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Research in the field of severe accidents performed by the European Union as cost shared projects

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ABSTRACT: Research activities are described which the European Union is carrying out in cost sharing under the Third Framework Programme for Research and Technological Development (1990-94) in the field of Nuclear Fission Safety, more particularly in the area of severe accidents.

1 INTRODUCTION

The European Commission is since many years performing research in the field of nuclear safety in its own Joint Research Centre as well as in cost sharing in laboratories of its member countries.

A main objective of this research is to develop a sound basis for the development of a common safety approach in the European Union. The Euratom Treaty (1957) and a series of Resolutions of the Council of Ministers set out the Commission's mandate in this regard, i.e. to protect the worker, the population and the environment from any damage that could result from the use of nuclear energy and to work towards a harmonisation of safety criteria and guidelines for nuclear installations.

2 PROGRAMME ORIENTATION

The programme on Nuclear Fission Safety that was approved by the Council of Ministers on 28 November 1991 is part of the Third Framework Programme for Community Activities in the Field of Research and Technological Development (1990-1994). It has two main activity areas: reactor safety and radiation protection. Here only the reactor safety part is considered. The budget for reactor safety is 13.5 MioEcu (1 Ecu is about 1,3 US $). Starting from its implementation in December 1992 the programme has a duration of two years and half, i.e. it comes to its end in June 1995.

The programme is carried out as a Reinforced Concerted Action, which means that the European Commission considers the main goal of its intervention the stimulation of cooperation between the member countries. Any financial contribution to research being performed in national research laboratories is subordinated to this principle. The total amount of research activities under this programme has actually a value of 62 MioEcu, out of which 20 % are contributed by the Commission. In total about eighty partners from all parties involved in nuclear plant development participate, i.e. universities, national research centres, regulatory bodies, electrical utilities and manufacturers of nuclear components.

The main theme of the reactor safety programme is the confinement of radioactivity under severe accident conditions in light water reactors. Accident progression analysis, the integrity of the containment system and accident management are the main problem areas being

addressed.

The programme objectives are a critical assessment of the current knowledge, the development of a consensus on how to treat outstanding problems, to perform theoretical and experimental investigations and, where necessary, to make recommendations for future research. The understanding of severe accident sequences and phenomena is seen as essential for the definition of prevention and mitigation measures by which the risk of severe accidents can be reduced.

The activities are grouped in eight projects that are carried out under the responsibility of a project coordinator. The projects are: core degradation, reactor pressure vessel response, hydrogen problems, molten fuel-coolant interaction, molten corium-concrete interaction, the radioactivity source term, containment integrity and accident management support.

3 TECHNICAL DESCRIPTION

The eight projects and the progress made in their execution are described in the following. Some indications are also given about possible further research activities.

3.1 *Core degradation*

The knowledge about when and how the core looses its geometry, how much hydrogen is produced, the rate of oxidation of the different core materials, the release of fission products and the processes by which solid or liquid core materials reach the lower plenum of the reactor vessel are fundamental for any severe accident scenario, because these processes determine the potential failure of the pressure vessel, subsequent ex-vessel phenomena, the threat to the containment and finally the radioactivity source term.

The objective of the Core Degradation project is to improve the physical- chemical understanding and the numerical modelling of the various phenomena involved and, where adequate data are missing, to perform separate-effects tests.

During the early phase of core degradation, i.e. after the onset of core uncovery, oxidation processes of the zircaloy cladding, which can start at about 1030 °C, lead to highly exothermic reactions such as hydrogen formation and to cladding embrittlement. Clad ballooning under low pressure conditions or clad-pellet mechanical interactions under high pressure conditions can lead to fuel rod rupture, especially in the case of fission gas bubble pressurization at high burnups. Clad failure models based on creep rupture and burst criteria are sufficient in the area of design basis accidents but not to describe severe accident scenarios involving complex thermo-chemical phenomena such as Zircaloy oxidation, fuel rod embrittlement, cracks in the zirconia (ZrO2) layers induced by thermal cycles and dissolution processes for zirconia and UO_2.

The modelling and validation efforts performed under the project are based mainly on a European data bank based essentially on the results of a series of integral in-pile PHEBUS-SFD tests carried out at CEA Cadarache and integral CORA out-of-pile tests made at FzK (Forschungszentrum Karlsruhe). The main effort of this project is devoted to the experimental validation and the benchmarking of models able to predict a wide range of thermo-mechanical and -chemical processes during the core degradation phase. One of the key phenomena is the oxidation behaviour of the core materials by steam and the resulting H_2 generation before, during and after core material relocation. In particular failure criteria of ZrO_2 layers have been developed and calculations have been performed for the relocation behaviour of solid and liquid materials (e.g. candling, slumping) and its influence on the debris bed formation.

Quenching tests, simulating cooling recovery under faulty ECCS or accident management conditions, were performed at FzK with emphasis on the behaviour of ZrO_2 layers on the zircaloy cladding and on materials interactions with stainless steel, Inconel and absorber materials (Ag-In-Cd and B_4C). Calculations using ICARE-2 at CEA-IPSN/Cadarache and MELCOR

or SCDAP/RELAP-5 at AEA/Winfrith are performed with regard to early and late phase phenomena, like oxidation of degraded core and molten-fuel/materials interactions. Complex melt chemistry aspects, including the modelling of low temperature eutectics formation, fission product release and debris bed behaviour, are examined by ENEA-ERG/Rome in connection with the ICARE-2 code developments. The University of Pisa focuses on the validation of H_2 generation models due to solid and molten zircaloy oxidation against a series of PBF, LOFT and CORA tests.

Calculations with the KESS code are performed at IKE/Stuttgart, in connection with ATHLET-CD (GRS) and ESTER(JRC/Ispra) predictions, to better understand cladding embrittlement and core thermal-hydraulics effects on the quenching behaviour (oxidation energy). Radiative heat transfer effects and late phase phenomena including debris bed formation are investigated by the University of Bochum.

Items of interest for future research in this area could be melt relocation and materials interaction phenomena during the late phase core degradation as well as reactivity insertion phenomena.

Besides the effects of cladding oxidation and hydrogen production during reflooding of the core, it is necessary to examine in particular the chemistry of the late phase core degradation with a special view to the fission product release and transport, especially in the case of high burnup and complex aerosols behaviour.

Core uncovery accidents have been shown in this project to be ultimately characterized by rather slow materials melt down processes with high in-core temperatures. On the contrary, reactivity insertion accidents, generated for example by an early control rod melt down in a BWR, are to be considered as rapid events where pellet-cladding mechanical and chemical interactions are primordial.

3.2 *Reactor pressure vessel response*

If the degraded core cannot be quenched in time, the molten corium can put the RPV under extreme stress-strain conditions, either mechanically because of in-vessel steam explosions (in wet conditions, usually under low pressures) or thermally because of high heat transfer between molten corium and RPV (in dry conditions, usually under high pressures). In the framework of the TMI-2 VIP (Vessel Investigation Programme) considerable research efforts were already devoted to this kind of problems.

In the latter case a crust may appear in the form of a large metallo-ceramic hard layer which influences the heat transfer. It may successively rupture due to thermal melt-through, thermo-mechanical stresses induced by large pressure and temperature gradients or eutectics formation. In the molten multi-phase multicomponent pool the decay heat may be dissipated through large natural convective heat transfers in the bulk of the molten corium and heat losses at the boundaries by irradiation or cooling by water. Separate-effects tests and numerical models are needed to better understand the molten pool behaviour, especially the onset of natural circulation in large molten pools, as well as the thermo-mechanical stability of the crust. As a consequence under typical conditions, such as inner vessel wall surface temperatures of 1300 °C, internal pressures of 17 MPa and gravity forces, high temperature plasticity and creep phenomena may lead to the failure of the hot RPV structures.

In the case of steam explosions and subsequent molten corium jet impacts on the RPV structures, highly non-linear dynamics models are required to predict the rapid lower and/or upper head rupture under extreme strain-rate and temperature conditions.

The major topic under this project is the behaviour of the lower pressure vessel head in case of core melt loadings. Phenomena like the flow and distribution of the core debris on the vessel bottom, and the thickness of the crust insulation layer are considered in various numerical simulation programs, as they determine the temperature distribution and the creep behaviour. These models enable to extrapolate first results to different vessel geometries.

The objective of the Reactor Pressure Vessel project is to establish a consensus on the present capabilities to describe the thermal

load and the structural response of the lower pressure vessel head in case of meltdown scenarios. An extension of the project consists of investigating impact effects on the upper vessel head. A major portion of the project is the generation of a material database for typical pressure vessel materials, with existing experimental results and with the results of specific new experiments for extreme values of temperature and loading.

Benchmark calculations are performed to determine the capabilities of existing models to describe the thermal loading and the structural response of the pressure vessel lower head and to assess the sensitivity of the models to certain input parameters.

The development of a material database takes into account the most recent experimental results relevant to pressure vessels, analyzing the data with respect to median values and uncertainty bands. The data are provided in ready-for-use format for finite element codes.

GRS/Cologne performs calculations of failure modes and time sequences of a PWR lower head without penetrations under various severe accident conditions. Slug impact tests on the upper vessel head are being performed in the BERDA facility of FzK: the calculations focus on strain-rate effects and energy dissipation by structural deformation. Dynamic loadings on a pressure vessel under severe accident conditions are computed in the framework of a benchmark exercise about plastic and creep failure.

Thermal calculations of a turbulent heated pool of molten corium are performed by CEA-IPSN/Fontenay. ENEA-DRI/Rome and ENEL-VDN/Rome are examining jointly ex-vessel cooling techniques on the basis of corium/RPV interaction calculations, with and without water in the vessel. They are using the computer codes TRIO-EF and CORIUM-2D for the thermal-hydraulics and CASTEM-2000 for the thermo-mechanics. They are also developing, together with the University of Pisa and with ANPA/Rome, dynamic structural loading models under in-vessel steam explosion conditions. This helps in particular in predicting stresses and deformations for the bolted flange of the upper vessel, needed for the precalculations of the BERDA experiments.

The experiments performed by CEA-DRN/Saclay include new creep and rupture data at 700 and 1000 °C. Moreover new constitutive equations under multi-axial conditions are derived from rupture experiments with non-uniform thickness tubes in connection with penetration tube failure analysis. The creep tests performed at KFA/Jülich focus on large specimen effects (Diameter=70 mm and height= 1000 mm) at 700 and 1000 °C. Effects of strain-rate as well as pressure and temperature drop are investigated by Siemens-KWU/Erlangen, together with radiation and heat flux distribution effects in the case of ex-vessel cooling.

Items of interest for future research in the area of reactor cooling system (RCS) integrity could be issues related to the primary piping behaviour, like temperature induced PWR hot leg failures and their potential for precluding direct containment heating (DCH) in the case of a RPV failure under high pressure conditions. Another issue of interest - even if outside the strict scope of this section - is the response of the secondary circuit to the primary one, e.g. the effects of temperature induced PWR Steam Generator Tube Ruptures (SGTR) and how to prevent or mitigate these events.

3.3 *Hydrogen problems*

During a severe accident in a LWR, water may interact with hot core materials and hydrogen may be produced and released into the containment. Hydrogen may react with the oxygen of the containment atmosphere and flames, deflagrations or detonations may jeopardize the containment integrity.

The current research efforts are concentrated on: the prevention of hydrogen combustion through pre- and post-accident inerting procedures, the prevention and mitigation of hydrogen combustion through H_2-O_2 recombiners and the mitigation of hydrogen combustion and explosion through deliberate ignition.

The objective of the Hydrogen project is to assess the present

knowledge of hydrogen explosion problems in LWRs, the improvement of modelling techniques and the investigation of measures to reduce the risk resulting from hydrogen. Attention is focused on various pre- and post-inerting techniques. Recombining and deliberate ignition techniques are considered as well. Computer codes are developed for the prediction of hydrogen distribution and combustion.

The chapter on Hydrogen Explosion Mitigation of the 1991 CEC/IAEA State of the Art Report "Hydrogen in Water Cooled Nuclear Power Reactors" (EUR 14037) has been updated. A state-of-the-art report about combustion phenomena and H_2 distribution has also been produced.

As far as R&D is concerned the emphasis is both on the prediction of hydrogen distribution and combustion and on the examination of inerting procedures, namely pre- and post-accident inerting. Different inerting procedures are compared taking into account the state of the plant (containment pressure, injection temperature, chemical corrosion, etc.) and the safety characteristics (whether active or passive). The University of Munich is implementing the code RALOC for post-accident inerting analysis calculations of liquid pressurized CO_2.

University of Pisa is examining, both experimentally (in their upgraded VIEW facility) and analytically, the burning rates and the H_2-O_2 recombining mechanisms of air-H_2 in connection with the techniques of deliberate ignition and passive catalytic recombination, respectively. FzK is developing a 3D model of fast turbulent deflagration in collaboration with the Kurchatov Institute and TH Achen. Buoyancy driven natural convection is also looked at through results from the experimental facilities of HDR, Battelle Model Containment and Phebus. More generally investigations are carried out by all partners to assess and validate the modelling of hydrogen distribution and combustion effects against experimental data.

The containment threat has been analysed by NNC/Cheshire in the frame of a level 2 PSA study, with emphasis on the influence of the H_2 distribution.

KFA/Jülich is investigating prevention techniques for deflagration-to-detonation (DDT) events by computing minimum energy levels as a function of turbulence and reaction rates.

Siemens/Erlangen, together with Framatome/Paris, are looking at H_2 explosion mitigation techniques using recombiners and igniters in a multi-compartment configuration: for that purpose they are developing a field code for the modelling of the H_2 stratification.

University of Madrid is investigating post-accident active containment inerting techniques by gas injection: they are also performing a sensitivity analysis of the MELCOR code.

Future research items in this area could be the improvement of continuum models for multicompartmental distribution effects and the better prediction of critical conditions for DDT phenomena.

3.4 *Molten fuel-coolant interactions*

The potential consequences of releases of large amounts of corium into the lower plenum of the reactor pressure vessel (e.g. steam explosions) have been investigated in many severe accident studies. First estimations from separate effects studies, like fluidization, concluded that some hundreds of kg could trigger a severe molten fuel coolant interaction (MFCI), while more recent results from multi-D codes showed quantities in the order of tens of tons.

The amount of melt involved in a MFCI is mainly determined by the premixing phase for which the necessary constitutive laws (fragmentation of corium jets and drag laws) are still poorly defined for three phase mixtures (corium, liquid water and vapour). After the initial premixing phase the MFCI can develop in the propagation phase due to some triggers, including fine fragmentation and subsequent steam explosions with detonation waves.

The objective of the Molten Fuel-Coolant Interaction project is to perform several analytical and experimental studies, going from simple to more complex configurations, to be used for code validation of premixing and propagation effects.

The study of the fuel fragmentation is undertaken with droplet experiments where the transition is analyzed from the thermal fragmentation, observed at the onset of the MFCI, to the hydrodynamic one, expected when a pressure wave travels through the mixture. Results of this and other specific experiments are being used to improve the initial conditions for detonation codes in the propagation phase modelling.

A state of the art report is being produced to cover some key areas, such as the premixing, the triggering and the fragmentation mechanisms as well as the fuel-coolant heat transfer and the achievable energy conversion rates obtained from the analysis of experiments.

The results coming from the premixing experimental programme are used to validate in three phases the constitutive laws needed to describe the premixing of corium and water.

CEA-DRN/Grenoble is validating the MC-3D code against specific premixing experiments in their BILLEAU-2200 facility (5 liters of solid oxide spheres at temperatures up to 2200 °C) and against global experiments of the FARO facility of JRC Ispra (up to 150 kg of UO_2+ZrO_2+Zr). A special model is also developed for the thermally induced fragmentation resulting from vapour film destabilization.

FzK is validating their IVA-3 code against premixing experiments at their QUEOS facility (10 kg of Mo spheres at temperatures up to 2300 °C), against jet experiments at their PREMIX facility and against propagation experiments performed with simulant materials at low and high temperatures, like those of the KROTOS facility of JRC Ispra (a few kg of UO_2).

IKE/Stuttgart is improving their IKEJET code (to be implemented into the COMETA code of JRC Ispra) by developing a jet break-up model under film boiling conditions using a 3-layer approach (melt, water and vapour film). They are also improving the hydrodynamic fragmentation model of their 1D-IDEMO detonation code.

Detonation calculations are also performed by ENEL-VDN/Milan in the IFCI code (from Sandia Nat. Lab.), using a new adaptative meshing technique, in connection with some KROTOS experiments.

AEA/Winfrith is further developing their codes for premixing (CHYMES) and for propagation (CULDESAC) on the basis of specific FARO and KROTOS tests at JRC Ispra. The University of Oxford finally is performing very fundamental experiments for heat transfer and drag laws in a 2-phase flow through a rigid array of hot spheres at 800 °C.

There are no doubts that molten fuel-coolant interactions are a crucial aspect of the late phase core degradation phenomena, especially if steam explosions take place under the appropriate pressure and fragmentation conditions.

Future research should concentrate in particular on the potential for strong detonation waves and on the assessment of the thermal-to-mechanical energy conversion efficiency.

3.5 *Molten corium-concrete interaction*

Extended studies on the safety of LWRs have emphasized the important role of ex-vessel melt behaviour during postulated severe core meltdown accidents. If the cooling mechanisms of the degraded core, either by flooding recovery or by engineered heat removal systems, fail to work, the RPV may fail and molten materials and solid debris may be released into the reactor cavity.

If water is available in the reactor cavity, the melt could be cooled down by evaporating water, but care should be taken for the risk of steam explosions. The containment walls could be heated up. Airborne fission products could contribute to heating the atmosphere, which could rise the pressure in the containment.

If no water is available in the reactor cavity, as it is the case for the current reactors, the melt is only cooled by radiation heat transfer to the atmosphere and by conduction to the reactor cavity structure, generating gases (CO_2 and H_2O) from the decomposition of the concrete. These gases in turn may be partly reduced when passing through the melt and become burnable gases (H_2, CO) in the containment volume.

There is a potential for erosion of the concrete reactor cavity with the possibility of reactor cavity penetration and of loss-of-containment integrity. During the interaction of the molten core material with the concrete floor, aerosols may be generated by evaporation from the hot melt surface and by the mechanical agitation of sparging gases.

The objective of the Molten Corium-Concrete Interaction (MCCI) project is to perform several analytical and experimental studies of the key phenomena associated with the corium coolability. Some efforts are dedicated in particular to the identification of R&D requirements for core melt retention devices (e.g. core catchers).

A State-of-the-Art Report has been produced summarizing the current knowledge of MCCI and of Corium Coolability, based both on experimental and theoretical investigations.

A series of experiments regarding the coolability of corium melts are conducted at FzK in the frame of their core catcher evaluation programme with emphasis on the optimization of sacrificial materials. The spreading of corium melts on large surfaces is investigated in the modified BETA facility by flooding the melt through openings from the bottom (COMET tests). A staggered-pan core catcher with ceramic spheres is developed in another facility. IKE/Stuttgart provides a theoretical support to the COMET experiments, using their computer code IKEJET to optimize the coolant water supply pressure and IDEMO to predict the pressure build-up and the resulting mechanical loads in the case of corium fragmentation.

Other research efforts towards a core catcher concept for future reactors are carried out at CEA-DRN/Grenoble in the CORINE facility with simulants and experiments with real materials are planned at CEA-DRN/Cadarache in the VULCANO-E30 facility. One of the scopes is to validate new models for heat transfer and axially varying viscosity in 2-D corium spreading computer codes like CORFLOW from Siemens-KWU/Erlangen.

Problems of radiation heat transfer, which contributes substantially to the coolability of flat melts, and corium/concrete aerosols formation are examined through a series of AEROSTAT experiments at CEA/Cadarache. The WECHSL computer code describing the melt/concrete interaction under dry conditions is being improved, especially with respect to the oxidic melt behaviour and the thermal boundary layer. The ESCADRE program also benefits from the above experiments.

The thermodynamics of multiphase multi-component equilibria of fission products and fuel/concrete aerosols is computed by ECN/Petten and AEA/Winfrith for a series of different thermal-hydraulics conditions. In particular chemical activities and thermal measurements for corium-concrete mixtures are provided in the cases of limestone as well as limestone-common sand concretes under various severe accident conditions. The ENEL-VDN/Milano contribution consists of applying the EDF-ENEL computer code CORIUM-2D to a series of heat transfer problems including radiation and boiling effects.

Future items of interest in this research area regard the understanding of corium accumulation and spreading phenomena through experiments of core catchers with real materials and with decay heat effects.

3.6 *Radioactivity source term*

The radioactivity source term of a severe accident is defined by the magnitude, chemical species, physical form and timing of the radioactive emission to the environment. A series of assessments, sensitivity studies and state-of-the-art reviews, in connection both with the primary circuit and the containment, are needed to identify uncertainties in the database of the source term.

The objective of the Source Term project is to establish a consensus on the state of knowledge of the radioactivity source term. Priority is given to research, both through appropriate experiments and modelling activities, on source term phenomena and radionuclides which from PSA studies were found to be most important.

A report outlining a European Source Term strategy has been

produced to review source term activities both within the European Union (EU) and in a wider international context.

AEA/Winfrith, in particular, is conducting experiments to quantify the release of FPs in an oxidizing atmosphere (data to be validated against the FPREM code). They are also investigating the behaviour of non-fission product materials like boron, Ag-In-Cd and vapours or aerosols from stainless steels and Inconel, using some results of the Falcon and Harrier experimental facilities. Research is also conducted on FP transport both in the reactor cooling system and in the containment, including revaporization and resuspension (VICTORIA code), iodine chemistry (e.g. radiolytic oxidation of CsI in the INSPECT code), and iodine behaviour in the sump and under H_2 detonation conditions.

The above developments of FPREM and VICTORIA are going to be implemented into the KESS code by IKE/Stuttgart, who is also examining the behaviour of the multicomponent aerosol systems using a PSA approach (MAAP code).

Aerosol/FP transport and resuspension problems are also investigated by ENEL-VDN/Milano in the frame of the STORM Project to be conducted at JRC Ispra, together with primary circuit thermal-hydraulics aspects of the radionuclide behaviour. CEA-IPSN/Cadarache is conducting experiments both at their PITEAS facility to validate a model for the hygroscopic effects on particle behaviour and in their TUBA facility to investigate the diffusiophoresis of CsI. Measurements of heat and mass transfer coefficients are also undertaken to understand better the volatilization process. Moreover some PHEBUS-FPTO data are provided for a common benchmark calculation exercise conducted with JRC Ispra.

The contribution of ENEA-ERG/Bologna and the Universities of Bologna and Pisa consists of interpreting thermal-hydraulics aspects of containment behaviour, like natural circulation and bulk condensation of sprays. The pool scrubbing analysis is performed by CIEMAT/Madrid with emphasis on the aerosol retaining capacity of the churn-turbulent region.

Long term release effects, like the low volatile gas release from deep pools, are investigated both by the University of Bochum and by Siemens-KWU/Erlangen. The Forschungszentrum Rossendorf (FZR) has been preparing a review of chemical vapour reactions with surfaces and aerosols. The effect of containment thermal-hydraulics on aerosols behaviour are investigated by NNC/Knutsford in view of MCCI and iodine chemistry calculations, by BATTELLE/Frankfurt using the VANAM-M2 experiments, by VTT using the AHMED and VICTORIA facilities, by Universita Politecnica Madrid to improve the ESTER code of JRC Ispra and by Riso National Laboratory to improve their source term modelling.

Further efforts in this area should concentrate on the full understanding of the complex series of Phebus-FP experiments.

3.7 *Containment integrity*

The containment is the last barrier protecting the population and the environment against radioactive consequences of a hypothetical severe accident. Therefore the assurance of its integrity is a key issue. The objective of the Containment project is to perform several analytical and experimental studies. Four aspects determining the containment integrity are examined: leak assessment, the consequences of hydrogen explosions, the impact of missiles on the containment and the containment coolability after a severe accident.The objective of the program is to reach a well-based common position on these subjects.

The experimental program of the leak assessment activity is aiming at the characterisation of the leakage of air/steam mixtures through cracks with rough walls in the SIMIBE facility of CEA-DRN/Saclay. Theoretical methods are developed by ElectroWatt/Horsham for predicting the 2D time-dependent behaviour of gas/vapour/liquid/aerosol mixtures in fine capillaries. Microcrack plugging problems by aerosols are examined by AEA/Winfrith with emphasis on the influence of the pressure drop and the water vapour content in the carrier gas. Aerosol deposition, resuspension and saltation phenomena are also investigated. CASTEM 2000 is used to compute cracking probabilities and concrete deformations.

The hydrogen detonation activities, both at CEA Saclay and FzK, are

oriented towards the experimental validation of computer tools for the prediction of deflagration-to-detonation transition (DDT) and of detonation effects, with emphasis on the hydrogen chemistry and on the treatment of shock waves in 3-D geometries, using the codes PLEXUS, D3D and DET3D.

The concrete impact activities are oriented towards the experimental validation of numerical models for concrete impact and perforation, with emphasis on nonlinear strain-rate effects and failure mode analysis. CEA Saclay is conducting a series of experiments on high resistance reinforced concrete slabs. Blind benchmark computations are performed by the participants for different impact energies.

University of Karlsruhe is developing a new constitutive stress-deformation law and a special "slide" finite element to treat the friction energy produced under high pressure impact conditions during the steel/concrete interaction, using experiments from the Fraunhofer Institute ICT.

In the framework of decay heat removal assessments ENEA-ERG/Rome is conducting experiments on the thermo-mechanical cracking behaviour of high-performance concrete at elevated temperatures (100 - 400°C) both in compression and in tension.

Concrete calculations, including strain-rate effects and material behaviour under very high pressure, are also performed by ANPA/Rome using DYNA-2D, with emphasis on structural identification techniques developed at the La Sapienza University of Rome.

Containment thermal-hydraulics assessments are being conducted by CEA/DRN Saclay in view of the preparation of the MISTRA test matrix, using thermal-hydraulics models like CONTAIN, JERICHO and TRIO for the stratification problems. Another series of thermal-hydraulics studies are performed by FzK in the PASCO facility (PASsive containment COoling by natural air convection).

Future efforts in this area should concentrate on the 3D modelling of detonation waves in real reactor containments and on the understanding of all loss-of-leak-tightness mechanisms.

3.8 *Accident management support*

In order to be able to control and mitigate an accident, the operators of NPPs should have in due time a realistic and correct picture of the accident and of its progress. Hence signal validation is a main issue under severe accident conditions, especially if instruments are working beyond their specification range.

The objective of the AMS project is to define, investigate and develop means and methods providing reliable information and diagnostics as well as support tools for accident management. A great deal of efforts are dedicated in particular to conceptual definitions of the degradation of the instrumentation under accident conditions.

Detailed investigations are being conducted in signal validation methodology and sensor modelling/signal processing. Two state of the art reviews (SOARs) have been produced on the current AMS practices in European NPPs, namely "Instrumentation and Signal Validation in Accident Situations" and "Operator Assisting Systems for Accident Management". Specific attention is given to identifying the safety functions of the different classes of operator support systems.

There are two basic approaches to match the requirements of reliable information about the validity of a plant signal: model-based methods realizing functional redundancy and noise diagnostic methods, using the signatures of inherent signal noise as "finger prints" of specific sensors and plant conditions.

Sensor modelling activities performed at CEA-DTA/Saclay are aimed at the improvement of a computer model for a fission chamber in an extended range of faulty operating conditions. Experiments are being run in the Ulysse test reactor of Saclay.

In the field of operator assisting systems for accident management, investigations about real-time monitoring and decision-making techniques, using neural networks, advanced modelling and noise analysis techniques, are carried out both by IST/Garching and by ECN/Petten.

Framatome/Paris is developing expert system based strategies for design and maintenance of emergency guidelines in connection with an "automatic operator model". CEA-DRN/Cadarache investigates the operators role with respect to automation.

The development of a knowledge-based operator aid system, able to interact visually with the operator, is developed by ANSALDO/Genova. ECN/Petten works also on computational aids for technical support centres to predict critical milestones.

NNC/Knutsford is developing operating instruction displays to optimise command/control under emergency conditions.

Siemens-KWU/Erlangen is developing the SIROG (Situation Related Operation Guidance) procedure which is based on dynamic documents for the operator, using adaptative algorithms and extrapolations from recorded plant data. An advanced expert system, called "OPerator Advisor" (OPA), is also developed by TRACTEBEL/Bruxelles with the aim of assisting the operator team for optimal recovery following an accident.

GRS/Garching together with IST/Garching is working on fast predictive models which help the operator in providing immediate answers for the evaluation of emergency operating procedures.

Future efforts in this area could concentrate on a better use of the phenomenological understanding, made available in the previously mentioned projects, in order to develop advanced instrumentation and control as well as fast running physical models, able to provide the operators in charge of the severe accident management with the right plant information in due time.

4. CONCLUSIONS

The present state of knowledge about severe accident sequences and fission product behaviour allows to extend the traditional defense-in-depth concept by introducing accident management procedures and in particular accident mitigation measures as an additional line of defense.

The Community action described has undoubtedly contributed to consolidate the knowlege for most of the severe accident phenomena. Further, the close cooperation between the different partners in the projects has essentially allowed to come to a common understanding. This should become visible at the occasion of the FISA-95 symposium (20-22 November 1995, Luxembourg), organised by the EC with the aim of concluding the present Reinforced Concerted Action by a presentation of the main results obtained.

Research in the area of severe accidents will be continued also under the Nuclear Fission Safety programme which has just been approved by the Council of Ministers and which is currently being implemented. Emphasis will be put to further improve the knowledge of severe accident phenomena and on the development of appropriate accident management techniques.

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