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Results and exploitation of FP-4 and FP-5 research in the area "Safety of Existing Installations"

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ABSTRACT

In this paper an overview is given of the most important achievements of the research programme cofinanced by the European Union (EU) in the area of LWR safety over the FP-4 and FP-5 periods from the end-users point of view. The end-users are : the contracting organisations (i.e. utilities and associated engineering companies, regulatory bodies and associated technical safety organisations, manufacturing industry and associated services), the non-contracting organisations (including decision makers and opinion leaders) and the European Commission.

Besides Community research strategy and programme implementation aspects in general, this paper is focusing on the S/T achievements obtained by multi-partner projects in the

- 7 clusters of multi-partner projects in Euratom FP-4 (1994-1998): AGE for structural ageing, INV and EXV for in-vessel core degradation and ex-vessel molten corium coolability, ST for radiological source term, CONT for containment integrity, AMM for accident management measures and INNO for innovative safety features - the total cost of the 67 mutipartner projects comprised in this Community research was € 71.3 million, out of which € 35.9 million was contributed by the EU budget

- 3 clusters of multi-partner projects in Euratom FP-5 (1998-2002) : PLEM for plant life extension and management; SAM for severe accident management and EVOL for evolutionary safety concepts - the total cost of the 71 mutipartner projects comprised in this Community research is € 85.4 million, out of which € 43.5 million is contributed by the EU budget.

The objectives of this Community research are discussed and a number of FP-4 and FP-5 projects are selected to demonstrate to what extent the proposed objectives were indeed met. Besides technological requirements, socio-economic aspects are becoming increasingly important due to the level of public and political acceptance and to the economic pressure of deregulated electricity markets : this is also discussed. Finally the needs for EU-wide nuclear knowledge management are discussed and some conclusions are drawn regarding past and future Euratom research of common interest in the area of reactor safety.

INTRODUCTION

PRIORITIES OF EURATOM RESEARCH IN LWR SAFETY

The fundamental safety objective for nuclear power plants (NPPs) consists in protecting the public and the environment from the harmful effects resulting from ionising radiations. Traditionally the technological problems of nuclear reactor safety are related to the 3 basic safety functions, namely: controlling the power, cooling the fuel and confining the radioactive material. In the relevant Euratom research programme, the emphasis was put untill now on these technological problems, leaving man-machine and organisational problems to national (plant dependent) research.

As a result of a long series of technological (both exprimental and theoretical) research programmes, connected to plant operational feedback, the 3-levels defence-in-depth strategy was developed against accidental radioactivity releases, i.e.: (1) prevention of abnormal operation and failures (quality control), (2) control of abnormal operation and detection of failures (surveillance and protection), and (3) control of accidents within the design basis (safeguards systems). Usually associated with this approach are the following concepts: (1) the multiple barrier design for the confinement of radioactive material, (2) the protection and safeguard systems to ensure the integrity of the barriers, and (3) the regulatory procedures (for example, the safety analysis reports) to ensure the health and safety of the plant workers and of the population. The multiple barrier design consists traditionnally in the following three elements : fuel pin (primary barrier), RPV and reactor coolant system (secondary barrier), and containment building (tertiary barrier).

From an industrial and regulatory point of view, most of the technological problems seem to have received a standard solution. Despite the historical safety record of NPPs of Western design, however, the key players believe that research is still needed to further increase both the safety and the performances of these power plants in line with the steadily growing pressure of regulatory and market forces as well as of the public opinion. Research is still needed, both at the plant dependent level (short-term results expected, national research) and at the generic level (medium- and long-term results expected, EU and/or international research). The time period corresponding to medium- and long-term is meant in terms of Euratom framework programme periods : medium is in the order of 4 years and long is twice as much.

The EU research activities discussed here are naturally of the generic type. From a defence-in-depth point of view, EU research is more specifically related to the above-mentioned level (2) and to a new level, say (4), in line with the increasingly stringent safety requirements. From an end-users point of view, EU research is focussing on the above-mentioned multiple barrier design under both operational (PLEM cluster) and severe accident (SAM cluster) conditions, while exploring new safety concepts of the evolutionary type (EVOL cluster).

What the contracting organisations in EU reactor safety research programmes are looking for, besides S/T discussions, is an international « neutral » platform, bringing together players who have not the opportunity in their own country to discuss S/T issues of common interest (e.g. industrial competitors or regulators versus utilities). Another benefit, appreciated in the current deregulated electricity market, is to hear from each other about technical implementation of international safety regulations (socio-economic aspects) and so to slowly come to a common safety culture.

The EU establishes research priorities on the basis of discussions with all international organisations concerned (e.g. IAEA [1] and OECD/NEA/CSNI [2],) and with all stakeholders within the EU (e.g. the regulatory authorities [3], the utilities [4], and the industry [5]). Decisions about framework programmes and implementation instruments (including calls-for-proposals) are then taken by the Commission after consultation of the various programme Committees.

OBJECTIVES OF THE END-USERS, I.E. THE CONTRACTING ORGANISATIONS (E.G. UTILITIES, REGULATORY BODIES AND MANUFACTURING INDUSTRY), THE NON-CONTRACTING ORGANISATIONS AND THE EUROPEAN COMMISSION

As far as level 2 of the defence-in-depth is concerned (i.e. surveillance and protection), the FP-4 cluster AGE (« structural ageing) and the FP-5 cluster PLEM ("Plant Life Extension and Management"), were/are focusing on the following items: integrity of equipment and structure; on-line monitoring and maintenance; organisation and management of safety. Most of this research is of the operational type (as defined by A. Alonso, FISA-2001 [6]) : no wonder then that industry (in particular, the utilities) – primarily interested in plant safety and performance - is taking most profit from this type of Community research, whenever the issues treated are not requiring a prompt solution (results expected in the medium-term). It is clear that Community research is not appropriate for technical problems requiring a quick solution - remember the well known slogan: « Industry is interested in good solutions in one year and not in excellent solutions in ten years ». Community research is not aimed at providing « good solutions in one year », not only because of the inherent inertia of international organisations but also because problems of the operational type are usually plant dependent and commercially sensitive in nature. On the other hand, Community research may aim at providing « excellent solutions in ten years », especially when it is related to advanced innovative (as opposed to evolutionary) technologies.

What <u>utilities</u> are primarily looking for in participating in Community research of the operational type, seems to be a common platform for development of scientific knowledge and exchange of experimental facilities with a twofold aim :

- EU integration : de-fragment the many existing governmental/industrial RTD programmes in order to share RTD costs related to large experimental and numerical simulation programmes (in line with the EU policy for FP-6 : the European Research Area) - « value for money ».
- EU knowledge management : give and take information about S/T developments of common interest amongst partners/competitors and discuss industrial applications throughout Europe (in particular, whenever new national and/or EU policies might affect their commercial strategies).

As far as level (4) of the defence-in-depth is concerned (i.e. severe accident management), the FP-4 clusters INV and EXV (in-vessel core degradation and ex-vessel molten corium coolability), ST (radiological source term), CONT (containment integrity), and AMM (accident management measures) and the FP-5 cluster SAM ("Severe Accident Management"), were/are focusing on the following items: assessment of severe accident risks; severe accident management measures. Most of this research is of the regulatory type (or confirmatory as defined by A. Alonso, op. cit.) : no wonder then that regulatory bodies (in particular, their associated technical safety organisations) – primarily interested in plant safety - are taking most profit from this type of Community research, whenever the issues treated are not requiring a prompt solution (results expected in the medium-term).

What <u>regulatory bodies</u> are primarily looking for in participating in Community research of the confirmatory type, seems to be again a common platform for development of scientific knowledge and exchange of experimental facilities, but with a slightly different twofold aim :

- EU integration : de-fragment the existing national/regulatory RTD programmes (in the sense of the European Research Area) in order to share the RTD costs needed to improve knowledge and practices of common interest, in particular, to get an « international confirmation » to consolidate solutions for safety cases
- EU knowledge management : give and take information about S/T developments of common interest amongst partners and discuss regulatory practices throughout Europe (in particular, whenever new industrial and/or EU policies might affect their regulatory strategies).

Finally, the activities dealing with evolutionary safety concepts are put together in the FP-4 cluster INNO (innovative safety features) and the FP-5 cluster EVOL ("Evolutionary Concepts"), focusing on the following items: evolutionary safety concepts (using, for example, passive safety systems); specifically in FP-5, advanced fuel technologies (high burn-up and MOX fuel). Most of this research is of the promotional type (or exploratory as defined by A. Alonso, op. cit.) : no wonder then that industry (in particular, manufacturers and the associated services) – primarily interested in plant safety, performance and innovation - are taking most profit from this type of Community research, especially when it comes to design new concepts far away from the market (results expected in the long-term). It is clear that this type of reactor safety research is usually more appropriate for technical problems that can wait « ten years » to receive a solution.

What <u>industry</u> is primarily looking for in participating in Community research of the exploratory type, seems to be a common platform for development of scientific knowledge and exchange of experimental facilities with a twofold aim :

- EU integration : build up new common national/industrial RTD programmes (in the sense of the European Research Area) in order to share the RTD costs for acquiring new knowledge of common interest and developing demonstration products or services
- EU knowledge management : give and take information about some S/T breakthroughs amongst partners/competitors and discuss potential practical applications throughout Europe (in particular, whenever new industrial and/or EU policies might affect market opportunities in the far future).

As far as research policy goes, the Commission interests, so to say, are to maintain industrial competitiveness within the EU. As a reminder, Title XV "Research and technological development" of the Amsterdam Treaty (1997), which provides the legal basis for Community research in general, states the following in its Articles 163 and 164 : « The Community shall have the objective of strengthening the scientific and technological bases of Community industry and encouraging it to become more competitive at international level, while promoting all the research activities deemed necessary by virtue of other Chapters of this Treaty ».

Under FP-4 and FP-5, the Commission was insisting on industrial competitiveness as the basis for Community research. Under FP-6, the Commission is adding another dimension : actually, it is proposing a frame (including implementation instruments), for the construction of the European Research Area (ERA / http://europa.eu.int/comm/research/era/leaflet/en/index_en.html). As a consequence, EU support will go preferably to proposals for RTD projects (or « common platforms ») demonstrating the three following qualities :

- European common benefit and added value (following the subsidiarity principle)
- integration at various levels (preferably durable - for example, cross-disciplinary work programmes, mix of governmental and industrial budgets), and/or
- contribution to other EU policies such as energy and internal market (« nuclear package »), enlargement of the EU (ex-PHARE programme), protection of man and environment, etc.

More generally, as far as other policies are going, the Commission interests are expressed in various « communications ». For example, regarding the energy policy, it is worth recalling the current debate about the "Draft Proposal for a Council (Euratom) Directive Defining the basic obligations and general principles for the safety of nuclear installations". On 6 November 2002, the Commission launched (amongst others) a proposal for a framework directive on safety, laying down the basic obligations and general principles concerning the safety of nuclear installations during operation and decommissioning. This proposal was formally adopted by the Commission on 30 January 2003 - see http://europa.eu.int/comm/energy/nuclear/nuclearsafety.htm. It will introduce common safety standards and monitoring mechanisms to guarantee that common methods and criteria will be applied throughout the enlarged Union. Worth mentioning in this context is also the interest of the EC for the WENRA activities regarding the harmonisation of safety approaches for existing nuclear power reactors (see, for example, their « Pilot Study on Harmonisation of Reactor Safety in WENRA Countries » / WENRA = Western European Nuclear Regulators Association / members = BE, FI, FR, DE, IT, ES, SE, UK + regulators from the CEECs). Community research might have a role to play as technical support in these debates about reactor safety standards.

In this paper, a number of projects are selected out of the clusters PLEM, SAM and EVOL in both FP-5 and FP-6 from the end-users point of view (i.e. the contracting organisations : utilities, regulatory bodies, manufacturing and service industry and the Commission itself). Some practical applications are given to illustrate the compliance with the (medium and/or long-term) needs of these end-users.

1. FP-4 AND FP-5 RESEARCH IN PLANT LIFE EXTENSION AND MANAGEMENT (main end-users = utilities and associated engineering companies)

BACKGROUND

Nowadays, a large number of nuclear reactors have been operating for longer than 20 years. In the EU only, a total of 65 reactor units were put in commercial operation before 1980 (and a total of 78 after this date). As a consequence, the nuclear industry is increasingly interested in research activities aimed at better understanding and managing materials aging phenomena (i.e. changes in microstructural and mechanical properties due to irradiation, etc.). More importantly, the optimisation of the operational conditions of aged reactors (using, for example, appropriate prediction tools for evaluating the safety margins) and the decision process about plant life management (involving, for example, replacement of equipment) are becoming key issues for those in charge of plant safety and performance.

Looking at the future of the nuclear power plants, there is a natural drive to extend their lifetime where this can be achieved safely, bearing in mind that the lifetime of a nuclear plant is definitely limited by the aging of non-replaceable components like the nuclear reactor pressure vessel (RPV). Knowing that operating license for most countries will broadly expire by 2015-2020, many utilities and vendors of equipments nowadays are very active in improving techniques to ensure both the performances and the safety of their plants until this expiration date and even beyond it. Indeed extending the life of units is usually considered a worthy investment, depending on license renewal obtention as well as replacement equipment and operational safety requirements.

RTD needs in plant life extension and management

Following the recommendations of the Euratom programme committees and in line with the conclusions of other international organisations (discussed earlier), it was decided to focus Community research on the safety and performance of the RPV and reactor coolant system (i.e. the integrity of the second barrier against radiological releases under operational conditions). Bearing in mind the general objective of FP-5, i.e. economic competitiveness of technological developments, – which is the challenge faced especially by the nuclear utilities – the following issues are requiring nowadays particular attention:

- plant availability (ensure integrity of equipment and structures through in-service inspection as well as repair, replacement and back-fitting whenever needed);
- core optimisation and increased plant efficiency (install on-line surveillance systems and new control processes) ;
- standardisation of industrial norms and of quality requirements.

Actually two types of research are proposed in Community programmes : knowledge driven and application driven. To understand the behaviour of the second barrier under operational conditions, research of the fundamental (phenomenological) type is needed. To monitor (and maintain) the integrity of this barrier under operational conditions, research of the applied (industrial) type is needed.

The knowledge driven (phenomenological) research can be broadly divided in two classes corresponding to the loading term and the response of the RPV and reactor coolant system, respectively, that is:

- research related to determining the loads to the second barrier, radiological, thermal (including irradiation induced heat effects) and mechanical (pressurisation processes), and
- research related to the response of this barrier to these loads, with the aim to predict the damage due to irradiation, stresses and strains.

To determine the loads to the barrier, it is necessary to investigate the thermo-hydraulical, thermo-mechanical and dynamic conditions upon the barrier, taking into account the various material properties with a view to predict the damage processes. To determine the response of the barrier to the above-mentioned loads, it is necessary to investigate the micro- and macro-cracking of the barrier subject to thermal and/or dynamic loads and the subsequent damage or loss of mechanical properties.

The application driven research can be broadly divided in two classes corresponding to detection of failures and to standardisation of quality requirements, respectively, that is:

- surveillance and protection (on-line monitoring) for the detection of failures
- operational safety guidance and EU standardisation for quality requirements (safety culture in general).

RTD needs covered by Community programmes

In the 31 projects belonging to the clusters AGE and PLEM, the emphasis of the work programme is on the reactor’s very high radioactive inventory which is perceived as a potential danger for people and environment. Community research is conducted in preventive measures to ensure plant integrity. Under FP-4 and FP-5, the following three key issues have been identified:

- (1)_harsh environmental conditions put on structures and equipments (for example, reactor pressure vessels with coolant at temperatures of 290 °C - 325 °C, pressures up to ~ 15.5 MPa and end-of-life irradiation doses of ~ 10^{19} n/cm^2)

- (2) inspections, tests, maintenance and supervision aimed to monitor the behaviour of components and structures and/or to control the radiation effects, while maximising the plant performances

- (3) dissemination of safety culture, with emphasis on the harmonisation of best practices through the drafting of European handbooks for the benefit of both industry and safety authorities.

No wonder that the main actors in the area of “plant life extension and management” are the European utilities or the associated engineeering companies, that is: EDF in France; E.ON AG and RWE AG in Germany; British Energy in the UK; ENDESA and IBERDROLA in Spain, VATTENFALL AB in Sweden; TRACTEBEL in Belgium; and FORTUM in Finland.

Under FP-5, the total EU budget to be spent for the 31 projects in the cluster PLEM amounts to EUR 18.2 million, which represents roughly half of the total value of these projects (EUR 35.4

million). This might be compared to the total EU budget spent under the 4th framework programme 1994-1998 (FP-4) for the 11 projects in the AGE cluster, which was EUR 2.1 million.

1.1 INTEGRITY OF EQUIPMENT AND STRUCTURES IN PWRS AND BWRS

Materials properties deteriorate during plant operation. Irradiation, corrosion and thermal fatigue are phenomena participating in the "ageing" of the structural components in NPPs decreasing thus the safety margins of the structures. Dynamic loads (e.g. waterhammers in piping systems) are also deteriorating the structures and equipments.

To prevent any in-service failure of the RPV and in general of the reactor cooling system, stringent operational rules are necessary, based on the understanding of the above phenomena, quantification of their impact and their synergetic effect, as well as on accurate evaluation of safety margins.

1.1.1 Measuring and understanding irradiation embrittlement, thereby contributing to the improvement of prediction tools for the residual lifetime of NPPs

One of the most important material properties for ensuring structural integrity is the fracture toughness, which measures the resistance of the material to the propagation of a hypothetical sharp crack which in the safety case is (conservatively) assumed to be present in the pressure vessel from the time of its construction.

In unirradiated steel, there is a transition, as steel is cooled down, from high toughness to low toughness. This transition is accompanied by a change in the mode of fracture from ductile to brittle. As the irradiation dose increases, the fracture toughness decreases, with a shift of the ductile-to-brittle transition towards the lower temperatures. At typical operating temperatures, it is necessary to be in the upper region, away from any risk of brittle failure. By predicting the extent of the irradiation induced shifts it is possible to modify the operating rules, and the temperatures at which the vessels actually operate, to ensure that they always operate in the ductile region.

Hence, during operation of a nuclear power plant, changes to the fracture properties of the steels (i.e. in particular, the fracture toughness of the pressure vessel material) are followed as a function of the dose rate received by testing at regular intervals surveillance specimens, which are exposed, inside the pressure vessel in locations close to the vessel wall.
Irradiation embrittlement is the subject of a number of different projects, focusing on reactor pressure vessel and internals ageing aspects.

- **Reduce the uncertainties associated with irradiation induced RPV embrittlement; validate measuring methods and produce data to evaluate the actual impact of irradiation on material toughness**

Demonstration of the continued integrity of reactor pressure vessels (RPVs) requires that the ductile-to-brittle transition temperature (DBTT) of the RPV materials can be predicted as a function of irradiation exposure. Irradiation-induced changes in the DBTT are conventionally monitored by measuring the shift in transition behaviour according the so-called reference temperature methodology - rather conservative - which makes use of Charpy-V notch impact testing. The FP-4 REFEREE project, (Relation Between Different Measures of Exposure-Induced Shifts in Ductile-Brittle Transition Temperature) undertook a comparison of methods for measuring irradiation-induced shifts in DBTT on four different materials that cover a range of different reactor systems of interest within Europe. The obtained results were compared with existing data and predictive trend curves to crosscheck with expectations. The discussion of the results focused mainly on the comparison between static and dynamic fracture toughness transition temperature shifts as well as between fracture toughness and Charpy–V shifts. FP-5 project FRAME complements this effort by applying the Master curve method that allows to measure directly the fracture toughness – as opposed to the current practice of indirect measurements. Data are produced based on steels for RPVs (BWR and PWR) but also on a series of model alloys creating thus an important database on irradiation induced lose of toughness.

Other projects in FP-5: PISA, PRIS, INTERWELD, complement this effort by providing data on the irradiation effect to phosphorous segregation on grain boundaries, internal stresses evolution, corrosion behaviour

- **Exploit better the surveillance materials, a concern of both operators and regulatory authorities facing the continuous decrease of surveillance materials availability**

Availability of surveillance material is a crucial problem both for operators and authorities that can complicate, at least, plant life assessment and even more life extension procedures.
On this subject, the FP-4 project RESQUE (Reconstitution Techniques Qualification & Evaluation to Study Ageing Phenomena of Nuclear Pressure Vessel Materials) succeeded a breakthrough by developing a set of 'Recommendations for Reconstitution of Non-Irradiated and Irradiated Charpy-size Specimens' strongly needed to both operators and authorities to cope with the unavoidable continuous decrease of surveillance material availability. The provided guidelines and recommendations through RESQUE allow a potential user to qualify their reconstitution equipment and develop a reconstitution methodology. They also identify a number of procedures and limitations for the reconstitution process.

Other projects in FP-5: deal directly or indirectly with the subject as the already mentioned project PRIS that has as a secondary objective to validate a method for optimised use of irradiated material for mechanical testing based on use of miniaturised test specimen that is of major importance for both handling irradiated specimens during testing and for surveillance material economy.

- **Evaluate the actual irradiation doses especially in reactors without surveillance programme, using relatively simple laboratory tests**

The irradiation ageing severity is function of the irradiation doses received. Accurate evaluation of the irradiation field is crucial for estimation of the material actual "age". Breakthrough in the area is expected by the FP-5 RETROSPEC project. A method is being developed and validated for retrospective dosimetry that is crucial for VVER reactors without surveillance programme. Codes of practices for use in laboratories with an average-level equipment are under development.

Other projects: AMES-Dosimetry, and MADAM (FP-4) projects supported the collaborative efforts of the AMES network on harmonising dosimetry practices and on developing a pilot data base for dosimeter parameters and REDOS (FP-5) that aims at improving dosimetry for irradiated steel and qualifying a methodology for radiation field parameters monitoring.

- **Propose a EU strategy for materials testing reactors (MTR), including scenarios for task sharing and renewal of their capacities**

Most of the materials testing reactors (MTRs) will be more than 40 years old by 2010. The objective of the FP-5 network FEUNMARR is to determine the future European irradiation needs in MTRs. The following items are covered in the final report: materials and fuel for current and future reactors, medical applications and productions, neutron beams for research, and other applications as neutron radiography and isotope production.

Worth mentioning here is the FP-5 network ITEM, aimed at developing multiscale numerical simulation tools for the prediction of radiation effects on materials. These tools known as "Virtual

Test Reactors" (VTR) are becoming increasingly important in view of the expected insufficient availability of test reactors and hot cell facilities in the future.

1.1.2 Monitoring and understanding corrosion and thermal-hydraulics, thereby contributing to the optimisation of operational conditions and to the improvement of prediction tools for the residual lifetime of NPPs

A review of plant concerns has revealed that besides embrittlement of the RPV, corrosion in synergy with irradiation becomes an important degradation mechanism especially for internals. For example, ageing of reactor internals is in almost all cases associated with irradiation assisted stress corrosion cracking (IASCC). For BWRs, in particular, the 2 major corrosion concerns at present are cracking of the core shroud/plate and possible cracking in the lower plenum region.

♦ **Reduce the uncertainties associated with corrosion mechanisms; validate testing methods and produce experimental data to evaluate the actual impact of corrosion on the integrity of RPV and internals**

The FP-4 shared-cost action DISWEC (Evaluation of Techniques for Assessing Corrosion Cracking in Dissimilar Metal Welds) evaluated the suitability of various laboratory test methods to study the degree of susceptibility of Bi-Metallic Weldments to an unacceptable environmentally assisted cracking (EAC) in service environments. It provided a number of recommendations. The availability of such recommendations is of benefit both, to plant operators, as guidance on the most cost effective means of obtaining the data necessary to validate plant components for service, and to regulatory authorities to aid their judgement in the suitability of data underlying plant safety cases.

Other projects : within the FP-5 project CASTOC, environmentally assisted corrosion (EAC) of low alloy steels, for RPV (PWR and BWR), under static and cyclic conditions is studied with the aim to improve service operation and code implementation. The already mentioned FP-5 projects PRIS and INTERWELD, by producing data on the evolution of the microstructure and the internal stresses of internals (austenitic steel) due to irradiation, provide important inputs for assessing their susceptibility to stress corrosion cracking after irradiation.

♦ **Develop predictive tools for thermo-hydraulics problems (best-estimate numerical codes for condensation-induced waterhammers and CFD for local boron dilution scenarios), thereby contributing to the acceptance of numerical modelling in the industrial community**

It is worth recalling that decisions about repair/replacement/back-fitting and operational performances under the usual high safety standard conditions depend not only upon the challenges of irradiation or corrosion effects on materials integrity but also upon the dynamic loadings during operation (generated, for example, by condensation induced waterhammers in pipes and open networks). The WAHALOADS project is producing a numerical tool for the prediction of dynamic fluid-structure interactions of direct interest to the contracting utilities and engineering companies concerned with piping systems subject to extreme conditions.

The FP-5 project FLOMIX-R is performing a set of mixing experiments that are supported by CFD calculations. Emphasis is on slug mixing phenomena relevant for local boron dilution scenarios and thermal fatigue. Improved measurement techniques with enhanced resolution in time and space are tested for mixing phenomena of direct interest to improve operational practices.

1.1.3 Evaluate the safety margins, thereby contributing to the optimisation of operational conditions and to the improvement of prediction tools for the residual lifetime of NPPs

Scientific knowledge progress but also feed back from field experience is continuously taken into account and implemented in reviewing the safety margins. This is of great assistance to both operators and authorities in charge of the evaluation of the plant safe remnant life.

- **Develop new structural integrity procedures to improve the prediction of structural safety margins**

The positive effect on materials behaviour of phenomena as constraint or pre-stress warming appears to be quantifiable today. The FP-5 projects VOCALIST and SMILE deal with these subjects able to provide arguments for life extension. Obviously of high industrial interest the projects involve also regulators in order to make best use of the results. A first issue of a “handbook of best practices on constraint based structural integrity procedures” has been produced.

- **Develop procedures for crack growth assessment, supported by operational feedback**

Bimetallic welds (BMW), connecting ferritic components with austenitic piping are used in safety class systems of all PWR and BWR plants. For PWRs, the BMWs of particular interest are those attaching the piping systems (made of stainless steel) to the various nozzles of the RPVs, SG and pressurizer. Because of their metallurgy, these weldments are particularly prone to localised cracking. The integrity of the BMWs without and with hypothetical cracks has to be justified in all conditions for the life of the plant.

Two projects were especially designed to support the assessment of the cracks identified in Bi-Metallic Welds (BMW) in different US and European plants contributing in the large effort of establishing safety margins. The FP-4 project BIMET (Structural Integrity of Bi-Metallic Components) and its FP-5 successor ADIMEW. BIMET contributed to the development and verification of analysis methods which describe the behavior of an external circumferential defect at the surface of a Bi-Metallic Weld. ADIMEW applies these methods to an industrial scale testing. Dissimilar metal welds, representative of a 16’’ diameter girth weld of a French pressuriser, have been fabricated with artificial defects, and are tested under operating temperature (300°C) and bending conditions. Different types of testing and analytical tools (fracture mechanics, residual stress measurement and calculation, finite element analysis, etc…) are used in order to evaluate the crack driving force. It aims at providing “Recommendations for codes and standards on flaw evaluation for Dissimilar Metal Welds”.

Another FP-5 project THERFAT**,** was initiated after the Civaux-1 incident (May 1998). Such events indicated that certain piping system Tee’s are susceptible to turbulent temperature mixing effects that cannot be adequately monitored by common thermocouple instrumentation, putting the reliability of integrity evaluation in doubt. THERFAT proposes to review field data and to perform advanced thermohydraulic flaw simulations and stress and fracture analysis. Critical elements of the procedure are investigated by targeted verification tests. “A European Methodology on Thermal Fatigue” is under preparation, containing proposals for improved thermal fatigue assessment procedures.

1.2 ON-LINE MONITORING AND MAINTENANCE IN PWRS AND BWRS

Developing new NDT methods, or enhancing the detection capability of existing ones, for early detect and following damage is on the benefit of both utilities and regulators. Both of them also, being the main concerned actors, follow with great interest the current debate on the potential development of a risk-informed regulatory environment. Finally, maintenance is not only a safety issue but also it is an important component of the economic performance.

- **Assess detection capabilities of existing NDTs for both reactor coolant system and containment building; benchmark and explore capabilities of new NDTs aimed at early damage detection**

  - **Reactor coolant system integrity (second barrier)**

The FP-5 project SPIQNAR aims to increase the sensitivity of ultrasonic methods on detecting cracking in austenitic steels mainly by applying modern techniques for signal processing to eliminate signal noise.

The FP-5 project LIRES is already on the point to develop a reference electrode able to operate in the reactor (BWR or PWR) within operational conditions to follow in situ the evolution of the electro-chemical potential and thereby the possibility for crack initiation. FRAMATOME and STUDSVIK are interested in its commercialisation.

Other projects : AMES-NDT (FP-4) supported the collaborative efforts of the AMES network in identifying non destructive techniques able to early detect damage and GRETE (FP-5) by applying those qualified and promising techniques to real ex-service materials.

  - **Concrete containment integrity (third barrier)**

The CONMOD project is improving the interaction of NDT and finite element calculations in order to optimise maintenance activities for concrete containments. Emphasis is placed on the identification of possible critical defects and damage mechanisms, using laboratory-scale and full-scale experiments (for example, in Sweden's Barsebäck-1 reactor). It aims also at standardisation of the concrete testing procedures throughout the EU.

- **Develop methods for quantitative risk analysis and evaluate their use as a decision support tool in risk-informed environments (industrial or regulatory)**

The EURIS FP-4 project supported the ENIQ network efforts to establish a European methodology for risk based assessment relevant for the needs of plant operators in order to identify safety-significant categories for power plant components, and to optimise the targeting of inspections whilst maintaining or even increasing the safety. EURIS arrived to the positive conclusion that the explicit use of risk as a basis for determining in-service inspection strategy within the European nuclear industry is in a viable way forward and recommended an integrated programme to develop the risk based philosophy for introduction within the European community. "Guidelines for a European framework for risk informed inspection" were developed within this utilities driven concerted action.

Other projects: The FP-5 project NURBIM develops improved procedures to identify where the highest likelihood of damage/failure is located in plant and provide quantitative measures of the associated risk. It focuses on the definition of best practice methodologies for performing risk-based analysis and establishing a set of criteria for the acceptance of risk quantities that can help regulatory bodies in Europe to accept risk based inspection (RBI) as a valid tool for managing plant safety.

The FP-5 concerted action SPI (Evaluation of Alternative Approaches for Assessment of safety Performance Indicators for NPP) reviews and evaluates the application of safety performance indicators – in combination with other tools, like PSA – in order to maintain and improve safety of NPPs. It also recommends methods that can be used in a risk-informed regulatory system and environment. An open workshop will be organised for regulatory bodies and utilities.

♦ **Develop tools for best maintenance procedures reducing unplanned outage time, thereby increasing the economic performance and safety of NPPs**

The FP-5 project ENPOWER develops weld repair procedures and alternative post weld treatments that minimise residual stresses and shorten repair time scales. Assessment methodology for treating defects in residual stress fields will be refined to give more accurate and informed sentencing of defects in aging plant. “Guidelines for optimising weld repair procedures” are being developed. Development of advanced weld repair and assessment technologies will reduce unplanned outage time and thereby increase the economic performance and safety of the nuclear power industry. This will also profit to other sectors of the industry.

Other projects : the FP-5 project VRIMOR developed innovative tools based on virtual reality support to the decision process related to inspection, operation, maintenance and repair of NPPs.

1.3 ORGANISATION AND MANAGEMENT OF SAFETY

♦ **Evaluate reliability and implement digital instrumentation and control tools**

From a technological point of view, one important challenge in NPP modernisation is the implementation of digital instrumentation and control tools (substituting the original analog systems) and the subsequent training required for the operating staff. Two projects are devoted to software modernisation for plant safety. BE-SECBS is dealing with computer-based systems embedded in a nuclear installation to support I&C functions important to safety. CEMSIS is aiming at developing a safety justification framework for the refurbishment of systems important to safety (SIS) that is acceptable to different stakeholders (especially licensing bodies and utilities).

♦ **Reduce frequency of safety incidents due to organisational factors**

Under FP-4, issues on organisational matters were investigated in several projects, in cooperation with some of the JRC driven networks. One example is the concerted action ORFA that looked at organisational factors and how they influence nuclear safety. In many studies it is recognised that organisational factors are often the root cause of incidents and accidents. However, there is unfortunately no agreed and validated method for their assessment. Important issues for short-term research are related to the identification and description of those factors which define good practice, the development of organisational self-assessment tools, the inclusion of organisational factors in incident analysis, the definition of methods of how to maintain the corporate knowledge, etc.

The main objective of LEARNSAFE is to create methods and tools for supporting processes of organisational learning at the NPP. This has become increasingly important for the nuclear industry in its adaptation to a changing political and economic environment, changing regulatory requirements, changing work force, changing technology and changing organisation of NPPs and power utilities. The focus is on the management of change.

1. 4 SUPPORT TO EU ENLARGEMENT POLICY (EMPHASIS ON VVER SPECIFIC SAFETY ISSUES)

As the EU enlargement process is now well engaged, a number of Central and Eastern European Countries (CEECs) become of particular interest, as far as nuclear power production is concerned. The Czech Republic, Hungary, Slovakia and Slovenia, which are particularly active in FP-5, are operating all together 16 nuclear units (15 VVERs-440 and 1 PWR) with a total capacity of 7.3 net GWe providing 30 % of their electricity. Safe operation of these installations is for the European energy policy maker at least as important as this of the western technology designed reactors. Development of a common safety culture is obviously a great benefit to the European integration. A number of concerted actions, by their specific conception focusing on integration, provide a specific support to the European research policy.

♦ **Harmonise VVER and PWR Codes and Procedures for plant life management and participate in the development of a common safety culture**

The FP-5 project IMPAM-VVER is addressing a safety relevant issue identified in recent studies on VVER safety. It investigates effective means and criteria for primary depressurisation during small loss of coolant accident (SBLOCA) including feed and bleed operation. The resulting knowledge will effectively contribute to the safety in all VVER countries.

Two FP-5 concerted actions are dedicated in harmonising safety culture within an enlarged Europe. The project VERSAFE brings together utilities from some of the Central and Eastern European Countries : common guidelines have been produced for the implementation of techniques in two areas, plant modernisation and severe accident management. VERLIFE is creating a "unified procedure for lifetime assessment of components and piping in VVER type nuclear power plants" based, in a first step, on former Soviet rules and codes. Later on, a critical analysis of possible application to some PWR type components will be done, with the aim to harmonise VVER and PWR Codes and Procedures.

♦ **Integrate governmental (e.g. national regulatory, Euratom FP and TACIS / PHARE, OECD/NEA) and industrial (in particular, private) funded R&D programs**

ATHENA, the AMES thematic network on aging, aims, within the enlarged EU, at reaching a consensus on important issues that have an impact on the life management of nuclear power plants. ATHENA creates a structure enhancing the collaboration between European funded R&D, national programs and TACIS/PHARE programs. This is increasing greatly the return from the individual projects and maximising the European added value.

More generally, the objectives of the FP-5 project EUROSAFE are to support the convergence of nuclear and radiological safety practices (safety culture) in Europe, while developing the idea of a European scientific and technical pool in the fields of reactor safety and radiation protection.

2 *FP-4 AND FP-5 RESEARCH IN SEVERE ACCIDENT MANAGEMENT* (main end-users = regulatory bodies and associated TSOs)

BACKGROUND

The fission products constitute the principal health hazard to the public, resulting from a severe accident. Therefore, the amounts and physico-chemical forms of those materials released from the reactor (the radiological source term) are of great safety significance. As a result, the regulatory authorities in some EU countries are requiring to take into consideration as much as possible the very unlikely severe beyond design-basis accidents (BDBAs). In the German licensing process, for example, BDBA evaluations are necessary since 1 January 1994 to ensure that even extremely unlikely events involving core melt-down would not require radical actions to ensure protection against the damaging effects of ionising radiations outside the fence of the installation site. BDBAs are also a concern expressed by the utilities and by the designers/vendors, as it is shown in the discussions around the European Utility Requirements (EUR Document, last release in April 2001) and in the MICHELANGELO initiative (started in December 1996).

As a result, it is envisaged by design to "practically eliminate" situations and phenomena which could lead to early failure of the containment system and subsequent uncontrolled large releases of fission products into the environment. Examples of such situations are high-pressure ejection of molten core (possibly leading to direct containment heating) and energetic in-vessel core debris interactions with water (possibly leading to hydrogen generation). For example, for such situations in the containment, a hydrogen strategy is proposed, based principally on passive autocatalytic recombiners (PARs) aimed at keeping the hydrogen concentration far from the critical deflagration-to-detonation transition (DDT) conditions. Other situations, then, such as low pressure core melt, should be dealt with – or "controlled" – by ensuring in the design that the decay heat of the molten core can be removed and that the vessel or containment can withstand the associated loads.

To better understand the source term behaviour and to develop appropriate prevention and mitigation measures, appropriate research is needed which combines experimental investigations and numerical modelling activities, supported by a robust scaling up strategy to extrapolate from simulant to prototypical materials and from small-scale laboratory to full-scale reactor conditions. Historically, since the accidents of TMI-2 (March 1979) and Chernobyl-4 (April 1986), many international RTD programmes have been focusing on the development of a kind of 4th level to be added to the 3 "standard" levels of the defence-in-depth strategy, mentioned in the Introduction. The international PHEBUS FP programme, in particular, that was launched by IRSN (France) and is co-sponsored by the Euratom (EU contribution of EUR million 4.5 under FP-5) and other partners, is the largest and most successful in-pile experimental programme devoted to the source term behaviour: it is aimed at bringing essential contributions to the knowledge on melt progression and fission product release. Based on a series of integral in-pile experiments using real core materials, the PHEBUS FP programme evaluates the amount and nature of radioactive products that could be released into the environment by occurrence of a core melt-down accident.

More generally, it is worth recalling that the Euratom research activities on severe accidents have interactions within various international frames, e.g. with CSNI of the OECD/NEA, CSARP of the U.S./NRC, IAEA and EC/ISTC. It is in particular noticeable that a new ISTC 'Contact Expert Group on Corium Management' (CEG-CM) was launched by the EC in 2002 with the aim to improve the interaction amongst EU and Russian severe accident research teams and to make recommendations on behalf of the EU for selection and funding of research Fproposals.

RTD needs in severe accident management

Following the recommendations of the Euratom programme committees and in line with the conclusions of other international organisations (discussed earlier), it has been decided to focus Community research on the integrity of the various barriers against radiological releases under extreme severe accident conditions.

Actually two types of research are proposed in these Community programmes : knowledge driven and application driven. To understand the behaviour of these barriers under extreme conditions, research of the fundamental (phenomenological) type is needed. To maintain the integrity of these barriers under these extreme conditions, research of the applied (industrial) type is needed.

The knowledge driven (phenomenological) research of the relevant projects can be broadly divided in two classes corresponding to the loading term and the response of each barrier, respectively, that is:

- research related to determining the loads to these barriers, both thermal (including irradiation induced heat effects) and mechanical (slow and fast pressurisation processes), and
- research related to the response of these barriers to these loads, with the aim to predict the damage due to stresses and strains, and ultimately the perforation of the barrier, and to develop appropriate mitigation measures.

To determine the loads to the various barriers, it is necessary to investigate:

- the thermo-hydraulics conditions within the barrier (e.g. stratification and potential for critical concentrations), taking into account the heat contents of the various materials with a view to predict the pressurisation processes, and
- the dynamic conditions upon the barrier (e.g. in-vessel or ex-vessel steam explosion, hydrogen explosion, missile impact).

To determine the response of the barrier to the above-mentioned loads, it is necessary to investigate:

- the thermo-mechanical response of the barrier, and
- the micro- and macro-cracking of the barrier subject to thermal and/or dynamic loads and the subsequent off-barrier release rates of radiological mixtures.

The application driven research of the relevant projects can be broadly divided in two classes corresponding to risk assessment and to accident mitigation measures, respectively, that is:

- PSA level 2 for the probabilistic evaluation of severe core damage and/or large early radioactivity releases
- Application of deterministic and probabilistic tools for the development of SAM guidance.

RTD needs covered by Community programmes

In the projects belonging to the clusters INV, EXV, ST, CONT, AMM and SAM, the emphasis is on the development of mitigative measures in the very remote case of severe accidents (level (4) of defence-in-depth). Under FP-4 a particular effort was devoted to the understanding of BDBAs through the 45 projects of the 5 clusters INV (= IN-Vessel core degradation), EXV (= EX-Vessel accident progression), ST (= radiological Source Term), CONT (= accident progression in the CONTainment building) and AMM (= Accident Management Measures). This effort on severe accident analysis is continued under FP-5 in the cluster SAM. Under FP-4 and FP-5, the following two key issues have been identified:

- (1) core degradation, corium formation in the reactor pressure vessel and its behaviour inside and outside the vessel (in particular upon a core-catcher). Research is needed with the aim of evaluating the coolability of the melt and ensuring the containment integrity. Criteria for deflagration and detonation processes in hydrogen/air/steam/dust mixtures also are needed to improve engineered safety systems and to better understand the capabilities of structures to withstand dynamic loads. Finally, understanding the release of radioactive materials from a degrading core into the cooling circuits and the containment, using in particular the PHEBUS-FP results, will enable to optimise mitigation measures and to better predict the source term.

- (2) improved methods and tools for severe accident management and operator training that make use of modern information and control systems and can handle uncertainties associated with man-machine interfaces in a structured way. Research is needed to develop safety systems for present and future reactors, which enable to extend the grace period, i.e.: the period during a severe accident when no active intervention is needed.

No wonder that the important actors in the area “severe accident management” are the European technical safety organisations, working for the national regulatory bodies, that is principally: IRSN in France; GRS in Germany; universities and CSN in Spain; NNC and HSE in the UK, SCK-CEN and AVN in Belgium, universities and SKI in Sweden and VTT in Finland.

Furthermore, the risks associated to those phenomena can be reduced through appropriate SAM measures that could be implemented through the improvement of new plant specific designs (e.g. ex-vessel core catchers) and for existing plants the development of both, engineered systems and backfitting measures (e.g. techniques for removing the hydrogen risk in the containment or mitigation processes against radiological releases). Development of specific operating emergency procedures is another expected result for current and future NPPs.

Generally, the results from experimental investigations and analytical studies on severe accident (SA) phenomena contribute to improve the phenomena understanding (e.g. corium behaviour, hydrogen explosions or radiological releases) and to validate SA models and integral codes, which have an impact on the quality of safety assessments, reduce uncertainties in the quantification of safety margins and maintain readiness to respond to emerging issues.

Under FP-5, the total EU budget to be spent for the 23 projects in the cluster SAM amounts approximately to EUR 15.3 million, which represents roughly half of the total value of these projects (EUR 28.2 million). This might be compared to the total EU budget spent under FP-4 for the 45 projects in the clusters INV, EXV, ST, CONT and AMM, which was EUR 29 million.

2.1 ASSESSMENT OF SEVERE ACCIDENT RISKS

The assessment of severe accident risks and strategies for severe accident management is addressed in several FP-4 and FP-5 projects.

♦ **Reduce the uncertainties associated with severe accident phenomena; produce experimental data to evaluate the actual impact of the severe loads on the various barriers, thereby improving the evaluation of safety margins and maintaining the readiness to respond to emerging SA issues.**

The results obtained from severe accident (SA) experimental investigations and analytical studies contribute firstly to improve the phenomena understanding and to validate SA models and integral codes, which are used to perform safety studies, to quantify safety margins and to maintain the readiness to respond to emerging SA issues or to specific SA scenarios. In this regard, deterministic and probabilistic (i.e. PSA) safety assessments have in particular a direct regulatory impact, while the reduction of uncertainties regarding safety margins permits optimised quantifications of them with a significant impact on the nuclear industry, also in economics terms.

This goal, the consolidation of the understanding of severe accident phenomena, is addressed by specific experimental programmes in various projects encompassing analytical support for a wide spectrum of phenomena, i.e. from core degradation, corium behaviour, hydrogen explosions to radiological releases.

♦ **Understanding of core degradation phenomena and prediction of threads to the RPV**

The PHEBUS-FP programme evaluates, through a series of integral in-pile experiments using real core materials, the amount and nature of radioactive products, which could be released in case of occurrence of a core meltdown accident in a LWR. In that frame, knowledge of various phenomena were prototypically improved, i.e. the cladding oxidation and hydrogen production were coupled with degradation processes; fission product release and aerosol transport and deposition with resuspension and vaporisation in particular of cesium; and late phase of fuel degradation. Other conclusions indicated the strong influence of iodine behaviour on source term mitigation strategies, i.e. the formation of gaseous iodine becoming insoluble in the sump water by reaction with oxidised soluble silver; organic iodines coming from reactions between gaseous iodine with paints; and various mechanisms governing the iodine volatility in the containment. Furthermore, a specific experiment was devoted to study the releases of low volatile fission products, actinides and uranium from a debris bed; the physics of debris bed and the transition from debris bed to molten pool; as well as their fission product releases. It was the first interpretation for fuel degradation evidencing the formation of an upper vault, a cavity and a molten pool.

Several FP-4 projects also supported the PHEBUS FP programme delivering gained knowledge of in- and ex-vessel fission product behaviour, vapour-aerosol formation and behaviour, and revaporization phenomena. In particular, the plant assessment benchmark exercise STU (FP-4) comprised many severe accident scenarios over eight reference designs revealing that the main uncertainties in PSA level-2 studies were the aerosol behaviour in the containment and late revaporization of the deposited radionuclides.

Knowledge on chemical interactions between core materials and corium structures, as well as on thermodynamic corium properties was gained in CIT, while COLOSS investigated various core degradation processes in the presence of high burn-up and MOX fuel with emphasis on the implementation in the various severe accident codes for PWR, BWR and VVER reactors. These processes included very specific knowledge on high burn-up and MOX fuel dissolution, clad rupture, oxidation of U-O-Zr mixtures, and bundle experiments concerning B_4C control rod degradation and oxidation. Furthermore, the B_4C-bundle degradation experiments for PWR/BWR were carried out with the same B_4C control rod design and the same bundle geometry as in the Phébus FPT-3 test, providing in that way a valuable support for the preparation of that test.

Moreover, specific late-phase phenomena in degraded core scenarios were also largely understood in line with the former COBE (FP-4), which extended the range of severe accident codes to be able to properly model debris-bed and molten pool behaviour. The experimental parts of this activities were supported by comprehensive phase diagrams of the elements present in both, in- and ex-vessel corium, based on the development of the unified thermodynamic database NUCLEA and its coupling with severe accident codes developed by the ENTHALPY.

In relation to the reactor pressure vessel (RPV), the REVISA (FP-4) developed new knowledge on damage, creep behaviour and size effects, concluding for various simulated LOCA scenarios that a delayed plastic collapse is the dominant vessel failure mode over creep damage. In this line, damage models and criteria for failure strain of essential reactor components were developed with emphasis on the ultimate deformation capacity by applying uniaxial and biaxial static and dynamic loads within the frame of the LISSAC, which showed that plastic strains up to 20% may not cause extensive structural RPV failure leading to corium release from the RPV.

As consequence of core degradation phenomena and molten fuel coolant interaction the Hydrogen behaviour in the containment became an issue. In this line, the hydrogen combustion behaviour and the corresponding loads in complex multi-compartment geometries are experimentally investigated in HYCOM by means of the large RUT facility (Kurchatov Institute, Moscow), whose combustion modes are ranging from slow to fast turbulent deflagration.

♦ **Developing an experimental/numerical code validation strategy for SA, based on a series of separate effects and integral models covering the entire spectrum**

In relation to the impact of modelling of new phenomena on severe accident calculations, numerous sequences (e.g. loss of SG feedwater, large break LOCA, station blackout) were performed in the frame of COLOSS for different plant designs with B_4C control rods, i.e. 1300 MW_e French PWR, BWR, VVER-1000 and EPR by means of various several SA codes. Such models are also used for a common code validation strategy for the European integral code ASTEC developed by EVITA, which also optimises SAM strategies in a variety of NPPs. Such development is taken also benefit from different models developed in core degradation experimental activities.

Furthermore, also specific PHEBUS-FP results are applied to SAM strategies in PHEBEN-2, which enables to improve the safety margin calculation tools by developing detailed models for separate-effect tests and integral codes for full-scale plant analysis. In this line and connected to the OECD/NEA International Standard Problem exercise (ISP-46) devoted to the PHEBUS FPT-1 test, THENPHEBISP is aimed at validating various specific and integral SA codes with emphasis on the identification of numerical model uncertainties.

♦ **Understanding corium stabilisation and coolability both in-vessel (RPV melt retention) and ex-vessel (containment melt retention)**

As initiating boundary condition for corium stabilisation, STRATIEX (FP-4) assessed, for both in- and ex-vessel scenarios, the conditions required for melt stratification by reviewing several experimental programmes and thermal-hydraulic and thermochemical models. In relation to basemat integrity considerations, the corium stratification plays a crucial role regarding the layer inversion expected in a reactor pit through melt interaction with sacrificial concrete layer and Zr oxidation, that reduces the liquidus temperature of the corium, in that way improving its spreading on core-catchers.

The efficiency and reliability of various core-catcher designs regarding the corium spreading and heat transfer issues was investigated in CSC (FP-4) through experimental and analytical activities. They were focused on corium spreading tests under dry and wet conditions with prototypical and simulant materials and with different substrata, as well as on tests oriented to long-term corium coolability by direct water contact, based on either flooding on the top of the corium or on an alternative water injection from the bottom. For the latter concept, the feasibility of the core-catcher cooling from the bottom through cooling tubes crossing the ceramic layer towards the sacrificial layer was demonstrated. Consequently, the core-catcher design concept for the European Pressurised Reactor (EPR) entered into a technical and economical optimisation phase by replacing those cooling tubes by different layers of porous concrete.

In this line, several large-scale spreading tests with prototypical molten corium of COMAS (FP-4) showed a number of interesting results: the spreading behaviour was dominated first by inertia forces and later by viscous forces; mixed melts separated into horizontal metallic and oxide layers even at small density differences, which is crucial for designing sacrificial layers; the front immobilisation was caused by melt bulk freezing and crust formation at the top and front surfaces; and the spreading length was independent of the type of substratum but dependent on the temperature and metallic/oxide ratio for mixed melts. Consequently, the conclusion for designers that spreading areas can be completely covered by a sufficiently high release melt rate is going to be demonstrated in ECOSTAR, where the technical feasibility of such ex-vessel mitigation measures (core-catcher approach) and the validation of spreading codes are the main goals. Its experimental activities are further addressing issues related to melt dispersion, jet erosion, large-scale spreading, corium solidification, interaction with structural materials, and coolability approaches as top and bottom flooding.

2.2 Mitigation of Severe Accident Consequences (SAM Measures)

♦ **Develop severe accident management strategies to reduce the SA risk**

The scope of severe accident management (SAM) is to contribute to the development of techniques, for example, to practically eliminate some of those phenomena or to develop mitigation strategies to control some of them. The associated risks to those better-understood phenomena can be reduced by means of appropriate SAM strategies. Such measures have a clear impact on the nuclear industry and could be implemented through design improvements of new power plants (e.g. ex-vessel core catchers), through the development for existing plants of both, backfitting measures or engineered systems (e.g. techniques for removing the hydrogen risk in the containment or mitigation processes against radiological releases) and implementation or optimisation of SAM guidelines or operating emergency procedures. In addition, the progress in numerical techniques as well as the availability of powerful and cost-effective information technology systems assures more reliable information and helps to improve the diagnostic means as well as the implementation of some accident management measures.

♦ **Assess recriticality risk in BWRs severe accident scenarios**

Since melting and relocation of the control rods during the progression of a BWR core degradation starts before the relocation of the fuel, there is a possibility of recriticality occurrence during the reflooding of a partly degraded core with mostly unborated water. In this regard, in SARA (FP-4) magnitudes and gradients of the possible recriticality transients were determined in order to assess the impact of such recriticality on the accident progression itself, especially on reaching a prompt critical state of the core, which could lead to a large power excursion. As consequence, various SAM measures were recommended, such as the increase of boron injection system capacities, the possible limitation of the maximum reflooding rate, the delay in depressurisation of the primary system, as well as the enhancing of the containment energy removal systems capacity in order to mitigate the long-term build-up of the containment pressure. This study of containment response with early filtered venting was based on a postulated recriticality event performed for Olkiluoto-1/2 BWRs.

♦ **Assess strategies for molten corium retention within the RPV**

The creep behaviour of prototypic reactor pressure vessels, timing and modes of its failure with and without penetrations, as well as the effects of the melt pool stratification were investigated in ARVI, whose results indicated that there is no evidence of gap-cooling effectiveness in a melt-pool scenario, as well as that the temperature is the dominating parameter (more than vessel penetrations) for the failure location. Furthermore, a reasonable basis does exist to accept IVMR (in-vessel molten retention) as SAM strategy for medium vessels, for instance VVER-440, in particular for Dukovany and Paks NPPs. This conclusion is supported by MVI (FP-4) that assessed pool stratification considerations and showed the influence of the melt focusing effect on the vessel.

Furthermore, assessment of various retention concepts, i.e. the pros and cons of internal and external reactor vessel cooling, were assessed in EUROCORE aimed at achieving consensus on feasible and reliable industrial corium recovery concepts connected to the mentioned IVMR, the corium-concrete interaction with water addition and ex-vessel spreading techniques. Particularly, a specific in-vessel core retention approach, including internal core-catcher as preventive SAM, was analysed in IVCRS (FP-4), where uncertainties linked to the involved physical phenomena were assessed and for which related patents were also registered.

♦ **Investigate impact on RPV of severe accident phenomena**

Further to the high-temperature database developed for various vessel steels, specific experiments of RPVSA (FP-4) confirmed the formation of oxide melt crust protecting the lower head and the penetration of corium into nozzles over long tube distances. Furthermore, it was demonstrated through specification of scaling rules to prototypic dimensions, that the reactor pressure vessel upper head, with and without control rod internals, withstands the effects of a molten slug impact caused by a postulated in-vessel steam explosion. On the other side, and further to the demonstrated relevance of thermo-physical, mechanical and chemical properties, MFCI (FP-4) showed that steam explosion due to molten fuel coolant interactions (MFCI) seems very unlikely with prototypical corium, and that the energetic interaction involving reactor material would not jeopardise the reactor structure. The corium does not show potential to induce spontaneous steam explosion.

♦ **Assess feasibility of hydrogen mitigation concepts**

Specific catalytic coating materials were experimentally defined in the project THINCAT as complementary strategy to the installation of passive autocatalytic recombiners (PARs) for hydrogen risk mitigation measures in containments. Such materials could be located on thermal insulation elements of the main coolant loop components. The feasibility and reliability of the mentioned PARs from an industrial prospect were examined in PARSOAR, which compared qualification tests and licensing procedures, relevant to the hydrogen risk, under various severe accident conditions. A « Handbook of best practices for installation of PARs» was produced.

♦ **Assess impact on PSA studies, emergency procedures and SAM guidelines**

The long-term behaviour of a solidified core immersed in a water pool were examined by LPP, which provided useful kinetics data concerning the release of fission products and core materials from in- and ex-vessel molten corium, as well as sensitivity studies addressing long-term coolability and impacts on radiological source term in plant calculations (EPR, Konvoi NPPs and VVER-1000 Temelin NPP). The latest included PSA-1/2 evaluation and selection of various sequences for analysis of late-phase phenomena and fission product behaviour and assessment of SAM strategies as well. Moreover, a better exploitation of volatile iodine mitigation processes was investigated by ICHEMM, which provides useful kinetics data for destruction and transmutation reactions of volatile forms of iodine. The impact on the iodine source term calculations regarding the formation of iodine formation and destruction, and the calculated iodine source term for various PSA-2 sequences defined for the VVER-440/213 Dukovany NPP was assessed. In this regard, improved calculations of the iodine source term were carried out for selected sequences (based on PSA studies) of Dukovany NPP (and possibly for VVER-1000 Temelin NPP), whose results are oriented to improve SAM strategies or plant safety with respect to radioiodine releases.

SGTR generated experimental data and validated transport models in order to assess and support accident management interventions in vertical and horizontal (VVERs) steam generator tube rupture –SGTR- sequences, including aerosol deposition, leading to severe accident conditions in PWRs and VVER-440. As result, two types of accident management procedures are foreseen: primary system depressurisation and water flooding of the secondary circuit. Also new orientations for the optimisation (Loviisa, Beznau and Borssele NPPs) or development (Dukovany NPP) of SAM guidelines related to the tube rupture scenarios of steam generators of current nuclear power plants have started.

Finally, the impact of the SAM measures on the predicted radiological releases, as well as the basis for an outline methodology to define representative source terms were evaluated in OPTSAM, where a total of 24 accident sequences and topical issues to predict baseline source terms were selected, e.g. venting strategy, hydrogen management system, in-vessel retention, ex-vessel flooding, by-pass release control, spray strategy, RCS depressurisation, long term decay heat removal from containment, and shutdown sequence. Those accident sequences were supported by several sensitivity analyses for each base-case of different NPPs: Cofrentes, Barsebäck-2 and Forsmark-1 as boiling water reactors, and Sizewell-B, Ringhals-2, CPI 900 MWe, Tihange, Konvoi, Loviisa-1/2, Paks and Temelin as pressurised water reactors. In this line, various categories of SAM objectives were defined in order to facilitate the interaction and cross-fertilisation amongst the mentioned different NPP types: strategies to protect the reactor coolant system integrity (incl. RPV); strategies to protect the containment integrity; hydrogen management strategy; and long term decay heat removal from the containment atmosphere.

The results are exploited for PSA studies, even as starting point for a PSA level 3, and for improving the supporting safety case for SAM guidelines as well. In particular, results were used to verify the applicability of some measures defined in the SAM guidelines under preparation for Temelin NPP. The SAM optimisation, together with the definition of realistic source term, is also used in the update of PSA level 2 for Temelin NPP. Furthermore, results for the in-vessel retention cases are used to reassess the SAM guidelines for the Loviisa NPP, where also the spray operation would be reassessed in the emergency operational procedures (EOPs). Also source term information is used in PSA2 evaluation for shutdown states. Furthermore, the feasibility of a computerised operator-supporting tool for elements in SA scenarios as the severe accident management guidelines is also being investigated in SAMOS.

3 *FP-4 AND FP-5 RESEARCH IN EVOLUTIONARY SAFETY CONCEPTS (main end-users = manufacturing industry and associated services)*

BACKGROUND

It is worth recalling that there are currently 20 reactor units in construction in Eastern Europe and the Russian Federation and 25 reactor units in construction in the rest of the world, with total planned capacities of 16 and 21 net GWe, respectively. Some of these reactor units are of the evolutionary type, i.e. with emphasis on design simplification and enhanced man-machine interface, with the aim to further reduce any (severe) accident risk. In evolutionary LWR designs, with their emphasis on design simplification and enhanced man-machine interface, the severe accident risk will be further reduced. Some of the innovative reactors rely mainly on passive prevention and mitigation features and systems.

Part of Euratom co-sponsored research is devoted to the investigation of phenomena associated with the use of passive systems in some evolutionary LWR designs for decay heat removal (from the core region and from the containment building) and for other safety measures (e.g. depressurisation and injection). The potential advantages of passive safety systems (e.g. independence from external energy sources, simpler design, less complex instrumentation and control) should be weighted against their potential disadvantages (e.g. reliance on small driving forces, limited operational flexibility, reduced in-service testing capability, difficult diagnosis of status, etc). This opens new areas of future research (e.g. passive systems reliability, systems interactions, risk quantification, cost/benefit assessment, etc.) whose results should help to make sound decisions.

In addition, the coupling of different neutronics and thermal-hydraulics computer codes, needed for this purpose, enables to improve and/or validate the numerical models used in the existing thermal-hydraulics computer codes, and to extrapolate the results of the small-scale experiments towards full-scale reactor conditions.

Another area of interest for Euratom co-sponsored research in evolutionary reactors is the use of MOX fuel which has been used on industrial level since 1982 in a number of EU power plants, for example: up to 50 % core loading in 9 German reactors, up to 30 % in 17 French reactors and up to 25 % in 2 Belgian reactors. High burn-up fuel is another matter of increasing interest – for countries like France who have achieved burn-up targets of nearly 50 GWd/t for standard fuel and 40 GWd/t for MOX fuel on an industrial scale.

RTD needs in evolutionary concepts

Following the recommendations of the Euratom programme committees and in line with the conclusions of other international organisations (discussed earlier), it has been decided to focus Community research on design simplification (including enhanced man-machine interface) and on a new generation of numerical simulation tools, with the aim to better estimate and further reduce the severe accident risk. Some of the innovative « simplified » reactors rely mainly on passive prevention and mitigation features and systems.

Actually two types of research are proposed in these Community programmes : knowledge driven and application driven.

To understand the behaviour of the passive safety systems, knowledge driven (phenomenological) research is needed. This is making use of the most important European thermal-hydraulics facilities in Europe covering large-scale integral, large-scale separate-effect and small-scale separate-effect tests. These programmes include both experimental and analytical activities directed mainly to investigate the phenomena associated with the use of passive systems for decay heat removal (both from the core region and from the containment building) and for safety measures (e.g. depressurisation and injection) in advanced LWRs. These projects have also provided a bank of valuable independent experimental data about different phenomena and aspects, ready to be used by reactor designers and code developers.

To test the performance and the reliability of these passive safety systems, application driven research is needed. The industrial (application driven) research of the relevant projects has contributed to demonstrate the feasibility and the efficiency of most of the passive systems being investigated for different types of advanced LWRs, and in particular the ESBWR and SWR-1000 designs. In particular, the following aspects have been thoroughly assessed:

- efficiency of pool immersed heat exchangers operating at low pressure
- efficiency of condensers with various geometrical tube arrangements for different steam/gas mixtures
- performance of gravity driven safety injection systems
- performance of single-stage steam injector systems.

The use of different neutronics and thermal-hydraulics computer codes (ATHLET, CATHARE, TRAC, RELAP5, etc.) for pre and post-test calculations has been an important part of the work programmes of these projects. It has enabled to define better the test configuration and parameter range extensions, to improve and/or validate the numerical models used in the existing thermal-hydraulics computer codes, and to extrapolate the results of the small scale experiments towards full scale reactor applications.

Finally, the results of two concerted actions of this cluster have been of strategic importance for the establishment of an R&D Network on Safety-Related Innovative Nuclear Reactor Technology (SINTER project) and for the identification of the Needs of the industry sector related to the Next Generation Reactors (MICA project). Another study (INNO-HTR) is devoted to a revisitation of the high temperature reactors (HTR) with a special view to co-generation and plutonium burning applications.

RTD needs covered by Community programmes

In the projects belonging to the cluster EVOL the emphasis is on a new generation of reactors which should be cheaper, safer and more simple to operate, using, for example, passive (self-acting) safety systems. Under FP-5, the following two key issues have been identified:

(1) cost and safety advantages of evolutionary improvements in currently used nuclear technologies, in particular those with the potential to significantly reduce the risk and consequences of human error and public concerns about nuclear technology (e.g. passive safety systems).

(2) understanding of the performance of high burn-up and MOX fuel under transient and accident conditions as a basis for lowering fuel costs, whilst maintaining or improving safety margins.

No wonder that the important actors in the area “evolutionary concepts” are the European manufacturing industry and the associated services.

As far as passive safety systems in evolutionary LWRs are concerned, the emphasis is on understanding decay heat removal processes (both from the core region and from the containment building) and developing safety measures (e.g. depressurisation and injection).

Research is also conducted to improve some technologies (of interest for both present and next generation reactors) that have potential cost and safety advantages. Under the economic pressure of the manufacturing industry in a deregulated electricity market, many efforts are naturally devoted to the improvement of evolutionary safety concepts adapted to the new operational conditions (e.g. related to reactor backfitting, power upgrades, high burn-up, MOX fuel, etc).

Under FP-5, the total EU budget to be spent for the 17 projects in the cluster EVOL amounts to approximately EUR 10 million, which represents roughly half of the total value of these projects (EUR 21.8 million). This might be compared to the total EU budget spent under FP-4 for the 11 projects in the INNO cluster, which was EUR 4.8 million.

3.1 EVOLUTIONARY SAFETY CONCEPTS

The projects in this area have all a strong “numerical” flavor as progress in numerical modelling, especially CFD (computational fluid dynamics) codes, is needed to catch up with the progress made recently in experimental investigations, especially for passive systems.

♦ **Reduce the uncertainties associated with passive decay heat removal systems related to reactor coolant system (second barrier) and containment (third barrier)**

The FP-4 project IPSS (Innovative Passive Safety Systems) produced a set of recommendations to vendors and utilities for possible improvements in the design of passive safety systems. As a complement, the project BWRCA implemented and qualified computer codes useful for future BWR plants.

The FP-4 project TEPSS (Technology Enhancement for Passive Safety Systems) has lead to a significant improvement of the technology base of the European Simplified Boiling Water Reactor (ESBWR).

The FP-4 project INCON (Innovative Containment Cooling for Double Concrete Containment) confirmed the viability and the soundness of the proposed Passive Containment Cooling System (PCCS) designs consisting of three main integrated components: inner heat exchanger located inside the primary containment, outer heat exchanger and intermediate loop connecting the inner and the outer heat exchangers.

The further use, improvement and applications of CFD codes for a wide range of conditions were highly recommended in the conclusions of some final reports of FP-4.

Four FP-5 projects are strongly related to the use, development and improvement of three-dimensional (3-D) and computational fluid dynamics (CFD) codes. The project ASTAR aims at the development of advanced numerical methods for 3-D two-phase flow simulation tools that might lay the scientific and technical basis for a new generation of thermo-hydraulics codes. This should improve the modelling of safety relevant phenomena related to the next generation of evolutionary LWRs. The expected outcome of the ECORA project is a comprehensive evaluation of CFD software for applications in the primary system and the containment of nuclear reactors, resulting in recommendations for Best Practice Guidelines and for necessary CFD software improvements. The project aims also at establishing a Network of European Centres of competence for applications of CFD codes to reactor safety. In the concerted action EUROFASTNET a critical assessment is made of the needs, in nuclear engineering, for thermal-hydraulics R&D, with emphasis on a coherent balance between advanced CFD modelling and experimental validation using innovative instrumentation.

Four other FP-5 projects address the improvement of analytical tools. Two of them address the issue of coupling neutronics and thermal-hydraulics codes. The main objective of CRISSUE-S is to elaborate a state of the art report about the coupling of neutronics and thermal-hydraulics codes for LWRs with emphasis on the so-called RIA (Reactivity Initiated Accidents). The project should provide recommendations to utilities and regulators about possible safety margins of existing reactors and optimized accident management. The VALCO project is specifically aimed at the validation (against experimental data) of coupled neutronics/thermal-hydraulics codes for VVERs. The work is based on results from an EU Phare project (SRR1/95) which analysed transients initiated by perturbations in the VVERs secondary circuit.

- **Prepare the ground for a European strategy for next generation reactors (Michelangelo initiative) with emphasis on HTRs**

The FP-4 project SINTER (Safety-Related Innovative Nuclear Reactor Technology Elements – R&D Network) has established on Internet a telecollaboration platform, which provides tools and databases for information exchange on R&D activities, innovation potentials, R&D facilities and documents –see http://w2ksrvx.ike.uni-stuttgart.de/sinter_neu/.

The FP-4 project MICA (Industrial needs in R&D for the Safety and the Competitiveness of the next generation of reactors) showed the need for a real stable and long-term partnership of European nuclear industry and research and the diversity of needs to be satisfied by nuclear energy and consequently the diversity of technical solutions to be found for that purpose.

The FP-4 project INNOHTR (Assessment of Safety and Innovative Technology for HTR Generating Plant) showed that there are both economic and social motivations for developing a modern HTR technology. They have concluded that modern HTRs have to be developed as soon as possible in order to keep this type of reactor competitive in the long term, in comparison with gas turbines which are still improving.

Three FP-5 projects are addressing operational practices and design improvement of LWRs. The goal of the NACUSP project is to enhance the basic understanding on BWRs thermal-hydraulics stability issues, under both forced and natural-circulation conditions, through generation of new experimental data, elaboration of guidelines, development of efficient models and validation of computer codes. The results of this project should improve operational flexibility and increase the confidence level on the safety margins of the operating BWRs and future designs.-The DEEPSSI project proposes to develop and test an innovative high-pressure steam injector design and to assess its potential application in an Emergency Feedwater System (EFWS) of PWRs steam generators. Applications to both western and eastern (i.e. VVER-440) types PWRs are considered. The purpose of the FABIS project to develop and test a diverse fast-acting boron injection system that can be used in case of an anticipated transient without scram in the existing and the future advanced BWRs. The main innovation of FABIS compared to similar existing designs is the use of steam for the pressurisation of the boron solution tank instead of nitrogen. In this project, the SWR 1000 concept from Framatome ANP has been selected as the reference reactor

The objective of the RMPS project is to propose a specific methodology to assess the thermal-hydraulic reliability of passive systems. The main activities are the identification and quantification of the sources of uncertainties, the propagation of the uncertainties through the thermal-hydraulics models and the introduction of passive systems unreliability in the accident sequence analysis.

3.2 HIGH BURN-UP AND MOX FUEL

This area presents projects of the FP-5 only, because the FP-4 did not include nuclear fuel related research.

♦ **Quantify safety margins**

The regulators need to know the safety margins by using fuels at higher burnup levels in terms of fission gas releases and their scientific and technical basis, as utilities request approval of new nuclear fuels and methods. Likewise, for utilities and vendors it is also very relevant to know which margins are available with respect to their manufacturing processes, as well as to understand their technical basis in order to identify in which direction further progress may be made to optimise fuel and fuel cycle as well. For these reasons, the impact of the three projects below is important for both regulators and industry.

Effectively, the FP-5 project MICROMOX includes experimental tests aimed at understanding to what extent the as-fabricated microstructure of the MOX fuel influences the gas release in normal and abnormal conditions at high burn-up. Different numerical codes simulating the thermo-mechanical behaviour of the fuel are benchmarked. This project is complemented by OMICO, whose main objective is to study and model the influence of microstructure and matrix composition on fuel behaviour under normal and abnormal LWR operating conditions for various oxide fuels. Furthermore, the FP-5 project VALMOX is oriented to the validation of MOX fuel calculations at high burn-up based on available experimental data from LWRs using the JEFF nuclear database as well as state-of-the-art neutronics codes. First indications on safety limits of the MOX fuels in terms of helium production are also expected.

♦ **Extend safety assessments of nuclear fuel codes**

The use of specific nuclear fuel codes for appropriate safety assessments has importance in licensing processes by comparing their application to submitted analyses of the vendors. In this line, the two projects below are dealing with further developments of determined nuclear fuel codes to address specific issues of the cladding materials of LWRs fuel assemblies.

In this regard, the FP-5 project EXTRA, is dealing with the extension of the applicability of the TRANSURANUS fuel code to VVER reactors that contain niobium in their fuel cladding. This extension is based on compilation and verification of experimental databases together with the development of new models to simulate high temperature behaviour, creep and fracture criteria. The use of this extended code has a special role in the licensing procedures of Hungary and Slovak Republic in the sense that fuel vendors carry out fuel safety computations, while the regulators demand independent analyses for comparison. Consequently crosscheck computations are going to be performed by the extended code in both countries. In that way, the simulation of the VVER fuel rods' performance under abnormal conditions and design basis accidents is promoted widening also the scope of licensing applications. On the other side, the FP-5 project SIRENA is developing, in the context of the previously mentioned ITEM network, a numerical simulation code to assess the integrity of LWR fuel pin cladding manufactured with Zr-Nb alloys.

NUCLEAR KNOWLEDGE MANAGEMENT (indexes, education, e-science and Eurocourses)

Improving knowledge or acquiring new one, as it has been discussed above, is naturally inherent in all FP-4 and FP-5 projects. From a programme management point of view, however, this basic objective requires nowadays what is called a knowledge management strategy. It is worth presenting first of all three FP-5 projects of general interest that are somehow dealing with nuclear knowledge management, stretching across the areas covered by the three clusters PLEM, SAM and EVOL, thereby contributing to the conservation of the nuclear expertise.

The principal end-users of these three projects are the decision makers interested in knowing who in Europe is doing what in reactor safety research, the academic community concerned with the decrease of financial and human resources and the developers of numerical simulation tools interested in virtual laboratories (e-science).

- **JSRI** (Joint Safety Research Index), a concerted action for the dissemination of information about the reactor safety research programmes in Europe (i.e. the so-called « centres of excellence ») and their main achievements in the areas covered by PLEM, SAM and EVOL (see homepage http://w2ksrvx.ike.uni-stuttgart.de/jsri/). Research projects from 16 EU and CEE countries (+ DG JRC) are described and numerous expert names are referred. This is a useful management tool for decision makers interested in the most recent RTD developments.

- **ENEN** (European Nuclear Education Network), an accompanying measure to pioneer the European area of higher education in nuclear engineering, using the instruments proposed by the Bologna declaration (1999). It is proposing a global strategy for quality control and mutual accreditation of Euromaster grades and performing pilot education sessions to test the proposed scheme (see homepage http://www3.sckcen.be/enen). A durable legal structure is proposed, comprising more than 22 universities, representative of EU and CEE countries.

- **CERTA** (Consolidation of the Integral System Experimental Data Bases for Reactor Thermal-Hydraulic Safety Analysis), a thematic network for the preservation of the integral system experimental data bases for reactor thermal-hydraulic safety analysis acquired by institutional and industrial research organizations throughout the EU and CEE countries (see homepage http://lunar.jrc.it/stresaWebSite/). This web-based informatics platform is a e-tool that can be extended to drive the exchanges of large datafiles needed for any joint distributed numerical development, avoiding the stiffness of the static traditional centralised databank systems.

It is clear, however, that a real nuclear knowledge management strategy at the Community level is still missing, i.e. a coherent durable EU approach and the appropriate implementation instruments for activities related to identification, acquisition, development, disssemination, use and preservation of knowledge, as it is shown in Table 1 (courtesy of NAGRA). To optimise dissemination, for example, a thorough discussion with the owners of scientific data and know-how is still needed, in particular to agree on access rights for carrying out research activities and/or for using the results (commercial exploitation and/or further research).

Worth mentioning in the context of dissemination are also a series of FP-4 and FP-5 Eurocourses, co-organised by DG Research and national hosting organisations, with the aim to disseminate specific research results and/or to discuss the latest achievements in a given area, thereby contributing to the objectives of “education and training”. The principal end-users of these Eurocourses are students (young and old !) as well as decision makers interested in the latest S/T breakthroughs.

Here is the list of Eurocourses organised under FP-4 with an attendance varying between 30 and 60 participants:

- in the area of PLEM :

- Symposium on FP-4 projects RESQUE and REFEREE (SCK-CEN, Mol, 5-7 September 2001)

- in the area of SAM :

- Analysis of severe accidents in LWRs (UPM Madrid, 13-17 October 1997)

- in the area of EVOL :

- Advanced Nuclear Reactor Design and Safety (GRS Garching/Munich, 17-21 May 1999)

Here is the list of Eurocourses organised under FP-5 with a similar attendance:

- in the area of PLEM :

- MASC: “Use and application of the master curve method for determining fracture toughness” (co-organised by VTT, Finland, 12-14 June 2002)