EUR 7916 EN

COMMISSION OF THE EUROPEAN COMMUNITIES

Multiannual Programme
of the Joint Research Centre
1980 - 1983

1981
Annual Status Report

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REACTOR SAFETY
1981

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Projects:
1. Reliability and Risk Assessment
2. LWR Loss-of-Coolant Accident Studies
3. Primary System Integrity
4. LMFBR Core Accident Initiation and Transition Phase
5. LMFBR Accident Post Disassembly Phase

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1. INTRODUCTION

The JRC reactor safety programme involves theoretical and experimental activities to analyse accidents and their consequences for LWRs and LMFBRs.

The first project deals with the improvement and the application of methodologies for risk and reliability assessment. This activity involves the identification and modelling of accident sequences and events and the analysis of fault trees.
In this project, the implementation of a centralized data bank system (European Reliability Data System) is foreseen, which should provide the information needed for risk assessment studies.

In project 2 a major effort on LWRs is centered on the study of the loss-of-coolant accident following large, intermediate or small breaks of the primary circuit. These accidents are simulated out of pile in the LOBI facility which represents in a 1 to 712 scale a four loop primary coolant system of a 1300 MWe pressurized water reactor.

In project 3 a contribution is made to solve material problems and to provide data and calculation methods for end of life predictions of reactor components. It involves a contribution to the programme for the inspection of steel components (PISC) as well as the study of fracture and creep fatigue properties of stainless steel.

In the project 4 and 5 a deterministic approach is adopted to solve some problems of large hypothetical accidents in an LMFBR. The calculation tools developed concern sodium thermohydraulics in fuel element bundles, fuel coolant interaction, whole core accident analysis, containment loading and response and post accident heat removal.

Looking at this programme as a whole one notices that the constitution of a reliability data bank is a public service and it is expected to help the Community in its role of harmonization. On the other hand, the risk assessment studies provide the scientific and technical expertise which may support formulation and implementation of guiding rules. Also it helps to allocate available resources for research in nuclear safety.

Project 2 to 5 are solving questions of a «central nature», sometimes involving the construction of large installations and the execution of expensive tests using real reactor materials. In these projects the JRC also acts as a focal point for the development and verification of large computer codes.

2. RESULTS

Project 1: Reliability and Risk Evaluation

The Reactor Safety Study (Rasmussen, Wash 1400) and more recently the German Safety Study represented the first attempt of supplying a coherent framework for the safety evaluations of nuclear power reactors. Previous analyses were indeed mainly directed, either to the assessment of the reliability of the principal safety-related systems or to the evaluation of the consequences of major hypothetical accidents (Maximum Credible Accident, Design Basic Accident).

These studies enlarged the safety evaluations to a wider class of accidental chains, taking into account their probability of occurrence and their consequences.

The indications and the results of these studies and subsequent discussions pointed out the necessity of better investigating some major items, such as:

- adequate data base for the probabilistic evaluations,
- completeness of the analysis with respect to accident initiation and development, and
- completeness of the analysis with respect to accident initiation and development, and
- adequate treatment of the uncertainties of the physical and operational parameters governing the accident behaviour.

Furthermore, recent occurrences stressed the importance of the operational aspects of reactor safety.

The «Reliability and Risk Evaluation Project» of the JRC Ispra aims at giving a valid contribution to resolve the open problems mentioned previously.

The project has been structured into two main sub-projects, closely interdependent:

- European Reliability Data System. The implementation of such a centralized data system on the operation and reliability aspects of LWRs and their components in Europe, should supply the adequate basis for probabilistic calculations and it should also help in achieving an analysis completeness that only a direct comparison with experience can provide.
- LWR Accident Sequence Analysis, in which a methodology is developed for a more adequate and complete modelization of reactor systems and possible indicent sequence.

The European Reliability Data System (ERDS)

Lack of organized collection and exploitation of operational records in nuclear power plants leads to large uncertainty ranges in probability estimation for risk assessment studies. It is sufficient to recall the great uncertainty band that affects the results of the most known risk studies, and the strong criticism on this issue from various review groups to understand how decision makers are often left in such a state of uncertainty that they do not rely on probabilistic methods, even if these are judged to be the most adequate ones for safety analysis.

Moreover, without the implementation of a viable continuous feedback between power plant operating experience and risk assessment, it is impossible to assure the necessary control, improvement and updating of risk estimates; thus the confidence of a decision maker on the technical assessment of the risk, made by experts, can be also matter of considerable uncertainties.

These considerations are at the basis of the European Reliability Data System (ERDS): a centralized system whose aim is therefore not only the procurement of reliability data for probabilistic risk assessments but also the setting up of an organized collection of information for assuring necessary feedback from reactor operation. This will improve the confidence in the safety standards of the reactor operation and it will allow a correct «risk management» by the plant operators.

According to the type and level of the information, the ERDS has been structured into four main data systems. This structure is given in Table I.

Table I

Component Event Data Bank (CEDB)
The purpose of this system is the merging of data on reactor components failures as collected in national systems, such as SRDF, GRS/RWE, ENEL with reference also to the American NPRDS and the Swedish ATV.

Abnormal Occurrences Reporting System (AORS)
The purpose of this system is the collection of information as contained in national abnormal occurrence systems, as a service and a tool for safety analysis.

Operating Unit Status Reports (OUSR)
Collection, organization and dissemination of productivity and outages data from European power reactors.

Generic Reliability Parameter Data Bank (GRPDB)
Collection and organization of reliability parameters for similar classes of components, by exploiting also the early data bank structure set up by the JRC in 1972.

According to the lines defined at the end of 1980 the activity on Component Event Data Bank has been concentrated on the extension of the reference classification for system components and failures.

With the data already available (supplied by national organizations or collected in specific data campaigns) a provisional informatic structure has been designed and the different classification systems are being compared. Procedures for the statistical treatment of the data are developed in this frame. The program BIPEDES has been implemented for the estimation of constant failure rate, repair rate and unavailability.

The Abnormal Occurrences Reporting Systems (AORS) is being developed: the informatic capabilities (interrogation procedures and automatic management of data) and the quantity of available data have been extended. Abnormal occurrences data from Netherlands has been sent to Ispra and good possibilities exist to receive also FRG data. A proposal for a reference European Abnormal Occurrences formal is under study and has been extensively discussed with all the interested people in a meeting held at Ispra in November 1981.

The Operating Unit Status (OURS) involves at present the collection, organization and distribution of the operating data of European Nuclear Power Plants; the improvement of the present system for data collection in order to increase the use of the information; the design of a first informatic structure for the treatment of these data.

LWR Accident Sequence Analysis

The main aim is to develop suitable methods to improve the completeness and to reduce the uncertainties in risk analysis.

The following main lines have been followed for the methodology development:

- Event Sequence Consequence Spectrum: an approach to model accident sequences by taking into account the interaction of the system behaviours with the physical and time progression of the accident itself;
- Response Surface Methodology and its application: an approach to study complex safety computer codes and to substitute them with simpler expressions;
- Fault Tree Analysis: linking its development to the implementation of component models which have to be coherent with the data collection procedures in the European Reliability Data System.

ESCS (Event Sequence and Consequence Spectrum)

The DYLAM (Dynamic Logical Analytical Methodology) code for the ESCS technique has been further improved to increase its efficiency in analysing in one run a set of top conditions which are well quantified values of physical variables such as temperatures between a given interval or stresses above a given threshold.

The technique has also been applied to the dynamic modelization of operator behaviour. In the study case, the operator must identify the occurrence of a LOCA and its location, attempt to isolate the break and actuate the high pressure injection system. The use of the ESCS technique makes it possible to evaluate critical operator response times and overcomes the drawbacks of static event tree representations.

Response Surface Methodology (RSM) Development and Application

The application of RSM methodology to calculation codes developed for reactor safety studies strongly reduces computing times and makes their implementation in probabilistic risk assessment practicable. Emphasis has been given to prepare an RSM hanbbook. In addition, the methodology has been applied to the RELAP code describing LOCA phenomena in the primary circuit and to the code ALMOD describing the whole system of loops in a nuclear plant.

Fault Tree Analysis

The seventh of a series of codes for fault tree analysis, SALP-MP has been released. MP stands for multiphase and indicates that the code handles systems whose function changes at a given time allowing the system to perform different tasks in subsequent mission intervals.

On-going activities concern the development of the SALP-NOT code for the analysis of logical structures (AND-OR-NOT trees), the construction of component models and the computer aided fault tree design.

Project 2: LWR Loss-of-Coolant Accident Studies

LOBI project

The LOBI project is performed in the framework of an R & D contract with the Federal Republic of Germany. A small part of the tests are performed for the FRG exclusively, all others are available for the Community as a whole. The test facility became operational in December 1978. The main objectives of this project can be summarized as follows:

- The design, construction and operation of a two-loop test facility simulating a four-loop primary cooling system of a 1300 MWe PWR reference plant with respect to its thermohydraulic behaviour during a loss-of-coolant accident.
- The performance of loss-of-coolant experiments by simulating tube ruptures of various break sizes (large, small, intermediate) at three different positions within the «broken» LOBI loop with the aim of investigating the influence of the thermohydraulic behaviour of the individual primary cooling system components on the course of a loss-of-coolant accident.
- The experimental investigation of the LOBI pump operation characteristics and discharge nozzles under two-phase flow conditions.
- The application of the experimental results for checking and improving blowdown computer codes and associated theories used for the safety analysis of LWRs.

In the reporting period the PREX exercise has beem terminated in which sixteen organizations participated in the prediction calculations of the first LOBI test. One of the main objectives of this experiment was to study the departure from nucleate boiling (DNB) of the heater rod bundle under large break LOCA conditions. As a conclusion of the comparison between predicted and measured data the following areas have been identified where existing blowdown codes should be improved:

- calculation of the critical mass flow
- phase separation in vertical channels due to gravitational forces
- release of the stored heat from the loop structural materials
- two-phase pump behaviour
- dry-out and film boiling heat transfer

The LOBI test programme in 1981 included a total of 10 tests in which down-comer gap width, ECC water injection mode, break size and location have been varied.

A typical result of three 2 x 100% cold leg break tests with different ECC injection modes is shown in Fig. 1. The Figure shows the cladding temperature (axial and radial distribution) during the transient for the times 13 secs. and 85 secs. after break. A complete quenching of the cladding has been reached after 85 secs. only in the case of combined cold leg and hot leg ECC injection.

The LOBI installation will be substantially modified in 1982 for the execution of small break tests. This will still increase its flexibility and allow eventually the execution of special transient tests.

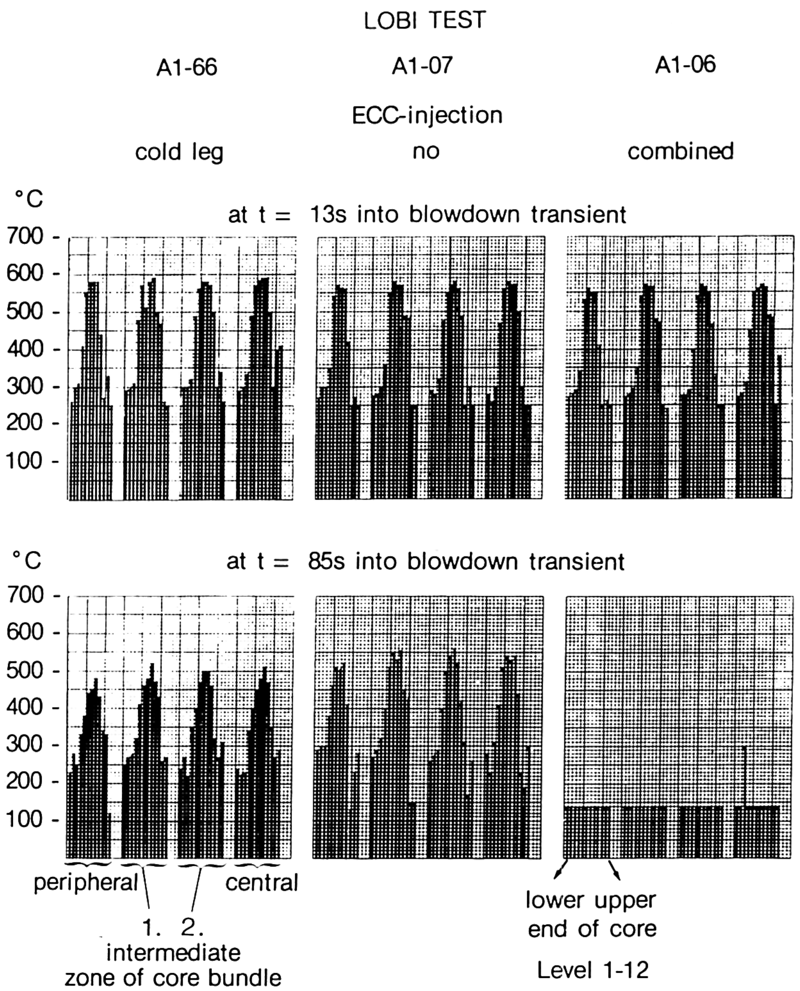


Fig. 1 Cladding temperatures at t = 13s and t = 85s after rupture

Project 3: Primary System Integrity

Reactor operating experience and safety requirements prove the need for an increased effort in materials research, particularly in the fields of defect detection, sizing and propagation. The JRC contributions in this research are:

- Failure detection in LWR primary circuit components,
- Models development to assess probability of failure of LWR primary circuit components,
- Fracture mechanics and creep crack growth related to LMFBR materials

Failure Detection in LWR Primary Circuit Components PISC II (Programme for Inspection of Steel Components)

The PISC I results have shown the need for improvement of reliable detection of faults. Alternative ultrasonic techniques, some of them already used in Europe, are systematically applied in the PISC II round robin exercise. In addition, a number of studies are conducted to optimize defect positions and geometry, characterize the measuring equipment, study the effects of vessel cladding and residual stresses. The JRC in its role as operating agent on behalf of CSNI is coordinating the activity of about 60 teams in 15 countries with the help of its NDT laboratory and of four working groups. Four plates will be used in this exercise, two of the present longitudinal welds and two have nozzles as shown in Fig. 2.

According to the present time schedule this activity will terminate by the end of 1984.

Primary Circuit Components Life Prediction

In order to get maximum operational safety of nuclear power plants, all reactor components are extensively controlled prior to and during service. Using the data periodically supplied by these inspections and estimating the load functions for a given time period, an updated calculation of the reliability (in terms of residual life) of pressure vessels and piping systems is provided. Due to the still limited experience in this field, an effort is being made to demonstrate the capabilities of presently available methodologies in predicting the behaviour of cracks

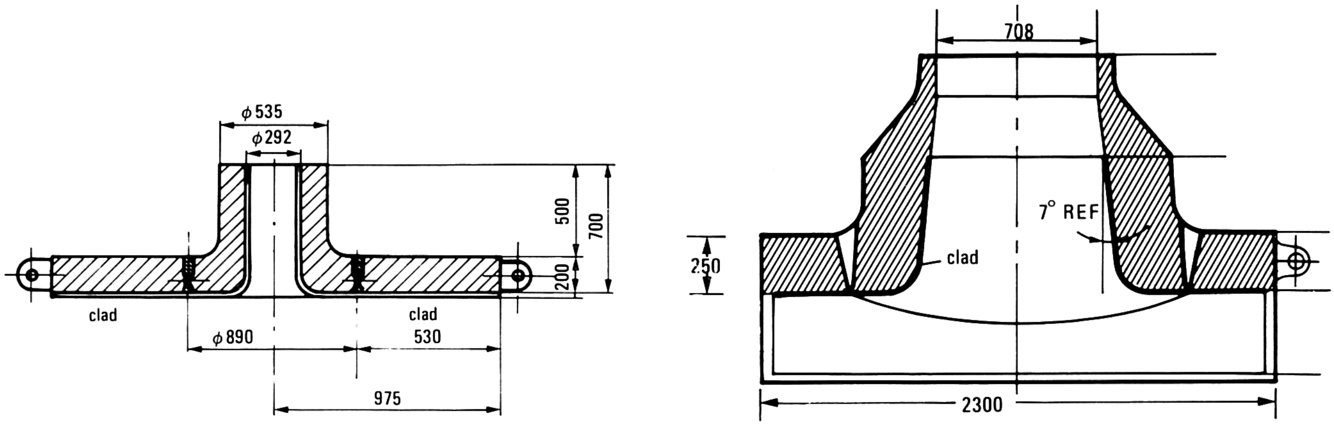


Fig. 2 PISC II nozzle plates

in 1:5 scale pressure vessel models. Three vessels as shown in Fig. 3, have been fabricated and work is in progress to install them in an appropriate laboratory for introduce fabrication defects only e.g. lack of melt flow, lack of penetration flow, solidification crack.

An activity on assessment of the structural reliability of the piping system is continuing in the frame of collaboration contracts. A statistical analysis of about 1000 data on piping failures has been carried out at the JRC in the frame of the CSNI Working Group, where the results of this exercise have been presented.

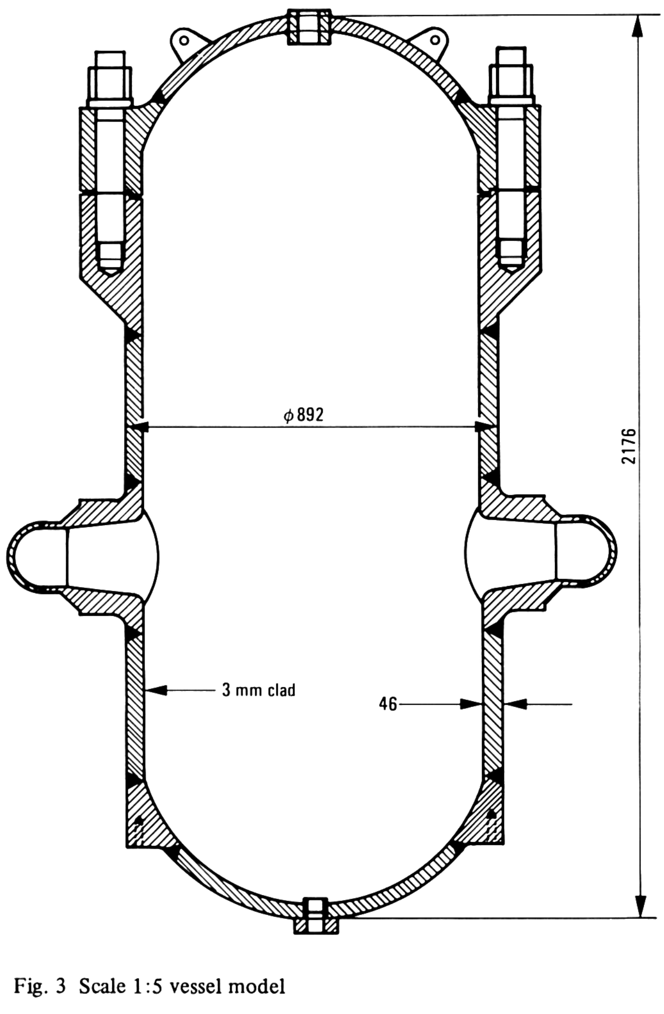


Fig. 3 Scale 1:5 vessel model

Fracture Mechanics and Creep Studies related to LMFBR Materials

The general objective of fracture mechanics is to assess the significance of defects under the aspect of structural safety. The JRC experimental activities are mainly directed to analyse materials previously irradiated in the HFR Reactor at Petten.

To define an experimental methodology for testing irradiated specimens in hot cells, screening tests on the unirradiated IDEAS 1 reference specimens have been made in 1981. For each testing temperature (350°C or 550°C), thermal aging (26 or 78 days at the afore-mentioned temperatures) and base, weld or heat affected materials, the J data have been obtained in the form of a J resistance curve. The data analysis procedures and the preparation of the experiments on irradiated specimens in nearly concluded.

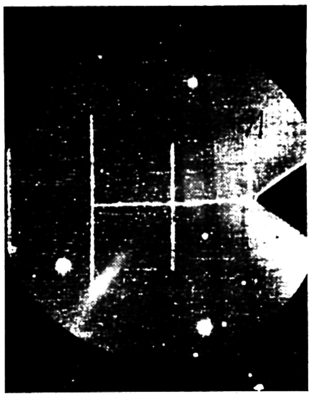
The analytical activity was devoted to:

1) analysis of the elastoplastic fracture mechanics round robin calculations (jointly organized by the Welding Institute (U.K.) and the JRC).
2) the improvement of FE techniques for the analysis of tests on irradiated specimens.

An important problem in nuclear plant design is the assessment of the role of creep deformation on different components. The maximum load which can be sustained by a component at the end of its working life has to be evaluated: obviously the magnitude of this loading will be affected by the material deterioration. Cyclic conditions induced by large thermal loadings, introduce fatigue mechanisms which are affected by the presence of the creep; the imperfections introduced during the manufacturing are related to the growth of cracks. The creep crack growth studies on austenitic stainless steels have been continued with tests on compact specimens (CT). Fig. 4 shows the crack length increase measured with a telemicroscope having 100 μm accuracy through a glass window on one side of the furnace.

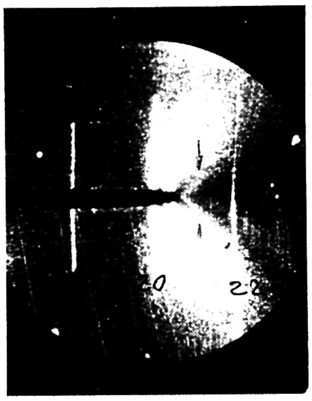
The results of an extensive programme of tensile tests at temperatures between 773 and 1073°K and strain rates between

$5.10^{-5}s^{-1}$ and $5.10^{-2}s^{-1}$ on AISI 316 stainless steel specimens have been terminated in 1981. Constitutive equations describing plastic behaviour at high temperature have been established.

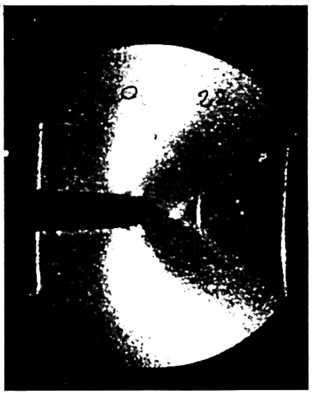


t = 2.3 hr
a = 20 mm

t = 42.8 hr
a = 20.2 mm

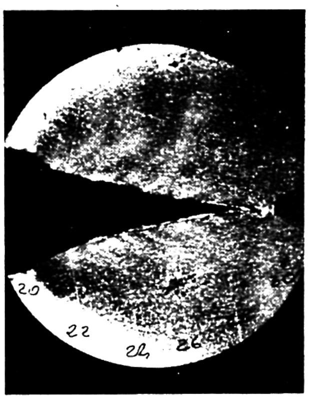


t = 44.1 hr
a = 20.8 mm



t = 66.8 hr
a = 21.2 mm

t = 87.9 hr
a = 24.1 mm



t = 99 hr
a = 29 mm

Fig. 4 Six successive pictures of crack growth at T = 550 °C

Project 4: LMFBR Core Accident Initiation and Transition Phase

This Project includes three sub-projects:

— Liquid Metal Boiling Studies
— Fuel Coolant Interaction (FCI)
— European Accident Code (EAC)

Liquid Metal Boiling Studies

The objective of these studies is to gain quantitative data on the coolant behaviour in a fast reactor in case of anomalous function provoked by blockages in the core bundle, flow run-down due to pump failure or power excursion.

During 1981 the activity was concentrated on the bundle experiment: the tests with the 12-pin test section (Fig. 5) with grid spacers was in a short test assembly (performed in collaboration with CEA-Cadarache) was successfully terminated. The foreseen boiling experiments had to be interrupted due to an accident which destroyed the test section and the instrumentation. The preparation of the new test section and the repairing of the test facility is in progress. The data acquisition and processing system has been tested and adapted for the boiling experiments.
The UK SABRE-3b code, now implemented on the JRC computer, will be used for precalculations and for test analysis.

Fuel Coolant Interaction (FCI)

The analytical and experimental investigation on possible molten fuel-coolant violent interactions (FCI) and on the consequences of vapour explosions continued in 1981. The main

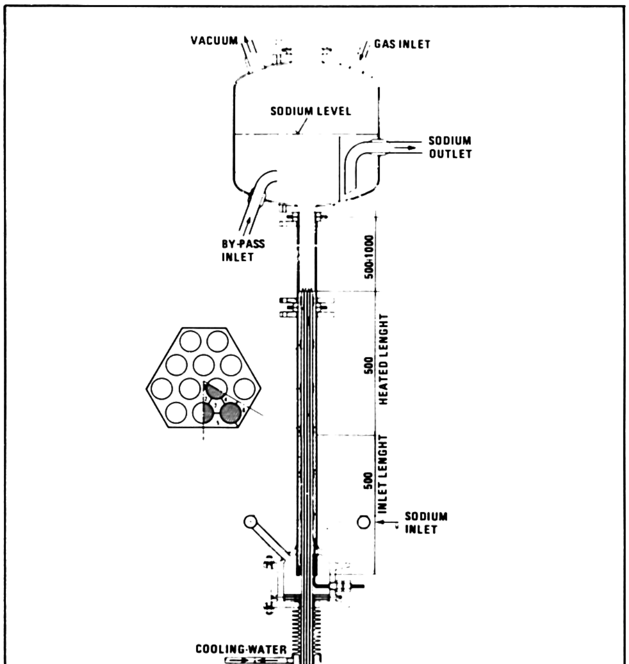


Fig. 5 Longitudinal cross section of the test bundle

results on thermal detonation, taking into account non-uniform temperature and mass distribution, may be summarized as follows:

— A non-uniform temperature distribution in the reaction zone provokes a reduced increase of entropy and consequently increases the available mechanical energy;

— The non-uniform temperature distribution allows for a wide «range» of critical initial mixing conditions of two materials which potentially may lead to an explosion. This explains why self-sustained explosions could easily be generated in some experiments.

The shock tube experiments have been evaluated and the conversion rate of thermal into mechanical energy has been determined, taking as a reference the heat really transferred to the coolant. Conversion rates as shown in Fig. 6 are obtained. The figures scatter largely but remain substantially below «Hicks and Menzies» values.

The simulation experiments with NaCl-H_2 vapour explosions at different pressure, performed in collaboration with USNRC and BMFT have been terminated and the complete analysis is being published.

Finally, stratification experiments with melt alloys are under way: preliminary results confirm that in case of a melt flooded by water, conditions for violent mixing can be established.

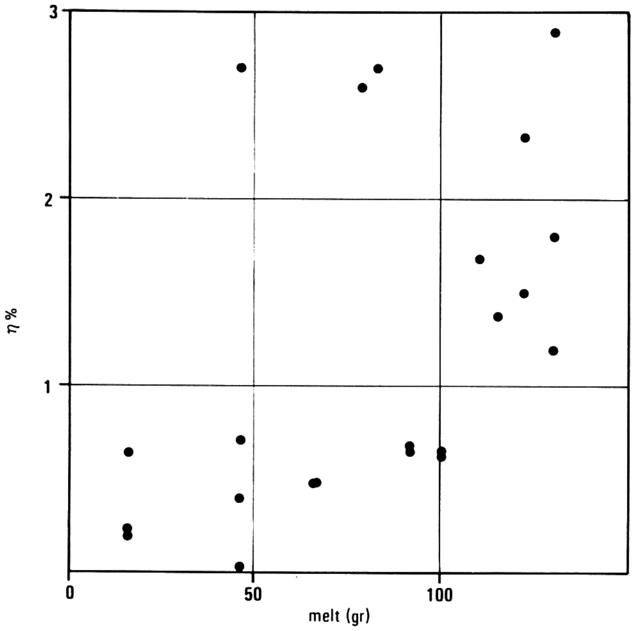


Fig. 6 Conversion rate of mechanical work to heat stored in interacting melt vs melt fragmented

European Accident Code

EAC is a modular system of computer codes allowing the description of the different phases of hypothetical whole core accidents. The pilot version of EAC is operational and available or request. Comparison with existing programmes has shown that EAC is competitive both from the point of view of physical results and computer time.

The system is designed to be able to receive modules developed and/or to be developed in different countries or at the JRC, for the description of the physical phenomena involved in different accident scenarios. So comparison of different models and numerical approximations are possible by means of the EAC.

A 1D two-phase flow module using the finite element method has been developed at Ispra: it is intended to extend this module to treat fuel motion in the pin after clad failure, fission gas and fuel motion in the channel.

During 1981 comparison of the different hydraulic modules available in the EAC have been performed to evaluate the effects of different sodium boiling modelling in various accidental situations.

As in the past, the EAC was successfully used for the international code comparison organized by the WAC Groups.

LOF and TOP situations are being considered. The general conclusions of a particular comparative TOP calculation are shown in Fig. 7 and 8. All programs showed a good agreement for what concerns the net reactivity time history, whereas the detailed analysis of the fuel pin behaviour shows discrepancies due to the different levels of development of models concerning fission gas release and thereby cavity pressurization.

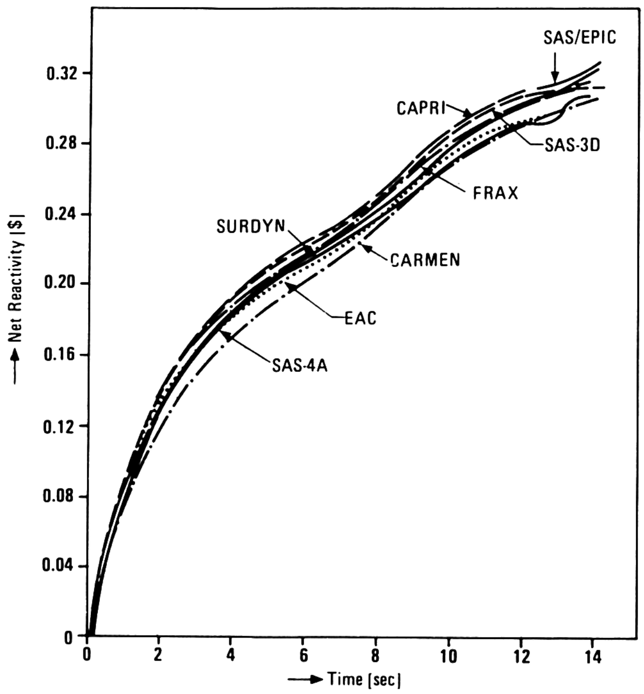


Fig. 7 Net reactivity vs time after accident initiation

Project 5: LMFBR Post-Disassembly Phase

This Project includes several activities related to the study of phenomena taking place as a consequence of a core disruptive accident. The main chapters are:

— Multiphase Multifluid Hydrodynamics related to CDAs
— Dynamic Structure Loading and Response
— Material Dynamic Properties
— Post Accident Heat Removal

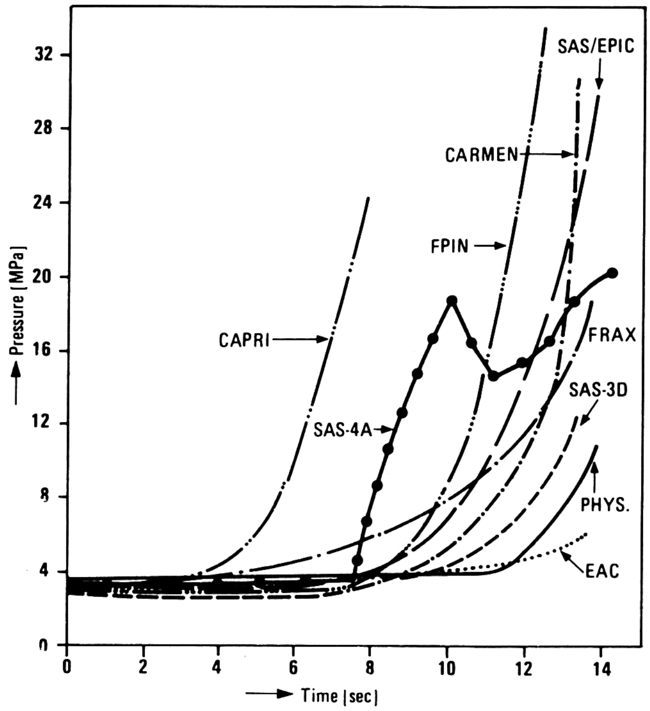


Fig. 8 Molten cavity fission gas pressure vs time after accident initiation

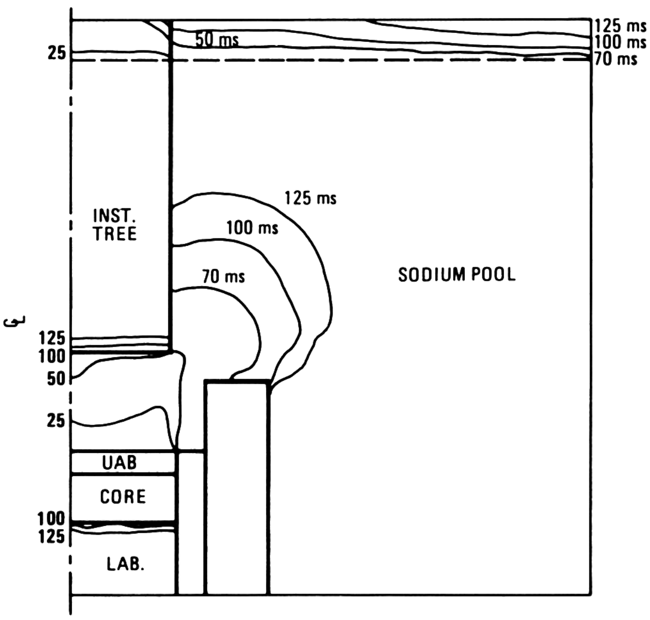


Fig. 9 Development of core bubble and cover gas volume in a pool type reactor

Multiphase Multifluid Hydrodynamics related to Core Disruptive Accident

During 1981 the study of the models able to describe in more detail the complex physical phenomena taking place during the post-disassembly phase has been continued. The objective of this study is a more realistic evaluation of the different accident scenarios and in particular of the mechanical energy released to the reactor structures during a CDA. The activity, still in a preliminary phase, has been mainly focused on the analysis and the use of the US SIMMER-II code available at Ispra. The analytical study of the multiphase conservation equations allowed interesting conclusions and indications for the improvement of modelling and numerical techniques. A first proposal and conceptual design of a series of simple validation experiments has been prepared in close contact with the national experts.

By means of SIMMER-II some interesting calculations have been performed, in particular the IT10 COVA test and a post-disassembly calculation for a pool type reactor (Fig. 9).

Dynamic Structure Loading and Response

The objective of this activity is the study of the response of containment structures during a significant mechanical energy release.

The COVA and COVAS programmes have been designed for the development and validation od 2D hydrodynamic-structural codes applied to the study of the response of the vessel, internal structures and subassemblies when a well-defined amount of energy is released. The COVA experimental tests have been completed in 1981; the SEURBNUK calculations of the last COVA tests will be terminated during 1982.

Some particular tests are calculated also by means of EURDYN/1M code. Fig. 10 shows on example of predicted and measured plastic strains of the internal vessel in IT12.

In 1981 improvement and modifications of models and algorithms in the SEURBNUK code (jointly developed by UKAEA and JRC) allowed to calculate the COVA experiments for a longer period of time and the treatment of more complex geometries. The detailed analysis and comparison of all COVA tests is under way and will be completed in 1982: problems seem to be open in the interpretation of the roof impact predictions; the need of additional research on material properties in dynamic conditions seems to be confirmed.

In the EURDYN-1M code (developed at the JRC) the simulation of axial flow for the analysis of subassembly clusters has been introduced and will be validated by means of the future COVAS tests on clustered hexcans. As a part of the EURDYN code series, a finite element programme has been completed for the analysis of 2D fast transient structural problems in the large strain regime. One of the objectives of this new code is to verify the validity range of the usual small strain assumption, in particular in the case of loaded hexcans.

Material Dynamic Properties

Classical elastoplastic flow theory together with isotropic and kinematic hardening is currently used in the fluid-structure coupled codes Seurbnuk and Eurdyn mentioned above.

Strain rate effects are idealized in that only the yield stress and eventually the hardening at given strain are strain-rate dependent. Strain rate history effects are taken into account in a purely qualitative manner.

It appears that further experimental and theoretical work is required to better understand the specific influence of the strain rate history effects.

To improve direct techniques and to overcome the difficulties encountered in defining the parameters involved in a specific

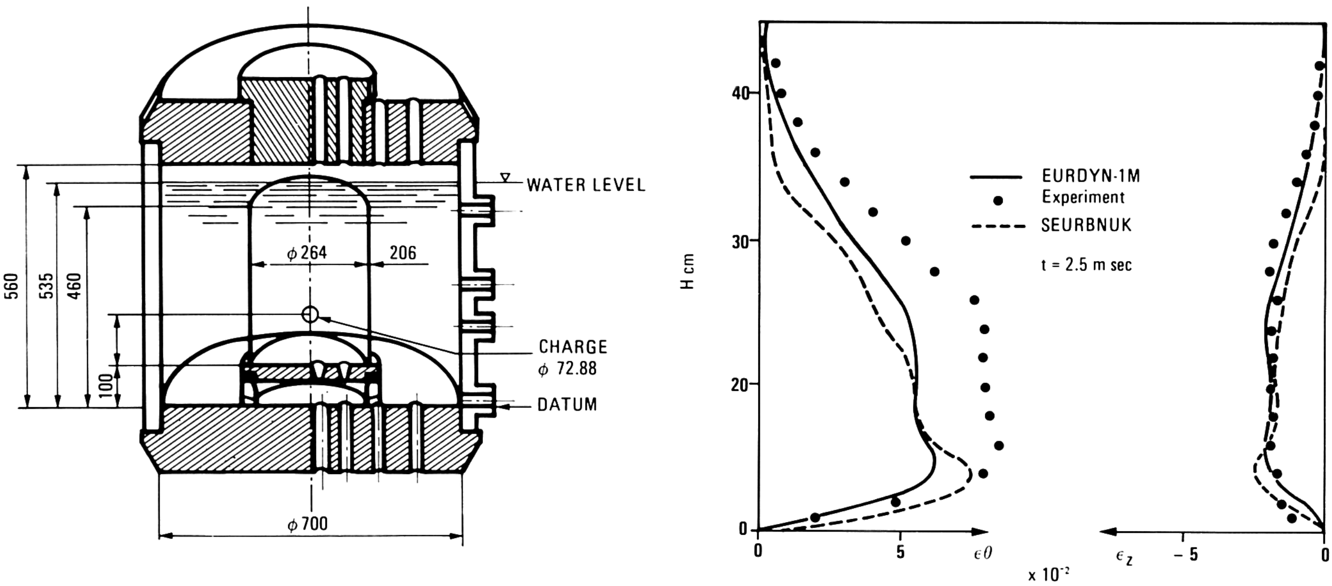


Fig. 10 IT12 lay out and comparison of innertank strains

constitutive law the application of system identification techniques, using the Pontryagin theory, was successfully initiated for a simple 1D test case. The excellent results show that this technique can be applied to more complex constitutive laws.

The experimental activity has been continued in several directions.

- Efforts have been devoted to the preparation of the experimental set-up in order to determine the dynamic mechanical properties of AISI 316 cold worked 20% and of Nimonic alloy FE 16 irrariated to very high dose, materials used for the construction of subassemblies (collaboration with CEA). Under preparation are the dynamic tensile tests on AISI 316 and AISI 304 K used for LMFBR vessels, previously submitted to low cycle fatigue (collaboration with PNC Japan).
- Definition of static and dynamic mechanical properties of the material (AISI 304) taken from a hemisphere of a COVA tank. This analysis showed important differences in the dynamic mechanical properties of the materials used for the hemisphere, for the cylindrical walls and for the weld region.
- The construction of the high load biaxial machine (5 MN), based on the loading principle of the modified Hopkinson bar, in the version adapted to uniaxial tests was almost completed. This machine is designed to study the behaviour of large specimens of steel, with creep and fatigue defects, real size welds. In the future the machine will be used also for the study of the dynamic behaviour of concrete.

Post Accident Heat Removal

This is one of the projects in the Reactor Safety Programme which has growing importance in the 1980-83 JRC activities.

All hypothetical severe accidents normally end with dispersal of fuel from the reactor cavity.

The main goal of the JRC studies in the post accident phenomenology is the analysis of coolability of core debris in appropriate catcher systems within the pressure vessel whose mechanical resistance has also to be assured.

The PAHR activities fall into two categories: out-of-pile studies and in-pile programme with the related supporting activities.

Out-of-Pile Studies

The complexity of post accident heat removal and fuel coolant interaction problems asks for experiments to be performed under realistic accident conditions using real reactor materials. A PAHR out of pile test facility is being built at JRC Ispra.

The main parts of this facility are a large fuel melting furnace FARO (100 kg of UO_2 are molten by Joule heating), connected via a release channel to one of three sections (BLOKKER, TERMOS, FRAGOR).

The FARO furnace has been assembled in 1981 and in the near future a series of experiments will be performed in which a molten pool of 12 liters UO_2 (100 kg) will be imbedded in 500 kg of UO_2 powder, the molten pool and the powder being separated by a crust. Temperatures and cooling fluxes will be measured. During 1982 and 1983 the BLOKKER test section now under construction will be used for freezing and blockage experiments in argon atmosphere. Channels or fuel element clusters can be introduced in the test section where freezing and blockage formation will be measured (y-ray absorption techniques) to verify the analytical models.

Melting front advance and catcher plate performation will also be studied in the collection vessel at the bottom of BLOKKER where different structural materials will be exposed to molten UO_2 jets.

The detailed design of the THERMOS test section (for tests on thermal and mechanical load on supporting structures, UO_2 particulate formation and settling) has been concluded in 1981 and the construction will be initiated in 1982.

Calculations are under way for the design of the FRAGOR test section in which violent interaction of large mass of fuel and coolant shall be produced.

Different codes are being developed for the description of the different phenomena: MACONDO (finite difference) and CONDIF (finite element) for heat conduction and convection, the code JOULE for the generation of a molten pool, the code PLUG for freezing and plugging description.

In-Pile and Supporting Activities

There is some experimental evidence that upon contact of molten fuel and sodium, the fuel will fragment forming particulates having a medium size of about 150 μm.
These particulates will settle on core catchers built in the bottom part of LMFBR primary containment vessels.
The objective of JRC studies is to provide criteria for the adequacy of particulate bed cooling by sodium within the bed and to develop models for the eventual transition of the bed into a molten pool following sodium boiling and dry out.

All phenomena of potential importance are considered: debris bed cooling by conduction, single phase convection or two phase conversion boiling of the coolant, particle dry-out sintering and remelting, upward and downward heat transport flux, onset of different heat transfer modes as a function of bed composition, bed geometry, heat source density, bulk sodium temperature.

The programme includes in pile experiments at SANDIA (US) started in 1980 and a European series of tests at Grenoble (Melousine Reactor) and MOL (BR2 Reactor) initiated in 1981. The European and US programmes are complementary. In the European programme, particle bed diameters and hights are larger and extensive remelting of fuel and steel will be investigated. An extensive out of pile back up programme is performed with the scope to provide the data necessary for pretest calculations and to prove the feasibility of each test. It comprises the following main items: fuel particle sintering and crust formation, PuO_2 particle segregation, migration of stainless steel in a UO_2 particle bed as a function of temperature and temperature gradients. All these phenomena change the bed geometry and its thermophysical properties.

An important part of the back up programme is performed in the JRC laboratories, in particular compatibility studies. Crucible development and phenomenological studies on specific aspects are funded by the JRC in different European laboratories.

The European in pile programme includes distinct phases each corresponding to two tests. In the first phase, bed temperatures slightly below melting temperature of steel will be reached. The local time-dependent progression of the dry-out zone will be investigated. In phase two, extended dry-out in the presence of melting steel at temperatures reaching 1800°C will be studied. The melting of stainless steel and UO_2 will be analysed in phase three.

3. CONCLUSIONS

1981 has been the second year of the 1980/83 pluriannual programme. Interesting progress has been made in all five projects which can be summarized as follows:

— The LWR reactor safety research is directed to substantiate results on probabilistic risk assessment by applying developed methodologies to well chosen reactor plant systems and by implementing national data in the European Data Bank.
The LOBI-LOCA investigations continued successfully. Comparisons of different emergency core cooling injection systems have been made and results are extensively analysed by JRC and national organizations.
In order to assure the integrity of the reactor loops, a broad materials programme is being performed, which has also lead to the execution of PISC II, an international exercise of non-destructive testing of steel plates and nozzles.

— In the field of LMFBR safety progress concerns primarily the release on request of the European accident code and the containment code development and validation programme which is almost terminated for 2D situations.
JRC successfully promoted cooperation with and between member countries which allows for quicker progress through a division of tasks and mutual help amongst interesting organizations.
The European PAHR programme concentrates on in-pile and out-of-pile tests using real reactor materials. It is integrating national efforts and is designed to prove the concept of in-vessel coolability of core debris.

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