REVIEW OF FAST REACTOR ACTIVITIES

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Commission of the European Communities

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1. Introduction

A description of some highlights of the activities performed by the Commission of the European Communities in the field of fast reactors is given. They fall into two categories :
- coordinating and harmonizing activities and
- research activities.

The former are essentially performed in the frame of the Fast Reactor Coordinating Committee (FRCC)*, the latter in the Commission's Joint Research Center and to some extent under contract in research centers of the Member States.

2. Coordinating and harmonizing activities

2.1. Strategy studies

A study performed jointly by the Commission and UNIPEDE ** on the "Role of the Breeder Reactor System" in the European Community which has been mentioned at previous meetings is now concluded and has been published.

Scope of the study was to analyse for an assumed electric energy demand scenario the penetration potential of nuclear energy in general and of fast reactors in particular in the European Community, taking into account the effect of different fast reactor and fuel cycle parameters. The study covers the period up to the year 2050. This rather long period was selected in order to be able to analyse the impact of the different scenarios on resources (uranium) requirements. (Let us not forget that the construction and lifetime of each station is approximately 30-40 years.)

The total natural uranium resources potentially available to the European Community over the period considered was a basic parameter for the study. The best estimate figure taken was 1,2 x 10^6t; it is based on the resources extractable at a cost of less than 30 $/lb (1975 price). Concerning the electrical energy demand evolution, two scenarios were considered assuming that 4000 and 7000 TWh/y will be reached assimptotically by the year 2050.

Assuming a penetration of nuclear energy of 80 % of the electricity production, it was found that the necessary natural uranium demand would exceed the estimated available resources by a factor of 3-5 if thermal reactors alone would be built. Recycling of Pu would not change this result fundamentally.

If the commercial introduction of fast breeders could be possible starting from the year 2000 with a rate depending on the Pu availability, then the available uranium resources would be sufficient for the nuclear programme with both electrical energy scenarios.

Several factors influence these conclusions :

- The cumulative natural uranium requirement could increase by a factor of 2 if the LMFBR characteristics would not be improved compared to demonstration plants (Creys-Malville).

- A ten years delay in the commercial introduction would lead to greater natural uranium requirements by 40 %.

- An increase of the fuel out-of-core time from 2y to 4y would also increase the uranium requirements by 40 %, whilst a reduction to 1 y would lower it by 10-15 % only.

- In order to provide the Pu necessary for fast reactors a continuous installation of thermal reactors is necessary and the availability of the reprocessing plants must be ensured.

- A rough economic assessment showed that LMFBRs have the potential to become competive with coal fired plants.

In conclusion, it should be stated that independently of the uncertainties in the assumptions made, especially with regard to the future electricity demand and the quantity of uranium likely to be available in the Community, the study showed clearly the need to continue the development of LMFBRs as otherwise it would not be possible to recover from an adverse situation in respect of either of those factors.

2.2. Safety

The Safety Working Group (SWG) which is an expert group of the FRCC made good progress in the elaboration of preliminary safety criteria and guidelines for fast reactors. Difficulties which were mentioned at the last meeting were overcome and it should be possible to conclude the still outstanding chapters of accident oriented criteria and guidelines in about one year.

Amongst the specific topics related to LMFBR safety, discussions on improved plant protection measures are to be mentioned. The group showed interest in the potential of microprocessors for accident prevention and supported the performance of a number of studies financed by the Commission to examine particular aspects of microprocessor application to LMFBRs.

Whole Core Accident Codes

The comparative calculations which the Whole Core Accident (WAC) subgroup of the SWG performed with different European and US codes (DOE, NRC) for a mild TOP accident in an irradiated core are terminated. The results will be published and presented at the Lyon Conference in July. In a next and probably last series of calculations, a LOF accident in an irradiated core will be considered.

The European Accident Code, which is developed by the Joint Research Center in close collaboration with the WAC group has now reached a stage which allows its use by external users. The code was already used in the comparative exercise mentioned before.

The WAC Group continued its discussions of special topics which are of interest for whole core accident analyses with the scope to find out if a common view on a particular issue can be reached. Amongst others the following topics were treated :

- The effect of design changes (homogeneous - heterogeneous) on whole core accidents
- The potential of subassembly accidents for a whole core involvement.

Containment Loading and Response

The Containment subgroup of the SWG pursued its activities in the field of the mechanical loading and response of the primary containment and internal structures during a hypothetical core disruptive accident. The group performed a critical assessment of the adequacy of the mathematical tools available.

The capabilities of existing codes were found to be generally satsfactory. However, further development was recommended in specific areas such as a more realistic energy source term description or as the roof impact treatment.

The group also followed closely on-going work to assess the consequences of a subassembly accident on the adjacent structures. In this field still some work remains to be done.

Finally it should be mentioned that the CONT group's mandate was extended to include also secondary containment problems. With the scope to come to a more realistic radioactivity source term, the group started to consider specific aerosol problems in more detail. A code comparison will be performed for a LMFBR benchmark case. In parallel an experiment will be performed to compare different aerosol measurement techniques.

2.3. Codes and Standards

The Codes and Standard Working Group (CSWG) of the FRCC continued its activities aimed at a progressive elimination of divergences existing between the various codes and standards applied in the European Community to the design of fast reactor components.

The different activities fall into four areas.

Manufacturing standards and quality control

The benefits and usefulness of a quantitative analysis of existing differences between national rules and standards has been demonstrated by comparative studies. Fields to which priority was given in these investigations are : welding processes, non-destructive testing and inspection.

Structural analysis

Benchmark calculations to asses the numerical accuracy of inelastic and elasto-plastic programmes have been performed. The results were summarized in a number of technical reports, one of which will be presented at the ASME conference foreseen in Orlando during June this year.

Materials

A Comparison of national data for steels used in the various projects has been performed. So far the creep-rupture and the tensile strength of AISI 316 were considered. In a next stage the mechanical properties of chrome alloys and the influence of defects in AISI 304 and 316 will be compared. Again some of the results were or will be presented at international conferences.

Classification of components

A first draft of a document proposing a classification system for LMFBR components has been elaborated and is now being circulated for comments amongst the different interested organisations. Scope of this document is to provide guidance to classify LMFBR components according to the importance, to safety and availability of structure failure of those components.

3. R + D activities performed in the Joint Research Center

In the frame of its 1980-83 multiannual programme the Commission's R + D activities related to LMFBRs fall into two main areas : safety and fuels.

3.1 Safety

The current activities are mainly focused on the study of the different phenomena occurring during hypothetical accident scenarios. These activities comprise work in the field of neutronics, thermohydraulics, fluid dynamics, continuum mechanics, thermomechanics and informatics.

The programme is subdivided in three main chapters :

- Accident Initiation and Transition Phase
- Accident Post Disassembly Phase
- Material Research

Accident Initiation and Transition Phase

The activities are concentrated on liquid metal boiling experimental and theoretical studies, on the development of the European accident code and on fuel coolant interaction research.

Liquid Metal Boiling

The objective of these studies is to gain quantitative data on the coolant behaviour in a fast reactor in case of anomalous operating conditions such as blockages in a rod bundle, flow run down due to pump failure, or power excursion. The main activities are :

- Boiling Code Development and Validation
  The UK code SABRE (release 3b) is used for precalculations of the boiling experiments foreseen
- Measurements of the boiling characteristics in steady state and transient mass flow conditions for single and bundle geometry
  In 12 pin bundle test sections, with grid and helicoidal spacers, single phase experiments and thermal noise measurements have been performed in 1981. The test sections are now being prepared for two-phase experiments.

European Accident Code (EAC)

EAC is a modular system of computer codes allowing the description of the different phases of hypothetical accidents. The pilot version of EAC is operational. The code is continuously improved by the insertion of new modules supplied by various countries and by modules developed by the JRC. A module describing 1 D two-phase flow 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 (with a variable cavity cross section) fission gas and fuel motion in the channel. Comparisons of the different hydraulic modules available in the EAC have been made to evaluate the effect of different sodium boiling modelling for various TOP and LOF accidental situations.

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

Fuel Coolant Interaction

The main objectives of this research are :

- development of physical FCI models and codes
- experimental studies of factors influencing the FCI process or triggering a vapor explosion
- verification of theoretically postulated mechanisms of vapor explosions
- simulation of reactor like conditions with respect to volume ratio and mode of contact.

The hydrodynamic fragmentation models (induced by shock wave propagation) have been developed to a very advanced stage for their incorporation in thermal detonation codes. The capillarity wave stripping model and also Taylor instability modelling allow the calculation of size and number of fragments, enlargement of surface area and fragmented mass as function of time.

In addition to the hydrodynamic fragmentation mechanism the thermal models of fragmentation (e.g. boiling, fragmentation mechanisms due to vapor bubble and film collapse) are studied.

Stratification experiments with melt alloys with different melting points and water are underway to increase the knowledge of the behaviour of melt-water stratified systems. In fact, in case of core melt down flooding of the melt is highly probable and a high vapor explosion potential is attributed to such configuration.

Accident Post-Disassembly Phase

The activities in this area are devoted to a more realistic description of the post-disassembly phase, to the analysis of the behaviour of the primary containment system and to post accident heat removal.

Multiphase Multifluid Hydrodynamics related to HCDAs

A limited effort has been set up in order to familiarize the JRC with the US code SIMMER-II, to evaluate the capabilities and models of this code in the context of post-disassembly calculations and to investigate the possibility of an experimental and theoretical programme for SIMMER validation.

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 experiments has been prepared in close contact with national experts. The first test could be initiated in 1982.

Containment Loading and Response

The experimental part of the COVA programme for the validation of containment codes was concluded in 1981. The calculations of the COVA tests and the complete analysis of the results will be terminated in 1982. To obtain better agreement between experimental and numerical results, the Finite Difference code SEURBNUK (jointly developed by JRC and UKAEA) and Finite Element code EURDYN (developed by the JRC) have been continuously improved with regard to fluid-structure modelling and numerical technique.

The preliminary analysis of the COVA series showed that the results of pressure and impulse prediction on bottom and lateral walls are generally satisfactory. Problems are still open in the interpretation of the roof impact predictions. Ad-hoc experiments would be needed to clarify better the phenomena occurring during the coolant impact on the roof. Activities in this direction could be decided after the conclusion of the COVA tests analysis. From the hoop and longitudinal strain prediction for flexible

internal and external structures, the importance of additional research on material behaviour in dynamic conditions became evident. The JRC is devoting a significant effort to those problems (see below).

The COVAS programme (Code Validation Subassembly) was mainly aimed at a validation of structural computer codes for dynamic plastic analysis by means of very simple structural experiments. A series of tests with clustered hexcans are planned in the near future.

Post Accident Heat Removal (PAHR)

The PAHR activities fall into two categories : out of pile studies to analyse specific PAHR phenomena with real reactor materials and in pile studies and related supporting atívities.

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 test sections (BLOKKER, TERMOS, FRAGOR).

The FARO facility is now being assembled 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 82 and 83 the BLOKKER test section will be used for freezing and blockage experiments in argon atmosphere. Channel or fuel element clusters can be introduced in the test section where freezing and blockage formation will be measured (y-ray absorption techniques) and calculated to verify the analytical models.
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Melting front advance and catcher plate perforation 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 of thermal and mechanical load on supporting structures, UO_2 particulate formation and settling) has been concluded and the construction will be initiated during this year.

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

In pile studies and supporting activities

To demonstrate the coolability of core debris which may form in a large hypothetical accident and settle on various parts of the reactor vessel, the Commission executes a comprehensive programme to develop and verify physical models and codes used to predict the temperature field in a debris bed and in debris retention devices.

All phenomena of potential importance are considered : debris bed cooling by conduction, single phase convection or two phase convection boiling of the coolant, particle dry-out, sintering and remelting, upward and downward heat 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) and a European series of tests at Grenoble (Melousine reactor) and MOL (BR2 reactor). 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 on high temperature urania and stainless steel. Crucible development and phenomenological studies on specific aspects are funded by the JRC in different European laboratories.

The European in pile programme is performed in three 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-dependant 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.

Material Research

Research in particular on stainless steel is performed at the JRC in several areas: fracture mechanics with particular emphasis on irradiated materials; study of creep crack growth for austenitic steel AISI 304 and 316 for typical operating conditions (load, temperature, creep-ductility etc.), studies on material dynamic behaviour and definition of corresponding constitutive laws. The latter studies were fundamental for the COVA and COVAS programme. Finally they should provide a full understanding of the response of real reactor structures under different loading conditions (temperature, stress state) and various degrees of degradation (welding, creep, mechanical fatigue, irradiation).

A large high load dynamic machine is under construction at ISPRA. It will be completed in early 1982 and will be used to investigate how the results obtained for small specimens (up to 20 mm^2 cross section) can be transferred to large structures of damaged materials (up to 5000 mm^2).

An interesting theoretical study for the application of the identification technique to the definition of constitutive laws is being performed at the JRC.

3.2. Fuel and Fuel Cycle Safety

3.2.1. Operation Limits of Plutonium Fuels

Advanced Fuels

Swelling of Advanced Fuels

In the previous years a description of the in-pile performance of advanced fuels was developed mainly on the basis of fast reactor irradiations of carbide, carbonitride, nitride and oxicarbide fuel up to 4 to 6 a/o burn-up and of out of pile property measurements. This led to the definition and investigation of four contributions to swelling (solid fission products, small fission gas bubbles or microscopic swelling, coarse porosity and cracks) and to the definition of four structural zones in the fuel.

For the microscopic swelling a burn-up dependent "critical temperature" was established below which microscopic swelling is small but above which this swelling contribution increases strongly with temperature and burn-up.

During 1981 this general picture was quantified in detail and extended for carbide fuel by the analysis of the four swelling contributions in Na-bonded carbide pins up to 12 a/o burn-up giving a complete picture of the in pile performance of a carbide over the whole range of temperatures and burn-up values of technical interest. At the same time a remarkable dependence of the fuel performance on details of the fabrication structure could be established by out of pile mechanical property measurements and by the first analysis of a relevant irradiated carbide fuel.

A more quantitative investigation of this fuel is under way and a critical comparison of these results with the previous ones will conclude the programme "Swelling of Advanced Fuels" during 1982.

Head-end Processes for the Reprocessing of Advanced Fuels

Two lines of activity are followed :

i. Oxidation of advanced fuels

This method has immediate interest since it allows the transformation of irradiated advanced fuels into oxide which can be processed further in operating reprocessing installations for oxide fuel.

After having defined the oxidation conditions such that a soluble mixed oxide powder is formed the oxidation products were transferred to a reprocessing installation. 2300 g highly enriched uranium and 560 g plutonium have been recovered from advanced fuels irradiated in the swelling programme.

ii. Direct dissolution of advanced fuels.

After establishing the dissolution kinetics as function of acid normality and temperature first experiments have been carried out with carbides of low burn-up. During the dissolution various organic compounds harmful to the TBP-extraction process are formed. An efficient way has been found to cope with these carbonaceus by-products.

Oxide fuels

Fuel fabrication and characterization

The fuel for the oxide transient irradiation programme are pellets whose feed material consists of particles prepared by the gel-route. This process allows the possibility to vary the amount and distribution of porosity within certain limits. Fracture stress has been employed to characterize the properties of such pellets as function of their fabrication structure.

Oxide Fuel Transients

This programme consists of two parts, i) the study of short fuel pins in in-pile transient experiments and by transient heating experiments in the hot cells and ii) the establishment of the microkinetics of the volatile fission products in the oxide by special experiments and sophisticated methods of analysis. An important feature of the transient irradiation experiments is the measurement of the fuel centre temperature by an ultrasonic thermometer. The first irradiation experiment TRANSON has been completed and is awaiting post irradiation examination. Investigation of fission product microkinetics has started.

Equation of State of Oxide Fuel

In the research on the vapor pressure of UO_2 exists at 5000 K a general disagreement between the measured and classically calculated vapor pressures of roughly one order of magnitude. This may either be due to a departure from equilibrium vaporization in the experiments or to inadequate input data in the calculations. Extensive experimental and theoretical investigations on the evaporation process induced by laser surface heating have been continued including mass spectrometry of the vapor species up to 5000 K as well as thermodynamic calculations in the $(U,Pu)O_{2-x}$ liquid system.

3.2.2. Actinide Fuel Cycle Safety

Minor Actinide Fuel Properties

Mixed uranium-americium oxide pellets have been prepared from gel-supported precipitation particles. Their behaviour in a radial temperature gradient has been determined, the thermal conductivity measured and a capsule irradiation of this fuel in the FR2 reactor in Karlsruhe has been performed.

Safe Handling of Plutonium Fuels

The measuring and analysis techniques for aerosol analysis have been further developed and applied to aerosols produced in an aerosol generator and sampled from glove boxes.

The installation of safety equipment and instrumentation for the study of aerosols generated by burning glove box material is nearing completion.