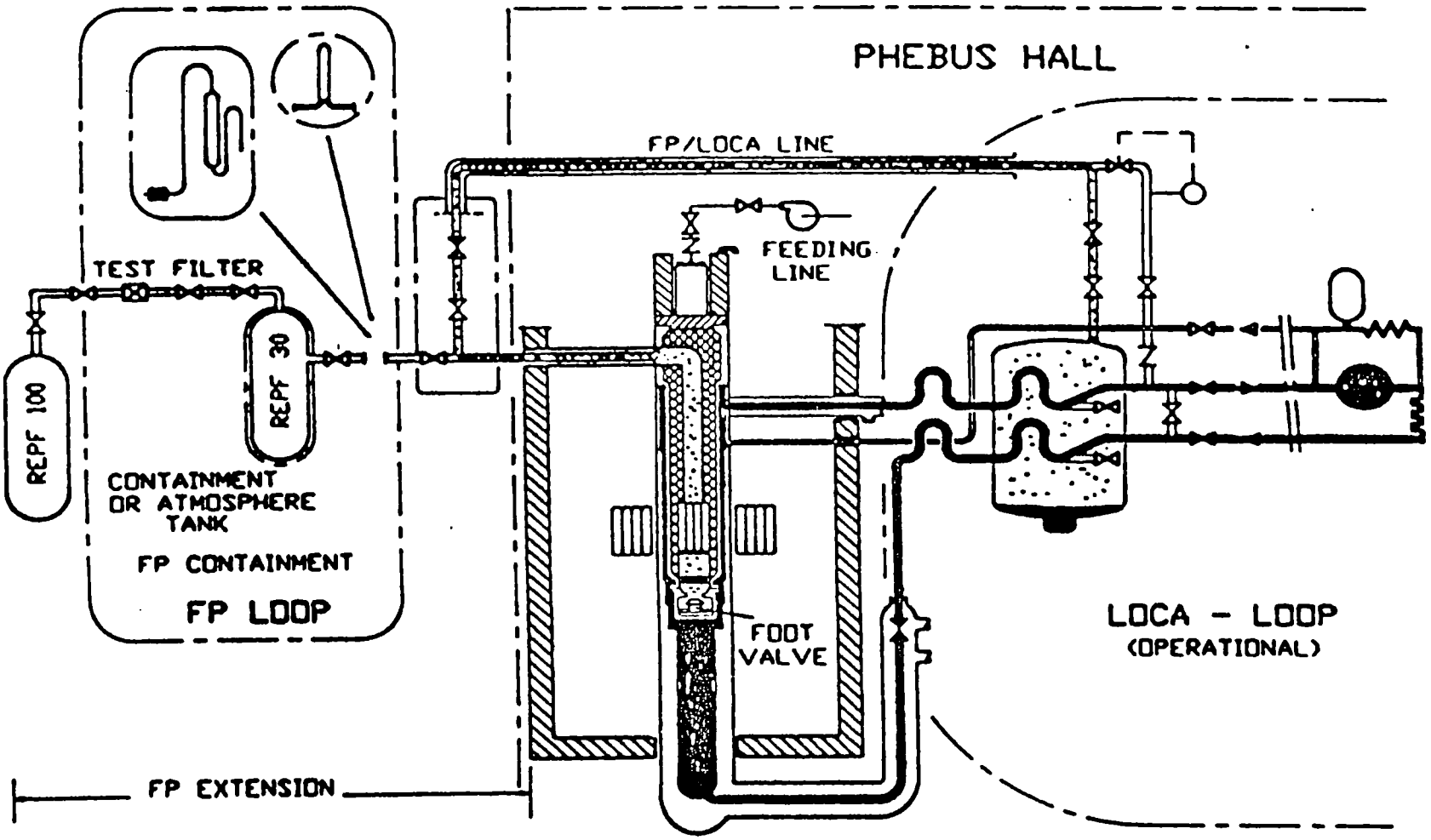


Fig. 1.6 : F.P. Experimental loops
EMPTYING PHASE

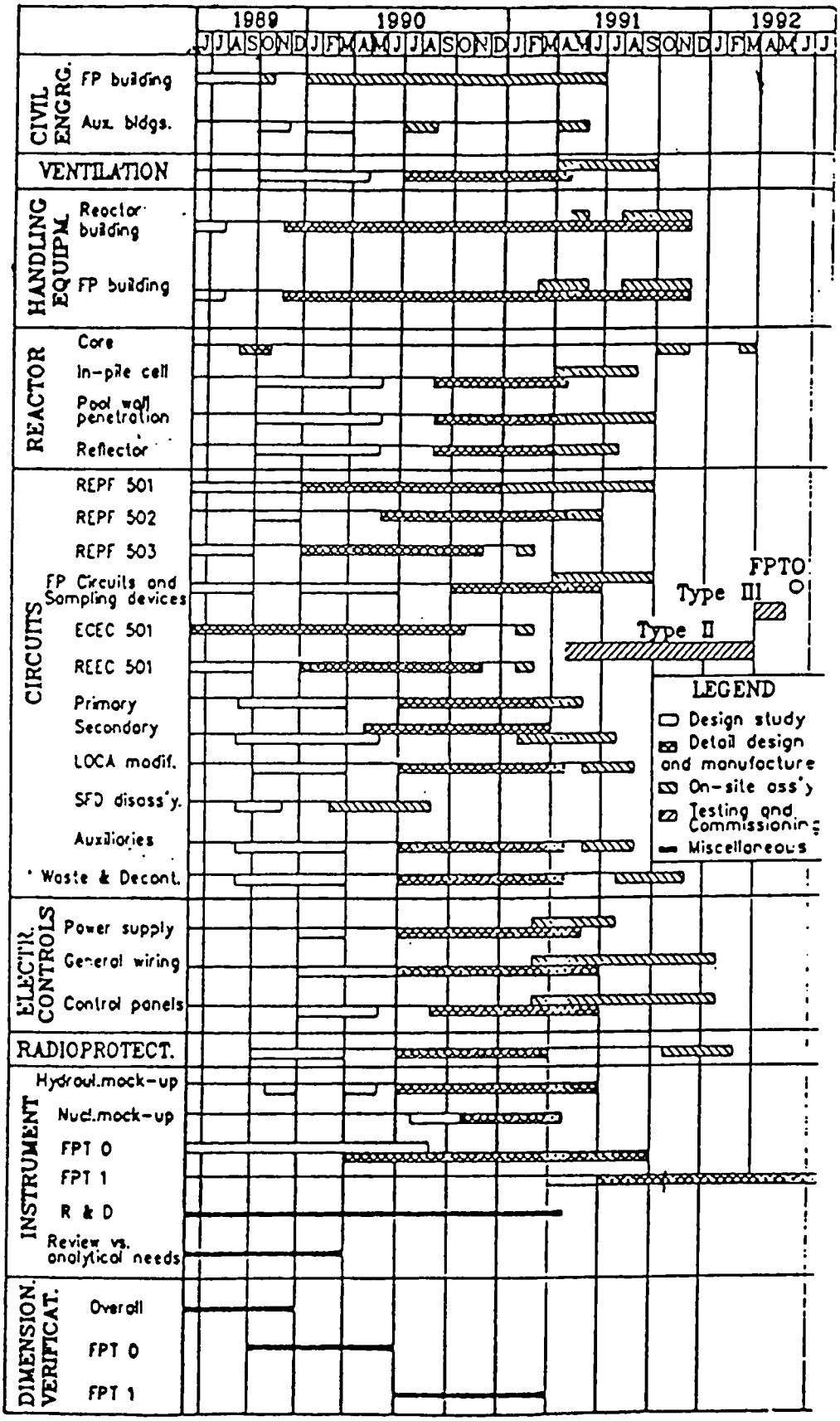


Fig. 1.7 : PHEBUS F.P. Planning Diagram

SIMPLIFIED TEST MATRIX

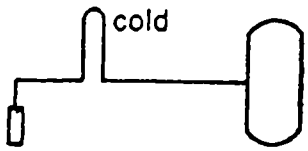
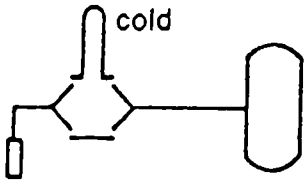
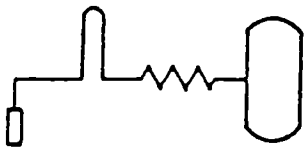
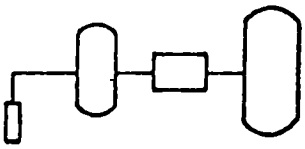
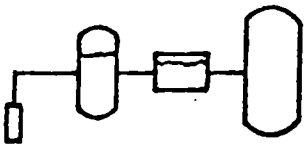
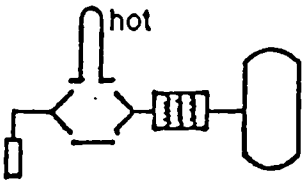
	Circuit config.	Pressure	Carrier gas nature	Other test parameters
FP-T-0		low	oxid.	- fresh fuel - low pre-oxid. - boric acid - control rod - vapour flush - revaporization - MCCI + spray
FP-T-1		low	oxid.	- pre-irrad. fuel - low pre-oxid. - boric acid - control rod - revaporization
FP-T-2		low	red.	- pre-irrad. fuel - high pre-oxid. - boric acid - control rod - vapour flush - MCCI + spray
FP-T-3		high	oxid.	- pre-irrad. fuel - low pre-oxid. - control rod - revaporization. - vapour flush - steam in cont. - MCCI + spray
FP-T-4		high	oxid. then red.	- pre-irrad. fuel - low pre-oxid. - steam in cont.
FP-T-5		interm.	red.	- pre-irrad. fuel - high pre-oxid. - control rod - vapour flush

Fig. 1.8 : PHEBUS F.P.

EUROPEAN ACCIDENT CODE (EAC)

Introduction

The work for the development of the European Accident Code (EAC) started at the JRC several years ago with the aim of creating a flexible informatic system for the prediction of the initiation phase of a hypothetical whole core accident in an LMFBR. The work was from the beginning strongly supported and guided by national organisations represented in different advisory and expert groups of the Commission. In a first period the emphasis was more on the development of an informatic structure where existing national modules could be easily included and compared. Subsequently the JRC effort was particularly spent in improvement of existing advanced modules and development of new ones in order to make the new version EAC-2 a best estimate tool. The objective of the present version EAC-2, which is being completed and released to the Member States, is to become a reference code to be used for comparison in national laboratories and in the longer term a code for licencing calculations.

EAC-2 still requires improvement and further validation work: the effort to be performed by the JRC will be strongly related to those countries still working on LMFBRs.

Achievements

In 1989 a first version of the European Accident Code-2 (EAC-2) [ref.5, 6] for the analysis of low-probability whole-core accidents in LMFBRs was completed. This first version of EAC-2 has advanced modelling in the areas of fuel pin behaviour, molten fuel motion inside the fuel pin and fuel motion in the coolant channels following pin failure. Moreover, the code now simulates properly the hexagonal geometry of an LMFBR and has a new detailed neutronics calculation for this geometry. In particular:

- A new version of the TRANSURANUS pin mechanics code [ref. 7] was developed (by JRC Karlsruhe) where the integral treatment within axial slices instead of a treatment at the node interfaces is included. This new approach does not require small meshes at the interfaces of fuel and blankets and simplifies the coupling with most other modules.
- The coupling of TRANSURANUS with the new version of the fission-gas behaviour module LAKU [ref. 8] has been improved through temporary smoothing of the very nonlinear effect of fission gas induced fuel swelling. The running speed of the new version of LAKU which also includes some new physical modelling is about an order of magnitude higher than the original LAKU and this is important for its use in whole-core accident calculations.
- For the coupling of the Material Dynamics (MDYN) model [ref. 9] to EAC-2 several steps were necessary. First, MDYN had to be coupled to the data management system (FIC). Then the data transfer from the CFEM boiling model to MDYN at pin failure time was made operational. For the dynamic coupling with TRANSURANUS, first the coupling with a separate TRANSURANUS cladding calculation was performed. The final step was the dynamic coupling with TRANSURANUS via the cladding-to-coolant heat fluxes (added over the MDYN time steps during the larger TRANSURANUS time step) and the channel pressure.
- The coupling of the 3D hex-z nodal diffusion and transport code HEXNOD [ref. 10] provides EAC-2 for each calculational channel with the axial power distribution and fuel, sodium, steel and Doppler reactivity worth tables. EAC-2 provides HEXNOD with the masses and temperatures of the different

components. The time-dependent HEXNODYN [ref. 11] code includes HEXNOD for static flux calculations. A good agreement with a time-dependent r-z benchmark problem simulating a rod withdrawal in an LMFBR has been achieved. This is shown in Fig. 1.9 which depicts the results of HEXNODYN together with the results of the CASSANDRE space-time kinetics code the results of which agreed very well with other codes participating in the earlier benchmark exercise. The power rises in this case due to an imposed reactivity ramp. It is first turned around due to the Doppler feedback arising from the fuel heat-up. Since the driving reactivity ramp continues until 15 sec, a second power peak arises and is turned around again by Doppler feedback. Considerable progress has also been made in reducing the computer running time. The case presented required only 5 min on the JRC AMDAHL-300.

- Another effort was the inclusion of the EAC-1 point kinetics model into EAC-2 in order to have also a simplified but well-tested time-dependent neutronics calculation in EAC-2.
- An important part of the work was the testing of the whole code for unprotected Loss-of-Flow and Transient Overpower Accidents. Furthermore validation attempts of the complete EAC-2 with in-pile experiments were made. One was the pre-test analysis of the MOL7C-7 local fault experiment [ref. 12] and another the post-test analysis of the CABRI BI-2 loss-off-flow experiment.
- A group of national experts (EAC Users Group) has been following in very great detail the progress of the work.

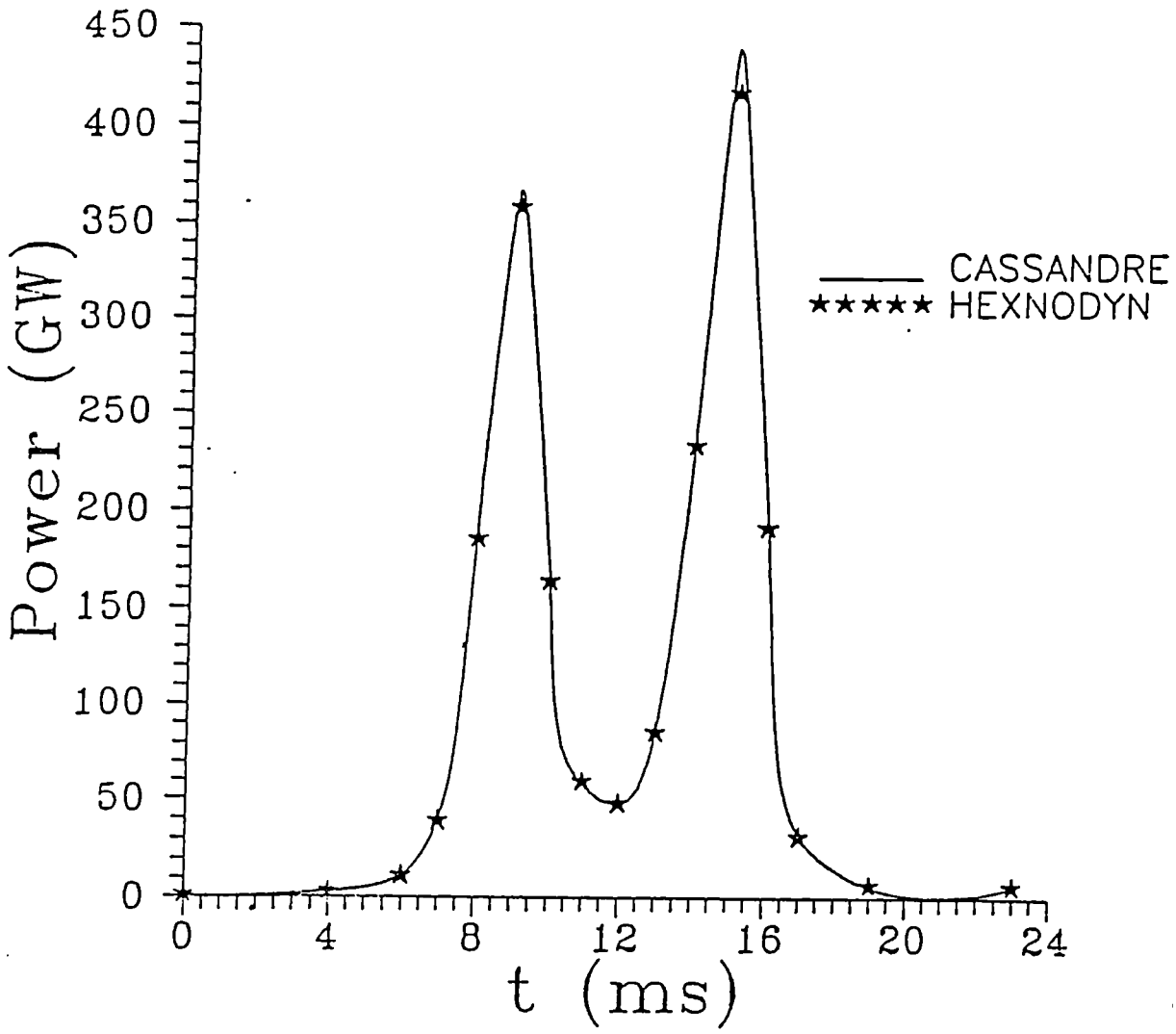


Fig. 1.9 : Comparison of Space-time Kinetics Codes

PISC III

In the framework of the PISC III programme, a microstructural investigation on the cracking of the Muhleberg reactor primary pipe "CH1" was performed by optical microscopy (OM), scanning electron microscopy (SEM) combined with energy dispersive x-ray spectrometry (EDX) and electron probe x-ray microanalysis (EPMA), [13].

The observations can be summarized as follows:

- the circumferential crack, extending more than 300°C and 3 ÷ 6 mm deep, was located in the base metal (AISI 304 steel), near the weld on the pump side, on the inner surface of the pipe, at the sharp edge of the counterborder
- the crack initiated and propagated in the zone which was heat affected by the welding, where fine intergranular precipitation of carbides (likely of chromium) was observed
- the propagation mode of the crack is typically intergranular, with extensive branching of the path
- the cracks are found to be either partly or completely filled with stable corrosion products, namely NiS and mixed oxides which are based on magnetite ($FeO.Fe_2O_3$) and iron chromite ($FeO.Cr_2O_3$), in addition to layers of mixed iron-rich oxides $(Fe, Ni, Cr, Mn)_x O_y$ with traces of Sulphur
- the source of sulphide ions for the precipitation of nickel sulphide consists of manganese sulphide which has been found as inclusions in the steel.

It is known from the literature that intergranular precipitation of chromium carbides sensitises austenitic steels and particularly the AISI 304 stainless steel, to intergranular stress-corrosion cracking (I.G.S.C.C.).

The fact that the crack started to propagate from the sharp edge of the counterborder strongly suggests that stresses were introduced by non appropriate mechanical machining, and/or that residual stresses were present near the seam weld.

Taking into account that crack propagation occurred in the heat-affected zone near the weld, were intergranular precipitation of carbides has been found, the conclusion is drawn that the crack initiated and propagated by stress-corrosion, as a consequence of residual stresses, and weld sensitisation.

The corrosive environment was likely originated or enhanced by sulphide ions, formed by dissolution of manganese sulphide inclusions in the crevices.

The complete intergranular character of the fracture, the diffuse crack branching and the fact that cavities are almost completely filled with stable corrosion products, indicate however that residual stresses were probably released in the first period of fracturing and that corrosion was predominant during the last period.

ATFI PISC PROGRAM

During the first 3 months of 1989 all the activities of the ATFI laboratory were centred on the non-destructive testing of samples made available by Switzerland.

More the 100 radiographs were made with the 2 MeV accelerator on 5 samples.

After this X-ray session all samples underwent extensive ultrasonic testing in the Purist installation.
Finally dye-penetrant tests were performed in all suspected areas of the samples.
A complete report of these tests is available.

The personnel of the ATFI laboratory is also working on Action 7 of the PISC III program entitled Human Factors.
For this action there is the need for a mobile laboratory for testing under extreme circumstances of temperature and humidity.
A specification was written and a call for tenders was made to which 5 firms responded positively.

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1.1.2 WASTE

PETRA

Progress in the construction of this facility is reported in chapter 2.2.2.

ACTINIDE MONITORING

The actinide monitoring activity is subdivided in the following main tasks :

1. construction of a fully automatised waste barrel monitor based on triple correlation analysis for the interpretation models;
2. improvement of the software due to theoretical and experimental work;
3. coordination of the european waste measurement intercomparison exercise;
4. contractual work for third parties;
5. presentation of a waste barrel monitor at various places.

Waste barrel monitor

The mechanical parts and the automatic control system of the barrel movements are completed and a new industrialised time correlation analyser has been tested successfully. Work on the counting chains has still to be completed before starting with in-field tests.

Theoretical and experimental studies

Special investigations were started to investigate the performance of the triple correlation for waste drums of various sizes and densities having more than one Pu source in unknown positions inside the waste drum. First experiments gave very promising results.

Intercomparison exercise

A final report was written on the results of Pu contaminated waste measurements performed with different techniques and at various laboratories as CEA-Cadarache, CEN/SCK-Mol, ENEA-Casaccia, KFK, KFA, UKAEA-Harwell and DOUNREAY and JRC-Ispra.
It turned out that for concrete waste the TCA gave the best results.

Contracts

About 1600 Pu contaminated waste barrels were measured at the fuel fabrication laboratory ENEA-CASACCIA. Both the VDC and TCA technique has been successfully applied.

Exhibition

The 220 l waste barrel monitor has been presented at the European Parliament at Strasbourg and at the Hannover Fair during 1988. The same technique has been presented at the First Inventors Fair EURISK 1989 at Paris where the CEC's exhibition won the first price

1.1.3 SAFEGUARDS

Development of non destructive essay methods for Safeguards

In the framework of the Safeguards and Fissile Materials Management programme exist the following main tasks:

1. Development of instruments and interpretation models and their tests in PERLA.

2. Operation of PrePerla and construction of PERLA.

1. Development of NDA Instruments and Interpretation Models.

   The following NDA techniques are being developed and tested . Most of the instruments are linked to personal computers based on MS DOS.

   a) Pu isotopic composition determination by high resolution gamma spectroscopy (Pu meter)
   The accurate knowledge of the isotopic composition of Pu material in different chemical and physical compositions is very important especially in connection with the Pu determination using neutron correlation measurements and calorimetry.
   For this reason several gamma spectrum unfolding codes developed by different laboratories in EC and US are under test in order to find out for the various applications the best tool. The tests are performed with different samples using the large variety of PERLA standards. The experimental programme has been terminated and a technical report comparing the performance of the various codes is in elaboration.

   b) Pu-mass determination by a determination of the spontaneous fission rate via neutron correlation counting techniques (Shift register, Correlation Analyzer, Fast time of flight multiplet analysis).
   The shift register instruments were equipped with personal computers operating with MS DOS. Software consists of a data acquisition, data elaboration and data interpretation system. The latter uses the JRC models for neutron multiplication and deadtime corrections of the measured singlet and doublets. First measurements were started in the PERLA laboratory to improve the interpretation models for dead time and other perturbation effects.

c) Pu-mass determination by calorimetry
A water bath calorimeter for the mass determination of Pu is being set up in the PERLA Laboratory and tests are being carried out with Pu samples having a heat output of up to 40 watts. A typical measurement has a duration of about 8 hours and accuracies in the order of a few % were achieved. Tests with a newly designed 40 W air calorimeter, designed by Imperial College (London) have now started.

d) U-mass determination by active neutron interrogation with a Sb-Be (gamma-n) source (PHONID-instruments) counting prompt neutrons
A compact version of the PHONID has been developed and the laboratory prototype has been constructed. Tests in PERLA are now being planned.

e) Statistical Modelling
Statistical modelling of the measurement errors in NDA has become a fundamental activity for the performance assessment of NDA methods. Presently error modelling has been completed for Pu isotopic measurement by gamma spectrometry measurements and PHONID neutron interrogation techniques.

2. Operation of PrePerla and Construction of PERLA

The PrePerla laboratory is now routinely operated and U and Pu standards representing important parts of the fuel cycle are now handled in view of performance tests and calibration exercises. The PERLA standards presently available are :

- a large number of homogeneous PuO_2 powder batches, MOX powders and pellets for low and high burn-up;

- a number of MOX fuel pins, resulting from normal production;

- samples with high and intermediate enriched U (powder, U spheres, metal plates, fuel elements).

Procurement schemes for low enriched uranium samples (powders, pellets, pins, short assemblies) have been prepared and characterisation of samples are expected to start in 1990.

PrePerla is used extensively for intercomparison exercises for PU isotopic measurements by gamma-spectrometry, Pu assey by calorimetry and neutron correlation techniques.

The construction of main PERLA calibration laboratory has continued and civil works have been completed. The commitment of resources has been made for the completion of the laboratory, expected for end 1990.

1.1.4. FUSION

As outlined in chapter 2.2. below a major item in the JRC Fusion programme is the construction and operation of the European Tritium Handling Experimental Laboratory (ETHEL). The activities described hereafter as well as the one outlined in chapter 2.2. are almost all designed to prepare the work in ETHEL.

TRITIATED WASTE MANAGEMENT

The assessment studies aimed at defining specific procedures and packages to be routinely applied for managing tritiated wastes in ETHEL has been continued during 1989.

They have been performed in close connection with the detailed design of the ETHEL Waste Conditioning Plant under development by the main Contractor (NNC, UK).

The design of this area had to be drastically revised in order to satisfy the enhanced requirements put forward by the Safety Authorities.

As a preliminary issue of these studies processing and packaging procedures envisaged for routine application to ETHEL tritiated wastes have been established.

Future experiments in ETHEL for verifying and improving waste conditioning procedures are being defined. To this end characterization tests on some water absorbent media have been started.

Tritium recovery from liquid breeding blanket

The recovery of tritium from the breeder blanket concerns closely the design of a fusion reactor. The utilization of the Pb17Li eutectic as breeding material for a D-T fusion reactor is limited by problems related to the tritium permeation from the blanket to the water-cooling system and to the environment.
In order to limit the tritium losses to acceptable levels, it is necessary to maintain a very low tritium inventory in the blanket.

Theoretical studies have shown that engineering process units usually applied in chemical industry such as the droplet spray extractors, the bubble columns and even a direct degassing process from stirred metal baths of Pb17Li eutectic might be suitable.
In order to control the validity of the applied parameters, the determination of the H/D desorption kinetics data from the Pb17Li and of the H/D mass-transfer rates through metallic walls wetted by the eutectic has been carried out in a pilot plant.
The experimental results have shown a hydrogen permeation rate which is much lower than that so far considered in the feasibility study.
The first results for the kinetics of hydrogen isotopic desorption, have been obtained for the deuterium removal from the Pb17Li eutetic by helium gas (fig. 4.1).

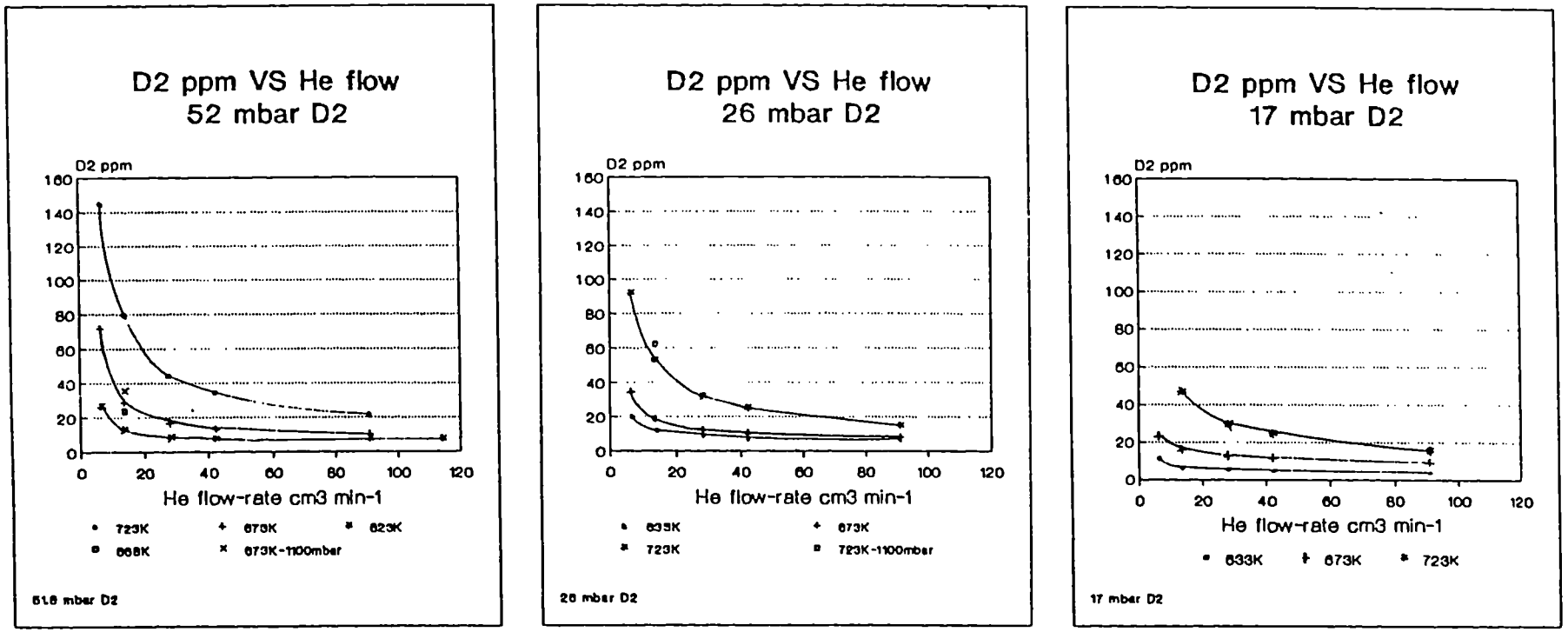

Fig. 4.1 : DEUTERIUM DESORPTION FROM THE Pb-17Li ALLOY IN THE DESORBER AT DIFFERENT TEMPERATURES AND He FLOW-RATES

In the course of this experimental work we have noted that there is:

a) Sensible variation of the D2 concentration in He gas at the lowest helium flow-rates. Above a ratio of 0.06 between the He gas and the Pb17Li eutectic flow-rate in the desorber, the D2 concentration decreases linearly indicating that the tritium recovery from the alloy is proportional to the helium flow-rate.

b) The D2 concentration increases by increasing the temperature and the deuterium partial pressure in the alloy, but not proportionally. The increase is sensitive for temperatures above 400°C while less affected by the D2 partial pressure.

c) The D2 desorption rate depends on the pressure in the desorber, as expected. Above 10% increase was observed when the pressure was increased from 1050 to 110 mbar at different temperatures and D2 partial pressures in the alloy.

From these results it seems that the overall process of deuterium desorption from the eutectic is represented by diffusion of solute deuterium atoms in Pb17Li eutectic and the heterogeneous reaction at the gas-eutectic interface with the deuterium atoms recombination.
This is in agreement with the rate-limiting steps obtained in hydrogen absorption process.

HYDROGEN ISOTOPES EXTRACTION, PURIFICATION, SEPARATION

The gas-solid separation processes, widely adopted industrially, can be utilized to resolve the problems of the Tritium Fuel Cycle for D-T fusion reactors under condition that suitable adsorbent materials are found.

A research was started at the JRC-Ispra Establishment in order to solve this problem.

The main items of this study are the following :

- Hydrogen isotopes separation.
- Argon purification from low Z impurities.
- Tritium extraction from large He flow.
- Fuel clean-up system.
- D_2O/H_2O separation.

Particular emphasis was put on the substrates of the mordenite family which, according to experimental results, appear to be a very good material for the purification of H_2 isotopes and their separation as well as for other purposes related to tritiated gaseous streams treatment.

Hydrogen isotopes separation

An experimental work has been carried out for the characterization of the Ba-, Ca-, Ni- mordenite as adsorbent material by gas-chromatographic techniques. From the experimental data obtained it results that a good separation of hydrogen mixture can be achieved at temperature of 160-170 K much better than

those carried out at 20-120 K by other authors using molecular sieves considered until now the more suitable material for this purpose.

In Tab.1 are reported as an example the retention times (t_{H2}, t_{D2} and t_{HD}),the separation factors (S_F) and the resolution factors an Na-, Ca-, Ni- mordenites for two different hydrogen mixture (H_2/O_2 = 50/50 and $H_2/HD/D_2$ = 25/50/25) under different experimental conditions.

To design the separation units at higher scale adsorption equilibrium constants and mass transfer coefficients have been obtained (Tab.2) by theory of the moments. The construction of a pilot plant based on Pressure Swing Parametric Pumping (PSPP) process for the separation of hydrogen isotopes mixture is progressing.

TABLE 1

RETENTION TIME (t), SEPARATION FACTORS (S_F) AND RESOLUTION FACTORS (R_F) OF TWO DIFFERENT HYDROGEN ISOTOPES MIXTURES UNDER DIFFERENT EXPERIMENTAL CONDITIONS

Na-Mordenite 80 : 100 mesh.

Q (cc/min)	T = 166.8 K				T = 161.6 K				T = 155.9 K				T = 150.4 K			
	t_{H_2} (min)	t_{D_2} (min)	S_F	D_F* (%)	t_{H_2} (min)	t_{D_2} (min)	S_F	D_F* (%)	t_{H_2} (min)	t_{D_2} (min)	S_F	R_F (%)	t_{H_2} (min)	t_{D_2} (min)	S_F	R_F
20	7.41	8.94	1.21	92.48	9.32	11.48	1.23	98.41	11.78	14.85	1.26	1.85	16.65	21.31	1.28	2
30	6.04	7.3	1.21	95.11	7.32	9	1.23	98.31	9.89	11.67	1.26	1.86	12.11	15.62	1.28	2.12
40	5.19	6.25	1.21	94.59	6.18	7.61	1.23	96.77	7.83	9.88	1.26	1.86	9.99	12.82	1.28	2.15
50	4.58	5.53	1.21	96.24	5.45	6.70	1.23	97.65	7.18	8.94	1.26	1.73	8.81	11.27	1.28	1.99

Ca-Mordenite 60 : 80 mesh.

Q (cc/min)	T = 189.4 K							T = 182 K							T = 175 K							T = 168.1 K						
	t_{N_2} (min)	t_{ND} (min)	t_{D_2} (min)	ND/N_2 S_F	ND/N_2 D_F (%)	D_2/ND S_F	D_2/ND * D_F (%)	t_{N_2} (min)	t_{ND} (min)	t_{D_2} (min)	ND/N_2 S_F	ND/N_2 D_F (%)	D_2/ND S_F	D_2/ND * D_F (%)	t_{N_2} (min)	t_{ND} (min)	t_{D_2} (min)	ND/N_2 S_F	ND/N_2 R_F	D_2/ND S_F	D_2/ND * D_F (%)	t_{N_2} (min)	t_{ND} (min)	t_{D_2} (min)	ND/N_2 S_F	ND/N_2 R_F	D_2/ND S_F	D_2/ND R_F
20	7.51	8.51	9.90	1.13	90.85	1.16	73.30	10.51	12.10	14.44	1.15	94.85	1.19	86.24	15.74	18.38	22.52	1.17	1.40	1.23	98.22	23.65	28.17	35.47	1.19	1.55	1.26	2.11
30	5.72	6.47	7.56	1.13	86.58	1.16	69.40	8.15	9.37	11.18	1.15	96.91	1.19	87.82	11.96	13.96	17.11	1.17	1.40	1.23	96.14	17.87	21.28	26.71	1.19	1.68	1.26	2.04
40	4.80	5.43	6.32	1.13	94.09	1.16	79.13	6.87	7.89	9.42	1.15	96.91	1.19	88.49	9.97	11.67	14.30	1.17	1.35	1.23	97.99	14.88	17.65	22.18	1.19	1.31	1.26	1.87
50	4.14	4.67	5.45	1.13	96.93	1.16	80.25	6.03	6.90	8.28	1.15	96.07	1.19	90.25	8.60	10.05	12.31	1.17	1.28	1.23	91.20	13.08	15.65	19.70	1.19	1.54	1.26	2.12

Ni-Mordenite 80 : 100 mesh.

Q (cc/min)	T = 213.1 K				T = 205.7 K				T = 189.5 K				T = 193.2 K			
	t_{H_2} (min)	t_{D_2} (min)	S_F	D_F* (%)	t_{H_2} (min)	t_{D_2} (min)	S_F	D_F* (%)	t_{H_2} (min)	t_{D_2} (min)	S_F	D_F* (%)	t_{H_2} (min)	t_{D_2} (min)	S_F	R_F
20	3.11	3.89	1.25	89.67	3.92	9.11	1.30	94.79	5.16	7.0	1.36	97.48	6.84	9.71	1.42	1.15
30	2.47	3.08	1.25	91.28	3.16	4.09	1.30	94.31	4.03	5.46	1.36	96.11	5.32	7.56	1.42	1.16
40	2.11	2.62	1.25	92.11	2.65	3.44	1.30	96.13	3.45	4.69	1.36	95.44	4.56	6.49	1.42	1.17
50	1.91	2.37	1.25	94.81	2.42	3.14	1.30	96.74	3.03	4.08	1.36	95.59	3.96	9.63	1.42	1.13

* D_F is the degree of resolution, where not reported D_F is equal to 100%

Table 2

Adsorption equilibrium constants, K, for H_2 and D_2 on Na-, Ca- and Ni- mordenite at 170 K

Component	Substrate	Adsorption equilibrium constant (cm 3/g)	Heat of adsorption (KJ/mol)
H_2	Na - M	17.0	8.31
	Ca - M	49.3	12.47
	Ni - M	132.2	16.22

Table 3

Values of axial dispersion and diffusion coefficient in macropores for two different adsorbent materials of 40 - 60 megh at 170 K

Component	Substrate	Axial dispersion coefficient (cm 2/5)	Diffusion coefficient in macroporus (cm 2/5)
H_2	Na - M	0.33	9.36×10^{-3}
	Ni - M	0.63	1.60×10^{-3}
D_2	Na - M	0.28	6.17×10^{-3}
	Ni - M	0.59	1.44×10^{-3}

Argon purification from low Z impurities by using modified zeolites

The goal of this research is to prepare a substrate which strongly adsorbs the low impurities as 0_2, N_2, CO, C_{H4}, $C0_2$ at temperatures as low as 170 K or less while the Argon and the $_{H2}$ isotopes pass through. This in the objective to purify the Argon used for the glove boxes. The second step will be the Argon purification from the traces of $_{H2}$ isotopes. Known techniques will then be applied. Many substrates of the mordenite family have been studies. The Ca-Na mordenite of which the channel diameters have been modified by using 4% of boric acid has given the best result as only the low Z impurities are strongly adsorbed at 160 K. The research is under progress in order to well characterize the substrate and determine the adsorption isotherms and the capacities for any component. Of course further attempts will be made by using other procedures of modification.

Tritium extraction from large He flow

The H_2, D_2, capacity of Na, Ca, Ni mordenite have been determined at 77 K and at other temperatures. The most promising material seems, actually, the Na mordenite that adsorbs about 80 cm^3 of H_2 per gram. A systematic program of work has been prepared in order to define for several promising substrates their strenght and adsorption capacity as well as the adsorption and desorption rates that appear already so different to allow a good separation among the H_2 isotopes by using the gas-chromatography displacement technique. An appropriate apparatus for this type of research is in phase of construction.

Fuel Clean-up System (F.C.S.)

The types of material for the successive traps to be used for the purification of the H_2 isotope from the low Z impurities, as 0_2, N_2, $_{CH2}$, $C0_2$ have been selected and only a better characterisation of the substrate is required. At present, the probable use of the Ni-Mordenite as the substrate for H_2 isotope separation at higher temperature than that foreseen if Na- or Ca- Mordenite were used, requires further determination of the isotherms and the kinetics of adsorption of any component in particular way the oxygen, on the Ni-Mordenite so as to verify the best temperature for a strong adsorption of the oxygen and a good separation of the H_2 isotope. If not, the Ca- or the Na- Mordenite will be used for a preliminary purification and separation of the H_2 isotopes followed by a complementary separation by a Ni-Mordenite column at higher temperature (-190 K).

Calculation of transport of hydrogen isotopes and helium in fusion reactor materials

Inventory, permeation and recycling of hydrogen isotopes and helium in fusion reactor materials have been calculated by a numerical code.

a) Liquid breeder blankets

The first blanket which has been studied is the water-cooled liquid breeder blanket concept of NET using Pb17Li, lithium of Flibe as a liquid breeder material. The time dependence of tritium inventory and of tritium permeation into the coolant or in the first wall boxes have been calculated. The influences of coatings at various surfaces of the structure material on tritium inventory and permeation have been studied.

The most important results are: a) The tritium inventory is low in a Li blanket, is relatively high in a Pb17Li blanket and relatively low in a Flibe blanket. b)Tritium permeation into the coolant is practically zero in a Li blanket, relatively high in a Pb17Li blanket and very high in a Flibe blanket.

The main conclusions are, if only tritium inventory and permeation into the coolant are considered: a) Li is most suitable and Flibe is not suitable as a liquid breeder material in a separately cooled blanket. b) Permeation into the coolant has to be reduced by permeation barriers in separately cooled Pb17Li blankets.

Another liquid breeder blanket which has been investigated is a self cooled blanket using the permeation of tritium from the blanket through the first wall into the torus chamber as tritium refuelling method. The calculations have been performed for a blanket with Pb17Li or Flibe as a breeder and V of Fe as a first wall material. It was found, that self cooled blankets with V as a first wall material are more suitable than those with Fe and that Pb17Li and Flibe are suitable breeder materials in this blankets.

b) First wall

The transport of hydrogen isotopes and helium has been studied in the first wall of fusion reactors. The influence of the presence of up to 10% of helium, on the transport of deuterium and tritium in the first wall of NET have been calculated. Qualitatively it was found that an increasing amount of helium reduces the recycling of hydrogen isotopes to the inner surface and increases the permeation to the outer surface and the inventory in the first wall.

Recycling of hydrogen and deuterium from first wall materials

An apparatus has been developed to study the recycling and outgassing of protium and deuterium from first wall materials under fusion reactor conditions. This apparatus is the test bed for a near future experiment in ETHEL, where permeation, recycling and outgassing of tritium and deuterium-tritium mixtures from first wall materials at very high fluxes and low energies will be studied.

Experimental studies of hydrogen recycling from AISI 316L surfaces under NET-conditions gave a characteristic recycling time of about 1 second, this is about 5 times longer than a theoretical value calculated by a numerical code for NET first wall with a temperature of 543 K.

Hydrogen interaction with fusion-relevant materials

The main goal of this activity is to study the interaction of gaseous hydrogen with fusion-relevant materials. The materials currently studied are TiC coatings on a metal substrate and TZM.

Several runs with bare TZM have been performed for temperatures in the range 400-600°C and with a loading pressures P_0 between 10^3 and 10^5 Pa. From a first processing of these data it turns out that hydrogen solubility S in TZM is very low, but higher than for Mo, the ratio $S/\sqrt{P_0}$ being in the range $5\text{-}7.10^{-3}$ mol/ (mol.$\sqrt{}$Pa). Diffusivity values of 2.10^{-11} $m^2.s^{-1}$ at 400°C and of 6.10^{-10} $m^2.s^{-1}$ at 600°C have been obtained for the highest pressure.

Some preparatory work for sample characterization by surface analysis techniques (AES and XPS) has been carried out.

Several samples of TiC_x and TiN_y with known C/Ti and N/Ti ratios x and y have been obtained and will be used as standards for surface analysis in AES and XPS work with our TiC coatings.

Modelling of hydrogen release due to diffusion and surface processes has been started.

INSTRUMENTATION IN SUPPORT OF THE FUSION PROGRAMME

Particle diagnostics

Some experimental work and feasibility studies are performed in support of the IGNITOR project. It is intended to measure the total neutron yield by classical techniques using fission chambers, BF_3, He-3 and He-4 counters. For thermal neutron detectors the counters will be surrounded by suitable moderator assemblies. Eventually solid state detectors will be used to discriminate between neutrons originating from (p,d) and (d,t) reactions. The neutron emission rate as function of time could be studied with the same detectors. For the study of the neutron emission profile two types of collimators are proposed. For the measurement of the neutron spectrum a telescope (n,p)-spectrometer and a telescope Li^6 (n,t)-spectrometer are considered.

Cold fusion experiments

Immediately after the communication of a "cold fusion ", the JRC Ispra has set up two experiments with electrochemical cells.
The cathode was a palladium tube with a diameter of 10 mm, a length of 50 mm and a thickness of 0,8 mm. For the anode a platinum ring electrode was used. The heavy water had an isotopic title of 99.7 Li 0D, 0.1 mm was used as electrolyte. In all experiments a constant voltage of 5 VDC was applied.
In the first experiment neutrons were counted using a scintillator detector having an estimated detection efficiency of about 5%. In the second experiment a cylindrical array of 18 He detectors imbedded in polyethylene moderator was used, with a detection probability of about 25% and a background rate of about

2 counts/s. In both cases no statistically significant neutron emission could be detected.
Other experiments are continued to repeat the measurements of Scaramuzzi. During the execution of the different experiments it became clear that the normal neutron counting equipment used for fissile material management purpose is not sufficient for low level neutron counting if counting rates close to cosmic radiation have to be measured. For this purpose a special neutron detector head is in construction, which equally serves for the measurement of low level Pu contaminated waste.

FUSION BLANKET SAFETY

One current concept in fusion blanket module design of NET is to utilize water as coolant and liquid lithium-lead as the breeding/neutron multiplier material. In case of heat exchanger tube rupture, water can contact liquid lithium-lead which can lead to a combined thermal and chemical reaction with intolerable high pressurization of the blanket module.

During the reporting period, two large scale tests (470 kg. $Pb_{17}Li$) have been performed in the BLAST blanket module test facility which is of 1/4 of the real design. The experiments simulated the rupture of a steam generator tube under 60 bar internal water pressure at 50°C subcooling and melt temperatures corresponding to the real design values. The effect of the presence of a heat exchanger bundle on the energetics of the melt/water reaction was simulated in the second test by a tube bank in the rupture zone.

Although mixing between the melt and the water intensified in the second test, similar results were obtained.

The results can be summarized as follows :

- The chemical reaction is self-limiting due to the hydrogen and $LiOH/Li_2O$ generation. Steam explosions are unlikely.
- The experiments confirmed that the pressure in the reaction vessel does not exceed the actual pressure in the heat exchanger.
- The temperature in the reaction zone of the test with the tube bank is significantly higher: 650°C against 450°C.

The characteristic pressure history of the reaction vessel of 0.575 m height and 0.330 m average diameter is shown in Fig. 4.2.

Both tests have been successfully precalculated with the SABA code, developed at the JRC Ispra

Independed tests of the behaviour of potentially radioactive species expected to be present in lead-lithium alloy, when used as breeder, were performed with the BLAST facility in collaboration with the Westinghouse Hanford Company (USA). Both tests were successful. The results will be presented at the 13th symposium on Fusion Engineering in Knoxville, Tennessee, October 1989

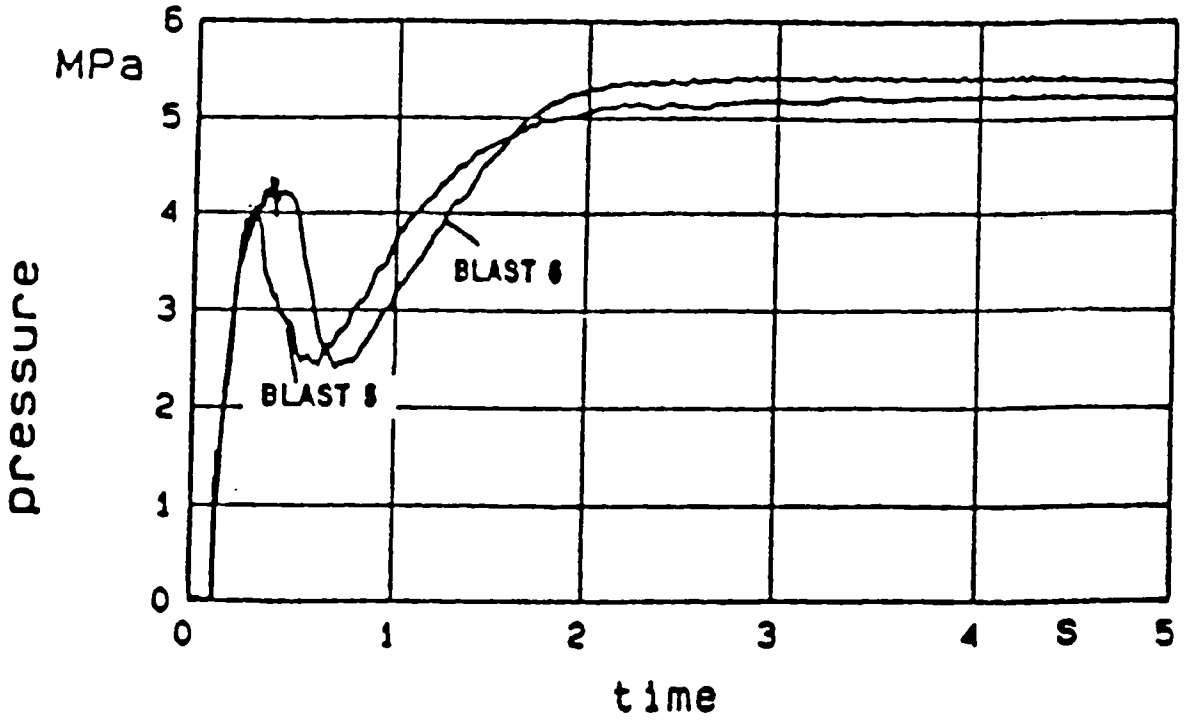

Fig. 4.2 : Pressure history of the BLAST reaction vessel

FUSION SAFETY ANALYSIS

The contribution of the Institute to Fusion Safety Analysis during 1989, was related to the following items:

- Development of the computer code THERM
- Safety analyses for NET and ITER

THERM is a general purpose, bidimensional, nonlinear program for both steady state and transient thermal analyses.

Main features of THERM are:

.Finite element technique
.Bidimensional geometry
.Nonlinear (Material and Radiation)
.Radiation in internal cavities
.Driven thermohydraulic analysis option
.Steady state and transient analyses

The program includes many iterative methods for reaching a good convergence even in difficult cases [1].

The program is strictly linked to CASTEM-2000 and uses its pre- and post-processing capabilities. The program has reached an advanced stage of development and has been applied to industrial cases.

The contribution to Fusion Safety during 1989 was essentially related to thermal and thermomechanical analyses of NET and ITER in cases of Loss Of Flow Accidents (LOFA) and of Loss Of Coolant Accidents (LOCA). Both system and component analyses have been performed.

The LOCA analysis of NET showed that the effects of residual heat can in general be controlled, except in the case of a generalized LOCA involving the whole reactor. The most common case of a generalized LOFA in the whole reactor (Case of electrical black out) has been studies and it has been shown that no major hazard occurs if the cooling circuits are designed to allow cooling by natural convection. Moreover, a LOFA limited to first wall/divertor circuit has been analysed. It was shown that an appropriate design of the inertia of the pumping system can avoid water boiling (due to heat stored in the graphite coverage) in the short term transient, whereas natural cooling can control the long term transient (due to decay heat) [2,3].

A tridimensional thermomechanical analysis of the JRC-Ispra proposed divertor plate for NET has been performed also [4]. It was shown that this component could work safely only below a surface heat flux of about 7 MW/m2 and for about 20,000 cycles.

A preliminary LOCA analysis of the ITER divertor plate has been performed [5] and presented to the ITER design team in Garching.

The following two geometries were considered:

a) Radiative gap between graphite and stainless steel

b) Thermal contact between graphite and stainless steel

The following three load cases have been considered:

-1) LOCA and instantaneous plasma shut-down
-2) LOCA and linear (5 s.) plasma shut-down
-3) LOCA and full power (5 s.) followed by linear decay (5 s.) plasma shut-down

A synthetic presentation of the thermal field obtained for case a-3 is shown in Fig. 4.3; the analysis was performed with THERM and the results processed by CASTEM-2000.

The ITER design team has asked for the continuation of this activity, its extention to LOFA analysis and for studies related to the first wall and breeding blanket also.

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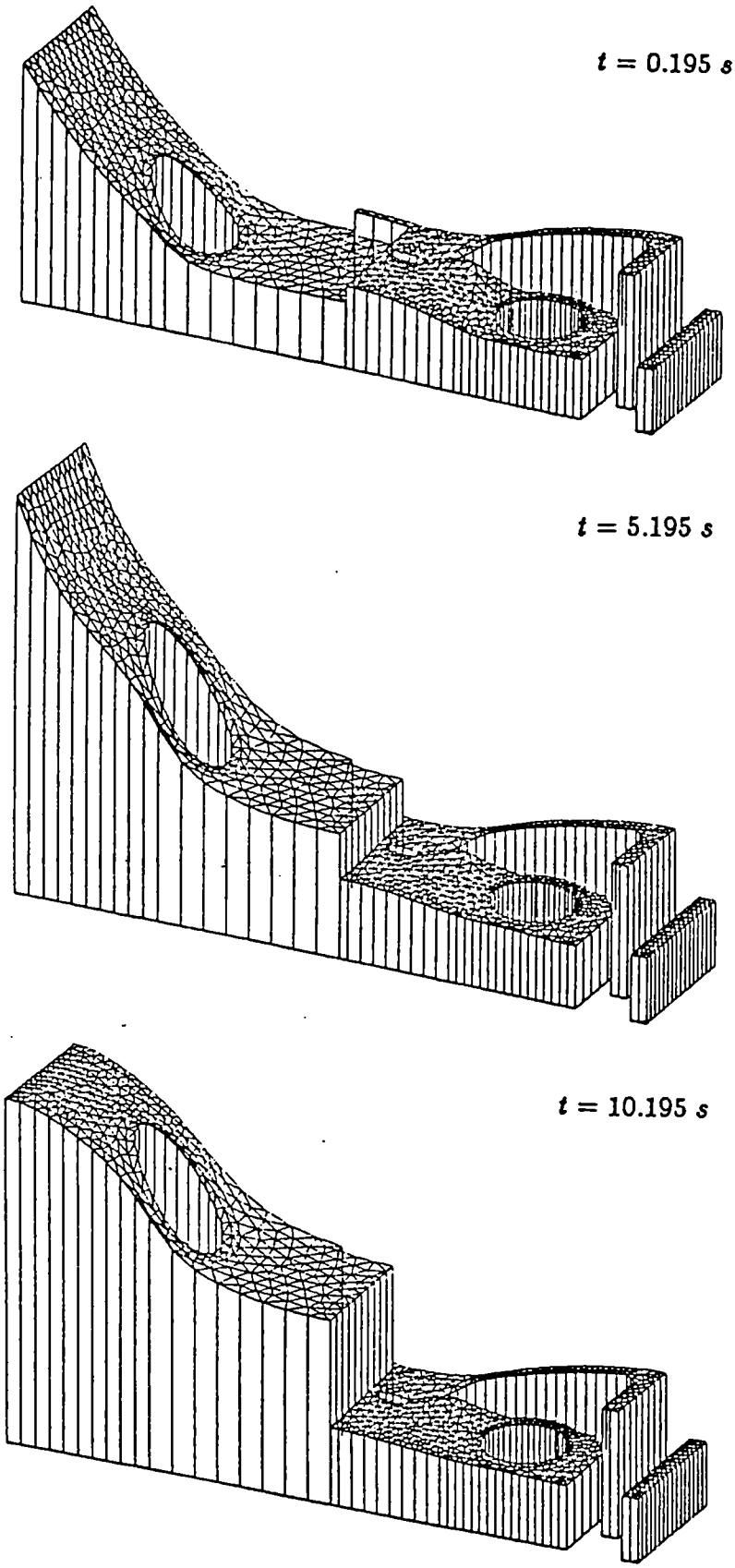


Fig. 4.3 : Load Hypothesis 3. Geometry A.

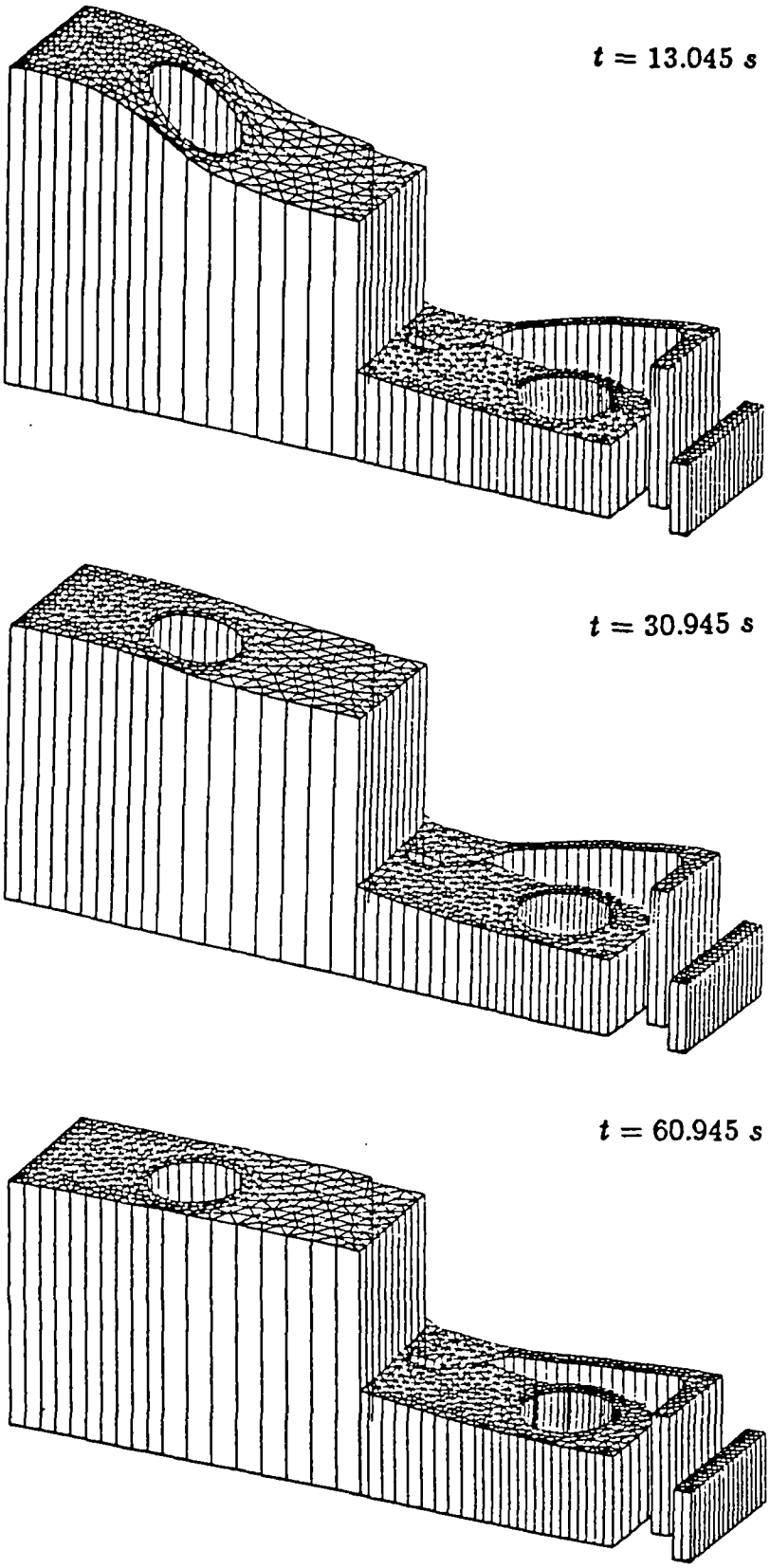


Fig. 4.3 : Load Hypothesis 3. Geometry A.
(cont.)

1.1.5. INDUSTRIAL HAZARD

RUNAWAY REACTIONS

Introduction

Research on chemical and fluid dynamic phenomena associated with runaway reactions was started at the JRC Ispra during the previous multiannual programme. The "Runaway Reactions" research programme has been formulated and is being pursued in close association with the "European Contact Group on Runaway Reactions".

This programme is part of the JRC "Industrial Hazards" programme, which provides scientific support to the Commission's Action on Environment.

The general objectives of the Runaway Reactions programme are:

- Prevention of runaway by development and testing of early-warning techniques and improved control systems.
- Assessment of techniques to stop an ongoing runaway event.
- Assessment and improvement of methods for the design of emergency pressure relief systems and of fluid discharge systems.

In the programme two main areas of research may be distinguished:

1) Process dynamics close to runaway.

   - Investigation in a batch chemical reactor.
   - Computer simulation of batch-type chemical processes.

2) Multiphase flow of chemically reacting multicomponent fluid mixtures.

   - Investigation of multiphase flow phenomena in reactor relief systems during venting.
   - Development and assessment of computer programs for emergency release of multicomponent fluid mixtures from chemical reactors and storage vessels.

A brief description of these research areas is given hereafter.

Process Dynamics Close to Runaway in Batch Reactors

The main research activity is that to be performed in FIRES (Facility for Investigating Runaway Events Safely [1]. This facility is presently under construction and is scheduled for completion by mid-1990.

The central part of FIRES is a cylindrical chemical reactor vessel with a volume of 0.1 m^3, equipped with sensitive measuring devices and provided with control, early-warning, and shutdown systems. For safety reasons the reactor is placed in a containment bunker.

The principle objective of the project is to study off-normal behaviour of chemical batch reactors, specifically:

- To develop criteria for the inherent safety of processes by studying the characteristics of reactive mixtures and determining the critical operating conditions.
- To study and develop measures to detect and stop a runaway: early detection and control interlocking systems to prevent and stop uncontrolled thermal excursions and the associated overpressurisation phenomena.
- To apply the knowledge gained for developing an expert system to assist (on-line) and to train (off-line) plant operators.

The chemical processes, on which attention is presently focused, are:

- Toluene mononitration by mixed acid.
- Suspension polymerisation of methyl methacrylate.

The main reasons for this choice are the industrial importance and high exothermicity of these reactions, and the fact that they gave rise to a relatively high number of incidents in the past.

The commissioning of the FIRES facility has been started. The first characterization tests have been initiated with a glass reactor (scale 1:1) in order to determine the influence of stirring effects:

- Surface wetted by the reaction mixture as a function of stirrer speed and liquid volume, for studying the stirrer influence on the heat transfer.
- Qualitative studies on stirrer performance. Determination of the homogenization limits.
- Determination of interfacial area for liquid-liquid systems.

In order to gain basic knowledge of the chemical processes prior to their investigation in FIRES, experiments are carried out in a small scale reactor with a volume of 2 litre (a Mettler RC1 reaction calorimeter [2]. The apparatus has been characterized (stirrer influences and heat transfer coefficients), and the following chemical studies have been done:

- Neutralization reaction between hydrochloric acid and sodium hydroxide at different temperatures, feeding rates and concentrations in order to study the dynamic behaviour of the reactor under well known kinetic conditions.
- Esterification reaction between 2-butanol and propionic anhydride with different concentrations of sulphuric acid (catalytic agent) in collaboration with the Health and Safety Executive, Buxton (UK).
- Toluene mononitration reaction in isothermal, isoperibolic and adiabatic modes in order to start the verification of the complex kinetics scheme formulated.

The OLIWA (On-LIne WArning) system, which measures the second derivative of temperature in the reactor and the first derivative of the temperature difference between the jacket and the reactor, has been connected to the reaction calorimeter in order to test its performance.

Concerning control aspects for batch chemical reactors, preliminary research has been done for evaluating the applicability of adaptive control systems to the FIRES installation.

Computer Simulation of Batch-Type Chemical Reactors

A mathematical simulator named FISIM (FIres SIMulator) has been developed. The aim of this simulator is to guide the FIRES experimental programme, to ensure optimal exploitation of the experimental facility and to facilite analysis of the experiments. The simulator has the capability to perform sensitivity analyses for all parameters of engineering interest, to determine safe ranges of operating conditions and to simulate the system behaviour for a certain number of anomalies.

The first version of FISIM has been modified in order to simulate the dynamic behaviour of the RC1 and to use the experimental data obtained for starting the adjustment phase. This modification includes the simulation of the temperature control system of the apparatus and the introduction of all the equipment characteristics into the data base.

Studies for the toluene mononitration, in order to feed the simulator with kinetic data, have been continued with the generalisation of the modelling of two partially miscible liquid phases [3].

- assuming thermodynamic equilibrium between the phases,
- defining a pseudohomogeneous reaction rate as a function of mass transfer and chemical resistances,
- including the phase inversion phenomenon.

The initial comparison between experimental data and results of the simulation showed a satisfactory agreement for heat exchange, temperature controller behaviour and instantaneous reactions. However, for the toluene mononitration more experimental study is necessary in order to improve the mass transfer model.

Investigations of Multiphase Flow Phenomena in Reactor Relief Systems during Venting

This activity deals with the experimental study of the flow behaviour of multicomponent fluids during emergency venting.

The general objectives are:

- To provide insight into the nature of multiphase flow phenomena associated with venting.
- To provide quantitative data for the validation of models and computer codes.

The available experimental multiphase-multicomponent flow facility (MPMC) allows experiments to be made in vessels with volumes in the range 0.02-0.05 m^3 and in vent lines with diameters in the range 10-50 mm, at pressures up to 15 bar.

Available instrumentation allows the measurement of temperature, pressure, mixture density, liquid phase velocity and liquid hold-up.

Venting studies carried out so far have concentrated on non-reacting aqueous fluids of which the viscosity was varied by a factor up to 6000 with respect to that of water (about 1 mPa s) [4]. For these experiments solutions of 10, 20 and 25% (wt) Luviskol in demineralized water were used, which have viscosities of 0.3, 3 and 6 Pa s, respectively. The initial pressure was 5 bar.

For reference purposes venting tests were performed with demineralized water, and with solutions of some ppm of Falterol in demineralized water. Small additions of this surfactant render the water foamy and decrease the surface tension to about half without effecting the bulk properties of the water.

Examples of the experimental results are presented in Figs 5.1 and 5.2. Figure 5.1 is a plot of the vessel pressure versus time, which shows that the depressurization of the high-viscosity liquids occurs slower than for pure water. It is interesting to note that the foaming low-viscosity water-Falterol mixture produces a depressurization similar to the non-foaming water-Luviskol mixture with a 3000 times higher viscosity.

Figure 5.2 shows the vented mass fraction (vented mass / initial mass inventory) versus time. There is again a qualitatively similar behaviour between the foaming low-viscosity and the non-foaming high-viscosity solutions, which both show a considerably larger vented mass fraction than pure water.

Basic investigation of flow patterns of high-viscosity liquids is underway, measuring the bubble rise velocity and the bubble size as a function of the superficial gas velocity [5]. It was noted that at high viscosities an "emulsion" of small gas bubbles in the liquid is built up with time.

Development and Assessment of Computer Programs for Emergency Release of Multicomponent Fluid Mixtures from Chemical Reactors and Storage Vessels

The general objective of this activity is to provide validated prediction methods for adequate and safe design of emergency relief systems for chemical reactors with runaway potential.

- **Development of the computer program VESSEL**

  In the past, within the Light Water Reactor Safety programme, significant experience has been attained in modelling transient two-phase flow situations associated with the depressurisation of coolant circuits with internal heat geeration. Among others, the computational tool MULTIVOL was developed [6,7] which models one-dimensionally the transient two-phase flow behaviour of a single component (water), within complex circuits under accident situations. This code has been used as the basis for the development of the code VESSEL [8,9] which focuses on the behaviour of multicomponent mixtures of chemicals during the emergency relief of a chemical reactor or storage vessel.

  During 1989 this code has been extensively altered and it is now capable of modelling the venting of batch-type reactors containing in principle any

number of components. The vapour and liquid equilibrium concentrations for the mixture are calculated at each time step, and the interphase velocity difference is modelled by a drift-flux approach. Up to now two categories of chemical reactions have been investigated, the first is typical of a decomposition reaction and the second represents a polymerisation type of reaction.

Parametric studies [10] have been carried out with the present version of the code to determine the influence of vent position, vessel size, interphase velocity difference (slip) and reaction type on vessel behaviour and it has become apparent that the ability of a code to furnish a one-dimensional description of in-vessel fluid behaviour during depressurisation is of crucial importance, as it allows the correct specification of vent line entrance conditions. As a consequence of this it becomes possible to model correctly the effect of different vent locations.

As an example of the code's present capability, a typical batch-type chemical reactor is considered (Fig. 5.3). Shown in Figs. 5.4 and 5.5 are some results that model a polymerisation type of reaction where the more volatile component is converted into the less volatile component. The relief valve was activated at a predetermined vessel pressure; this occurred some 40 seconds into the transient. The calculation highlights the significant influence that interphase slip has on the course of the transient. If assumptions of homogeneous (no slip) two-phase flow are applied then a strong thermal runaway is predicted, whereas thermal runaway is not predicted when interphase slip is modelled.

In parallel with this activity has been the development of a one-dimensional model of the vent line where critical two-phase flow of multicomponent chemically reacting fluids is being treated. Basic supporting work on non-equilibrium effects is being done in conjunction with the University of Louvain.

Development of RELAP5-MF

This code is an extension of the nuclear reactor safety code RELAP5-Mod1. It has been extensively improved at JRC in terms of computational efficiency and stability and is now able to handle fluids other than water [11]. At present the fluid mixture is treated as a one-component pseudo-fluid. The code has the capability of conveniently describing complex interacting circuits and has already been successfully used to predict the behaviour of a petrochemical plant under off-normal conditions [12].

Assessment of the codes against experimental data

Test runs have been made with the U.S. computer codes SAFIRE and DEERS for reproducing the experimental results of the MPMC facility [13,14]. SAFIRE is based on the representation of the reactor vessel by a single computational cell for which global balances of mass and energy are formulated. Two-phase flow phenomena in the vent line are treated one-dimensionally. DEERS provides a one-dimensional two-phase fluid dynamic treatment for the vessel, vent line and dump tank. Because of the excessive computing time when run on a PC, the code has been implemented on the Amdahl mainframe computer of the JRC. The necessary input and output satellite programs have been attached to the original PC version provided by the code developer.

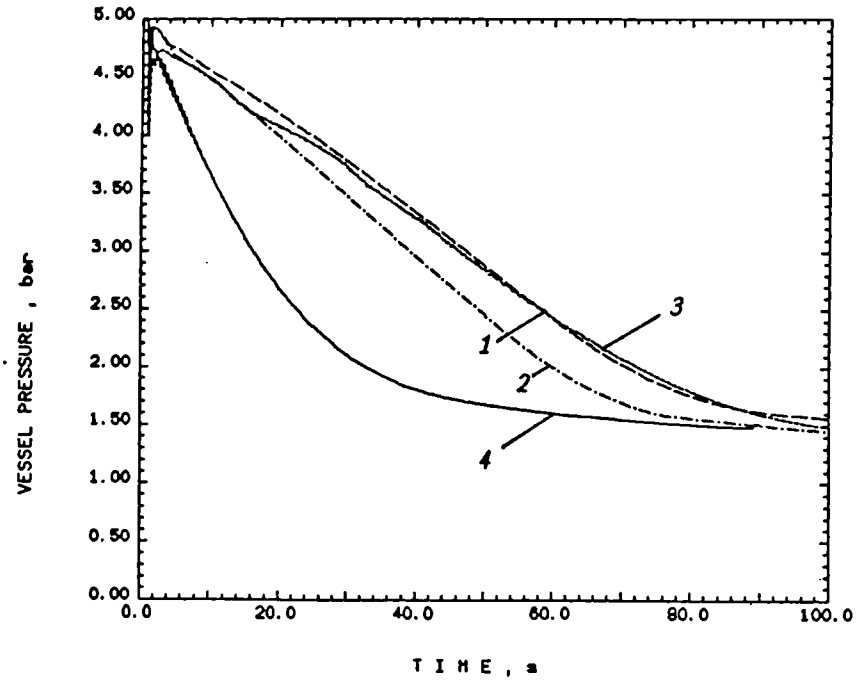


Fig. 5.1 : *Influence of viscosity and foaming on the depressurisation characteristics.*
1: foamy water (with Falterol), 2: high-viscosity 10 % Luviskol solution (0.3 Pa s),
3: very high-viscosity 20 % Luviskol solution (3 Pa s), 4: water

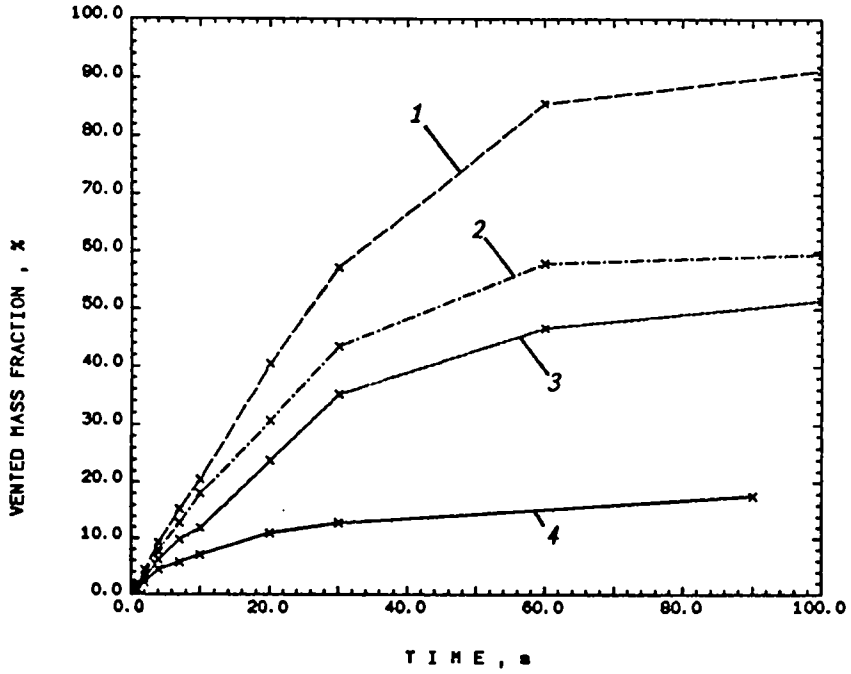
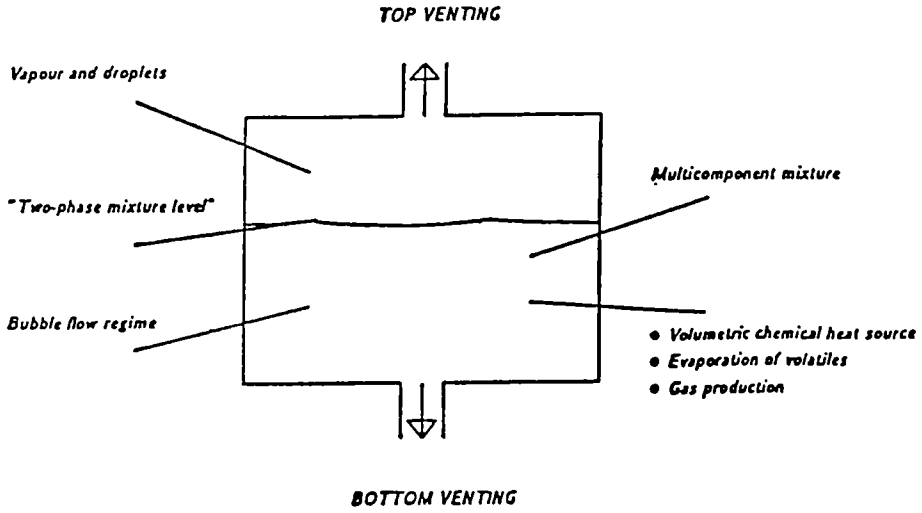


Fig. 5.2 : *Influence of viscosity and foaming on the vented mass fraction.*
1: foamy water (with Falterol), 2: high-viscosity 10 % Luviskol solution (0.3 Pa s),
3: very high-viscosity 20 % Luviskol solution (3 Pa s), 4: water



Key phenomena that have to be modelled

Typical phases during top venting

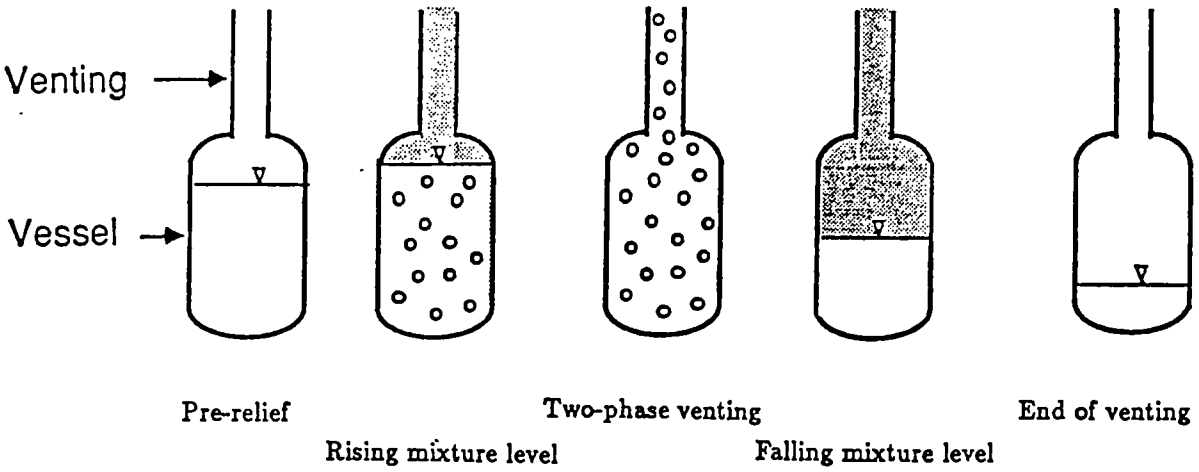


Fig 5.3 Schematic representation of the venting of a multicomponent mixture

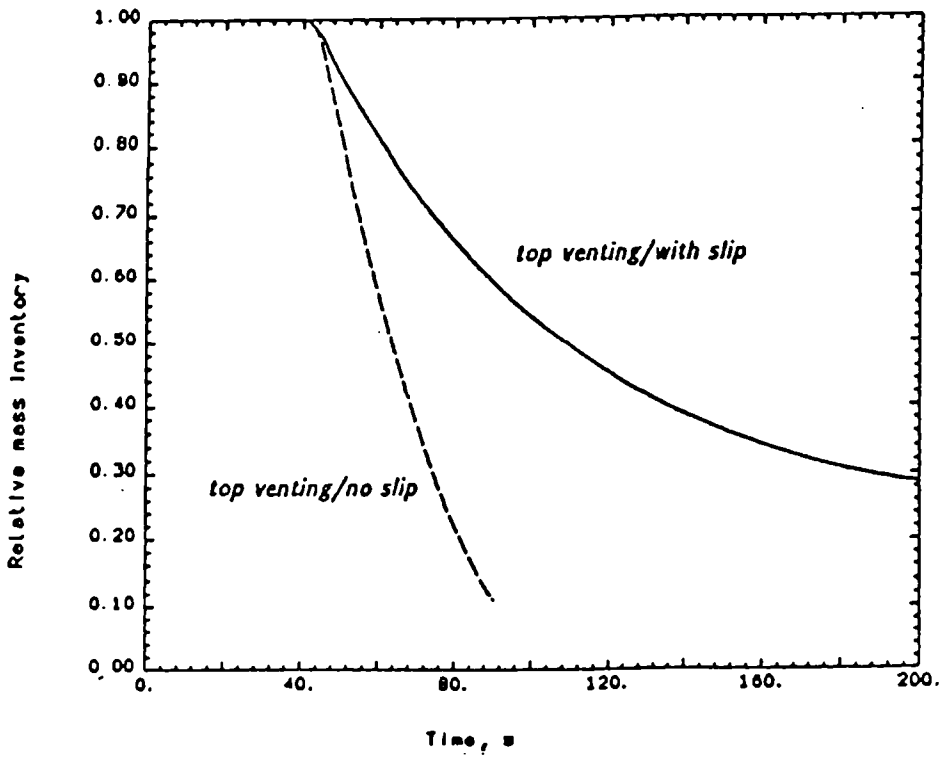


Fig. 5.4 : Influence of slip on the mass inventory during venting of a chemically reacting fluid mixture

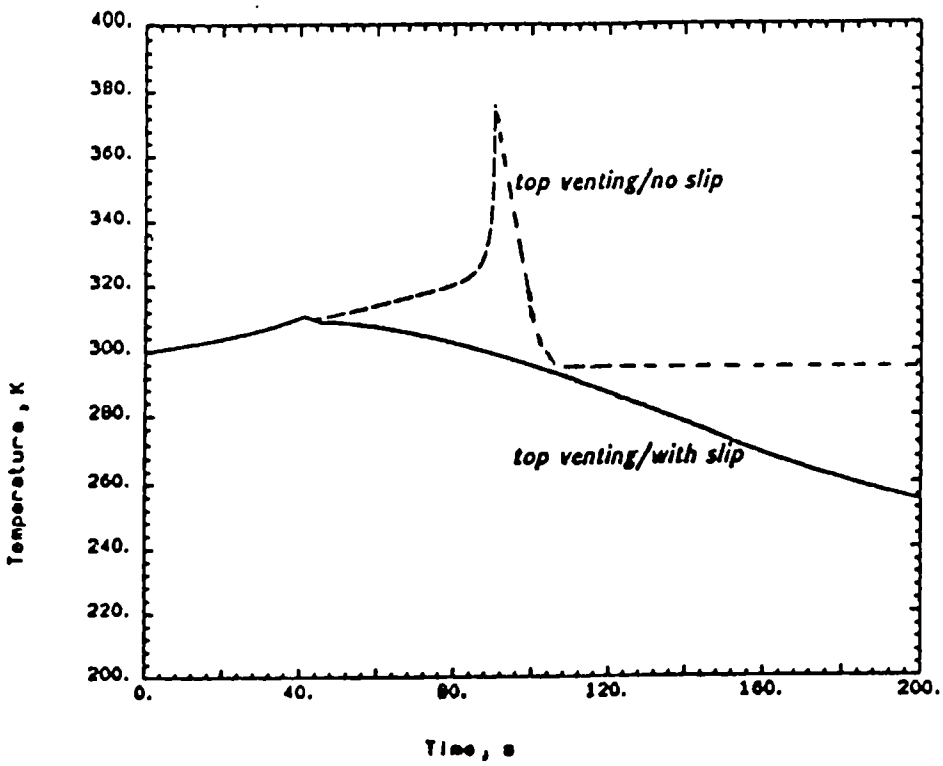


Fig. 5.5 : Influence of slip on the evolution of the temperature during venting of a chemically reacting fluid mixture

In order to produce reasonable results, SAFIRE was calibrated via a post-test calculation of a standard experiment. The calibration included setting a (constant) two-phase friction factor, a vessel flow regime, a radial distribution parameter (for the drift flux model) and the rate of heat transfer from the vessel to the fluid. SAFIRE was able to match the chosen standard experiment. However, when changing the venting onset pressure, reproducibility is limited.

During 1989 a benchmark exercise has been pursued where the codes currently being developed at the JRC together with the U.S. code SAFIRE have been compared with vessel depressurisation experiments carried out in the MPMC test facility of the JRC and other test facilities [15]. Preliminary conclusions indicate that a one-dimensional description of the vessel and the vent line is necessary to predict all the system parameters correctly.

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1.1.6. APPLICATION OF REMOTE SENSING TECHNIQUES

Modelling of Marine Transport Processes

On request of the "Institute for Remote Sensing Applications", in 1988 a small group has been built up for mathematical-numerical modelling of marine transport processes.Its purpose is to give support to the programme "Monitoring of the Marine Environment". The work regards the development of computer models for linking remote sensed surface concentration pattern of typical parameters (chlorophyll, total suspended matter, sea surface temperature, ...) to the occurring marine transformation and transportation processes.

The activity of the group started at the beginning of this year. Work is performed under two headings :

1. In depth-analysis of the numerical properties of several 3D computer models available at JRC, systematic evaluation of the application limits of model specific simplifying assumptions, comparative assessment of submodels for describing the turbulent transport properties.

2. Comparative application of the available computer models for simulating typical situation in the Adriatic Sea and confrontation with in situ measurement and with remote sensed data.

Presently three models of different origin are available at the JRC. Their main features are described in reference [1,2,3].

Under the first heading, an extended study has been performed dealing with the vertical diffusion of momentum in a homogeneous sea of an infinite horizontal extent. The results are described in ref. [4].

Under the second heading, the three computer models are being applied for simulating the circulation pattern in the Northern Adriatic during a winter-typical episode of a strong NE wind (bura), characterized by essentially homogeneous water. Work is underway for coupling a Lagrangian tracer model with the hydrodynamic models for visualizing the flow pattern, especially the Po plume. The results will be confronted with RS images. The elaboration of the RS images is described in a separate report [5].

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Report in preparation

1.1.7. REFERENCE METHODS FOR EVALUATION OF STRUCTURAL RELIABILITY

This programme has as a focal point the construction of a large reaction wall (see chapter 2.2.3.). The studies described below are preparatory and supporting activities for this large project.

Development of the Pseudodynamic Testing Methods (PDT)

The PDT-technique will be applied in performing tests with the Reaction Wall.

In the simplest form of the PDT technique, which applied stricly to loading-rate-independent structures representable by lumped-pass systems, the equations of motion under prescribed time-dependent loading (the ground motion in the case of an earthquake) are solved step-wise by a computer. The (nonlinear) reaction of the structure does not need to be assumed, but is measured directly by means of load cells attached to the actuators. At each time step the displacements of the masses are computed and then applied by the actuators which furnish the reaction forces to be used in the next step. By incrementing this procedure, the complete response is simulated, avoiding the difficulty of estimating the stiffness of the structure and its degradation due to damage.

At this time no 'turnkey' system for the PDT method is available; that to be used at the JRC will be built in-house and improved continuously in line with current developments. This will require a number of supporting tests. A specific objective in this respect is the development of a flexible system including the implementation of an implicit time-integration algorithm allowing the use of larger time steps than affordable with the normally used explicit algorithms. Secondly new forms of the PDT servocontrol loop, based on digital concepts, will be studied with the view of reducing the errors associated with improperly imposing specified displacements and possibly avoiding interruptions in the motion of the hydraulic actuators.

Tests to support the setting up of the PDT system will be started shortly. Initially these will use simple flexible steel specimens and existing equipment (a servo-hydraulic testing machine) and will be extended to include larger and stiffer specimens of reinforced concrete and larger actuators. The machine itself will be used as a shaking table for the comparison testing of the small steel samples, whereas an external table will be necessary for the concrete trials. As the actuators and hydraulic system components for the reaction wall become available larger scale PDT trials will be made using temporary reaction frames. These, mainly in-house, studies will be backed up by extra-mural tests on study contract where appropriate.

In the meantime the data acquisition and data treatment capabilities of the group is being completely overhauled and updated. Several alternative packages are being tested as to reliability and performance; the preference being directed towards PC-run packages providing blocks of 16 or 32 channels and high resolution (typically 14 or 16 bit). These are easy to run and will allow reliable data transfer to the new HP 9000 series computer now used for the storage and manipulation of data. Work has also begun on the digital generation of the ramps which will be required for the input loading signals in the experiments.

Cyclic Tests on Reinforced Concrete

In order to gain experience in testing reinforced concrete structures, and to make a start in supplying data for the development of global "member models" of reinforced concrete elements under cyclic loading, a series of tests has been initiated. To date five 1.5m. column-slab joints have been constructed and tested on a reaction frame. Uniaxial cyclic bending tests were performed at frequencies of 0.002 and 0.2 Hz. and with displacement amplitudes increasing to cause considerable damage and stiffness degradation. An example of the load/deflection relationship found is shown in Fig. 7.1. From the results so far, an increased load-carrying capacity of the order of 15% is suggested at the higher frequency. The potential implications of this rate-effect for PDT testing which of necessity is carried out slower than a real-time seismic loading, needs to be assessed. As a first approximation it suggests that the PDT method should be conservative but this may not be true in all respects and under all conditions. Further experimental and numerical studies are to be carried out to this end.

The cyclic testing of RC columns will be extended also to perform biaxial bending in the presence of an axial compressive load simulating the weight of a structure on the column. Data from this type of test is scarce in the literature and the understanding and modelling of RC members under multi-axial cyclic loading is not very advanced. A reaction frame has been constructed which will allow such testing of specimens, again of up to 1.5m. length.

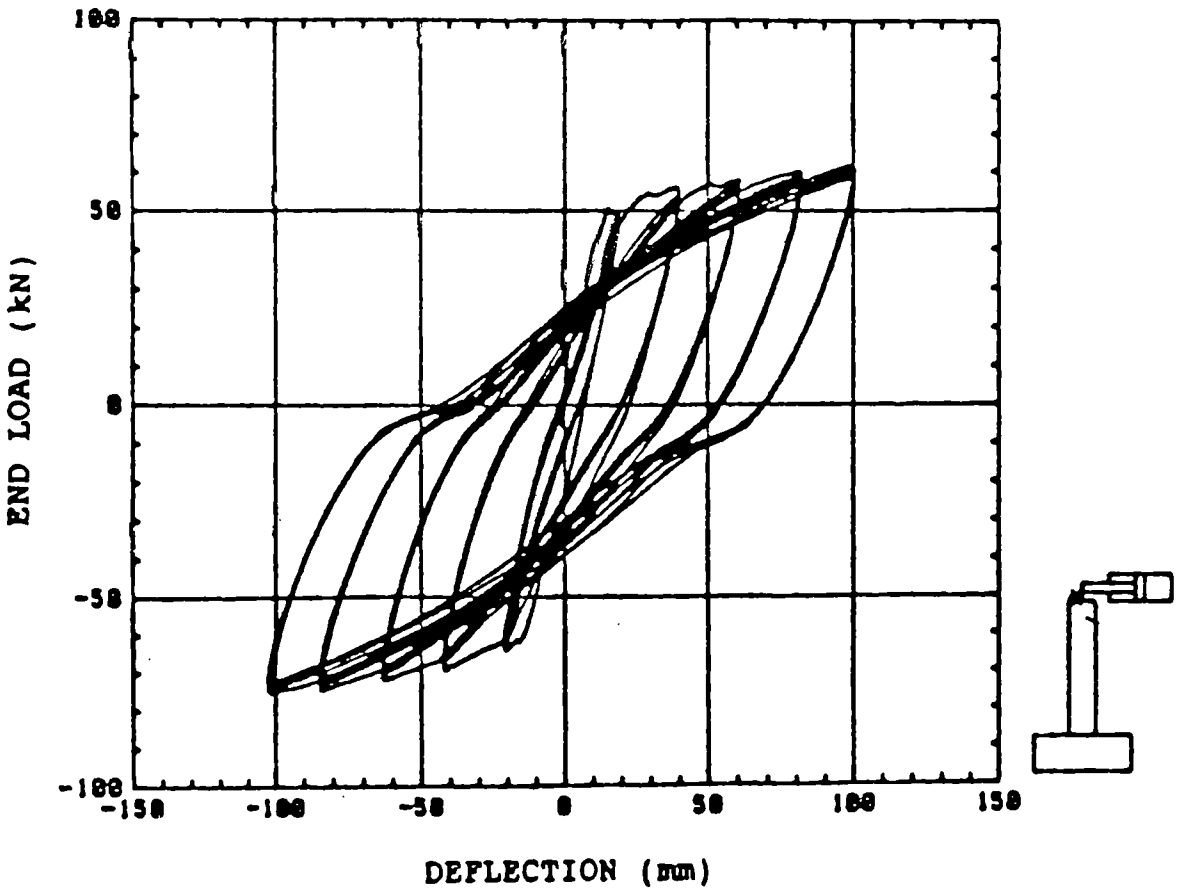


Fig. 7.1 : Cyclic Bending of Reinforced Concrete Column (0.002 Hz)

Dynamic properties of structural materials

- Studies on steel

  Efforts have been devoted to the determination by biaxial experiments of dynamic material response under combined loading. Two biaxial testing deviced developed in the JRC laboratory have been used for this purpose:

  1. Tension-tension or tension-compression loading of a cruciform specimen at strain rates ranging between 10^{-3} and 10^{3} s^{-1} (Fig. 7.2).

  2. Tension-torsion or double shear loading of thin-walled tubular specimen or of a specially developed specimen [1].

  The results obtained with cruciform specimens of AISI 316 stainless steel at strain rates up to 10 s^{-1} were reported in [2,3,4] and can be summarized as follows:

  . The experimentally determined initial yield surface at room temperature is in good agreement with the classical von Mises criterion (Fig. 7.3).

  . However, after initial yielding plastic strain increments do not appear to obey the usual normality rule, but rather show a dependence upon the straining direction . A directionality measure has been proposed and a viscoplastic model has been suggested in [4] which takes such measure into account.

  The equivalent flow curves at low and medium strain rates have shown strain hardening and strain rate hardening (Fig. 7.4); furthermore, the case of symmetrical loading allowed for larger fracture strain and slightly higher flow curves.

  Preliminary experiments performed in torsion on tubular specimens made of the same material as the cruciform specimens (AISI 316) produced different flow curves, indicating that the concept of a unique yield locus independent of the deformation mode and direction of straining seems to be questionable (see [5-6]). This will be further checked by performing tension-torsion experiments.

- Studies on concrete

  In parallel with the above activities, the dynamic tensile properties of plain concrete have been investigated using a Hopkinson bar in aluminum with a 6x6cm cross section. The concrete specimens were of the same cross section as the bar and their length varied between 2,5 and 15 cm. The experiments worked well and the results are under evaluation. This work will subsequently be extended to multiaxial loading conditions.

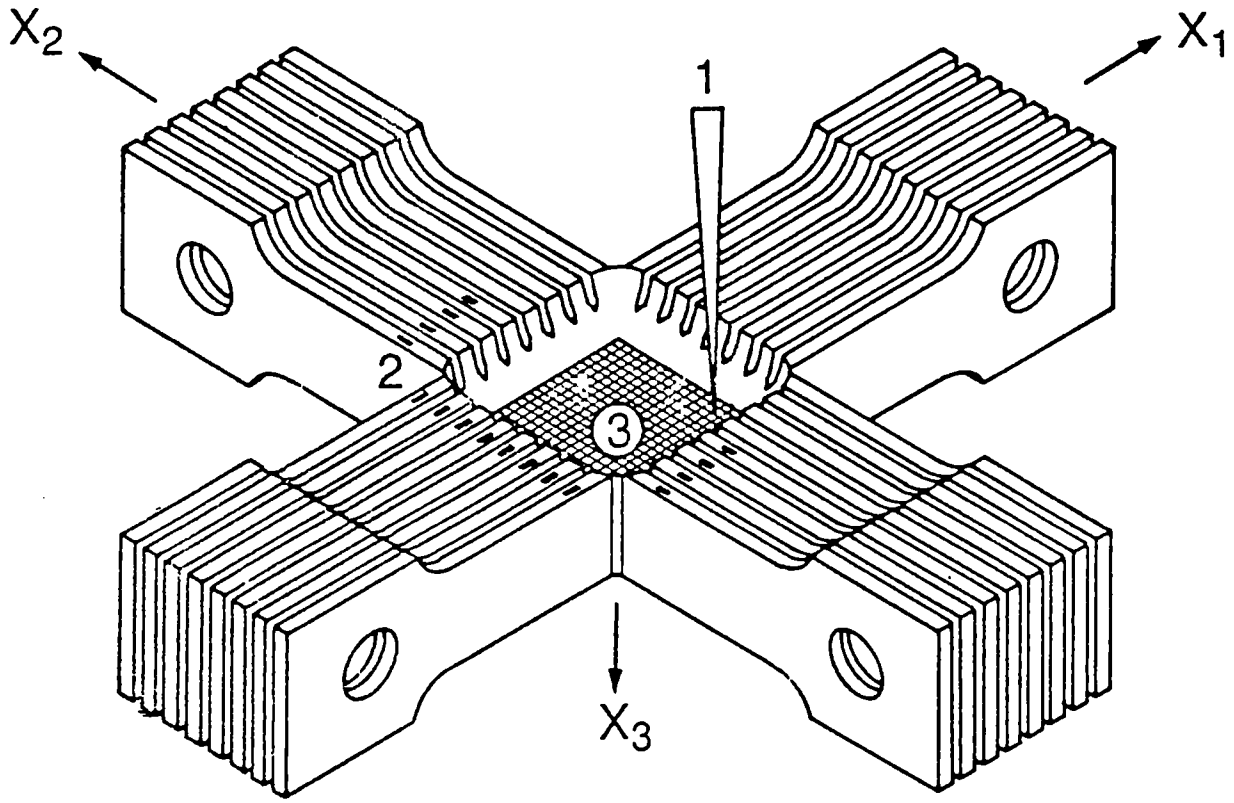
- Preparation of third-party work

  Modifications of the existing experimental equipment were studied to allow investigations on ceramics and composite materials. Research on these

materials will be funded by a contract under negociation with an industrial partner.

A research programme is also in preparation in support of the automotive and aircraft industries. A first study will be concerned with the measure of the dynamic properties of thin-sheet metals (Fig. 7.5). The goal here is to increase the stamping-speed and, more generally to improve forming methods. The second study is concerned with the precise evaluation of the load-deformation characteristics of thin-sheet box-girders used to mitigate the consequences of car accidents. This study will be performed using the Large Dynamic Test Facility which will be adapted to this purpose (Fig. 7.6).

The upper view of the cruciform specimen



MEASUREMENTS

1. Change of thickness
2. Distributed forces
3. In-plane dispiacements
4. Temperature
5. Strain gages

The bottom view

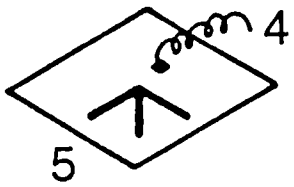


Fig. 7.2 : *The cruciform specimen*

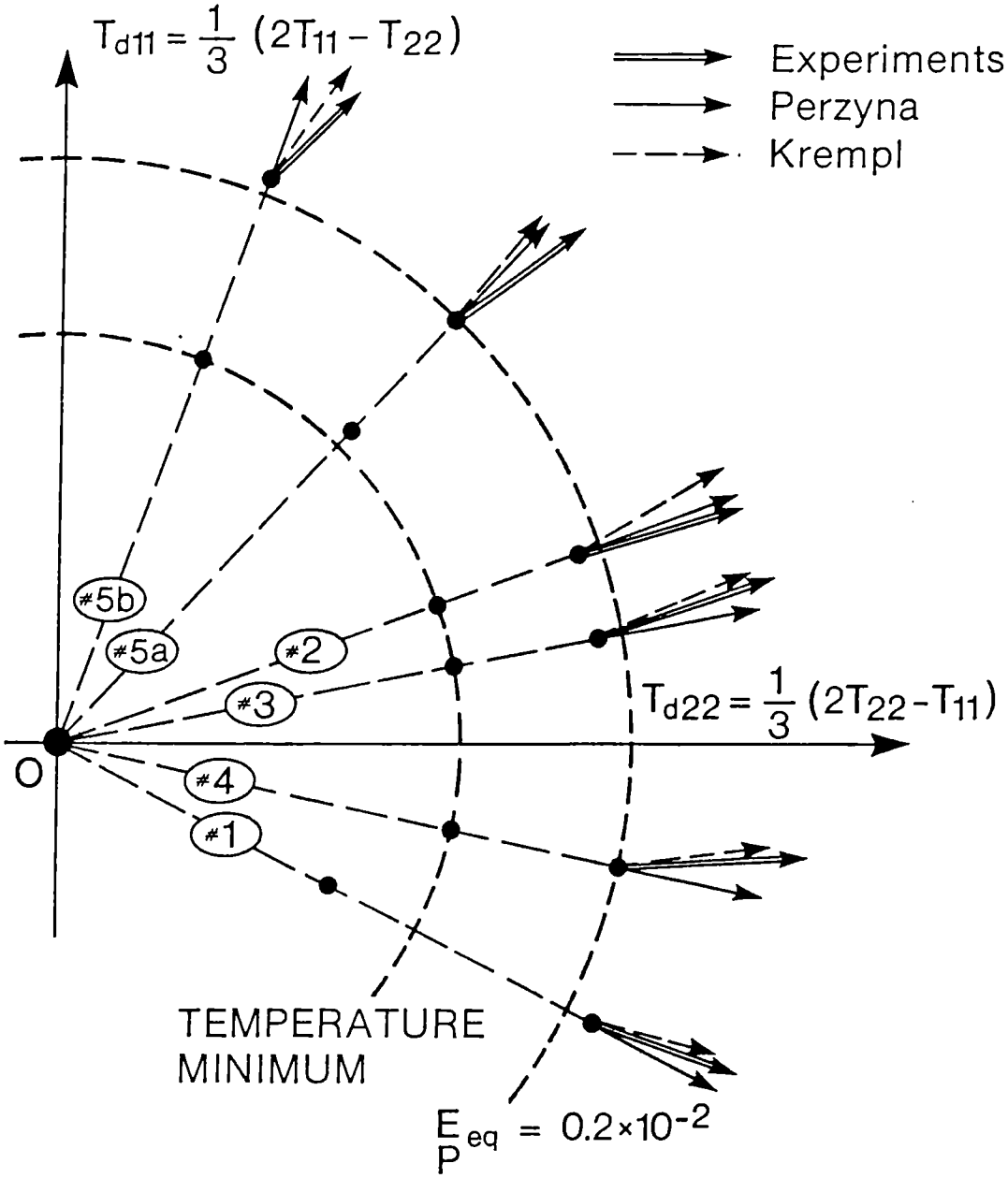


Fig. 7.3 : Relation of plastic strain rate direction with respect to experimentaly found initial yield surface

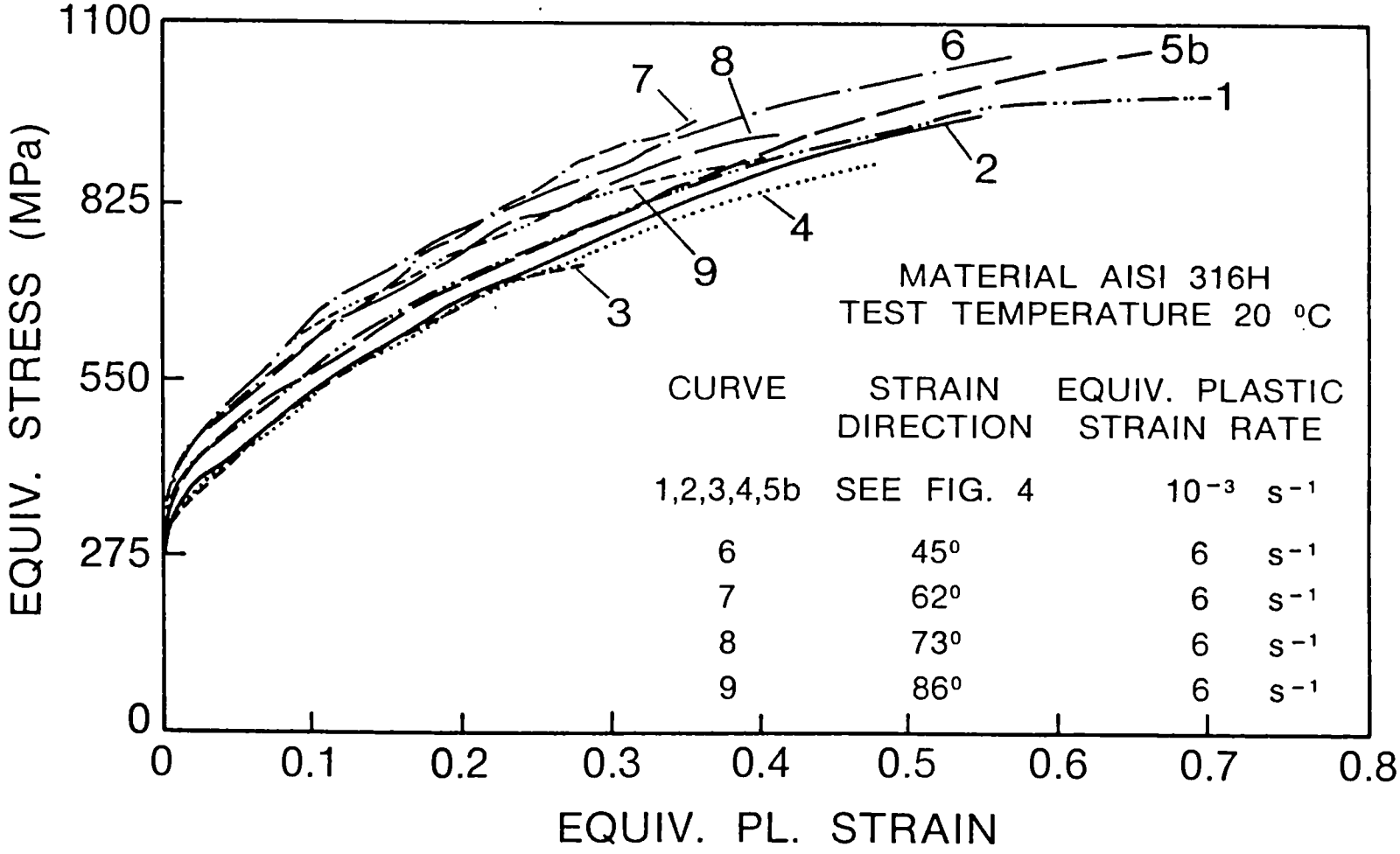


Fig. 7.4: EQUIVALENT STRESS-STRAIN CURVES AT LOW AND MEDIUM STRAIN RATE FOR AISI 316H

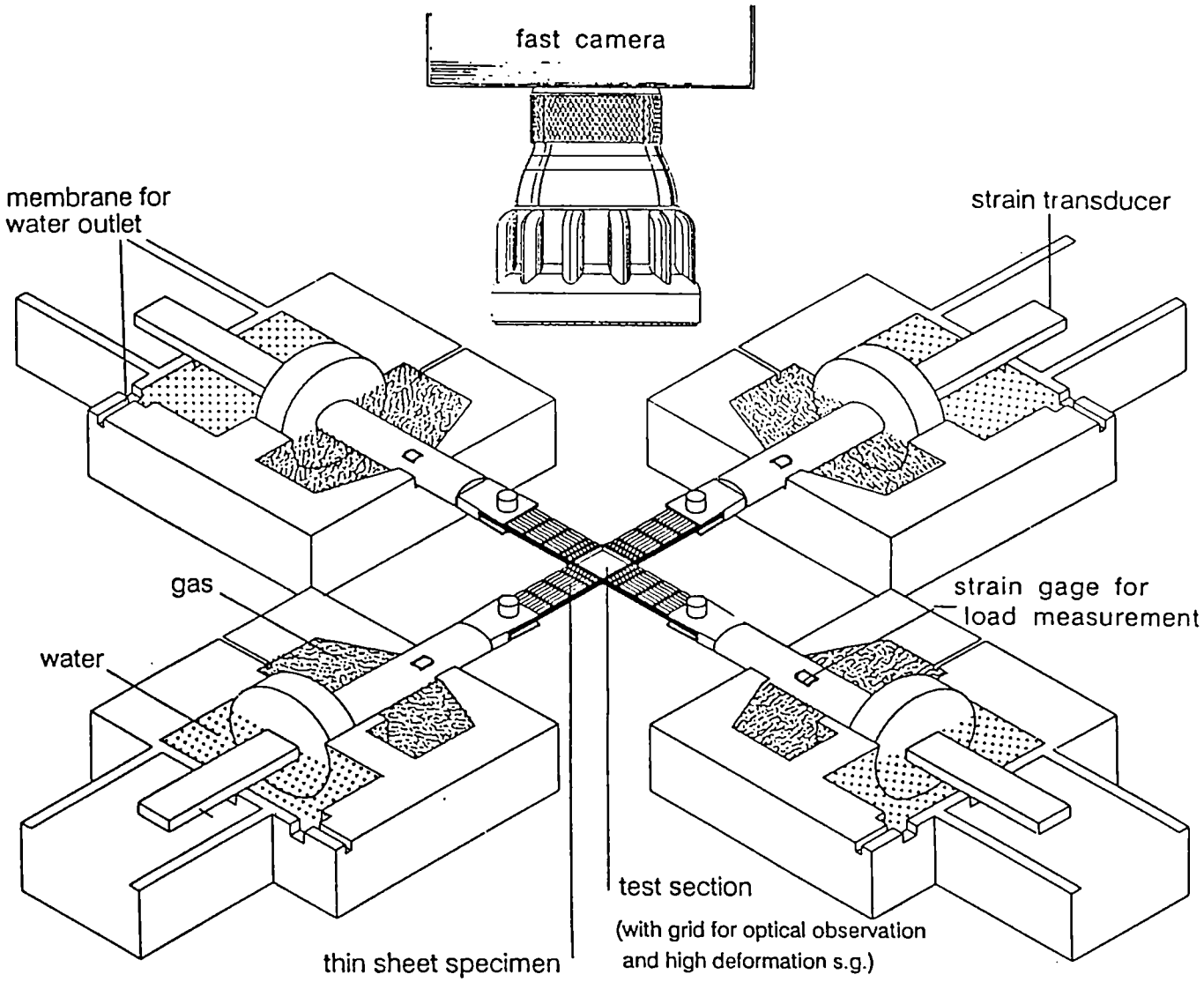


Fig. 7.5 : Apparatus for biaxial testing of thin sheet: layout

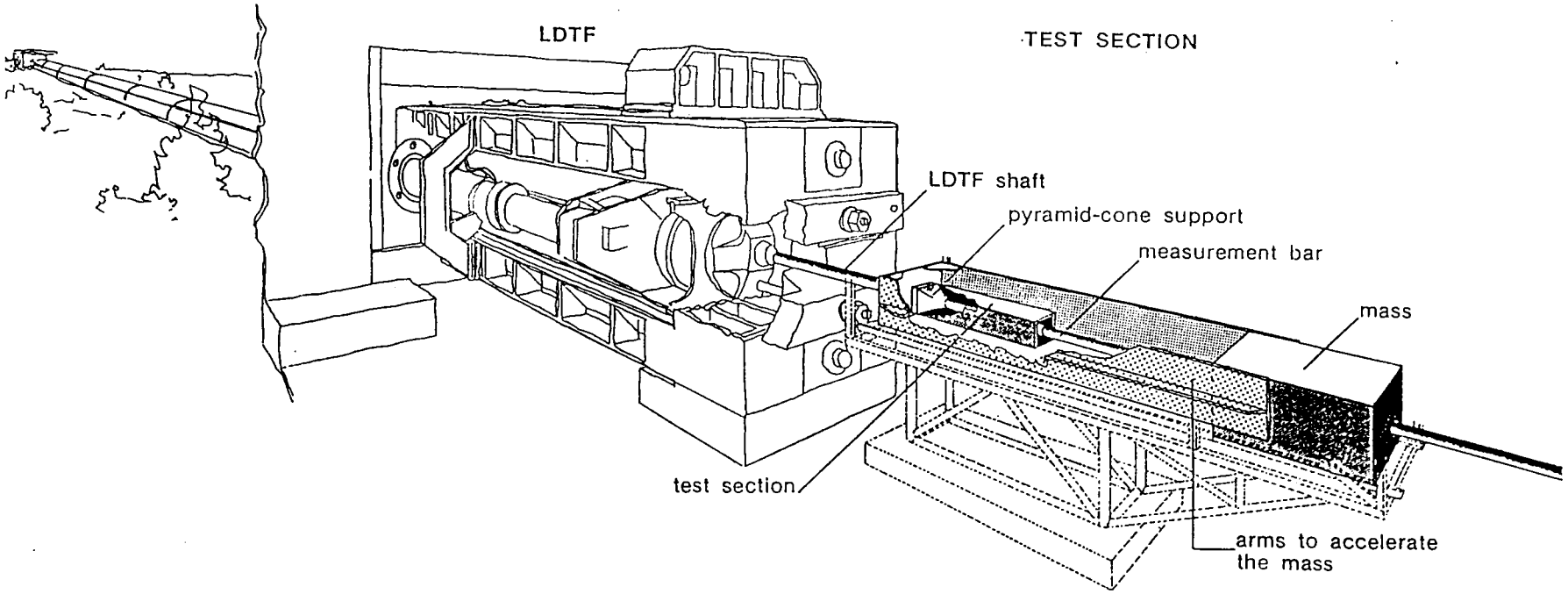


Fig. 7.6 : Layout to test thin sheet box specimen

Graphics developments

A testing activity aimed at assessing the possible uses of advanced graphics tools offered by scientific workstations in our software developments and applications has been started. Existing graphics interfaces of our major software (PLEXIS-3C, CASTEM 2000, ALICE) have been extended to take advantage of the new graphics concepts. To prepare exploitation of the most advanced graphics features such as 3D solid modelling, interactive mouse and menu-driven applications, use of shaded colors to represent iso-values of a scalar field, etc., a graphics module is being developed. This module will then be adapted in order to enhance the interactive capabilities of our software.

Informatic Support

In addition to the modelling and code development activities, the computational mechanics sector provided informatics support under various aspects:

- Complete system administration of an Ethernet local area network serving the Division and composed of 5 UNIX machines and 15 PCs distributed in three buildings.

- Full standardization: all software (more than half a million instructions) running on open system (UNIX), use of NFS (TCP/IP) for file sharing, exploitation of graphics workstations (X-window and PHIGS) and approach to the C language and to the object oriented version C + +.

- Integrated environment for editing and printing (scientific and office)

- Automatic links for compatibility with IBM mainframe computers

- Consulting and educational services

- Information system manager (ISM) activities at the Institute level.

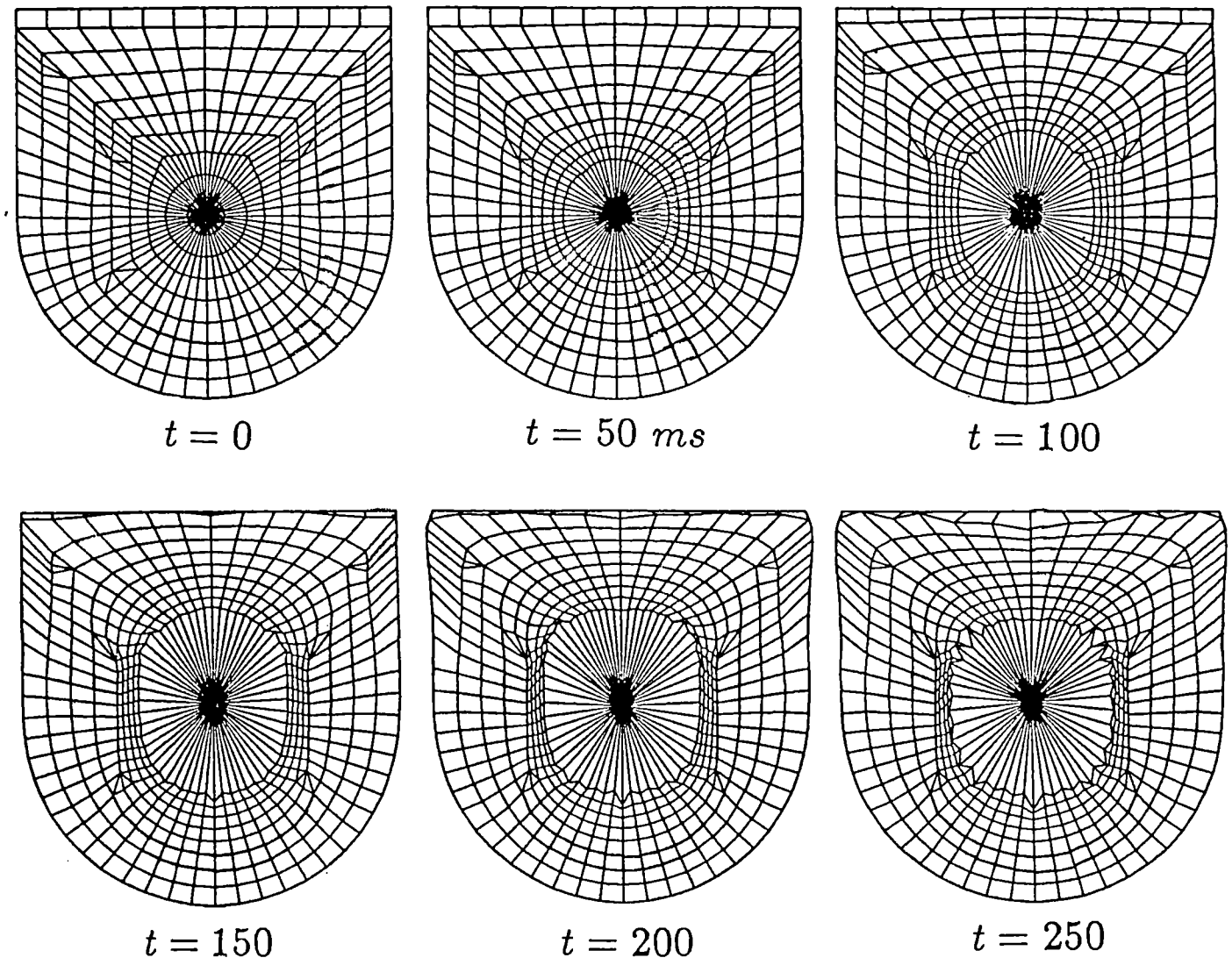


Fig. 7.7 : **Computer simulation of a hypothetical accident in a fast nuclear reactor (CONT exercise)**

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1.2 EXPLORATORY RESEARCH

From the proposed exploratory research activities only one was retained. It concerns H/D extraction and purification activities described in chapter 1.1.4.

1.3 S/T SUPPORT TO COMMUNITIES POLICIES

1.3.1. SAFEGUARDS

SUPPORT PROGRAM FOR DG I

(CBNM, Centre for information Technologies and ELectronics, Institute for Environment, Safety Technology, Systems Engineering, Transuranium Elements and Administration Directorate)

Technical support is provided to the IAEA safeguards inspectors in a large variety of disciplines such as nuclear measurement techniques, sealing and surveillance techniques and accountancy methods.

The main achievements are the following:

- an engineered prototype of the laser surveillance system has been installed at a spent fuel storage pond for field tests in air; other two systems have been produced for further investigation in underwater operation;
- in the framework of support programmes between several countries and the IAEA, JRC has presently provided and automatic system for video reviewing to IAEA headquarters, La Hague reprocessing plant and AECB (Canada) for CANDU reactors;
- a knowledge based system for the automatic transit matching of nuclear material movements has been developed and installed at Vienna headquarters; extension of this evaluation system is now being undertaken for both international and domestic (EC countries) transfer;
- simulation of reprocessing input measurements are performed (in collaboration with ENEA). The computer simulator has been adapted to the MITA facility (of ENEA) and to the CALDEX facility (prototype of a large throughput reprocessing plant tank);
- well characterized metallic spikes of U-Pu alloys for reprocessing input analysis have been prepared and interlaboratory measurement evaluation programmes are conducted on U and Pu materials;
- the Transuranium Institute (Karlsruhe) has contributed to the evaluation and the automation of analytical techniques in support to the Safeguards Analytical Laboratory of Seibersdorf. The tasks which are currently being out cover the development of software for quality control of analytical results on input samples, field-testing of a K-edge densitometer for reprocessing plant samples and the automatic conditioning of samples by robots. A joint evaluation and exchange of experience made with the robot will be done shortly. A cooperative field test of on-site sample conditioning by robots will be executed at the pilot plant a Gatchina near Leningrad. The scope of the experiment and the design of the robot are at present under discussion;
- a compact active neutron interrogation system has been designed and a laboratory prototype is now ready to be tested at PERLA;
- procurement schemes for NDA PERLA standards are being developed with IAEA experts, to assure their acceptability for international safeguards;

- three campaigns have been performed at PERLA with IAEA staff in view of the calibration of NDA instruments for Plutonium isotopic measurements and U assay in MTR fuel elements by gamma ray spectrometry and U analysis with an active neutron assay system;
- an intercomparison exercise of calorimeters for the measurement of large Pu samples is now being set up and will be conducted in 1990;
- the second physical inventory exercise on highly enriched uranium samples has been held at PERLA for IAEA inspectors.

SUPPORT PROGRAM FOR DG XVII

(CBNM, Centre for information Technologies and ELectronics, Institute for Environment, Safety Technology, Systems Engineering, Transuranium Elements and Administration Directorate)

The technical support provided to the Safeguards Directorate covers a wide range of application in the field of nuclear measurement techniques, sealing, identification and surveillance techniques and information technology.

The main achievements are the following:

- a computer aided video surveillance system (CAVIS 1) has been installed at Luxembourg headquarters for long term tests and inspectors training; full documentation has been provided and functional specifications for next generation (CAVIS 2) are being defined;
- a back-up system for image archive of E metal seals with increased capacity and improved data retrieval has been implemented. Automatic recognition of seal is under development;
- a material accountancy data evaluation system, installed at headquarters has been extended for parallel treatment of different material balance areas and/or different categories of nuclear materials;
- since 1986, a number of JRC MARK-II ultrasonic sealing-bolts for spent fuel containers undergo a long term underwater test in a storage pond in Sellafield (UK). Various demonstration campaigns have indicated that the seals and the portable reading equipment work properly. In 1989, a particular effort has been devoted to simplify the reading technique by merging the "identity" and "integrity" features of a seal so that only one reading would be used;
- the know-how for the fabrication of multipurpose ultrasonic cable seals on an industrial basis has been transferred to a French company and a technical support is being given as well. The seal embodies a built-in transducer and could simplify largely the reading of ultrasonic seals;
- after the redesign of the active neutron interrogation system already used routinely by inspectors, three such measurement devices were ordered by the Safeguards Directorate and are now being constructed. They will be delivered during 1990;
- a new PC based beutron correlation instrument has been demonstrated to the Directorate and is now being tested in Harwell (UK). Furthermore a MTR gamma scanning system has been designed and constructed and will be made available to inspectors in 1990;
- extensive Monte Carlo calculations have been performed to assess the possibility of measuring underwater LWR spent fuel in Multi-Element-Bottles;

PARLEMENT EUROPEEN BIBLIOTHEQUE

- in the framework of the Euratom network of analytical laboratories, a large number of destructive analysis has been performed on samples taken by inspectors in different parts of the EC fuel cycle. Also several measurement campaigns have been performed in bulk handling facilities with transportable analytical chemistry instrumentation; quality control exercises have been performed with well characterized reference materials;
- several development activities have been conducted to automise some analytical methods (e.g. titrator, K-edge densitometer) using on-line computers for controlling the analytical procedures, identification of samples and data evaluation;

- a new method is being studied for the analysis of solid residues in HAW or input solutions using inductively coupled plasma/mass spectrometry employing the technique of laser ablation directly on the solid material;
- a conceptual design is being conducted for the installation of an on site analytical laboratory to be situated at a nuclear fuel reprocessing plant and capable of analyzing the input and output material;
- training courses for safeguards inspectors have been organized on NDA techniques (gamma spectrometry-neutron counting), on the physical inventory taking in highly enriched uranium plant and on radioprotection

1.3.2 Expert analysis in support of DG XXI for scientific apparatus imported from Non-Community Countries

This activity deals with the study of scientific instrument dossiers imported from non-Community Countries, which are exempt from custom duty when they are used for scientific research and have high scientific value and provided there is no equivalent instrument with the same characteristics made in the Community.

The dossiers analysed are those for which the Customs of the Community Countries refused the freedom of tax on imported instruments.

The Institute's Nuclear Exp. Division makes available technical support and participates in meetings of the Custom Duty Free Committee, in which controversial cases are reviewed.

The decisions of the Committee are published in the Official Journal of the Community and become operative for all customs of all Member States.

In case of controversy between the Countries, the scientific opinion expressed by our experts is determinant for the decision of the Committee.

The decision of the Committee can be contested at the Court of Justice in Luxembourg. In such a case a scientific support is supplied to the Juridical Services of the Commission.

During 1989, 25 working days have been spent during 11 meetings. The work necessary to examine the dossiers required the full time efforts of 2 A and 1 B.

In addition a continuous support was provided for the legal Service in connection with last developments of jurisprudence in the field of UNESCO exemptions.

1.4 WORK FOR THIRD PARTIES

These activities can be subdivided into 3 groups:

- activities in progress since 1988
- activities which started in 1989
- activities which are being prepared

In the first group extensive use is made of existing nuclear installations and related competences. It includes a contract for increasing the capabilities of the LOBI installation, the delivery of special safeguards equipments, experiments for the incineration of low activity resins and decontamination studies of LWR components.

In the second group a contract was signed which foresees the use of a nuclear installation under construction. Smaller activities were agreed in the development of finite elements codes.

Contracts which are being prepared concern both nuclear and non nuclear activities. It is envisaged to launch larger projects with participation of several contractors for one particular task in which the multidisciplinary nature of the Institute is valorized.

1.5 ASSOCIATED LABORATORIES

EUROPEAN ASSOCIATION OF STRUCTURAL MECHANICS LABORATORIES

An association of laboratories has been set up in order to integrate the work of the new structural mechanics laboratory of the Institute and that of the national laboratories of the European Member States. Response to this initiative has been favourable with about thirty laboratories showing a positive interest.

The overall aims of the association are:

- to enhance the competitive position of the Community's construction industry world-wide,
- to reduce potential hazards to the public in the case of natural disasters, such as earthquakes, tornados and tsunamis,
- to improve analysis methods, design provisions and construction standards for structures subject to severe dynamic loading, in particular by resolving uncertainties associated with the nonlinear behaviour and propagation of damage up to complete failure, and
- to collaborate and coordinate the complementary range of resources available at individual European Institutions, in particular specialist manpower and high performance testing facilities.

The immediate objectives are:

- the identification of research needs and priorities within pertinent engineering fields such as civil/structural, mechanical, offshore and geotechnical,
- the development of detailed collaborative research proposals to be offered to public organisations sponsoring technological innovation and safety-related research,
- the identification of customers from industrial organisations involved in large projects, and
- the encouragement of inter-European cooperation.

As the first exercise of the Association a detailed collaborative research programme is under development which will study the response of civil engineering structures to severe earthquake loading. Four working groups have been founded within the Association to cover research on reinforced concrete, masonry and steel/concrete composite structures and the development of the appropriate testing techniques respectively.

This first collaborative research programme will thus involve the complementary range of resources available at individual European Institutions and is expected to culminate in tests of full-scale structures on the JRC reaction wall.

It has been decided to place study contracts with the lead Institutes of the four working groups referred to above. These Institutes will coordinate the work to be executed jointly by the Laboratories associated with each working group. A self-participation of the research cost will be contributed by each group. Two such contracts - concerning the reinforced concrete and the steel/concrete composite groups - have been placed; the remaining two contracts will be placed next year.

2. LARGE INSTALLATIONS

2.1 OPERATION OF LARGE INSTALLATIONS

The operation of the large facilities :

- the LOBI representing a 1:700 volume scaled PWR cooling system and

- FARO, an installation to melt up to 100 kg of UO_2 at about 3000°C are described in chapter 1.1.1.

Apart from operating these installation for the purpose of verifying and improving calculation tools, their potential use and related necessary modifications were analysed in the frame of Safety studies for new reactor concepts.

2.2 CONSTRUCTION OF NEW INSTALLATIONS

2.2.1 EUROPEAN TRITIUM HANDLING EXPERIMENTAL LABORATORY (ETHEL)

During 1989, the ETHEL project, as outlined in the attached figure, made significant progress with respect to completion of the executive design and construction of the facility. Concerning the building, all structural concrete was cast up to the laboratory's roof. Initial delays were experienced with the design of the Heating & Ventilation and Electrical Systems. However these were completed and installation of the plant is expected to commence early in 1990 as the finish of the building's interior is concluded. Similarly, while the layouts of the Fire Detection & Protection system and general piping services are known, no installation has, as yet, been possible.

After successful air distribution tests at works, the four elements of the large caisson, i.e. a 350 m^3 double skinned stainless steel room weighing a total of 50 t, were transported to site and installed in ETHEL's Hall for Process Development (HPD) prior to final welding and leak testing. The other primary containment, a small caisson, a 5 m^3 vessel envisaged for development work, is still under construction. Also, following exhaustive testing at works of the prototype experimental glove-box suite, clearance was given to proceed with the construction of the remaining secondary containment assemblies. These have now been completed and are awaiting to be dispatched to site. Linked to the large caisson and glove-boxes are the various types of gaseous detritiation systems used for purifying the containment atmospheres. The first of these units has undergone performance testing thereby allowing the rest of the units to be fabricated. The large buffer tanks associated with the gaseous systems were installed in the first floor of the building.

Little progress has been made with the Liquid Waste piping system although the related tanks were installed in the basement of the building. The Waste Conditioning Plant which is now under construction following design changes to

increase operating flexibility. The Solid Waste Store crane is partially completed with the main support beams already situated in the store.

The construction of two specialised systems covering the Tritium Magazine and Radiological Protection is progressing although installation of these systems will not occur until 1990.

Whereas the automatic control systems of the various parts in ETHEL are well established, significant problems were encountered in computer software selection at the Control Room level. These difficulties appear to have been resolved although it remains to successfully demonstrate effective communications between the individual plant and their control systems in the laboratory.

As indicated in the table, it is anticipated that installation of all plant will be completed during the third quarter of 1990. This will be immediately followed by cold, i.e. non-nuclear testing and commissioning on an individual plant basis and then as an overall operating facility, an exercise which will span approximately 12 months.

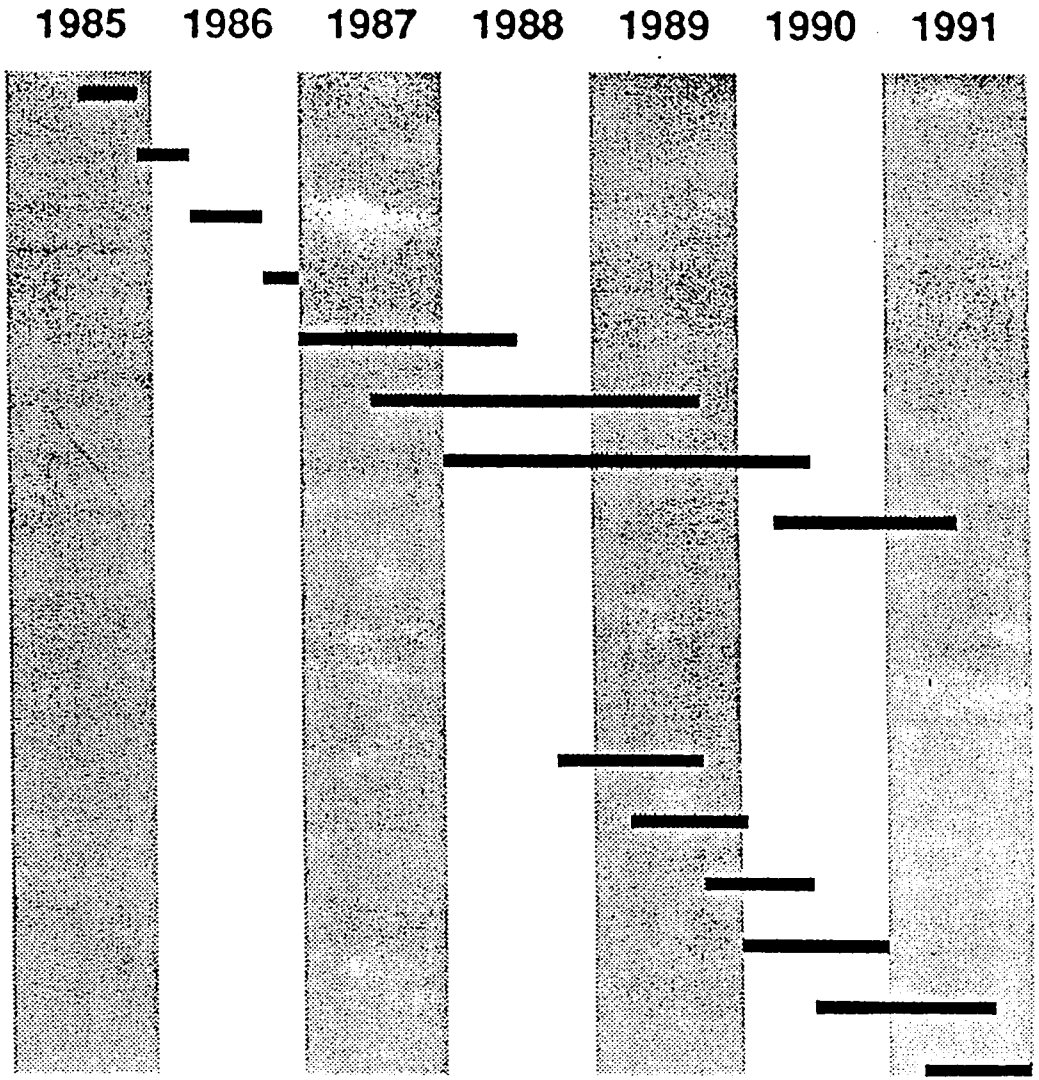
PLANNING OF ETHEL AND INITIAL EXPERIMENTS

LABORATORY

BASIC DESIGN SPECIFICATIONS
CONTRACTOR PRE-SELECTION
PRELIMINARY DESIGN STUDY
CONTRACTOR SELECTION
DETAILED DESIGN
EXECUTIVE DESIGN
CONSTRUCTION & INSTALLATION
TESTING & COMMISSIONING

EXPERIMENTS

QUESTIONNAIRE
DETAILED DESIGN
CONTRACTOR SELECTION
EXECUTIVE DESIGN
CONSTRUCTION & INSTALLATION
TESTING & COMMISSIONING



2.2.2 PETRA

The PETRA hot-cell facility at the JRC Ispra Establishment has undergone a series of commissioning tests since the formal hand-over and closure of the contract with the Consortium in January. The facility includes the typical high active processing steps involved in nuclear fuel reprocessing in order to produce significant quantities of various types of wastes. These wastes will be used primarily for optimisation, verification and demonstration studies at full radioactivity levels in relation to the safety and reliability of waste confinement and disposal concepts.

Successful testing of the essential auxiliary services namely, cooling, vessel, off-gas and pneumatic transfer systems have been carried out. The cell ventilation system has been completed including the modifications to improve the reliability of the system. Preliminary tests of the latter were performed in January and at the end of February in the presence of the Italian Safety Authority.

All vessels and pumps in the cells have been accurately calibrated and recorded on the process control system (PCS). Testing and implementation of the PCS has continued, in particular on the verification of instrumentation and the operational procedures installed for the various unit operations.

Testing of the plexiglass mixer-settler has been performed and subsequently new ones constructed in stainless steel. A number of modification to the facility have been implemented namely :

- replacing the heaters in order to adjust them for remote operation;
- the decision to eliminate the absorption columns after laboratory trials indicated problems in gas formation, caking and column flow;
- installation of further dip tubes in vessels, replacing some TDR level measurement devices for both plant control and safeguards purposes;
- the hot analytical box and glove-boxes have been installed together with the lead shielding and the associated ventilation system;
- the pneumatic shuttle line service has been connected and tested;
- the fuel and experimental equipment in the dismantling and working cells 4304/4411 have been dismantled, cleaned and prepared for housing the new fuel cutting machine which has been designed in-house;
- a new remotely operated sampling device has been designed, constructed, and successfully tested at the JRC;
- civil work modifications resulted in the PETRA cooling circuit equipment and associated cable runs becoming contaminated. The mechanical components were decontaminated and repositioned, whereas the cabling and cable runs replaced. The incident resulted in a delay of normal progress;
- experimental campaigns have been identified to be carried out by the end of 1990 to satisfy the Users Group recommendations;

in parallel to the functional system testing and preparation for the first experimental campaigns, the necessary document preparation as foreseen by the Italian Law is being prepared. The most realistic date for hot testing is Spring 1990;

various meeting have been held throughout the year with the Italian Safety Authorities,DISP, concerning the PETRA documentation relevant to the licensing procedure and authorisation.

2.2.3 REACTION-WALL PROJECT

The final phase of the design study of the reaction-wall facility was completed in May 1989. A call for tender for the construction of the facility was then made. The offers were assessed in September and a contract was placed to enable the construction work proper to start by the end of the year. The reaction-wall/strong-floor system will form part of a new Structural Mechanics Laboratory designed to house all the work of the Applied Mechanics Division of the Institute.

The reaction-wall facility is designed for large-and full-scale testing of components, sub-assemblies and full structural systems. It is available to industry to support the development of innovative design concepts by performing qualification and demonstration tests on large-scale prototypes.

Test Scope
- Quasi-static, cyclic and pseudo-dynamic testing of heavy models.
- Real time dynamic loading of lighter models (e.g. piping systems).
- Multi-dimensional, multi-point loading.

Research potential for the facility has been identified in various engineering fields: civil/structural, mechanical and geotechnical, and work with the reaction-wall will be performed within the framework of an integrated European programme making full use of existing facilities and expertise in the Member States. The current programme is being developed in collaboration with the European Association of Structural Mechanics Laboratories (see section 1.5). It concerns the investigation of the behaviour of civil engineering structures subject to severe earthquake loading and concentrates on the nonlinear regime between the conventional design limits and final collapse.

The research to be performed is of a combined experimental/analytical nature in that the experimental data will be used to develop and validate improved computer models for predicting the ultimate response capacity of civil engineering structures under earthquake excitation.

These models will be used for improving the current procedures for aseismic design and developing repair and strengthening techniques for damages or potentially vulnerable structures. Moreover, the results of large scale test conducted on the reaction-wall will be directly applicable to the updating of building codes, for example Eurocode No.8 for design of civil engineering structures in seismic areas.

CHARACTERISTICS OF REACTION WALL SYSTEM

REACTION WALL:		REACTION FLOOR:	
Width (m)	21	Length (m)	25
Height (m)	16	Width (m)	21
Max unit load (MN/m2)	5	Max unit load (MN/m2)	1
Bending Moment (MN.m/m)	5	Bending Moment (MN.m/m)	3

ACTUATORS:	Load (MN)	Displacement (m)
a)	1	± 1
b)	1	± 0.6
c)	0.5	± 0.3

It is an essential pre-requisite to validate fully the methods to be used in this programme because they have not been used to any great extent previously. In fact the JRC reaction-wall will be unique in Europe, comparable facilities existing only in Japan and in the USA. Similarly, the potential to simulate real earthquake loading via the "pseudo-dynamic test" (PDT) method is an exciting new technique still under active development.

3. HUMAN RESOURCES

Status December 1989

	Scient./ Techn. Staff	Admin. Staff	Authorized Recruitm. 1989		People who left in 1989		Grantholders		Visiting Scientists		Experts Seconded		Auxil. Agents
			ST	Adm.	ST	Adm.	present	addit expected arrivals	present	addit. expected arrivals	present	addit. expected arrivals	
Direction (1) Techn./Adm.Support & Marketing (2)	3 11	3 13	-	-	-	1	-	-	-	-	-	-	4
Thermodynamics and Radiation Physics	53	5	2	1	3	1	-	1	2	-	1	1	1
Process Engineering	83	5	3	1	3	-	10	-	3	2	-	3	3
Applied Mechanics	29	2	5	-	2	-	3	-	-	1	-	-	-
Nuclear Fuel Cycle	86	6	-	-	2	-	1	-	2	1	-	2	1
Nuclear Experiments	46	2	3	-	6	-	-	-	-	-	-	-	2
In Pile Tests	3	1	-	-	-	-	-	-	-	-	-	-	-
	314	37	13	2	16	2	14	1	7	4	1	6	11

(1) Includes QA/QC
(2) includes: purchase of materials, contracts, infrastructure planning, fissile material control and transportation

ANNEX A

List of publication

I. Contribution to periodical and monographs

FUSION

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A.Albermann, R.Diercks
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L.Meda, C.F.Cerofolini, R.Dierckx, G.Mercurio, M.Servidori, F.Cembali, Anderle, R.Canteri, G.Ottaviani, C.Clayes, J.Vanhellemont
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C. Addabbo, B. Worth
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Garching, 12. - 14.6.1989

C. Addabbo, A. Annunziato
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B. Worth, M. A. Viviande (CEA), J. S. Duffield, A. H. Scriven (CERL), C. G. Richards (UKAEA)
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F.Reiter : "Tritium production and extraction from liquid breeders" - Course and Workshop on Tritium and Advanced Fuels in Fusion Reactors - Varenna, Italy, September 6-15, 1989

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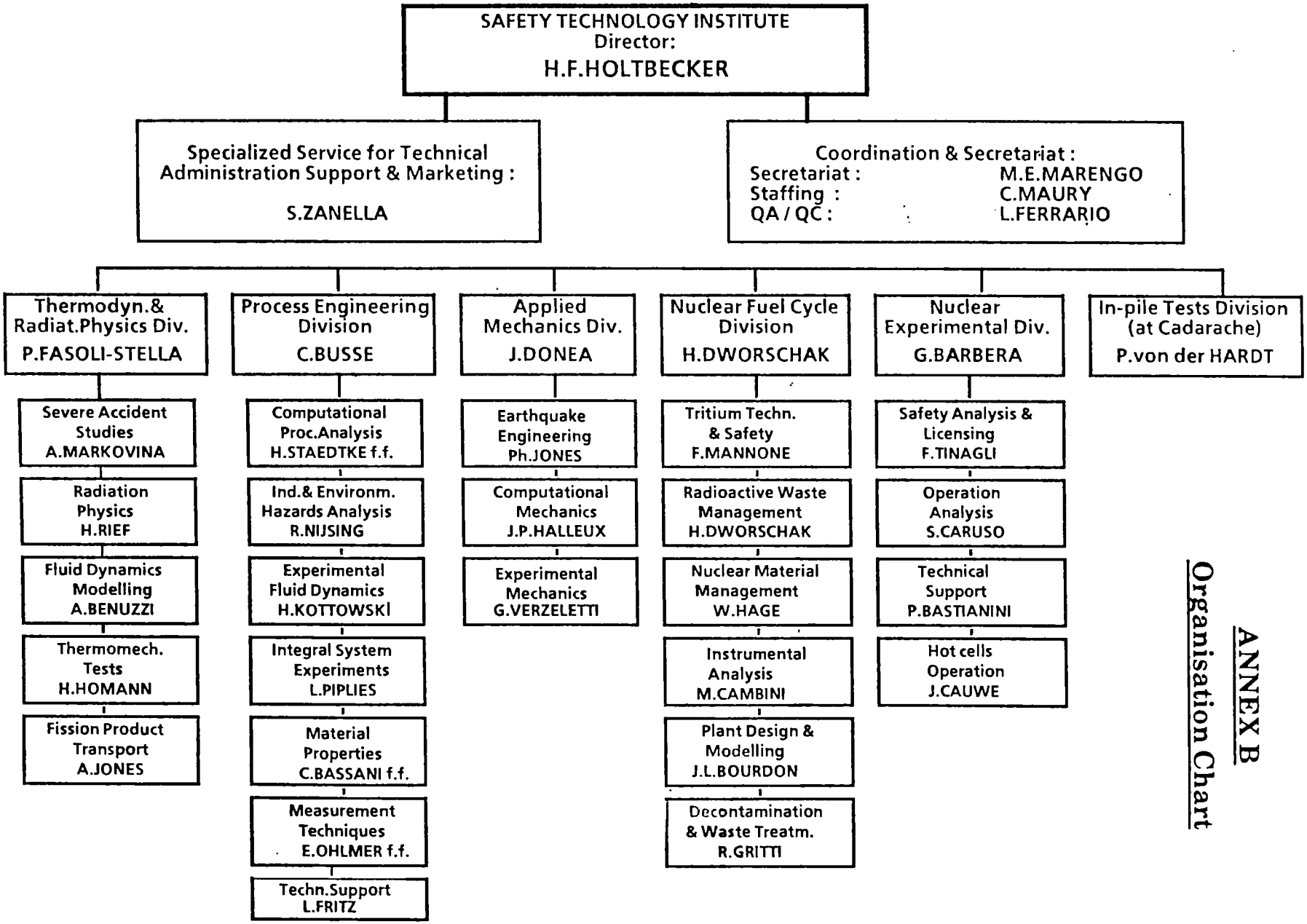
ANNEX B

Organisation Chart



ANNEX C

List of Tables and Figures

Reactor Safety

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Large Installations : ETHEL

ANNEX D

Glossary of Acronyms and Abbreviations

BMFT	Bundesminister für Forschung und Technologie
CATHARE	French Large System Thermohydraulic code
CEA	Commisariat à L'Energie Atomique
CONDIF	Computer Code Name (Molten pour behaviour)
DRUFAN	German Large System Thermohydraulic code
EAC	European Accident Code
EC	European Commission
EDX	Energy Dispersive X-ray
ETHEL	European Tritium Handling Experimental Laboratory
ENEA	Comitato Nazionale per la ricerca e per lo sviluppo dell'Energia Nucleare e delle Energie Alternative
ENEL	Ente Nazionale Energia Elettrica
FARO	Experimental Facility for Fuel Melting
FIRES	Facility for Investigating Runaway Events Safely
FISIM	FIres SIMulator
IAEA	International Atomic Energy Agency
IGSCC	Intergranular Stress-Corrosion Cracking
ITER	International Thermal Nuclear Experimental Reactor
KFK	Kernforschungsanlange Karlsruhe (FRG)
LDTF	Large Dynamic Test Facility
LMFBR	Liquid Metal Fast Breeder Reactor
LOBI	LWR off Normal Behaviour Investigation (installation)
LOCA	Loss-of-Coolant Accidents
LOFA	Loss-of-Flow Accidents
LWR	LIght Water Reactor
MDYN	Material Dynamics
MOX	Mixed Oxide Fuels
NDA	Non Destructive Analysis
NET	Next European Torus
OM	Optical Microscopy
PDT	Pseudodynamic Testing Methods

PERLA	Performance and Training Laboratory (Nuclear Safeguards)
PETRA	Facility for Treatment of Radioactive Waste
PISC	Programme for Inspection of Steel Components
PWR	Pressure Water Reactor
SCA	Shared Cost Activity
SEM	Scanning Electron Microscopy
UKAEA	United Kingdom Atomic Energy Authority
USNRC	United States Nuclear Regulatory Commission

European Communities - Commission

EUR 12885 — Annual Report 1989 - Institute for Safety Technology

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Luxembourg: Office for Official Publications of the European Communities

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Results achieved in the areas Reactor Safety and Industrial Risk, in which the Institute has a leading role, are reported.

Activity performed for the programs Fusion, Safeguards, Waste and Teledetection, in which the Institute makes available its expertise, are discussed.

The Commission Support activities are presented and indications are given of Third Party Work executed by the Institute.