REVIEW OF FAST REACTOR ACTIVITIES, MARCH 1980

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1. As in previous years, a short outline of the major achievements made since the last IWGFR meeting is given in the following.

   First I want to mention that on 18 February 1980 the Council of Ministers has approved a resolution in which they recognise the strategic importance of fast breeder reactors and the need to continue the efforts towards maintaining an effective fast breeder option in the Member States (°°)

2. Activities performed in the frame of the Fast Reactor Coordinating Committee

   Commission - UNIPEDE (°)

   The study performed jointly by the Commission and UNIPEDE on the penetration of fast reactors in the European Community was continued. From the variation of a number of parameters, especially with regard to the fuel cycle, the importance of an industrially well developed fuel reprocessing for the commercial introduction of LMFBRs is becoming evident. The conclusion of the study is scheduled for the end of this year.

   Safety Working Group (SWG)

   The Safety Working Group has made good progress in the elaboration of preliminary safety criteria and guidelines. Three accident categories have been treated until now : primary reactivity accidents, general cooling accidents and subassembly cooling accidents. The still outstanding items are accidents outside the core and external accidents.

(°) UNIPEDE = Union Internationale des producteurs et distributeurs d'énergie électrique

(°°) See Official Journal of the European Communities, N° C 51, 29.2.1980

In collaboration with experts from the Member States, the SWG reviewed the outstanding problems in the field of post accident heat removal and examined the current R + D activities with the aim to evaluate if all important areas were adequately covered. At the time being, the results of the review are analysed and possible ways to increase the activities in some areas are discussed.

Whole Core Accident Code Subgroup (WAC)

The results of a comparative study for a TOP accident using different codes from the Member States and the US NRC were published* and presented at the Seattle Conference (August 1979). In a subsequent round of calculations, a mild overpower transient (10 c/s) is treated for an irradiated core. Steady state and transient calculations up to clad failure have been carried out so far. The results showed generally good agreement in the early phase of the transient, as soon as phenomenological models come into play the results are more diverging. More analysis will be performed with regard to the models used to explain the discrepancies.

The European Accident Code (EAC) which is developed by the ISPRA Establishment of the Joint Research Center in the frame of the WAC group activities was further improved, in particular with regard to calculation time. The present code version will be documented by the middle of this year.

Containment Loading and Response (CONT)

The group continued the exchange of information on the current activities performed by the JRC - ISPRA and by national organisations in the field of development and validation of codes used to assess the consequences of a severe accident on the reactor tank. Besides this, increasing effort was devoted to the analysis of the consequences of subassembly accidents. The development of simulant material to be used in subassembly experiments, as well as code development were discussed.

* EUR report 6318

The group has followed with interest the APRICOT code comparison programme to which several member countries contributed. A reviewer has been nominated by the group who participated in the analyses of the results.

Codes and Standards Working Group (CSWG)

During 1979 the Council of Ministers has approved additional funds allowing to increase the Commission's financial support to the group's work. In 1980 an amount of 300.000 EUA is foreseen.

As had been said at former meetings, the main objectives of the group's activities are to draw up an inventory of existing codes and standards applied to fast reactors in the Member Countries, to identify points of similarity and to evaluate points of dissimilarity with the aim to reach a consensus on a possible elimination of the existing dissimilarities.

Main topics of the 1980' activities are .

- quantitative analysis of national manufacturing standards,
- performance of benchmark calculations using different structural analysis codes,
- comparison of structural material properties,
- elaboration of opinion papers on particular topics.

3. R + D activities performed by the JRC

The Council of Ministers has approved in March 1980 the JRC programme for the period 1980-1983. This programme is a logical continuation of the preceeding one and takes into account the recent development in the field of nuclear safety. The budget foreseen for LMFBR safety research in the frame of the general reactor safety programme is in the order of 70 MEUA and the budget for the LMFBR fuel and fuel cycle safety research is approximately 40 MEUA for the four-years period.

Some highlights from the two main areas of the JRC LMFBR activities - safety and fuel - are reported.

3.1. LMFBR Safety

Whole core accident code

The JRC activities on the European Accident Code (EAC) have been concentrated on streamlining of the code in order to reduce the calculation time. Various modules describing physical phenomena were cleaned up. A databank has been implemented which replaces the algebraic expressions used for empirical material properties by tabularized data.

The JRC participated in the comparative calculations performed by the WAC Group, using the presently available EAC version. The results were in good agreement with those from other codes.

LMFBR subassembly thermohydraulics

The mathematical modelling and code development (THARC-S) for the treatment of a loss-of-flow (LOF) accident in subassembly geometry up to the early phase of boiling has essentially been terminated. A user manual for the code THARC-S has been prepared. The code has been successfully used to analyse the thermohydraulic conditions of a LOF accident in SUPER PHENIX subassemblies (271 rod).

The VALESCO-3D computer code which treats steady flow conditions in a failed subassembly was developed further and validated by an experiment with a tight blockage.

Sodium boiling experiments in simple geometry have been terminated; the results are used to validate the Na-boiling code ESSO.

Boiling experiments in porous blockages formed of pure UO_2 or pure stainless steel particles have showed important differences in the boiling pattern. As a consequence, additional boiling tests in porous blockages using mixtures of those materials were initiated.

Fuel-coolant interaction

The experimental and theoretical work has been continued with the main objectives to investigate the thermodynamic processes of interaction and to estimate the consequences of vapor explosions.

A code (SAMI) allowing the FCI treatment in multichannel geometry was developed further and used for a parametric study. Blanket pressure, coolant temperature, coolant vapor distribution in the channel, non-uniformity of motion in subchannels and the length of the sub-assembly were varied to study their effect on pre-pressure, voiding and heat transfer as a function of time.

Measurements of the vapour explosion and mechanical energy were performed in different facilities using UO_2 and stainless steel melts in water. The main characteristic of these experiments was the forced impact energy which was changed over a wide range. A number of empirical correlations between impact energy and melt mixed with the coolant as well as the vapour explosion work could be established. Upper conversion factors of 3.3 % referred to the heat stored in the melt were measured.

The effect of vapour pressure cut off by system pressurisation was studied using a molten salt and water system. A cut off was observed for a pressurisation of 0.2 to 0.3 MPa. Also the effect of subcooling was investigated; a pressure cut off was observed changing the sub-cooling from 10° to 150°C.

Post accident heat removal (PAHR)

Experimental and theoretical work is performed with the aim to understand the different phenomena involved : release of molten material from the core, the formation of particulates, their settlement, cooling and possible remelting and the formation and cooling of a molten pool.

From consultations with experts from the Member States it has been concluded that, in order to show that molten core material can be safely cooled inside the reactor tank, in-pile experiments are desirable. In such experiments, besides the particulate cooling capabilities, their transition into a molten pool as well as molten pool cooling capabilities will be investigated.

In collaboration with national research centers, a feasibility study for a European in-pile experiment is performed. The choice of a suitable crucible material for such tests is the most urgent problem. Experiments to determine the heat source distribution in a molten pool are under preparation. Measurements of material data, such as the viscosity and the thermal diffusivity of molten uranium-plutonium oxide are to be reported.

Theoretical work is mainly centered on the development of codes necessary in support of the experimental work envisaged.

The computation model ASPAB for predicting the heat transfer behaviour of fuel particulate beds has been further developed. Parametric calculations were carried out for particulate beds with a loading of 600 kg/m2 and a variable decay heat range of 0.2 - 5 KW/kg.

Code development to study the molten pool behaviour and the interaction with its supporting structures was pursued. The code MACONDO describes in two dimensions natural convection in the molten pool, crust formation and growth of pool boundaries and heat transfer with change of phase in supporting structures.

Dynamic loading and response

Theoretical and experimental work to develop and validate 2D non-linear dynamic finite element codes for coupled hydrodynamics and structural codes was continued.

The codes SEURBNUK and EURDYN-1M were submitted to a series of tests for comparison. With regard to the structural behaviour almost perfect agreement was reached in the elastic field, but discrepancies appeared in the plastic deflection of the cap. By the introduction of the full bending theory instead of membrane theory in SEURBNUK the results were improved, but further analysis is still necessary.

The codes EURDYN-1M and SEURBNUK were validated by interpreting experiments of the COVA-Programme. The codes were also used to analyse the French validation experiments. The results of the codes agreed very well with the experimental data.

Further experiments of the COVA programme were performed as well as a firing in a 1:6 scale SNR 300 reactor tank model. SEURBNUK was used to perform pre-shot calculations. Overall examination of the experimental results indicated that the energy released experimentally was much smaller than that expected from the equation of state used in the calculation for the explosive.
A conclusion from the COVA programme was that additional effort is necessary to improve material data and material behaviour modelling. An adequate material testing programme was elaborated in contact with national experts to overcome the difficulties encountered.

All work described is executed in close collaboration with the Containment Expert Group (CONT) of the Fast Reactor Safety Working Group (SWG).

Safety related material properties

The activity on constitutive laws of austenitic stainless steels was continued and extended to include AISI 304 and AISI 321 besides AISI 316. A large number of dynamic uniaxial tensile tests were performed in support of COVA activity. The main negative effects of defects created by welding and irradiation on the dynamic flow curves were investigated.
In order to extend the investigations to materials damaged by mechanical and thermal fatigue or creep under multiaxial state of stress, as well as to include large test pieces where the statistical distribution of defects determines the strength of the real structure, a high-load dynamic biaxial device for specimens having a cross section of up to 20 mm^2 was developed and tested, and another machine for large specimens with a cross section of up to 5000 mm^2 was designed.

The activity on constitutive laws of subassembly internals were continued with the construction of a new test rig for the measurement of the relationship pressure/volume for a subassembly pin bundle in a hexagonal matrix with undeformable walls. Experiments are under way.

In the field of fracture mechanics the thermal shock experiments and the elasto-plastic fracture mechanics studies involving the use of "J" resistance curves methodology were terminated. Work on mixed mode type of fatigue crack growth was carried out. In order to assess the significance of defects, viz. irradiation damage, under the aspect of structural safety of operating reactors, an irradiation experiment to be carried out in the HFR reactor of the JRC Establishment Petten was prepared.
The work related to creep fatigue damage in stainless steel structures was concentrated on the effects of frequency, hold time and load cycle combinations (AISI 304), and on the study of creep crack growth by characterisation of the plastic deformation process of defects growth near a crack tip (AISI 304 and AISI 310). Finally, the creep modelling work in order to develop a phenomenological model of the creep damage in AISI 304 - taking into account experimental results obtained at temperatures of up to 650°C - was succesfully continued.

3.2. LMFBR Fuels

Utilisation limits of fast breeder fuels

New experimental results for advanced fuels - obtained in out-of-pile and in-pile experiments - on fission gas kinetics, fuel mechanical properties and irradiation induced diffusion have contributed to perfect the models of microscopic swelling and the concept of a critical temperature for swelling. A new model for the in-pile performance of advanced fuels based on four standard fuel zones was established and tested against experimental data.

In-pile temperatures of advanced fuels have been studied by means of ultrasonic thermometers developed by the JRC Establishment Karlsruhe. Previous irradiation data have been analysed in detail.

Work on equation of state of nuclear materials has been continued : studies have been extended to the advanced fuels systems, starting on UC. The gas dynamics study of the evaporation jet have been extended to binary mixtures of monatomic and polyatomic gases. The various theoretical models for the prediction of critical point and thermodynamic data of nuclear fuels have been assessed.

The activity on the corrosion of stainless steel cladding of mixed oxides approaches its end. Besides laboratory investigations of detailed aspects of the proposed quantitative corrosion model and further development of a microcell for measuring oxygen potentials in irradiated fuels, work was concentrated on the design and the analysis of an in-pile test. The purpose of this experiment is to test the various predictions of the model developed at the JRC Establishment Karlsruhe.

Plutonium and actinide aspects of nuclear fuel cycle

Determination of the fission yield of selected actinide isotopes and work on the cumulative fission yield measurements have been carried out. The previously determined integral cross sections of some heavy elements have been validated. Moreover, isotope correlation techniques have been used to predict actinide build-up, and for neutron emission studies.

Activities related to aerosol research has been continued : various techniques of sampling and analysing aerosol were compared. Furthermore, a special aerosol laboratory is being established.

Finally, in the field of head- end processing of mixed carbide fuels, experimental and theoretical studies were oriented towards thermochemistry and phase relations, kinetics involved in dissolution processes and in oxidation reactions, and controlled oxidation of irradiated fuels.

REVIEW OF FAST REACTOR ACTIVITIES AT OECD (NEA)

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The Committee on the Safety of Nuclear Installations (CSNI) has recently increased its activity in LMFBR safety, under the guidance of its Group of Senior Experts on LMFBR Safety R & D. This Group, formed in 1978, consists of CSNI delegates (or alternates) from Member countries sponsoring major research in the field, and the Commission of the European Communities.

The Group now oversees the preparation of international status reports on relatively well-developed areas of LMFBR safety technology, and the convening of specialist meetings, expert groups and task forces to aid in investigating and resolving problems in less-evolved safety subjects.

Three status-of-technology reports are to be published during 1980, on:

(i) the role of fission gas release in the propagation of fuel failure;

(ii) increasing the reliability of fast reactor shutdown systems, and

(iii) reactivity monitoring in an LMFBR at shutdown.

Preparation will begin in late 1980 of reports on:

(iv) local cooling disturbances in sub-assemblies of sodium-cooled reactors;

(v) interactions between sodium and concrete, including the effect of defective liners; and (possibly)

(vi) the consequences of local clad defects.

Expert meetings were convened in March 1980 on three subjects of current research interest:

(i) an ad hoc meeting discussed the relationship of fuel failure consequence modelling to the planning of confirmatory fuel experiments. The meeting recommended that an exchange of detailed information be conducted in 1980 on key accident phenomena, the computer codes used to model them, as well as uncertainties in related experimental data and work planned to reduce them.