EUR 7249 EN

COMMISSION OF THE EUROPEAN COMMUNITIES

**Multiannual Programme
of the Joint Research Centre
1980 - 1983**

1980 Annual Status Report

Reactor safety

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REACTOR SAFETY
1980

Research Staff: 308 *
Budget: 28,378,000 ECU *

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

* Including Super-Sara Project, Reperted Under Separate Cover

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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.

Projects 2 to 5 are solving questions of a "central nature", sometimes involving the construction of large installations and the execution of expansive 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 I: 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

— 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 incident sequences.

Table I gives a view of the project structure and of the logical relation between the different activities.

The European Reliability Data System (ERDS)

Lack of organized collection and exploitation of operational records in nuclear power plants leads to large uncertainty ranges for accident probability estimation; therefore, decision makers are left in such a state of uncertaintly that they sometimes do not rely on probabilistic methods, even if these are judged to be the most adequate ones for safety analysis.

In 1977, the JRC has started a feasibility study for implementing a centralized European Reliability Data System (ERDS) that would collect and homogenize all information of components, systems and reactor operation behaviour for LWRs in the European Community. ERDS will be based on data collected at power stations and also on the existing national data files.

The ERDS structure consists of four main data systems according to the type and level of information. This structure is given in Table II.

The positive result of the feasibility study have constituted the basis for the start of the 1980-83 programme. The preliminary activity in the ERDS has involved the problem of the information harmonization, that is the set-up of reference classifications (both for events identification and component-system classification) and has led to wide-scope compatible classifications being also considered by other international organizations.

Another important point was the recognition of the importance of structuring data acquisition systems according to the specific use in the reliability analysis and in the same time of the effort needed in order to maximize the use of the information content of the data.

FLOW CHART OF RISK AND RELIABILITY PROJECT ACTIVITIES

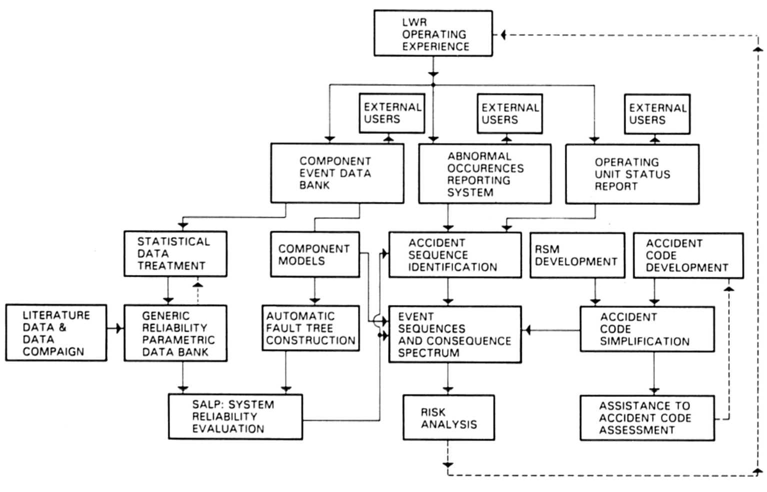


Table I

Table II - European Reliability Data System (ERDS)

Component Event Data Bank (CEDB).

The purpose of this system is the merging of data on reactor component failures as collected in national systems, such as SRDF, GRS/RWE, ENEL with reference also to the American NPRDS.

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 (OURS).

Collection, organization and dissemination of productivity and outages data from European power reactors.

Generic Reliability Parameter Data Bank.

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.

Certainly, not all the problems connected with this information system have been solved; we are facing at the moment a situation in which the internal structure of various national component reliability data banks is still under testing and revision, or is sometimes lacking; difficulties are furthermore encountered in the acquisition of data.

Anyway, the consensus reached on reference classifications leads us to predict the overcoming of these problems.

LWR Accident Sequence Analysis

One of the most important activities performed at the JRC in the past years in certainly constituted by the analysis exercise accomplished at the Obrigheim power plant (KWO). This activity concluded in 1980 has shown how the operation's historical records and more generally, the experience of a particular power plant, can be optimally exploited for complementing and validating risk assessment methods.

The development and the implementation of component data acquisition systems cannot follow independently of the modelizations required by systems reliability analysis and by the improvement of the description of the real logical and probabilistic behaviour of the actual systems.

To this aim the existing fault-tree series of codes SALP has been extended. The SALP-MP (Multi-Phase) code, capable of analysing also phased missions accomplished by reparable systems has been released during 1980.

A major problem for incident progression evaluation is to satisfactorily take into account the dynamic aspects of the random interaction between the "physics" of the transient and the "logic" (states) of the systems. Indeed, current techniques separate the system fault analysis from the real dynamic development of the accident. Actually, an accident moves in a specific direction according to the values assumed by physical parameters such as temperature, pressure, etc. (which control the intervention of protection and/or mitigating systems) and to the occurrence of events, such as lack of intervention of the demanded systems, delays, partial failures, etc., which can occur at random times. Indeed, the principal drawbacks of the event tree techniques are due to their practical impossibility to deal with time-dependent incident development and uncertainty distributions on physical and operational parameters. Moreover, it is very difficult to ascertain the completeness of an analysis based on the construction of a limited number of accidental paths and initiating events.

The Event Sequence and Consequence Spectrum (ESCS) methodology developed at Ispra allows to generate all the physically possible indicent sequences with their probabilities giving also a complete spectrum of the possible consequences consistent with the assumptions made for the system components. Each generated sequence is the result of dynamic and random interactions between the physics of the transient and the time evolving aspects of the component states.

A considerable effort has also been devoted to the study of specific problems posed by the application of the Response Surface Methodology (RSM) to the exploration of nuclear reactor safety codes, when a direct use of the code is impracticable for probabilistic assessments, due to the costs of the computing time. Together with the resulting methodological improvements, this study is aimed at producing a computer program library and handbook for the safety analyst, covering all aspects of the methodology.

This development is also strictly linked to real case applications: by using the routines already included in the RSM library during 1980 some relevant studies have been achieved: Analysis of Anticipated Transient Without Scram (ATWS) consequences in PWR reactors, analysis of loss-of-coolant accident experimental simulation in LOBI and assessment of structure behaviour due to seismic occurrences.

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.
A schematic view of the facility is given in Fig. 1.

During the reporting period nine large break loss-of-coolant experiments were successfully performed looking especially at the effect of the influence of the heating power input and the emergency core cooling injection mode and rate.

The first LOBI blowdown test was used to perform test prediction calculations (LOBI PREX exercise) by different organizations (16 participants from Europe and USA). The results of this international standard problem calculation exercise will be discussed and analysed by the participants of a special workshop in January 1981. A typical example of the results is given in Fig. 2 showing the comparison between the experimental data and the prediction calculations for the mean heater rod temperature in the mid level of the heated length as function of blowdown time.

More than 50 similar comparison plots for the main thermohydraulic parameters have been prepared by the JRC which participated in this exercise with test prediction calculations using RELAP4/MOD6 code. The 16 participants executed the calculations with seven different codes.
In addition to the 9 large break tests of the foreseen test matrix three small break scoping tests with rupture sizes, of 10%, 1% and 0.4% have been performed.

The evaluation of the two-phase characteristics of the LOBI pump investigations as well as the evaluation of the LOBI discharge nozzle calibration test results were continued. This information is important for the evaluation of the LOBI test results and hence for the improvement of the blowdown computer codes.

The revision of the LOBI experimental programme, based also on the lessons learned from the TMI-2 reactor accident and on the experience gained during the first operation period led to significant modifications of the facility especially with a view to its improvement for small break tests.

Design and performance specifications for new steam generators have been established, a high pressure emergency core cooling injection system and an auxiliary feedwater system will be added; the measurement instrumentation as well as data acquisition and process control systems are being adapted. Concluding it can be stated that the successful operation of the LOBI facility during the reporting period ended up with interesting results which are now evaluated and analysed.

The test facility itself has proven to be flexible enough to meet all unforeseen changes of the initial test matrix.

The LOBI project represents a valuable support of the JRC in PWR safety research and constitutes also a significant worldwide contribution of the Community in this field.

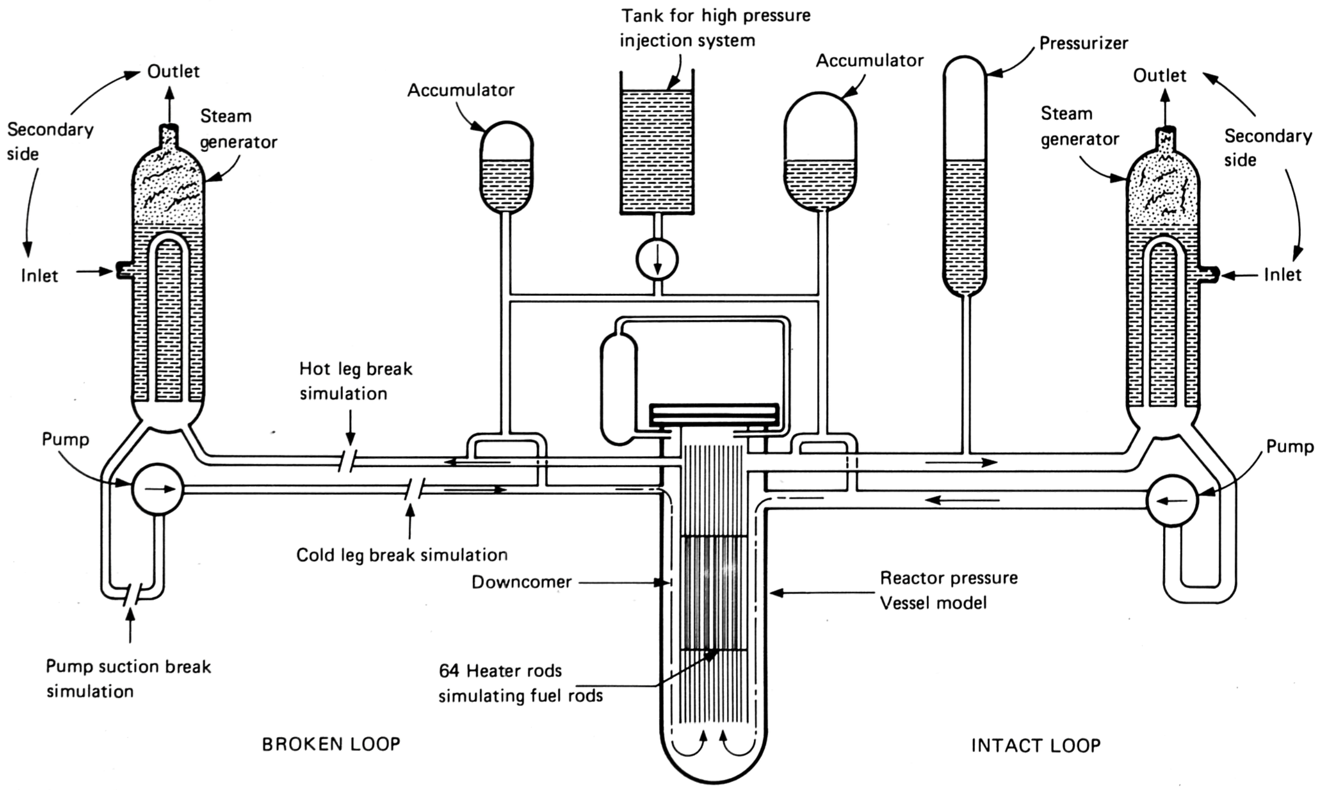


Fig. 1. LOBI test facility (schematic)

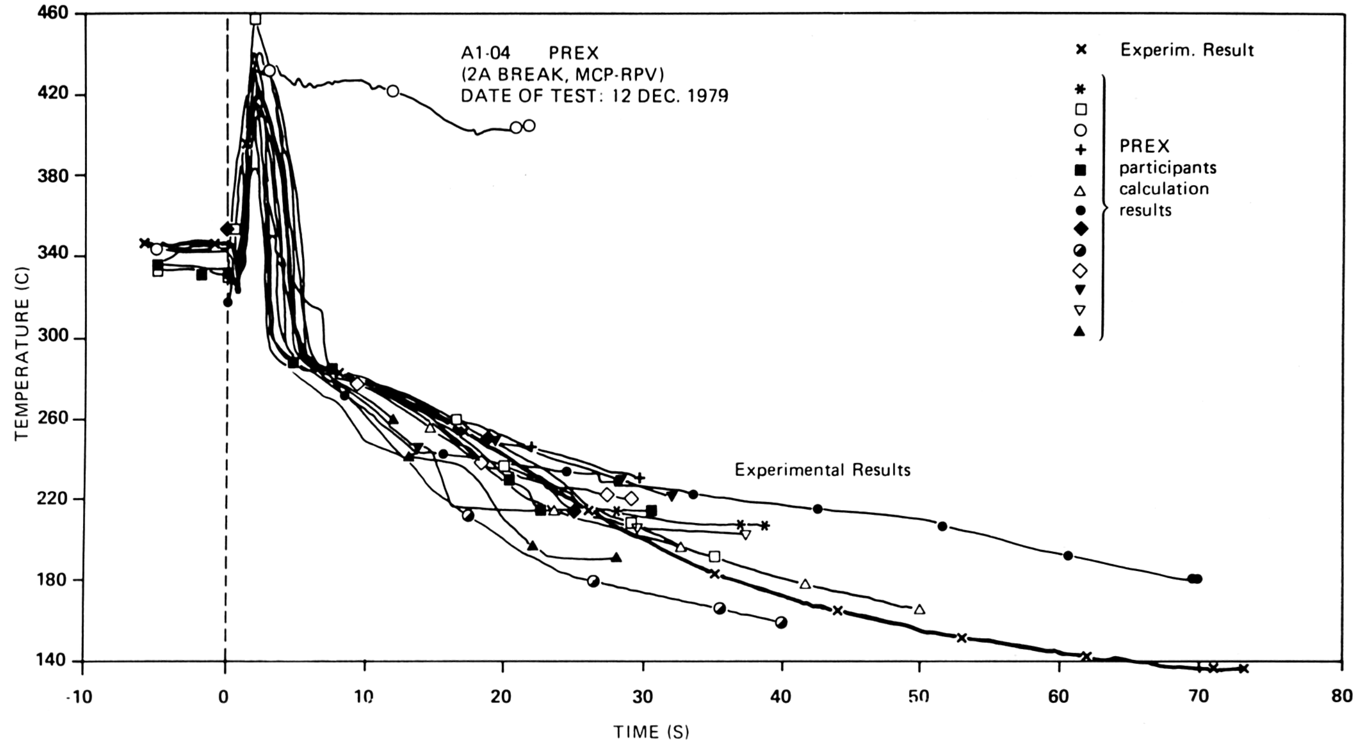


Fig. 2. Mean heater rod temperature level 6

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

The PISC I Programme for the inspection of steel components trials were primarily conducted to assess the capability of the ultrasonic procedure prepared by the Pressure Vessel Research Committee and based on the ASME code Section XI to detect, locate and size flaws or discontinuities in welds or heavy section steel. The results have shown the need for improvement in reliable detection of faults. Alternative ultrasonic techniques, some of them regularly used in Europe, have already been applied by some of the PISC programme participants showing that high sensitivity techniques, focussed beam probes, multiple orientation of the ultrasonic beams improve the detection probability and correct sizing of defects substantially.

Based on these results, the PISC II has been prepared in 1980 and endorsed by the CSNI (OECD). The JRC is "operating agent" of the programme and assures the management. The objectives of this new activity can be summarized as follows:

— evaluate the effectiveness of current and advanced non destructive techniques for plate inspections and in service inspection of reactor pressure vessel components with respect to in-service induced flaws,

— identify techniques for pre service inspection and in service inspection (ISI) which could be generally accepted, and

— bring the conclusions of the programme to the attention of the Code, Standard and Regulatory bodies concerned with ISI.

As a complement to the round robin tests, a number of parametric studies will be conducted, addressing in particular the questions of: defect position and geometry, equipment characteristics, effects of vessel cladding and residual stresses. An indispensable help to this exercise is being given by the qualified Non-Destructive Techniques laboratory at Ispra, which is engaged in equipment characterization.

Models Development to Assess the Probability of Failure of LWR Primary Circuit Components

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 continuously 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.

To perform these calculations, the JRC is developing the code COVASTOL. Work has been concentrated in 1980 on the complete calculation of probability for the onset of unstable crack propagation in the welds of a PWR vessel for defects having widths from 3 to 18 mm and lenghts from 8 to 2000 mm, located in any position through thickness. The probability of existence of the defects and the probability of occurrence of LOCA accidents have been considered.

Due to the still limited experience in this field, an effort has been made to design a programme in which, in addition to the study of separate effects, an attempt is made to show in a few integral small scale reactor vessel tests, the validity of the overall end of life prediction procedure. Acoustic emission techniques are being developed for continuouos monitoring of crack growth and available NDT techniques will be used for periodic monitoring of crack localization and sizing.

Fracture Mechanics and Creep Crack Growth 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 directed to analyse materials previously irradiated in the HFR reactor at Petten. In 1980, 4 so-called TRIO thimbles have been irradiated at nominal temperatures of 350 and 550 °C and at a nominal fast fluence of $1.5 \cdot 10^{20} N/cm^2$.

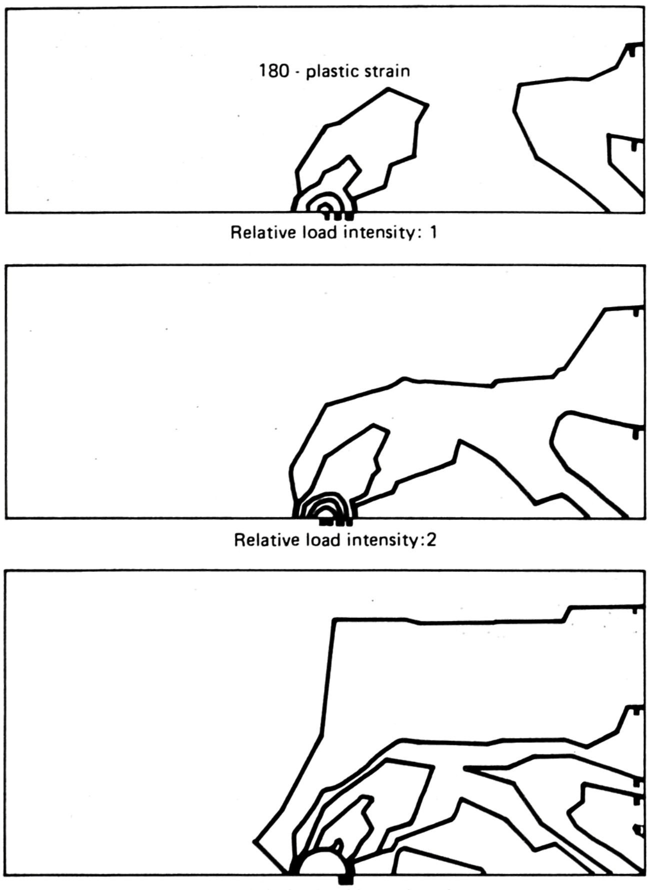


Fig. 3. Plastic zone sizes in the vicinity of the crack plane of a 3 PB specimen

At the same time, methodologies have been tested to get maximum information of the limited number of specimen available, which is obviously the case for irradiated ones.

The analytical work concentrated on the development of finite element calculation methods to determine values of the so-called J-integral which are calculated on the basis of definition of stress and strain fields near the crack tip.
Three different methods were successfully applied in a round robin on a three point bend specimen, jointly organized by the Welding Institute (UK) and JRC.

Creep crack growth studies are performed on austenitic stainless steels AISI 304 and 305 at 550 °C in argon atmosphere with compact tension specimens, 15 mm thick.

The understanding of creep phenomena is essential for LMFBR pressure vessel reliability studies, but it is far from being complete. The current analysis procedure consists in correlating data of creep crack growth rate with a controlling parameter called J* obtained by analogy from time independent fracture mechanics study. This view is not generally accepted, but preliminary results show that the agreement obtained between calculations and tests by laboratories in Japan, USA and JRC are fairly good.

In figure 3, plastic zone sizes around the crack tip for three stress levels are given. The complete results and conclusions of the round robin exercise will be available at a later date, but it is evident already now that work on Elastic Plastic Fracture Mechanics needs to be pursued.

Project IV: LMFBR Core Accident Initiation and Transition Phase

This project is composed of four subprojects:

— Subassembly Thermohydraulics, Code Development and Validation
— Liquid Metal Boiling Studies
— Fuel Coolant Interaction (FCI)
— European Accident Code (EAC)

Subassembly Thermohydraulics, Code Development and Validation

The main objective of this work is to provide understanding on thermohydraulic LMFBR subassembly behaviour for two types of accidental situations:

— Loss of coolant flow (LOF) or transient overpower (TOP) accidents. For these accident situations the analysis should provide adequate models for the incoherency of boiling and voiding in fast reactor subassemblies and assist in design and analysis of pump run-down experiments;
— Cooling deficiencies inside subassemblies under normal reactor operating conditions, that means at steady state flow and power. For this situation, the study should provide undestanding on thermohydraulic phenomena affecting the detection of incipient fuel failures and the failure propagation.

To describe the transient thermohydraulic rod bundle situation from TOP and LOF accidents, a special version of the code THARC named THARC-S has been developed. This code describes events until inception of coolant boiling and allows the computation of axial and radial void propagation. Further developments have been done to calculate the boiling boundaries and to describe the transient liquid coolant flow redistribution.

To study the steady state thermohydraulic situation of a subassembly with failures, the VELASCO-3D computer code has been developed and is in the validation phase by means of visualization experiments in rod bundles with blockages and different spacing devices (vertical wires and grids).

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 rundown due to pump failure or power excursion. The main areas of activity are:

— Measurement of the boiling characteristics at steady state and transient mass flow conditions for tubular, annular and bundle geometries.
— Construction of 12 pin boiling test sections for the investigation of the thermohydraulic behaviour of grid and wire spaced fast reactor subassemblies;
— Thermal noise analysis in bundle geometry;
— Investigation of the boiling pattern in porous blockage formed of stainless steel, UO_2 and mixture of stainless steel and UO_2 particles.

The measurements of the boiling characteristics at steady state and transient mass flow conditions and the related pressure drop measurements for simple geometries (tubular and annular test section) have been terminated and evaluated.

The fabrication and mounting of the 12 pin bundle test section with grid spacers has been terminated and its mounting in the test facility is under way. A measuring device for temperature noise detection developed and manufactured by CEA-Cadarache has been mounted at the outlet of the test section and is expected to give interesting results especially for boiling inception detection.

An important effort during the reporting period has been devoted to the evaluation of the experimental results of the boiling pattern and critical heat flux in porous blockages as they may occur in a LMFBR accident scenario (see test facility and test section on figures 4 & 5).

Fuel Coolant Interaction (FCI)

The main objectives of the FCI research programme are the investigation of the thermodynamic processes of interaction between molten reactor core materials and coolant, and the estimation of the consequences of vapour explosions.
This research includes the following activities:

— development of physical FCI models and appropriate codes;
— parametric experimental studies using simulant materials;
— verification of theoretically postulated mechanisms of vapour explosions.

During the reporting period, the analytical description of molten fuel coolant interaction has been continued with the

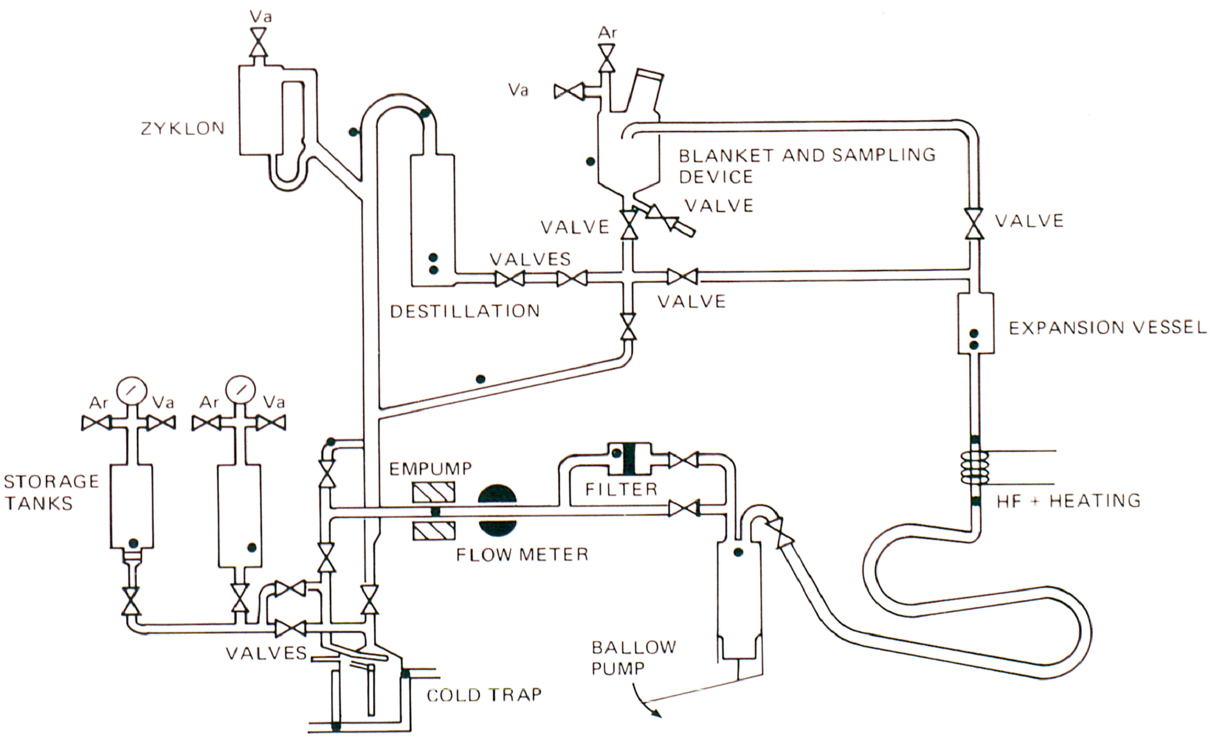


Fig. 4. Test facility - Boiling in porous blockages

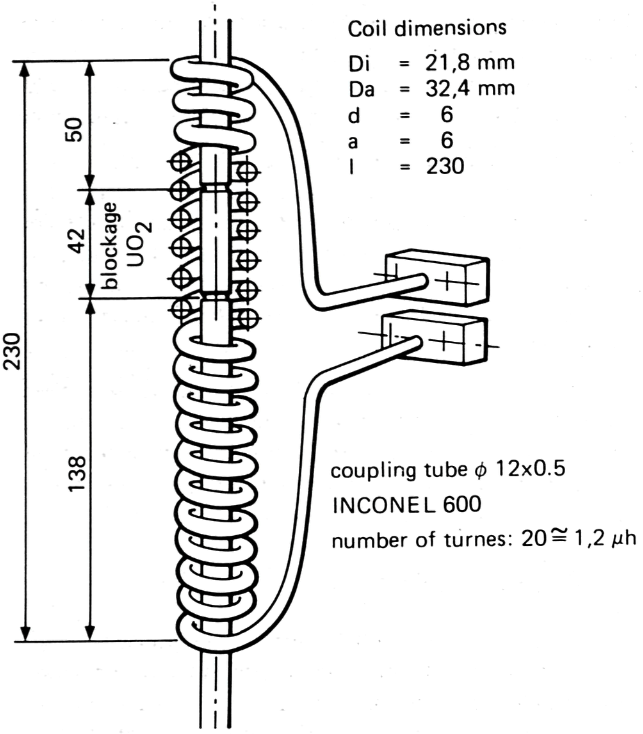


Fig. 5. Test section - Boiling in blockages

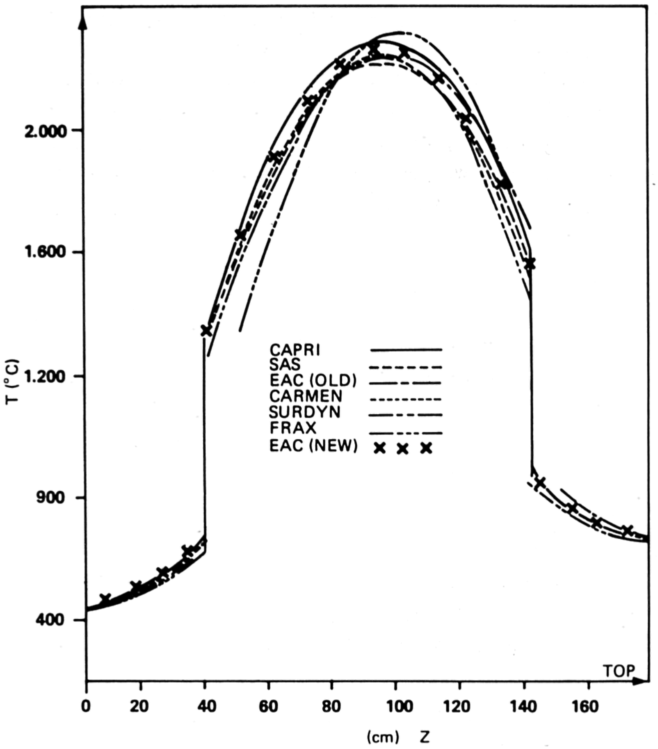


Fig. 6. Axial fuel temperature distribution at cavity radius (CH 2) (Typical comparison between EAC and others)

model development of shock wave induced hydrodynamic fragmentation processes.

The multichannel FCI code SAMI has been tested and is now operational. It is able to deal with heterogenous temperature distribution of the coolant and the vapour in a subassembly. The coolant motion, the vapour volume growth and collapse and the pressure history versus time can be calculated. The experimental part of this activity continued with the execution of NaCl-H_2O vapour explosions at different system pressures. Stratification tests with tin-lead alloy at high temperatures in contact with water were initiated. The accompanying theoretical study of instability mechanisms in a system consisting of high temperature melt covered by a less dense liquid with low boiling temperature has been started.

European Accident Code (EAC)

The activity is aiming to develop a modular system of computer programs to study postulated hypothetical core accidents in LMFBRs. The system under development, now called EAC (European Accident Code), is designed to be able to receive modules describing physical phenomena, developed and/or to be developed in different European and US laboratories.

The pilot version of the EAC, has been presented in June 1980 and is available on request.

Comparison with existing programs have shown that EAC is competitive both from the point of view of physical results and computer time.

An example of the results of comparative calculations for a TOP type accident performed with different European and American computer programs is given in fig. 6 where the axial fuel temperature distribution over the reactor core lenght is shown.

Accident initiation phenomena and fuel motion description during the accident are now neing implemented in the code. Intensive use of the EAC code is foreseen by the national establishments as early as its capabilities are increased.

Project V: LMFBR Post Disassembly Phase

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

— Structural Loading and Response
— Post Accident Heat Removal (PAHR) studies.

Structure Loading and Response

An important point of this project is the study of structural behaviour of the primary circuit and of the vessel during a significant mechanical energy release. This mechanical energy release has been evaluated in the past by making very conservative assumptions. Therefore a part of the JRC effort is now devoted to define that energy more realistically, an attempt which is being followed also in US and FRG.

During 1980 an exploratory study has been started on the possible physical models (such as those employed in the US-NRC code SIMMER) able to describe the energy dissipation mechanisms during the post-disassembly phase.

Since several years a considerable part of the structure loading and response activity has been devoted to the development of the COVA and COVAS programmes conceived for the validation of 2D hydrodynamic-structural codes applied to the study of the vessel, internal structures and subassemblies behaviour when a well defined amount of energy is released. By means of this study in which the results of clean model experiments are compared to the corresponding numerical results of the codes, an attempt is made to verify if:

— the physical phenomena and the various features of a typical reactor have been adequately modelled,
— the numerical approximation of the equations are adequate,
— the properties of the coolant and of the reactor materials and components are correctly represented.

The COVA and COVAS experimental results give continuous support for the improvement and further developments of the codes SEURBNUK and EURDYN 1M.
The COVA programme is being completed. During 1980 two tests have been successfully performed and the attention has been focused on numerical improvements of the SEURBNUK code in relation to the complex geometries encountered in the last COVA tests, to the treatment of interface motion in Eulerian coordinates and of the coolant impact on the plug.
The study on transient pressure drop through perforated plates is continuing: the realization of a new test section for the two dimensional validation of the models is foreseen.
It has to be mentioned that in addition to the COVA programme initiated by the JRC in collaboration with UKAEA research has been performed directly related to national programmes such as model tests and calculations for the German SNR-300, calculations of the French tests MARA, model experiments for the Italian PEC reactor.

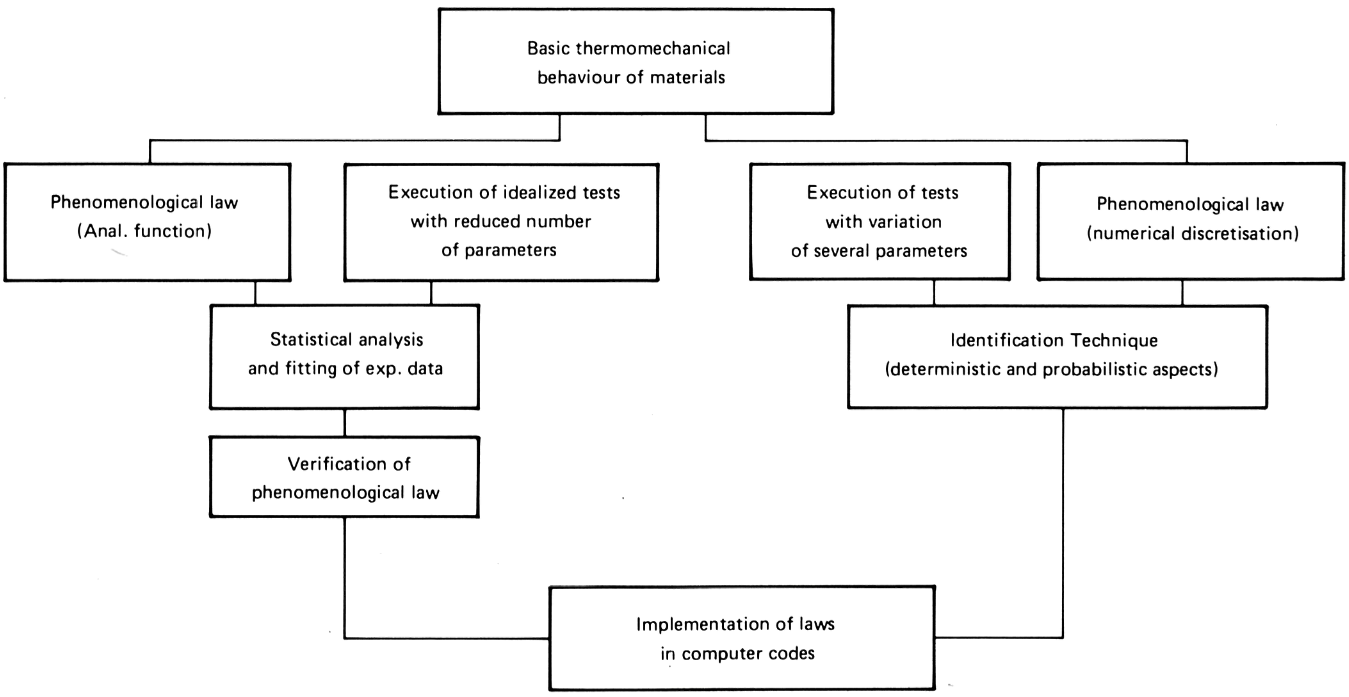
The COVAS programme (validation of hydrodynamic-structural codes for the subassemblies after a postulated local energy release) has been extensively rediscussed during 1980: the code EURDYN/1M has been improved for a more realistic treatment of fluid structure interaction in hexcan corners.

As already mentioned an important point for this research is the availability of an adequate description of the structural materials under dynamic conditions.
To this aim a specific activity is devoted to the development of constitutive equations for materials submitted to dynamic loading. Possible approaches to this study are illustrated in table III.

The so called ''Phenomenological laws'' are normally taken from the literature. They are based on both experience and theoretical models but more detailed and fundamental studies would be necessary to explicitly describe effects of temperature, radiation damage, work hardening, strain rate. Due to the complexity of the problems involved rather than performing the above mentioned fundamental studies usually a combined use of experimental tests and of numerical techniques is employed.
At the JRC special attention is given to the identification theory which consists mainly in identifying the constants of polinominal law in an integrated numerical procedure which models the behaviour of the specimen and makes use of the measures taken at specific measuring points.

The experimental activity in this domain is continuing with the aim of defining the dynamic mechanical behaviour of austenitic stainless steel in different conditions (temperature and strain rate) and various degrees of damage (due to welding, creep, mechanical fatigue and irradiation).

Table III. Determination of constitutive equations



More in particular the 1980 activities concentrated on the determination of uniaxial stress strain curves in support of the above mentioned COVA and COVAS activities, on uniaxial tests using irradiated AISI 304 L specimens and on scoping tests in a biaxial loading machine.

In addition work continued on construction of the high load (5MN) biaxial machine which is designed to investigate for metals the effects of material thickness, to test real size welds and also to provide data for the development of constitutive laws for concrete.

PAHR studies

All hypothetical severe accidents normally end with fuel dispersal from the reactor core cavity.
The main goal of the JRC studies in the post accident phenomenology area 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 two main lines of research are the following:

— Cooling of a particulate bed and its eventual transition to a molten pool
— Cooling of a molten pool and study of thermomechanical behaviour of core catchers.

Cooling of a particulate bed and its transition to a molten pool

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.
In-pile experiments are being performed at Sandia Laboratories (U.S.) and planned in Europe to verify these models and provide experimental assurance that fuel debris is safely cooled in the vessel even after partial reactor core melting.

In the reporting period a series of sensitivity studies have been performed with the ASPAB computation model. As an example fig. 7 shows the variation of phase composition versus bed power density, for a bed with 20% mass content of stainless steel cooled on top and bottom. The stainless steel is homogeneously mixed with the fuel particles. There are a series of assumptions underlying these calculations such as the sodium boiling and dry out, the distribution of melting steel, the porosity of sintering UO_2, the crust formation, which are being verified in out of pile and in pile studies.
The partition of work between the Sandia and the European tests is indicatively shown in fig. 7 as a vertical bar. In the ACRR reactor at Sandia the melting point of bed materials will be reached whereas in Europe substantial remelting of particulates in foreseen. In 1980, two experiments have been executed. The phenomenon to be investigated was that of dry

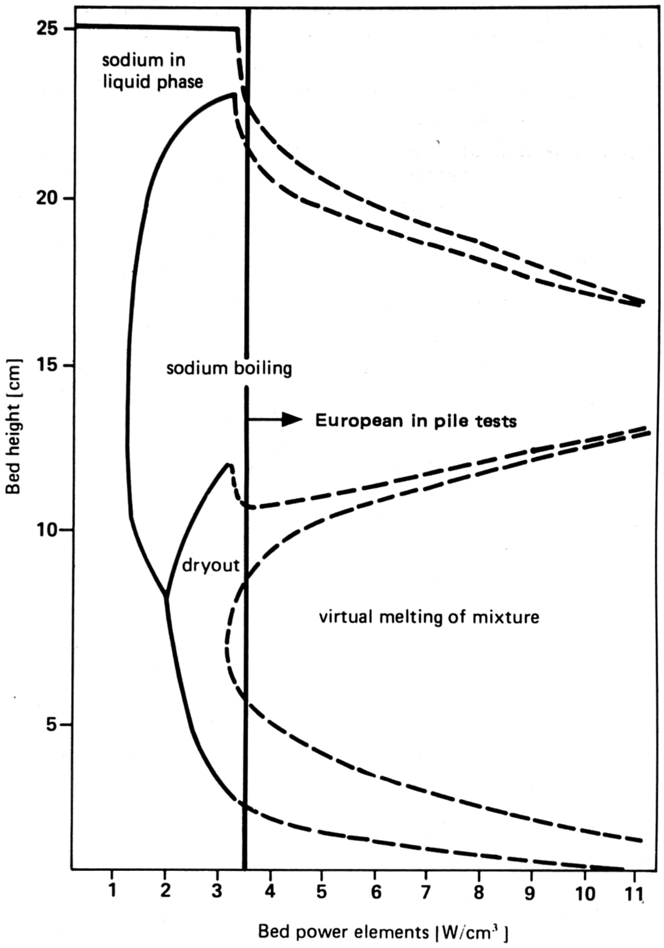


Fig. 7. Phase composition versus bed power density

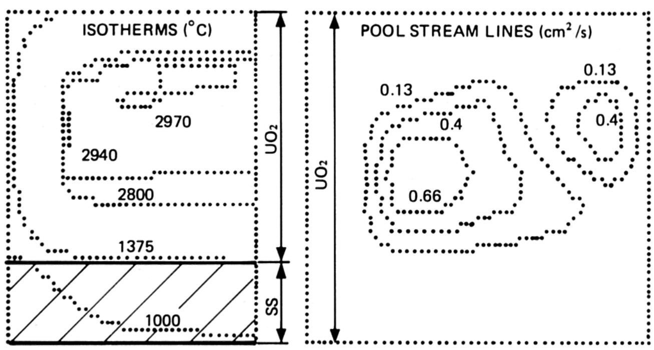


Fig. 8. UO_2-pool supported by stainless steel plate

out in beds of 450 and 600 kg. UO_2/m^2 bed loading.
Concerning the European in pile tests, feasibility studies provided by CEN-Mol, AERE-Harwell and CEN-Grenoble have been compared.
One more reason to perform tests in a European reactor in addition to SANDIA is the availability of larger bed diameters, ~ 150 mm instead of ~ 80 mm.
A decision for the choice of the European site is expected for early 1981 and the first two tests are scheduled for 1983.

Cooling of a molten pool and study of transient and steady state thermomechanical behaviour of core catchers

As already stated above a particulate bed might partially remelt and form a molten pool.
Also if large masses of molten core material are istantaneously released from an LMFBR core in hypothetical accident conditions the fuel fragmentation process may not become effective and molten pools may immediately form on core catchers.
Theoretical studies performed so far consider a molten pool interacting with catcher structures. The 2D code CONDIF and the 2-D code MACONDO developed at JRC consider both the formation of a solid crust at the interface between pool and structure whose extension varies in time and prevents mixing between fuel and molten support structural material. A typical example of pool isotherms and natural convection pool stream lines are given in fig. 8. In the reporting period MACONDO has also been applied to calculate catcher systems as designed for real reactors and the code has been modified to calculate Joule heating of a molten pool in the FARO installation.

The FARO installation currently under construction at JRC is a multipurpose plant in which the following problem areas will be investigated:

a) Behaviour of a 100 kg UO_2 molten pool and related structural problems.
b) Freezing and plugging of fuel jets.
c) Transient loading of core catchers during fuel impact.
d) Large mass fuel fragmentation and fuel settling tests.
e) Large mass fuel-coolant interactions.

In 1980 the furnace housing has been built and the detailed design work of the first test section related to problem areas b, c, and d has been started.

3. CONCLUSIONS

The main difficulties which appear in reactor licensing and operation have a strong impact on the JRC research programme. They may be sinthetised as follows:

- For almost all countries there are still difficulties to adopt the probabilistic risk assessment procedure. This reluctancy is partly due to the fact that a valid reliability data bank system is only now being implemented. Also, in the JRC risk and reliability project new methodologies are developed and applied to real plants showing that they lead to a comprehensive and systematic approach for reactor licensing and safe operation.
- A point which has become more and more evident in recent years is that a reactor, similar to other large installations, suffers periodically of small malfunctions which may propagate if they are not detected and removed. The main contributions given by JRC to the solution of this problem are included in the materials properties research as well as in the study of initiating events e.g.: in the sodium thermohydraulics studies.
- Finally, reactors are designed to survive large accidents, during which the release of fission product from the secondary containment has to be avoided. To reduce conservatism in the analysis the JRC is engaged in the development and verification of computer codes and in the execution of large and expansive tests e.g. LOBI.

Budget and staff of the JRC reactor safety programme are increased in the present pluriannual programme.
In order to disseminate results and receive data from national programmes, an efficient cooperation with European organizations has been set up. As a result research of common interest such as the development of large computer codes is performed in collaboration with national institutions and a division of tasks is adopted for the experimental studies where JRC is often considered also for reason of costs the right place for the construction of large installations e.g.: LOBI and FARO.
In addition JRC performs work under contract for particular national installations e.g.: SNR 300, PEC, or national programmes (LOCA, studies in the FRG).
The JRC reactor safety programme, performs an efficient cooperation with USNRC in particular for in pile PAHR studies.
Further tasks to be mentioned are the JRC leadership in the international PISC II programme and the collaboration with PNC (Japan) on dynamic properties of stainless steel previously submitted to low cycle fatigue.

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