Nuclear Safety

Nuclear Safety Research in OECD Countries

Major Facilities and Programmes at Risk

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

FOREWORD

In 1996, the Senior Group of Experts on Nuclear Safety Research Capabilities and Facilities (SESAR/CAF), which was investigating the ability of OECD Member countries to sustain an adequate level of research, identified a number of facilities and programmes that were important for continuing research needed by the safety community during the coming decade. It also pointed out that many of these facilities and programmes were facing increasing budgetary constraints, and that some would cease to be supported at the national level in the near or medium-term future. Some of the facilities were of interest to more than one country. It therefore seemed logical to investigate the possibility of operating the facilities in an international context, in order to share the costs and the expertise, and to promote quicker and deeper international consensus on safety issues.

It was in this context that in 1997 the NEA Committee on the Safety of Nuclear Installations (CSNI) decided to set up a Senior Group of Experts on Nuclear Safety Research Facilities and Programmes (SESAR/FAP). The new Senior Group of Experts was asked to identify facilities of potential interest for present or future international collaboration, to make specific recommendations regarding facilities, research programmes, and joint projects, and to discuss other possible forms of international collaboration. For efficiency, the group was restricted to the countries running the widest and most advanced research programmes.

This work was a continuation of the work performed over the period 1992-1997 by the CSNI Senior Group of Experts on Safety Research (SESAR) which had issued three reports: the first one entitled *Nuclear Safety Research in OECD Countries* reviewing the research being carried out and setting down views on likely future requirements and priorities; the second one entitled *Nuclear Safety Research in OECD Countries: Areas of Agreement, Areas for Further Action, Increasing Need for Collaboration* addressing the future direction of nuclear safety research and international collaboration in the field; and the third one entitled *Nuclear Safety Research in OECD Countries: Capabilities and Facilities* discussing existing and planned capabilities and experimental facilities to fulfil the needs identified in the two previous publications.

This report presents SESAR/FAP's findings as well as its recommendations to the CSNI. A condensed version of this report has been published under the title *Nuclear Safety Research in OECD Countries: Summary Report of Major Facilities and Programmes at Risk.*

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SUMMARY

In the field of nuclear power safety, OECD Member countries' Government Agencies have broadly similar responsibilities. In several Member countries, the funding levels of national Government safety research programmes have been reduced over recent years, and increased reliance placed on private sector or co-operative endeavours to maintain an adequate level of safety research. Care is needed to ensure that this does not have an adverse impact on the ability of Government Agencies to fulfil their safety responsibilities, especially since the reduction in Government direct or imposed funding in this area may not have been offset by concomitant increases from the industry. The CSNI has expressed concern that dwindling budgets and support as well as stagnant or even reducing programmes may lead to the untimely shut down of large facilities and the breaking up of experienced research teams. This will result in a consequent loss of competence and the reduced capability to deal efficiently and in a timely manner with future safety problems. In addition, research and educational programmes play a key role in attracting, training and retaining new talent and in nuclear safety. As an expression of these concerns, the following is a statement of the overall goal of the SESAR/FAP activity: "To ensure timely CSNI action is taken, as needed, so that an infrastructure of safety research facilities and programmes is maintained that assures the safe generation of electricity via nuclear power now and in the future."

Objectives and Terms of Reference

The SESAR/FAP group was set up by the CSNI to address the immanent loss of Facilities and Programmes (FAP) which may be considered to be at risk and are crucial to nuclear safety research and to recommend actions required. The group's terms of reference may be briefly summarised as:

- to identify key facilities potentially interesting for present or future international co-operation (whether this has been proposed by the host country or not);
- to make specific recommendations regarding facilities and research programmes, joint projects, etc., including, if necessary, recommendations regarding new facilities;
- to consider, where appropriate, other forms of international co-operation. For example, data banks, exchange or sharing of experts, establishing "Centres of Excellence", joint development of computer codes and to find ways of sustaining and refreshing key expertise through training and education[1] in nuclear engineering and the physical sciences;

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1. It is important to note that a number of other initiatives are underway concerning the future needs of technical education and training for nuclear capabilities. Recent summaries are available in "Workshop on Assuring Nuclear Safety Competence into the 21st Century" [NEA/CNRA/R(2000)1], Budapest, Hungary, 12-14 October 1999, and in "Nuclear Education and Training: Cause for Concern" (OECD, June 2000).

- to utilise the information from the current position of FAPs to give a view on future issues of strategic importance, and to make recommendations accordingly; and
- to make recommendations concerning the possible future activities of the group.

The group's work is based on a number of previous activities which have been reported in SESAR documents.[2] This means that it benefits from this experience and its recommendations should be seen in the context of an analysis of the situation that is continually being refined. Because of the rapidly changing situation, the report was "frozen" at December 1999. A few updates have been made, but none of the principal recommendations have been compromised.

In addition to these objectives, one of the critical requirements for future planning is the establishment of minimum programme needs to help the SESAR draw a benchmark of the necessary future activity and thereby give the CSNI a baseline against which to judge its recommendations. Such a set of minimum programme needs has been introduced and is discussed in detail in the individual chapters, and is put into the context of the whole project in the main introduction. We believe that this is a potentially important contribution from the group and that it should be used by others, especially the CSNI Working Groups[3] in their work.

Definition of "Centres of Excellence"

In this report we use the term "Centre of Excellence" rather loosely. Such centres have proved successful in the past, for example the Halden Reactor Project, PHEBUS-FP, and earlier the LOFT Project.

The basic constituents of a Centre of Excellence as defined for this work are as follows:

> *"Joint undertakings and international research programmes centred around internationally recognised expert teams operating a major facility which participating countries have agreed forms an international resource which they will jointly support for a reasonable period or internationally recognised research programmes focused on major topics which participating countries use as a means of maintaining their capability in that technical discipline, or a combination of these two"*.

We note that the usual model for a Centre of Excellence would be a team of internationally recognised experts centred upon a major experimental facility. However, we would also include focused analytical/theoretical teams and, increasingly on, "virtual" Centres of Excellence where members are not usually co-located, but communicate via modern telecommunications.

Under some conditions, networks of experts and facilities operating in a co-ordinated manner to investigate focused problems can be considered to form a Centre of Excellence. An example of such

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2. "Nuclear Safety Research in OECD Countries"; OECD, 1994, "Nuclear Safety Research in OECD Countries: Areas of Agreement, Areas for Further Action, Increasing Need for Collaboration"; OECD, 1996, "Nuclear Safety Research in OECD Countries: Capabilities and Facilities"; OECD, 1997.

3. The structure of the CSNI's support groups was changed at a very late stage in the development of this document; for practical purposes we use general terms such as "CSNI Working Groups" or "CSNI Expert Groups".

a network Centre of Excellence was the PISC (NEA Programme for the Inspection of Steel Components). Similarly, under appropriate conditions, data banks and the associated expertise required to maintain them can also be considered Centres of Excellence. These conditions include high quality advanced work as well as focused topics and an internationally supported strategic plan with well-defined long-term objectives.

We recognise the practical difficulties with funding (especially continuity of funding), mobility of research workers, international recognition of academic qualifications and, of course, the questions of security of national strategic needs.

It is worth noting that similar discussions are underway in other international organisations, in particular the European Commission in the framework of the Communication on European Research Area (ERA). This is proposing a number of objectives of common interest, such as mobility of researchers, opening up of national programmes and networking of Centres of Excellence in addition to measures more specific to the European Union countries.

Method of working

The method of working follows that utilised in previous SESAR reports. Senior experts from Member Countries took responsibility for specific technical areas[4] and have written the individual chapters. This accounts for some differences in scope and detail between the various chapters. Once the chapters were completed, a consensus on recommendations was then obtained from the whole group. In addition a peer review was also undertaken. In this particular activity there were no areas where consensus recommendations could not be reached and so there is no need for any minority reports. The detailed recommendations are given in the chapters whilst for this summary they have been paraphrased as necessary.

While the methods of working are discussed in some detail in the main introduction, we concentrate here on the criteria we have adopted in order to "screen" our recommendations. The criteria have been applied using the so-called "essential requirement templates" (abbreviated to "templates"), which are now incorporated into each chapter. The 'questions' in this mechanism are summarised in the following generic template. The aim here has been to be as transparent as possible in exposing how the group reached its recommendations. We should note that the minimum programme needs act as a base line against which to judge requirements. They do not in themselves identify facilities and capabilities at risk.

The generic template in the following figure summarises the questions to which each technical area has been subjected in the evaluation processes and, according to the different positions in the template, the conditions that must be fulfilled to support a recommendation for further action. Those issues which satisfy the criteria and secure a "yes" response in the "need for future CSNI action" column of the table are then expanded upon in the list of recommendations. This approach was applied to approximately 97 technical areas and issues.[5] It therefore represents a comprehensive coverage of a large number of relevant issues, but is, inevitably, not exhaustive.

The SESAR Group has considered what is the meaning of "essential" in this context and has used its judgement to form the basis for its recommendations by building upon its experience with

4. The technical areas follow the same definition as in previous SESAR documents.

5. As defined in the templates in the individual chapters.

respect to what type of information is necessary to resolve a safety issue. Accordingly, in the absence of existing national or international programmes, SESAR/FAP has developed a limited number of recommendations to maintain or enhance co-operative effort, or to launch new programmes where it has been judged essential needs are in danger of being lost. We have tried, through our use of these criteria to be as transparent as possible in giving a rationale for our decision for these recommendations.

Generic template for determining essential facility and programme needs and need for CSNI action in specific technical areas

Chapter	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	CSNI action needed?
Technical areas.	List the technical area or phenomenon still unresolved or of high uncertainty. Source of issue could be from: • SESAR reports • operating experience • industry initiative • other	Rank the safety significance of the technical issue based upon deterministic analysis, PSA information and/or qualitative judgement. This is expressed in the simple categorisation as being High, Medium or Low.	Is there already sufficient knowledge to resolve the issue? This is a simple yes or no based on expert judgement.	Is a facility or programme needed to resolve the issue?	Criteria for inclusion[6] as a recommendation. *Overriding Criteria:* • Is there a major short-term safety issue, and is it relevant in the overall context of future needs? and • Is the FAP currently under threat? *Additional Criteria:* • Is the project risk important? • Is the technical area still open? • Is the FAP unique to the nuclear industry? • Is the FAP flexible?

6. In order to emphasize that these criteria were used to filter our recommendations, they are repeated before the appropriate sections of each chapter.

Principal Recommendations

The SESAR Group has arranged its recommendations into two categories:

1. Strategic Recommendations (i.e. those that address broad crosscutting or policy matters) and

2. Specific Recommendations (i.e. those that cover specific activities). These are discussed in turn below.

Strategic Recommendations

- That immediate consideration is given to setting up Centres of Excellence, on an international basis, as the key focal points for the safeguarding of key Facilities and Capabilities in the future. Examples of these are given for specific technical areas in the following recommendations and in the chapters. However, there is now an urgent need for practical steps to be taken to begin to identify potential areas and to establish what must be done to create viable Centres.

- That the future role of the CSNI Working Groups in this context should be to provide points of focus for the specific technical areas which can give easy access to necessary up-to-date technical information. This should be set up so that such information can be made readily available, and not be linked only to annual meetings.

- It is vital to find ways of drawing the Industry into this future activity. The means for this may well be different in different technical areas, but a feeling of "ownership" by the Industry is necessary.

- With the closer ties between the OECD and New Independent States (NIS) of the Former Soviet Union and Central and Eastern European Countries (CEEC), it is now timely and important to perform a full review of the facilities and capabilities potentially available in those countries.

- It is necessary to develop and factor into CSNI activities a strategic vision of the needs of the research community and Industry, based upon information available from, for example the CNRA work,[7] the NDC study[8] and the recent CNRA Budapest Workshop.[9]

- CSNI should have a strategy to ensure recommendations and actions remain current.

Recommendations for Immediate Action on Facilities and Programmes

The following recommendations have been taken from the individual chapters as being the Group's view of the highest priority and those most deserving of action. Not every chapter is

7. Future Nuclear Regulatory Challenges; OECD, 1998.

8. Nuclear Education and Training: Cause for Concern?; OECD, 2000.

9. Assuring Nuclear Safety Competence into the 21st Century; OECD, 2000.

represented here, just those leading to the highest priority actions. The detailed justification for these recommendations (using the template already described) is given in the individual chapters. Near term and longer-term actions are presented. Near term refers to the next one to five years and longer term, > five years.

(a) Near-term actions on facilities and programmes

According to the arguments in Chapters 2 and 4, we previously recommended action by CSNI to prevent the near-term loss of key thermal-hydraulic and severe accident facilities. These recommendations are in direct response to our first two terms of reference.

Thermal-Hydraulic FAPs

Three major Thermal-Hydraulic facilities, in danger of being closed in the near term (1-2 years), are recommended for international collaborative programmes. In priority order, they are PANDA, PKL and SPES.

We continue to believe that it should be the aim of the research/industry community to maintain LWR and PHWR thermal-hydraulic facilities, both because of the potential need for future confirmatory experiments, and to support code development and provide educational opportunities – that is to underpin the future of capabilities in this area. Therefore, SESAR/FAP also recommends that the CSNI monitor the status of other key thermal-hydraulic facilities, including APEX and PUMA (in the USA), RD-14M (Canada) and LSTF (Japan), to ensure that they (or successor facilities, as appropriate) remain available to Member organisations.

Severe Accident FAPs

The situation on FAPs in the severe accident area is moving quickly. During the period when this report was being prepared, the FARO facility at the JRC Ispra has been announced for closure, and most recently the possible closure of the ALPHA facility in Japan, throwing the whole area of Fuel/Coolant Interactions (FCIs) and its supporting FAPs into a difficult position. Events have overtaken the Group, clearly illustrating the need to continuously monitor and respond to events. The Group's recommendation is based on our understanding of the current position.

As a necessary response to the current situation, the Group recommended and CSNI approved a meeting of technical experts[10] in the area of FCI and Core Debris behaviour. The objectives of such a meeting would be:

- to consider the impact of the loss of the FARO facility and its associated expertise. The meeting should note that the loss of FARO is serious as this was a unique facility in both in and ex-vessel FCI and its closure leaves key safety issues unresolved.[11] This now leaves the field with few facilities (principally ALPHA – but note the warning above –

10. We note that by the time of finalisation of this document, a meeting on Core Debris Behaviour had taken place. A meeting on FCI was under consideration.

11. By unresolved, we mean in this context that remaining uncertainties in the physics process make it impossible to either rule out the mechanism as a threat (albeit of very low probability), or be able to calculate with sufficient accuracy the implications if such an event were to occur.

and MAGICO) and, if they are to be safeguarded, the meeting must recommend suitable actions.

- to consider the programmatic needs of the area and the availability of FAPs and other resources required to provide them.

- to address the case for a future strategy for an international Centre of Excellence in FCI, to consider its location and potential resource needs.

- to consider the programmatic needs and action necessary to obtain experimental results in debris coolability needed in order to be able to understand in-vessel and ex-vessel debris coolability and the viability of core catchers.

These considerations should include proposals for the RASPLAV continuation and, on the basis of a host country proposal, the MACE facility should be considered for a co-operative OECD project.

The SESAR/FAP group also recommends that a data bank for Fuel/Coolant Interaction and Debris Coolability be established.

In addition, many national programmes on iodine chemistry and fission product behaviour under accident conditions have been terminated or dramatically reduced over the past several years, with the prospect of further decreases in the near term. To ensure that adequate expertise and capabilities remain available to the international nuclear community in this area of reactor safety, and to facilitate international collaboration among Members.

SESAR/FAP recommends that a Centre of Excellence on Iodine Chemistry and Fission Product Behaviour be established immediately.

Integrity of Equipment and Structures FAPs

There exists a substantial quantity of data and information on ageing, but it resides in different organisations, different countries and in diverse formats. It is necessary to collect this data and to make it consistent and readily available. Most facilities for materials testing, including standard and medium scale testing machines, hot cells and EAC (Environmentally Assisted Cracking) rigs, exist in Member countries and there is no need to identify central facilities that might be operated co-operatively. For critical needs in the area of structural integrity assessment (ageing of Nuclear Power Plant components with safety significance), information and knowledge should be available. The highest priority recommendations are given below:

- to co-ordinate and develop an overall international programme in materials ageing issues relevant to the nuclear industry, especially those with safety implications, and

- to begin a study of the development of an international data bank and Centre of Excellence for materials ageing data and analytical tools with the aim of attracting Industry and Government Agency participation. The NEA Nuclear Data Bank may be seen as a model, but the Group would not wish to limit any future Centre of Excellence activities. This recommendation directly addressed our third term of reference as it is strategic in nature.

This is revisited below in our recommendations concerning actions for the working groups and the generation of future technical advice. Other near-term recommendations for action were also developed. These are summarised in the following table and discussed further in each chapter.

(b) Longer-term actions on facilities and programmes

Longer-term actions at this time involve monitoring facility status and ensuring certain key programmes are maintained. These are summarised along with the near-term actions in the following table and discussed in detail in the individual chapters. They also include specific requests to the Working Groups for action. We have also included in the table all of the immediate actions discussed above.

SESAR/FAP recommendations

Area	Immediate	Near term	Longer term
Chapter 2: **Thermal-Hydraulics**	PANDA, PKL and SPES should be considered for a co-operative OECD project; needed to support code development and provide educational opportunity.	Experts to: • define experimental needs • co-ordinate T/H research programmes • collect, maintain and service key experimental data. Monitor the status of the APEX and PUMA facilities in the USA, RD-14M in Canada and LSTF in Japan.	Evaluate feasibility of reduction to one major facility for each reactor type worldwide.
Chapter 3: **Fuel and reactor physics** Chapter 3.1: Fuels		Experts to: • define experimental needs • promote exchange of information about codes through user groups. • promote industry involvement, including cost sharing	Monitor hot cell and reactor availability for RIA and LOCA exper-iments for high burnup fuels.
Chapter 3: **Fuel and reactor physics** Chapter 3.2: Physics		CSNI Working Group to: • strengthen ties to NSC (NEA Nuclear Science Committee) work on physics for future systems.	Monitor facility availability for measurement of nuclear data and integral reactor physics parameters for modern fuel and core designs.

Area	Immediate	Near term	Longer term
Chapter 4: **Severe accidents** Chapter 4.1 In-vessel Chapter 4.2 Ex-vessel	• Conduct a meeting or meetings of experts to consider the situation concerning ex-vessel FCI in the absence of FARO and the needs for experimental data in debris coolability. • On the basis of a host country proposal, MACE should be considered by a group of specialists for a co-operative OECD project. – Include proposals for the RASPLAV continuation project.	Experts to: • develop data base, • assess impact of up-coming losses of expertise, and • consider the needs for further data on core catchers.	• SOAR on Severe Accident issue resolution and research needs. • Generation of a database on debris coolability. • Consider the needs for further data on core catchers.
Chapter 4: **Severe accidents** Chapter 4.3: Fission Products	Establish a Centre of Excellence in Iodine Chemistry and Fission Product Behaviour.	Monitor status of iodine and fission product behaviour facilities (associated with FP release, transport and containment phenomena)	• Monitor status of facilities capable of performing fission product release experiments with active, irradiated materials. • Establish benchmark exercise on iodine and fission product behaviour.

Area	Immediate	Near term	Longer term
Chapter 5: **Human factors**		CSNI Working Group to: • define research needs for human performance data collection and modelling, • pursue the participation of industry, including non-nuclear industry, and • monitor and update human factor research strategy.	• Maintain the Halden Reactor Project as a Centre of Excellence. • Promote availability of HF expertise. • Monitor the effects of Utility restructuring on management and organisational performance research needs.
Chapter 6: **Plant monitoring & control**		CSNI Working Group to: • monitor and communicate to Members I&C developments in non-nuclear industries on safety, control and monitoring systems, • monitor developments on qualification of non-nuclear software for possible nuclear application; encourage Member collaboration on software qualification for safety systems, • monitor developments on introducing software-based monitoring and decision-aid tools into control rooms, and • define research needs to ensure safety of replacements of ageing I&C.	• Maintain I&C initiatives with Halden Reactor Project • Strengthen ties with NSC work on software

Area	Immediate	Near term	Longer term
Chapter 7: **Integrity of equipment and structures**	Establish a Centre of Excellence in materials ageing (data, methods, and conditions for life management.)		Monitor hot cells for mechanical testing and material testing reactor availability
Chapter 8: **Seismic**		CSNI Working Group to: • promote data exchange, including earthquake observation data.	Monitor status and identify opportunities for co-operative programmes using large shaking tables. Establish a "virtual" Centre of Excellence for test data and analysis results.
Chapter 9: **Risk assessment**		CSNI Working Group to: • promote the sharing of methods, criteria and experience and strengths and limitations.	To pursue co-operative efforts to develop methods, standards, criteria, data and analyses.
Chapter 10: **Fire risk assessment**		CSNI Working Group to: • define experimental programme needs based upon fire PSA needs.	Experimental work should be done co-operatively.

Chapter 1

INTRODUCTION

1. Background

The NEA Committee on the Safety of Nuclear Installations (CSNI) has a central role in providing OECD Member countries with authoritative advice on matters relating to nuclear safety. Because of this pivotal position and having available input from the most senior safety experts from the Member countries, it has been involved with and been concerned about the maintenance of essential research capabilities and facilities[1] for some years. In particular, it has noted the decline in available resources in many Member countries and the inevitable piecemeal loss of physical and human resources. In some Member countries this has been exacerbated by economic and structural changes. Since its aims include engendering international collaboration and the identification of areas of agreement and areas for future action, it undertook some time ago to gather information, analyse it and develop an international strategy for the efficient husbanding of essential nuclear safety research. The first work was commissioned in 1992, when a Senior Group of Experts on Safety Research (SESAR) was set up to review research being carried out and draw conclusions on likely future requirements and priorities. Its work was reported in the document "Nuclear Safety Research in OECD Countries".[2] A follow on report was also commissioned which took the arguments further. It reported in the document "Nuclear Safety Research in OECD Countries: Areas of Agreement, Areas for Further Action, Increasing Need for Collaboration". The result of these studies was to raise a concern in the CSNI about the ability of Member countries to sustain an adequate level of safety research capability individually, and potentially collectively, even though there was an international consensus in the majority of technical areas on research needs and objectives. This concern extended from the loss of important experimental facilities and the failure to replace them, to the loss of critical competencies and hence threatened to undermine the ability to adequately regulate and support operating reactors and in the development of new designs. Therefore, in November 1995, the CSNI decided to set up another SESAR, this time to focus more specifically on research capabilities and facilities (SESAR/CAF). The results of this activity led to a wide-ranging review of available capabilities and facilities in the document "Nuclear Safety Research in OECD Countries: Capabilities and Facilities". This was presented to the CSNI in December 1997. The report contains a very large amount of information on capabilities and facilities in Member countries, although it is not claimed to be exhaustive. It provides an overview of the SESAR's projections for both the short and near-term

1. The terms capability and facility are used in this report to cover, respectively, "the ability to perform a given task or to fulfil a given requirement; it implies a combination of expertise and access to specialised equipment" and "a semi-permanent installation, generally larger than bench scale; this may also include analytical equipment which has been made especially for a given type of study".

2. Nuclear Safety Research in OECD Countries; OECD, 1994.

(< 5 years) and long-term (> 5 years) time scales. It also contained specific recommendations to the CSNI for future action. In summary these were:

- That the CSNI take a proactive role in organising and implementing co-operative projects to provide key safety related facilities in the future.
- That the CSNI act as a forum at which facilities threatened by closure are identified and any possible international support programmes initiated.
- That the CSNI continue to monitor the situation with input from its Working Groups and receive advice from a small group of suitably qualified experts able to encourage co-operative actions in the light of evolving needs.
- That the CSNI seek a solution to the continuing problems arising out of commercial and other sensitivities and intellectual property rights.

As a direct result of this, the CSNI took the view that further and more specific action was justified in order to address this concern which is perceived as serious for the Industry and the Regulators, and which is getting worse. It decided, therefore, to continue the work of the SESAR, but with very significant changes to its mandate. The key tasks assigned to the group may be summarised as:

- To identify facilities potentially interesting for present or future international co-operation (whether this has been proposed or not by the host country).
- To collect and discuss cost information (operation and maintenance) about these facilities, staffing needs and in some cases the needs for upgrading or refurbishment.
- To make specific recommendations regarding facilities and research programmes, joint projects, etc., including, if necessary, recommendations regarding new facilities.
- To discuss, where appropriate other forms of international co-operation (e.g. data banks, exchange or sharing of experts, the establishment of Centres of Excellence, joint development of computer codes, etc.).
- To present an annual report to the CSNI.

In this, the setting up of the group can be seen as a response to the recommendations of the previous work, especially in the need for a much more proactive mode of operation; it will move the debate ahead by suggesting practical solutions. Also, the terms of reference recognise that there is an evolving situation and that "snapshots" of the current position (or at least the current position when the data was gathered) are inadequate in the dynamic situation pertaining today. This all reflects the view that the heightened importance of optimal international research programmes requires a more formal and focused approach on the part of the CSNI.[3]

The terms of reference for the SESAR also call upon the group to provide costing information for FAPs that are requiring action. We have tried to do this as well as possible, but such information is notoriously difficult to obtain, and has to be treated with great caution to avoid

3. We note that the recently initiated CSNI Programme Review Group will carry this forward as part of its terms of reference.

differences of accounting convention. Where possible we have made estimates of costs, but they must be seen as coming with a health warning.

It is the dynamic nature of the situation that has brought into sharper focus the need for a more structured approach to achieving a strategy for international collaboration. The process which has been adopted in this work in order to achieve that which is discussed in section 3 of this introduction. We begin by highlighting some of the key events and changes which are affecting current decision making in the nuclear context and hence directly impact the issues driving the need to improve collaboration in the safety research area.

2. Context

Since the establishment of the previous study in 1995, and certainly since the first work was initiated in 1992, there have been a number of key changes to the milieu in which the nuclear industry in general and nuclear safety research in particular finds itself. These affect the basic assumptions on which the "foresight" process for determining the future needs of capabilities and facilities depends. The assumptions used in the previous work were rather simple, i.e.:

- plants currently operating will seek life extension and some will seek uprating;
- new designs will continue to be developed; and
- operational problems will continue to occur.

In fact, it is now, of course, clear that the situation is more complex than this and that a number of sometimes conflicting pressures must be taken into consideration. Forecasting under these conditions is notoriously difficult, especially for the longer term. Nevertheless, difficult or not, it is essential to use the best information available and the judgement of experts since the alternative of no strategic planning is not acceptable. All decision making is done under uncertainty and it is our aim to reduce the associated risks by bringing together information and expert judgement to give the best possible advice. Clearly, focusing on shorter, rather than longer-term developments will give a better chance of success. Nevertheless, one of the key challenges facing the group is to rationalise the short-term requirements against the current best estimates of the medium and long term trends.

There are many pressures on the Nuclear Industry today. In principle they are well known, i.e. economic, environmental and socio-political. However, the balance between positive and negative influences changes, as do the time scales over which the changes occur. One of the most significant changes in recent years has been the international recognition of the need for action over global warming. The arguments accepted at the Kyoto conference in 1997 mean that the burning of fossil fuels will become increasingly unattractive for the heavily energy dependent industrialised countries. The aspirations of emerging economies add another dimension to the difficulties involved in implementing the objectives. In some camps this is seen as a positive challenge to the nuclear industry as one of the few technically secure ways of guaranteeing future electricity supplies without increasing CO_2, or other greenhouse gas, production. It is also true that the burning of natural gas has accelerated, as have the identification of new sources and the evolution of technology that allows much more economical recovery of existing difficult gas resources. Add to that the large reduction in costs for oil and gas recovery from tars and shales and a picture is emerging of a period of at least several decades before security of supply will be of serious concern for most Member countries. Of course, the additional complication of the possible trading in greenhouse gas credits offers another route by which the pressure for a decision on replacement technologies for energy supply could be lessened in the short term. This means that the economic and political pressure to order new plant is, if anything, lower now than three years ago, and, with the exception of Eastern Asia[4] and perhaps Finland and Turkey, and the special case of the CEEC and NIS countries, no new plants are currently being considered. We therefore believe that there will be an extended period before arguments relating to sustainability of supply will prevail and, therefore, this enters the decision making process as a further reason for the need to safeguard expertise over what may be an extended period before any expansion of the nuclear generating capacity is to be expected. Sustainability of supply in itself is not a strong driving force over the near term and hence for this present activity.

There are additional and more immediate factors that have an important bearing on the present context of nuclear power. Not all of them apply to all Member countries, but they are certainly having very significant effects in some. Many of these are discussed at some length in the introduction to the "New Future Nuclear Regulatory Challenges" document[5] and are not repeated here.

In the field of nuclear power safety, OECD Member countries' government agencies have broadly similar responsibilities. In several Member countries, the funding levels of national government safety research programmes have been reduced over recent years, and increased reliance placed on private companies to maintain an adequate level of safety research. Care is needed to ensure that this does not have an adverse impact on the ability of government agencies to fulfil their safety responsibilities, especially since the reduction in government direct or imposed funding in this area may not have been offset by concomitant increases from the industry. The CSNI has expressed concern that dwindling budgets and support as well as stagnant programmes may lead to the untimely shutdown of large facilities and the breaking up of experienced research teams with the consequent loss of competence and the reduced capability to deal quickly and efficiently with future safety problems.

It is understandable that the Industry's response to the situation outlined above has been to reform and regroup to give itself the best chance of survival. The supply side, i.e. plant constructors and the infrastructure of supporting companies has had to switch from new projects to services and upgrading. Their requirements for innovative research have, therefore changed considerably. The nuclear generating utilities likewise have had to reconsider all aspects of their operations in response to commercial pressures and increased competition.

This indicates how volatile the "context" of the nuclear industry is today. Of course not all of the most difficult situations apply to all Member countries and they are by no means homogeneous in their own "national context". However, in attempting to derive international actions, the overall situation has to be taken into account. Planning in this environment is clearly complex and difficult. However, it is possible to extract the key technical issues facing the industry in light of these conditions. These reflect the basic driving forces now acting on the nuclear industry:

- very few new plants will be built soon;
- asset protection and life management;
- end of life and waste disposal; and

4. The economic downturn in these economies has brought even this assumption into question, again demonstrating the dynamic nature of the current situation.

5. "New Future Nuclear Regulatory Challenges" OECD 1998.

- general economic pressures, including deregulation, ownership consolidation and particularly in the USA the move towards risk informed regulation.

From these concerns, we believe that the following are the key technical issues that underpin the needs of both industry and regulators in the current context:

1. Plant life management, including:

   - ageing of components, systems and structures;

   - ageing of analytical tools and documentation;

   - application of modern standards to old plant (and the ageing of the standards themselves in the light of new developments);

   - life extension; and

   - backfitting and upgrading.

2. Reduction in operating margins, including:

   - fine tuning plant for greater efficiency, for example by power upgrading as has been done in Finland and Sweden;

   - extending fuel burnup;

   - responding to the economic pressures of deregulation; and

   - using risk – informed regulation through expanded and improved applications of PSA.

3. Severe Accidents, including:

   - the continuing need to develop practical and robust severe accident management schemes;

   - the optimisation of design solutions for next generation plant; and

   - the improvement of the capability to quantify accident progression and radioactive source terms.

Therefore, our working assumptions are that the key issues summarised above will provide the main driving forces for the research which is required, and that the dynamic period of change will continue, making it necessary to focus planning primarily on long (up to ten years) and short term (up to five years) issues.

In conclusion, we believe that the dynamic change currently being experienced by all parts of the industry world-wide and the potentially conflicting demands for securing the necessary safety research demands a proactive, “managed” approach to the future provision of nuclear safety research capabilities and facilities. It is in this “context” that this work has been done and which has led to the definition of the objectives discussed below.

3. Objectives

The above descriptions of the tasks provide the outline of the objectives of this work. We note in particular that the overriding objective is to identify areas for collaboration that would give a higher efficiency at the international level. We might add that this collaboration is seen as providing a bridge to cover the expected complex period of evolution and change currently being experienced. More specifically we can identify a number of areas where co-operation would be beneficial. These are:

- to resolve specific current safety issues;
- to safeguard key facilities through collaborative experimental programmes;
- to sustain key expertise through combined research and applications programmes;
- to provide training and education[6] in relevant nuclear engineering and other physical sciences for the refreshment of the nuclear safety research community;
- to develop and maintain databases of relevant information both from previous research output and in underlying materials properties; and
- to determine the need, justification and scope of new facilities which would lend themselves to collaboration from the outset.

A crucial element in all of these is to provide recommendations on the practicality of the suggestions, including an estimate of costs, both capital and ongoing.

In addition to these objectives, one of the primary requirements for future planning is the establishment of minimum programme needs[7] so that the SESAR/FAP could draw a benchmark of the necessary future activity and thereby give the CSNI an anchor against which to judge its recommendations. Such a set of minimum programme needs is addressed in this introduction and in the later chapters.

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6. The group is aware of an action currently underway under the auspices of the NEA's Committee for Technical and Economic Studies on Nuclear Energy Development and Fuel Cycle (NDC). This is entitled "Survey and Analysis of Education in the Nuclear Field: Infrastructure for the Uses of Nuclear Energy". This is essentially an information gathering exercise to obtain a clearer picture of the current situation. We note that this is much broader than the nuclear safety field and would not cover the means for attracting good physical scientists into the nuclear field. Nevertheless, its input will be useful in giving a quantitative "yardstick" as to the potential supply of trained "nuclear knowledgeable "engineers in the next generation (see reference 8 of this report). In addition, there are activities under the auspices of the NEA Committee on Nuclear Regulatory Activities (CNRA), including a recent major meeting in Budapest (reference 9). These serve to highlight the importance now being given to this topic by different interested parties. Other activities include the so-called SOFRES study being undertaken by the European Commission.

7. Minimum programme needs act as a baseline against which to judge requirements. They do not in themselves identify Capabilities and Facilities at risk, but rather would expose any which are and would still be needed to fulfil the minimum programme requirements.

4. Method of Working, Scope and Organisation of the Report

The method of working follows that utilised in previous reports. Senior experts from Member countries have the responsibility for appropriate technical areas and gather the necessary information from the whole Membership. Consensus on the implications and recommendations is then obtained from the whole group at plenary sessions. In the rare cases of disagreement on priorities or recommendations, the report identifies differing views.

4.1 Criteria

Because this work intends to provide practical, and as far as possible quantitative advice, it is important that its recommendations are transparent and that the criteria underpinning them are well established. Whilst this will always have a strong element of expert judgement, the group has used the following criteria in their work. The overriding criteria are:

- is there a major short-term safety issue and is it relevant in the overall context of future needs?

- is the capability or facility currently under threat?

These reflect the overall objectives of the group, but are not specific enough to define practical projects based on combinations of quantitative data and expert judgement. We have therefore agreed the following set of more detailed criteria against which we have judged the suitability of projects for international collaboration:

- *Risk importance*: this may be quantitative, being derived from either PSAs or on system based assessments, or it may be judgement, based on the identification of key issues in uncertainty reduction.

- *Closure*: it is essential that the international community agree that the technical issue addressed by the facility is still open.

- *Uniqueness to the nuclear industry*: resources should be focused on capabilities and facilities where there are no other front line industrial or research interests. Examples of uniqueness are: core melt progression experiments, specific thermal-hydraulic test rigs, test reactors and critical facilities and hot cells. Examples on non-uniqueness are: instrumentation, heat transfer, basic Computational Fluid Dynamics (CFD) developments and some aspects of human factors.

- *Applicability to a broad range of conditions*: capabilities and facilities need to be flexible and able to accommodate different users needs. It should also be relevant to scaling criteria, if the technical demands of the subject area require it.

- *Responsibility*: the ownership of the capability (either Government or Industry) must be clearly established and there should ideally be clear commitment from the owner to support any future international programme in their facility. However, for specific and important FAPs, the group will raise the possibility of an international action even if the host country is not initially supportive.

- *Credibility*: the management provisions must be acceptable against modern standards, e.g. including proper financial, quality and technical control.

- *Size*: in order to limit the number of possible projects, an initial cut off point will be USD 1 million or more.

4.2 Scope

In terms of reactor types, the work will continue to be limited primarily to Western LWRs (PWRs and BWRs), and to PHWRs. Where relevant, this can be extended to VVER reactors.

The technical areas covered are as before, and for continuity we have retained the categories used before. These are:

- Thermal-Hydraulics.
- Fuel and reactor physics.
- Severe accidents.
- Human factors.
- Plant control and monitoring.
- Integrity of plant and structures.
- Seismic analysis.
- Risk assessment.
- Fire risk Assessment.

On the Risk Assessment topics we note that it is very much a “compendium subject” which draws on input from many other areas. Nevertheless, the basic methods and issues (e.g. the treatment of uncertainties, common cause failures, completeness and the inclusion of human factors) are separate, well defined technical issues in their own right. In the previous report this heading included both fires and seismic events. These are in fact distinct topics in their own right and so this time they are included as separate categories.

Where there are implications beyond strictly adhering to technical safety issues the group will consider them on a case by case basis and provide its comments and recommendations to the CSNI.

The geographical coverage will, of course, extend to all the Member countries,[8] but in addition will include capabilities and facilities in the Russian Federation. The rationale for the latter is the availability of large scale, relevant facilities that are potentially able to make an economically attractive, technically sound and sometimes unique contribution to reactor safety research. It has the

8. We have included in this category those CEEC countries currently enjoying membership of the NEA.

additional value of providing a means of safeguarding some of the key nuclear capabilities and facilities in that country, an action identified as necessary in the continuing support from Western countries to the improvement of nuclear safety in the New Independent States of the Former Soviet Union. At present the coverage in the Russian Federation is not comprehensive and it is the aim of the group to expand this in the next phases of the group's work.

The time frames will also be as in the previous report, with short-term meaning < 5 years and longer term > 5 years. As we have said above, we would expect priority to be given to those activities pertaining to the short term, but with the added requirement that short-term priorities are set against longer term considerations. Also, because this is now a rolling programme, issues may arise as they come into its planning horizon.

5. Summary

The general approach adopted in this work may be summarised as follows:

- To build on the areas categorised and reviewed in the earlier SESAR/CAF report, as updated in the technical chapters here.

- To define a set of criteria for judging the suitability of projects for international collaboration.

- To highlight particular concerns in the short term established in the broader context of longer-term trends.

- To obtain technical reviews and advice from experts via specific questions generated out of the SESAR/FAP's considerations and from special ad hoc specialists' meetings concerning potential programme planning for capabilities and facilities.

- To develop a set of practical recommendations for the CSNI.

Chapter 2

THERMAL-HYDRAULICS

2.1 Background

The SESAR/FU report [1] has recognised that a common technical position exists in that "extensive knowledge is available in the field of transients and major system computer codes have achieved a high degree of maturity". However, there is continued need for some additional experimental and code work.

Experimental programmes are needed to fill any lack of knowledge on physical phenomena and the course of transients and accidents, and to provide the experimental database for code validation. Computer codes are required for the simulation of full-scale reactor plant behaviour under these conditions. Best-estimate methods are to be preferred to the conservative approaches taken traditionally for design basis analysis. Best-estimate analyses are increasingly used in certification and licensing of new plants or power upgrading of existing plants. Rigorous validation and quantification of uncertainties is required for acceptance by regulators. It is imperative to ensure that findings of experimental programmes are properly used to improve the models in these codes. It is also important when defining experimental programmes to keep in mind the modelling needs of computer codes.

Various experimental programmes covering Separate-Effects Tests as well as Integral Tests are carried out in OECD Member countries, the CEEC and NIS. Tests in sub-scale and full-scale test facilities are being performed, addressing safety issues of plants in operation, as well as open questions related to new reactor concepts. Table 2.1 shows the correspondence between the priority areas already identified by SESAR/FU and examples of current research programmes.

An overview of the large number of separate-effects test programmes that have been carried out or are in progress is given in [2]. The results available provide a sound basis for model validation for the current system codes, although the available data are insufficient for future multi-dimensional codes.

For the investigation of current LWR system behaviour, large integral test programmes have already been carried out or are still underway in the USA, Japan, France, Germany, Italy, and Sweden. The tests already performed are presented in [3] together with a systematic selection of the best openly available data for code validation

For PHWR (CANDU-type) reactors, experimental programmes have been conducted in specialised facilities (e.g. Large Scale Header Facility, Cold Water Injection Facility, End-fitting Characterisation Facility) using full-size components such as headers, fuel channels and end fittings. In addition, a comprehensive test programme on emergency core cooling system (ECCS) effectiveness, natural circulation and shutdown cooling is being undertaken in the integral test facility RD-14M, a full-height, multi-channel representation of a CANDU reactor cooling system. Very recently, studies on the rate of channel voiding have also been initiated in RD-14M to address the issue of channel void generation relevant to power pulse during the early stages of a LOCA. For VVER reactors, experimental programmes have been conducted or are underway. For the VVER-440, the existing Finnish PACTEL and Hungarian PMK facilities can fulfil most needs; in addition, some large-scale separate-effects horizontal steam generator tests may appear necessary. For the VVER-1000, the Russian PSB facility has recently become available. There are international activities under the auspices of the OECD underway to help complete it. A systematic overview of the VVER situation was compiled using the same approach of relating the experiments to physical phenomena for various accident types as in the CSNI code validation matrices for Western type reactors [4]. This report is being updated and extended by the OECD Support Group on VVER TH Code Validation Matrix. The new report [12] will show the database now available for code validation and will allow the identification of any gaps in the database applicable to the VVER reactors. Research needs may arise from the analysis of thermal-hydraulics in VVER containments, see Chapter 4.2.

For advanced LWR (ALWR) designs, both PWR and BWR, experimental programmes are in progress. The tests are focused on new components and systems and address the passive safety features of the new designs. Most of these experimental programmes are being carried out with international co-operation. Results from some of these tests are already being used in the design certification process for ALWRs. The need for further tests will depend on the future evolution of the corresponding reactor programmes.

Code improvements (e.g. addition of specific models addressing phenomena such as condensation with non-condensable gases and mixing) addressing the needs of ALWRs and validation of existing system codes with ALWR test data are underway in the US, Italy, Germany, Japan, Switzerland, the Netherlands, and other Member countries.

Similarly for advanced PHWR designs, experimental programmes are underway in Canada to test new components and new passive safety features, and codes are being extended to model design enhancements, notably the higher temperatures and pressures being considered for efficiency improvements.

2.2 Needs and Challenges

Although extensive knowledge is available in thermal-hydraulics, there is a need for continued experimental research and code development.

For existing reactors, existing safety margins have to be evaluated considering flexible plant operation, possible power uprating, higher fuel burnup, and plant ageing. This assessment has to be based on best-estimate code calculations including quantification of uncertainties. Code capabilities, especially for multi-dimensional flows and for coupled thermal-hydraulic and neutronic processes have to be enhanced and thoroughly validated. The experimental database for code validation has to be completed.

For new reactor designs, additional requirements are emerging regarding passive safety features and containment phenomena.

2.2.1 *Near-term needs*

The actions to be taken within the next five years are of two different kinds:

- urgent decisions: Save key experimental facilities from imminent closure;
- preparatory efforts towards long-term goals: Establish a common experimental programme in support of next generation computer codes.

Experimental facilities and programmes

Future experimental programmes should significantly reduce uncertainties regarding phenomena, processes and system behaviour and extend the required database for model improvements and code validation of the currently used system codes such as APROS, ATHLET, CATHARE, CATHENA, TUF, RELAP5, TRAC, etc.

The SESAR/CAF report [7] has stated the needs for new experiments. For current LWRs, these are related to applications in areas extending beyond the current model/code development and validation areas; they include:

- start up and cooldown transients;
- low power and shutdown operation modes;
- accidents involving multiple system failures; and
- development and evaluation of preventive accident management procedures.

For advanced LWRs based on evolutionary, innovative or passive concepts, experimental programmes are necessary which address in particular the following phenomena, processes or system behaviour:

- low flow, low pressure regimes;
- reactor core and containment cooling and decay heat removal by natural circulation;
- passive containment cooling systems; and
- nuclear-thermal-hydraulic coupled instabilities.

For advanced PHWRs, similar experimental programmes are required to investigate proposed design improvements, system performance under loss of coolant conditions, and natural circulation in the reactor core and passive containment cooling systems.

For RBMK reactors, additional separate effects tests and integral test programmes have to be carried out to provide the required database for model development and code validation to bring the situation into line with that for other reactor designs. Such activities have to be seen in the context of the lifetime of these reactors.

The existing experimental facilities are capable of performing the necessary investigations, although some upgrading of components and instrumentation might be necessary. Table 2.1 relates important facilities to the research needs.

However, important facilities were closed while this report was written (BETHSY and MARVIKEN), others are threatened by lack of financial support in the future. To ensure industry's support which is vital for some facilities, future programmes must address their issues, e.g. power upgrades or optimised procedures for operational transients.

Industry sponsored research and parts of the national public research are tied up with specifics of a plant type or specific procedures. National programmes, therefore, are more likely to support existing integral test facilities representing such a plant type, rather than a general-purpose facility.

A possible OECD co-operative project, however, should consist in experimental investigations that:

- comprise generic tests of common interest; and

- support the further development of codes.

Following a proposal at the CSNI meeting in December 1998, a Technical Expert Meeting was organised on 8-9 November 1999 to prepare recommendations for such a project. For preparation of this meeting, the owners of PANDA, PKL and SPES were asked to submit draft proposals showing their capabilities to carry out parts of a common programme and offering specific tests to the community.

This procedure could result in a distribution of tasks over several facilities, all contributing to a common goal. This approach has several advantages:

- Carrying out an OECD-recommended international programme should be helpful for facility owners to acquire national support.

- EU members could finance parts of the programme through the Commission's 5th Framework Programme on Nuclear Fission.

- The scientific value of a distributed programme is higher since facility specific and scale specific effects will be partly compensated. This is in line with the validation strategy of computer codes where a variety of facilities are considered.

It is therefore recommended to establish an OECD co-operative experimental programme in thermal-hydraulics that addresses common research needs and makes use of existing facilities.

Code Development

The near-term needs in relation to code development can be summarised as follows (see also Table 2.2):

- Application of best estimate codes entails the need for quantification of uncertainties of important output parameters; these uncertainties result from:
  - limitations in physical models, e.g. lack of multi-field models, correlations and parameters not sufficiently supported by experimental data;
  - simplifications in plant models, e.g. representation of multi-dimensional flows by a network of one-dimensional flows;
  - assumptions on initial and boundary conditions of the accident; and
  - code user effects.

A recent study [11] demonstrated the availability of various methods for the quantification of uncertainties. An urgent need is recognised to increase the effort towards harmonisation and practical applicability of these methods.

- In relation to scaling effects, the simulation of real-scale plant behaviour with models and system codes validated on experimental data from small or reduced-scale test facilities may suffer from scaling distortions whose effects have to be assessed; model validation based on large scale tests will become more important in this respect. For PWRs of current design, the multinational 2-D/3-D Project (Germany, Japan, and USA) [5] has provided valuable results. For new reactor concepts, additional large scale tests may be required.

- The development of a new generation of thermal-hydraulic system codes, based on advanced multi-dimensional/a multi-fluid model requires a more detailed insight into the physical phenomena (e.g. transitions of flow regime). Separate-effects tests with advanced measurement techniques (possibly probing the flows down to the level of microscales) will be required to provide data needed for such models. The advances in measurement techniques were recently evaluated [10].

2.2.2 Longer-term needs

Availability and maintenance of research capability and staff competence are indispensable for the continuing safe operation of the present generation of reactors and for developing the next generation of reactors with advanced safety features. As the pool of international expertise ages, long-term programmes are needed to keep safety analysis tools up-to-date with the best available scientific information. Ageing of analytical tools has been recognised as of regulatory concern [8]. Code and model developers should be aware of developments in non-nuclear related fields and incorporate new ideas and data when available. Eventually, empirical correlations should be replaced by mechanistically based models. Detailed, advanced experimental work will be needed to provide the information needed for this.

In order to achieve these goals, both analytical and experimental capabilities have to be kept available. For thermal-hydraulics, this means specifically the continuous validation and user experience for computer codes and the operation of sufficiently large test facilities. It is not anticipated that computer codes will replace experiments in the foreseeable future.

It is important to recognise the obligations of both industry and public. As the recent CNRA report on New Future Nuclear Regulatory Challenges [8] stipulates with respect to research:

> "monitor the availability of research facilities and associated staff to ensure that an appropriate level of research is sponsored by the nuclear industry while arranging for regulators to have access to research in a way which maintains their independence."

For reactors of Western design, active operation of at least one large integral test facility for PWRs, one for PHWRs and one for BWRs is required to support the resolution of new safety questions that may arise. In addition, the experimental capability should exist in each world region (America, Asia, and Europe) to respond quickly to events with safety implications occurring in actual power plants. Indeed, learning from past experience, the TMI accident occurred at a time when the SEMISCALE facility was ready and could immediately run a series of small leak tests that helped clarify key issues. Looking into the future, incidents that can happen any time may require experimental investigations to answer questions such as:

- Under which conditions would this incident develop into an accident?

- Are additional preventive measures necessary in similar plant?

Further scientific-technical problems requiring continuation of experimental and analytical research activities are related to the following:

- The general need to speed up the performance of system codes; this is becoming increasingly possible by the introduction of advanced computing technologies. Real-time, or high-speed computations will be needed for real-time simulation of transients.

- The analysis of certain classes of transients and accidents (e.g. coupled neutronic thermal-hydraulic instabilities in BWRs, boron dilution in PWRs, power pulse and void reactivity in PHWRs, ATWS transients, etc.) is characterised by interaction between reactor physics and fluid dynamics and requires system codes based on coupling of 3-D-neutronics models and system thermal-hydraulics models; the need for improved multi-dimensional reactor kinetics is addressed in Chapter 3.2 on Reactor Physics. There is also the perceived need to couple thermal-hydraulic codes (that have traditionally been limited in their application to design basis accident (DBA) like situations) with severe accident analysis codes.

- The vast improvements in computing speed will make multi-dimensional calculations of primary system and containment behaviour increasingly possible. This is also being addressed by the NEA Nuclear Science Committee Advanced Computing Group. The analysis of novel reactor concepts reveals the need to perform (at least locally) detailed, multi-dimensional calculations with system codes. The need for such multi-dimensional computing capability was addressed by the November 1996 CSNI Workshop on Transient Thermal-Hydraulics and Neutronic Codes [9].

- The adaptation and validation of codes existing in Western countries for the specific design characteristics of VVER and RBMK reactors; these activities call for long term close collaboration between Western and Eastern organisations since differences between the reactors give rise to specific types of accident sequences and the database addressing specific VVER design features has to be extended, especially for accident management procedure development. For this reason, the existing CSNI Code Validation Matrix should be appropriately supplemented to allow code validation for VVER specific phenomena.

- Any emerging new safety issues that might ask for additional confirmatory tests.

Finally, scientific and technological know-how and technical skills cannot appropriately be preserved only in documents and data banks. Effective preservation of personal skills can only be achieved through active research programmes with ambitious goals, clear targets and secure funding. Motivation of the personnel involved depends on the definition of challenging tasks. At present, the following can be identified as being of the type required to meet these challenges:

- The development of a new generation of codes with physically well founded models derived from separate-effects tests with advanced instrumentation (capable of probing the microscales of the flow) and exploiting the fast growing computer capabilities; such an effort could provide multi-dimensional thermal-hydraulic, neutron kinetic and Balance of Plant (BOP) models in an integrated simulation tool.

- An ambitious goal is the linkage of faster than real time simulators to power plants with the ability to contribute to the prediction of the further course of an incident, or of the consequences of an operational decision by incorporation of information from the actual plant conditions. A first step would be the linkage of an integral test facility with such a simulator. Such capability could possibly assist in accident-management decision making, for example, to help assess several alternative action paths. The Halden CAMS project is probing some aspects of this problem.

2.3 Summary and Recommendations

There is a real danger that Member countries will discontinue operation of existing large-scale facilities or code development efforts needed for the resolution of present safety issues and for maintaining competence in the future. In view of this situation, the following is recommended.

2.3.1 Facilities potentially interesting for international collaboration

Experimental programmes are needed to reduce uncertainties regarding phenomena, processes and system behaviour and extend the database for code validation of the currently used system codes and for possible future codes. Table 2.1 shows the most important of these facilities in relation to the research needs. Some of these facilities are potentially interesting for international collaboration. A variety of experimental facilities have to be kept in operation, because:

- Each of these facilities has at least one important unique feature; loss of that feature cannot be compensated by other existing facilities.

- Code validation strategy is based on a matrix covering a variety of phenomena, parameter ranges and scales that cannot be covered by a single facility.

- Large experimental facilities with their research environment are centres of competence; maintaining this competence at several locations is necessary.

While this report was written, important facilities came to the end of their programme (BETHSY, MARVIKEN). Most of the remaining programmes will be completed in the next two to five years, and the facilities are threatened by lack of financial support in the future. To maintain an adequate level of experimental research requires keeping the majority of these facilities operating. A common OECD co-operative experimental programme could substantially support this goal.

2.3.2 Cost information for potential international facilities

The annual operating costs for large thermal-hydraulic experimental facilities vary depending on the size, pressure range and complexity of the facility and the number of tests to be performed. Typical cost estimates range from 350 000 United States dollars (USD) /a for the 1:2000 scale VVER model PMK, over USD 500 000 /a for the 1:400 scale passive BWR model PUMA, up to over USD 1 000 000 /a for a 1:100 scale PWR model, e.g. PKL. These cost estimates reveal that:

- exploiting existing facilities for the investigation of current and possibly arising new safety questions is much less expensive than the construction of any new large facility;

- experimental research is a cost-effective means to maintain and enhance capabilities; and

- maintaining a variety of facilities is affordable. This variety gives more trust to code validation, accounts for different reactor designs and provides experimental competence in different locations.

2.3.3 Specific recommendations

Experimental research capabilities should be maintained almost at the present level. Member countries cannot rely on the survival of certain key facilities elsewhere.

Member countries operating major facilities should be encouraged to offer specific tests as contributions to the common programme. In return, they would have access to results obtained abroad. Being part of a consistent international programme should strengthen the facilities position in obtaining national or EC support.

Recommendation 2.1: Three major facilities have been identified that are threatened by closure in the next two years and that should be of primary concern to OECD Member countries. In priority order these facilities are:

- PANDA: flexible large-scale facility for 3-D effects, multi-compartment containment behaviour and passive heat removal. Integral system behaviour investigations.

- PKL: 4 loop facility in 1:145 scale, hot and cold leg ECC with 8 accumulators, for boron dilution accidents, accidents under shut-down conditions, and preventive accident management (AM) procedures.
- SPES: 3 loop full pressure integral facility for flow regimes and system behaviour of new design reactors, optimisation of EOP and AM procedures.

(BETHSY was deleted from the original list because it will no longer be available and the LSTF was not considered to be at risk when the list was drawn up.)

It is recommended to establish an OECD co-operative experimental programme in thermal-hydraulics that addresses common research needs and makes use of these unique facilities.

The US experimental facilities APEX and PUMA are major contributors to the design certification of the advanced light water reactors. Similarly, Canada's RD-14M thermal hydraulics facility supports both current and advanced CANDU designs. Although not under immediate threat of closure, the availability of these facilities in the future should be of concern and monitored.

Recommendation 2.2: Monitor the status of APEX, PUMA and RD-14M to ensure their future availability (or timely replacement, as may be required to support future advanced reactor designs). This should apply equally to all facilities mentioned in Table 2.1.

2.3.4 Other needs for international collaboration

Best-estimate analyses are increasingly used in licensing. This entails the need for quantification of uncertainties. Several methodologies exist for this purpose but practical applications are few.

Recommendation 2.3: Harmonisation of the various methods for quantifying the uncertainties should be promoted by OECD/NEA. Because TH codes are validated using data from different scale facilities, extrapolation to full reactor scale requires a consistent approach.

Recognising the needs for improving calculational capabilities for present reactors and for modelling of phenomena in advanced reactors, USA, France [13] and Germany are clarifying their positions towards a future code. NEA Member countries should develop a common position about needs for and desired features of a possible future code.

Recommendation 2.4: A coordinated approach to development and validation of future thermal-hydraulic codes should be promoted by OECD/NEA.

Data from past experimental programmes are being lost due to closure of the facility, retirement of specialists and lack of funding for maintaining the data and the associated documentation in a form amenable to today's computer software. The necessary database for the validation of present and future computer codes is eroding. Once data from large-scale experiments is lost it will be impossible to recover it in the future.

Recommendation 2.5: Essential experimental data from previous programmes should be maintained for code validation. We recommend that the NEA Data Bank be made the focus for an international project to preserve this important data.

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Table 2.1. **Research needs and experimental facilities**

Research need	Test facility (country)	Special features	Annual costs USD	Programme completed
Current LWRs • multiple system failures/preventive accident management • abnormal start-up/cooldown • low power and shutdown • pressurised thermal shock • ATWS • containment response to TH in cooling system, PSS	BETHSY (France)	full height 1:100 volume scale integral PWR full pressure		1999
	LSTF (Japan)	full height 1:50 volume scale integral PWR		ongoing
	PKL (Germany)	full height 1:145 volume scale integral PWR, 4 symmetrical loops, hot and cold leg ECC, boron concentration measurement	1 100 000	2001
	UMCP (USA)	1/5 height 1:50 volume scale upper plenum model of B&W PWR	250 000	ongoing
	Marviken (Sweden)	nearly full scale conditions, representative geometry	550 000	moth-balled
Advanced LWRs • passive decay heat removal and containment cooling • passive core cooling • coupled instabilities • boron transport	APEX (USA)	¼ height 1:192 volume scale AP600 model 2x4 U-tube configuration	600 000	ongoing
	PANDA (Switzerland)	full height large scale advanced BWR model with passive safety components	1 000 000	
	PUMA (USA)	¼ height 1:400 volume scale SBWR model with passive safety components, key components of operating BWRs	500 000	ongoing
	SPES (Italy)	full height, 1:400 volume scale AP600 model, full pressure, and full power	650 000	
	KMS (Russia)	1:5 volume scale model of VVER-640, coolant circuit and containment		under construction

Research need	Test facility (country)	Special features	Annual costs USD	Programme completed
PHWR (CANDU) • ECI performance • natural circulation • shut-down cooling	RD-14M (Canada, at AECL-Whiteshell Laboratories)	full elevation, scaled facility with 10 channels in two passes. 10 MW in electric fuel element simulators. 1:100 in volume scale	1 500 000	RD-14M (Canada, at AECL-Whiteshell Laboratories)
VVER • gaps in validation matrix • preventive accident management	PACTEL (Finland)	1:300 volume scale multi-loop model of VVER-440	400 000	
	PMK (Hungary)	1:2000 volume scale multi-loop model of VVER-440	400 000	
	PSB (Russia)	full height VVER-1000 model		started 1999
	KS-1-VVER (Russia)	VVER core model.	250 000	

Table 2.2. **Research needs and computer codes**

Research need	Relevant programme	Comments
Computer codes • model development • validation • real-time simulation	APROS (Finland) ATHLET (Germany) CATHARE (France) CATHENA (Canada) RELAP5 (USA) TRAC (USA)	In progress: • model improvements • completion of validation • graphical user interface • speed-up of calculations Action needed: Consider needs and features of future code
Code uncertainties • unverified correlations • scaling effects	International Standard Problems, Benchmarks	Harmonisation of methods needed
Code user effects • nodalisation • user options • one-dimensional model limitations	International Standard Problems	In progress: • user guidance Action needed: • develop 3-D-models • benchmark with CFD-codes

Table 2.3. **Template for Thermal-Hydraulics**

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 2	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	Action needed?
			Current LWRs		
Large PWR integral	• Multiple system failures	M	Yes[4]	Yes	No[3]
Test facility in each world region	• Low power and shut-down states mid-loop operation	H	No[a]	Yes	Yes[1]
	• Natural circulation with or w/o non-condensable gases	M	No[5]	Yes	Yes[1]
	• Boron dilution and mixing	H	No	Yes	Yes[1]
	• ATWS	M	Yes[b]	Yes	No
	• Pressurised thermal shock	M	No[6]	No	No
	• Preventive accident management	H	Yes[4]	No	No[3]
	• Response to plant incidents	M	No	No	No[3]

Chapter 2	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	Action needed?
Current LWRs cont'd					
Large BWR test facility	• Containment response to TH in cooling system	H	No	Yes	Yes[1]
	• Pressure suppression system response	M	Yes[6]	No	No
	• ATWS	H	Yes[2]	Yes	No
Advanced LWRs					
Test facility with passive components and large volumes	• System interaction among components, instabilities	H	No	Yes	Yes[1]
	• Stratification in large pipes and tanks (3-D-effects)	H	No	Yes	Yes[1]
	• Passive containment cooling	H	No	Yes	No[2]
	• Gas stratification, mixing, venting	H	No	Yes	Yes[1]
Integral test facility for coupled phenomena	• Passive decay heat removal	H	No	No[2]	No[2]
	• Natural circulation at low pressure	M	No[5]	Yes	Yes[1]
PHWRs (CANDU)					
Integral test facility	• ECI performance	H	Yes[4]	No	No
	• Natural circulation	H	No[5]	Yes	Yes[1]
	• System Interaction among components	H	No	Yes	Yes[1]
	• Passive containment and moderator cooling	H	No	Yes	No[3]
	• Channel voiding	H	No[2]	Yes	No[3]
	• Shut-down cooling	H	No	Yes	No[2]

Chapter 2	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	Action needed?
VVERs					
Integral test facility	• System effects with horizontal SGs	H	Yes[4]	Yes	No
	• Natural circulation in VVER loop configuration	M	No[5]	Yes	Yes[1,2]
	• Preventive accident management	H	No	Yes	No[3]

a. Horizontal phase separation is not fully validated in the codes.

b. New needs may arise in future for modified operating conditions.

1. See Recommendation Nos. 2.1 and 2.2.
2. Design specific aspects involved.
3. Main responsibility of Member countries.
4. Additional information needed if thermal-hydraulic behaviour is coupled to other processes, e.g. neutron kinetics, containment response etc.
5. Two-phase and low flow to be investigated.
6. Results of component tests to be linked to system behaviour.

Chapter 3

FUEL AND REACTOR PHYSICS

Although the areas are closely linked, the research requirements for fuel development and reactor physics are quite different. Hence in this chapter they are presented in two independent sections.

3.1 Fuel

3.1.1 Background

The concerns of SESAR in the area of fuel are mostly associated with the behaviour of fuel at high burnup [l]. Although the behaviour of high burnup fuel under operating conditions is relatively well understood, most of the current understanding of fuel behaviour has been acquired from experiments on low burnup or fresh fuel. In the meantime, there has been a general trend towards higher burnups with the assembly average discharge burnup in LWRs now approaching 55 GWd/tU. It has therefore become necessary to re-visit the various fuel damage limits for high burnup fuel. These limits are used to determine clad integrity, fission product releases, and to ensure core coolability. Also, it is necessary to upgrade and validate fuel behaviour analysis codes to account for changes in fuel properties and physical phenomena observed at high burnups. These codes are used to determine the initial conditions for transient analysis.

MOX fuels utilisation in LWRs has already begun in some countries and will be expanded in the future with the aims of promoting the fuel cycle and for plutonium burning. In particular, in the USA, a decision on the use of excess weapons grade plutonium as MOX fuel in commercial nuclear power plants was made in January 1997. The fuel would be utilised in a once through cycle. MOX fuel behaviour is of concern especially for high burnup ranges and is dependent on fabrication procedures and Pu content. Also, Canada is assessing the feasibility of burning weapons grade plutonium (as MOX fuel) from the USA and Russia in CANDU reactors via demonstration irradiations in the NRU reactor at AECL-Chalk River Laboratories (the PARALLEX programme). Recent work on Reactivity Initiated Accidents (RIAs) in France (CABRI) and Japan (NSRR) indicates a reduction in failure threshold at high burnup which is related to changes in clad properties with burnup due to oxidation, hydrogen uptake and irradiation damage. This underpins the need for research to establish fuel damage limits under these conditions and for clarifying the consequences of fuel failure.

In Russia, a new type of zirconium alloy is being developed for high burnup use, since significant irradiation growth has been observed at high burnup for current fuel cladding material (Zr-1%Nb) of VVER.

The main area of concern identified is fuel damage limits at high burnup. Therefore priority should be given to establishing limits that cover the full range of both burnup and possible transients (reactivity insertion, LOCA, etc.). The limits should ensure fuel integrity based on the appropriate parameters (e.g. enthalpy, cladding oxidation).

A number of specific areas of research are identified for which further discussion is necessary:

- Cladding embrittlement at high burnup.
- Fission gas release.
- Fuel dispersal upon cladding failure.

A summary of the research needs and the programmes addressing them is given in Table 3.1.

3.1.2 Needs and Challenges

3.1.2.1 Steady-State Fuel Behaviour

The OECD Halden Reactor Project is generating data on high burnup fuel and MOX. This data is required to improve and validate fuel behaviour analysis codes at high burnup. Data from the Halden Project was used as part of the IAEA's FUMEX programme on the improvement of fuel behaviour analysis codes [2]. The Halden Project is expected to continue to provide the facilities for the generation of high burnup fuel properties data in the future.

International programmes on steady state and slow transient fuel behaviour for MOX-fuel have been conducted for more than 10 years in the Belgian high flux material testing reactor BR-2. Fuel materials irradiations have also taken place in the HFR reactor and the PIE facilities (LSO) at Petten. Re-irradiation tests of fuels from commercial reactors are conducted under steady state and transient conditions in a BOCA capsule in the JMTR reactor at JAERI in Japan.

AECL has been conducting high burnup irradiations of CANDU fuel in the NRU reactor at Chalk River. There are on-going programmes on both extended burnup and advanced fuels (e.g. MOX, low void reactivity fuels, natural and U-235 enriched CANFLEX fuel bundles). The NRU reactor is reaching the end of its life, but AECL is planning to replace NRU with a MAPLE type research reactor (i.e. the Canadian Neutron Facility), which will facilitate continued fuel development activities in Canada.

Other reactors are available in several countries for steady-state irradiations as indicated in Table 3.1 at the end of this chapter.

Several countries continue to develop their fuel behaviour analysis codes to model high burnup and advanced fuels.

3.1.2.2 Transient Fuel Behaviour

There are two active projects investigating the area of Reactivity Insertion Accidents (RIAs): at the NSRR reactor in Japan [5] and at the CABRI reactor in France [4]. Water capsule tests are conducted in the NSRR while the CABRI experiments are conducted in a sodium loop.

The NSRR has been extensively used for research on the behaviour of LWR fuels under transient conditions, including irradiated fuel at burnups of up to 50 GWd/tU. Tests with irradiated fuel are limited to cooling conditions of stagnant water at atmospheric pressure. Tests with high temperatures and pressures have been conducted on un-irradiated fuel. The reactor can be operated under two conditions; pulse mode (a few ms pulse) or power ramp mode (a few tens of seconds ramp). The current NSRR program is focusing on the behaviour of both PWR and BWR irradiated fuel under RIA conditions and is expected to continue for the next 10 years.

The CABRI facility was initially constructed to investigate the fuel behaviour under rapid power transient conditions in Fast Breeder Reactors. One of the current series of experiments is focusing on the behaviour of high burnup LWR rods under power pulse conditions. The tests are conducted in flowing sodium rather than water. However, this should not affect the fuel response in the early stages of the transient initiated from the hot condition. The power pulse width in these experiments has varied between 10 and 100 ms. The current programme under sodium cooling conditions is mostly complete, and the next programme with water loop is under planning.

Both the USA and Japan are making modifications to the fuel behaviour analysis codes FRAP-T6 and its successor FRAPTRAN to take into account high burnup effects on fuel behaviour in RIA conditions [3,6]. In Japan, the particular action is to include a cladding deformation model due to fission gas loading. In addition the USNRC is also developing a steady state fuel behaviour code FRAPCON-3 for high burnup fuels. Both countries and Finland intend to collaborate on these developments. Basic models to describe material properties at extended burnup are also being implemented in the transient code. The French programme takes a similar approach including international collaboration with computer codes such as TOSURA (steady state) and SCANAIR (transient) under development [4]. The key phenomena to be modelled are currently thought to be enhanced cladding deformation and fission gas release as observed in the pulse irradiation tests. Cladding ductility and corrosion (including hydrogen pickup) should be well predicted by the steady state codes for pre-transient conditions. Fuel degradation limits under reflooding and quenching conditions are also needed.

Many vendors and utilities are developing and performing reactor kinetic calculations of their NPPs with 3-D reactor kinetics codes to have better estimates of fuel enthalpies in high burnup fuels under accident conditions.

Pulse irradiation tests with fresh MOX fuels in the NSRR and SPERT programmes indicated no significant difference in the fuel behaviour from UO_2 fuel even with artificial Pu spots, but effects of irradiation may be different especially for high burnup conditions. Recently tests with irradiated MOX fuel have been initiated in the CABRI and NSRR programmes. Further experimental studies including pulse irradiation are needed for utilisation of MOX fuels in high burnup ranges.

Pulse irradiation tests under typical cooling conditions at slower power transients are also needed and are being performed, for example, in the BR-2. The new EU project MICROMOX is studying the impact of MOX fuel microstructure on fission gas release in transient conditions at high burnup (60 GWD/tonne) in the HFR Petten reactor.

The Blowdown Test Facility (BTF) in AECL's NRU reactor [9] was one of the last remaining facilities investigating fuel behaviour under LOCA conditions. The current series of experiments on LOCA and LOECC behaviour of normal burnup CANDU fuels have been completed, and no more experiments are planned for this facility. However, the CNF reactor, which AECL plans to build to replace NRU when it is shutdown later this decade, has been designed to be able to accommodate a BTF-like loop for in-reactor severe fuel damage tests and fission product release and transport experiments, should future programme requirements indicate the need for additional work in this area.

The safety criteria for LOCA were defined to ensure that the reactor core would remain coolable. Since the time of LOCA experiments, which were largely conducted with fresh fuel, changes in fuel design, the introduction of new cladding materials and in particular the move to high burnup have generated a need to re-examine these criteria and to verify their continued validity. For this purpose, hot cell programmes concentrating on embrittlement of high burnup cladding, so-called separate effect tests, have been initiated in some hot laboratories. They are the hot cell experiments at ANL[8], EdF[7] and at JAERI. A similar experiment is also planned at KAERI. Integral tests devised to demonstrate the extent to which high burnup affects the overall fuel behaviour in LOCA conditions are planned at the Halden Project. An experimental loop for integrated LOCA tests on irradiated fuel bundles is under study at the BR2 reactor in Belgium.

3.1.2.3 Additional Efforts Needed

Neither of the current facilities (NSRR and CABRI) for studying RIAs have typical cooling conditions (both temperature and pressure). In addition, power transients in the CANDU reactor are much slower than those in these research reactors. To resolve some of the outstanding issues associated with RIAs, the fuel damage limits need to be investigated under more realistic conditions. Plans are being considered to simulate more realistic cooling conditions with experiments at CABRI. In France IPSN is considering a test programme with a water-cooled loop with support from the OECD Member countries. In the NSRR program, the development of a high pressure and high temperature capsule to realise more realistic test conditions is also underway. In the past, the rod ejection event has been taken as the limiting RIA. However, there are lower speed transients such as boron dilution and power oscillations in BWRs that may require future work. This is also reflected in Chapter 2, which covers the thermal-hydraulic aspects.

Studies are needed to define the limits of general degradation of fuels, especially under reflooding and quenching conditions following LOCA. The value of in-pile experiments on the behaviour of high burnup rods and bundles under degraded cooling also need to be investigated. There is also the need to determine the operational limits of new fuel designs. Whilst facilities (NRU in Canada, BR-2 in Belgium and HBWR at Halden in Norway) currently exist that have the capability to conduct such studies, it is not clear how much longer such facilities will be available. For example, it is unlikely that NRU will be operated beyond 2005. However BR-2 and Halden will remain available at least until 2010 as the former has undergone refurbishment and the later obtained a license extension.

3.1.2.4 Longer-term Needs

Testing of new or advanced fuels under normal and off-normal operating conditions is necessary to confirm that there is no new safety issue. Facilities for the irradiation of prototype fuels are therefore required to demonstrate that the fuels are safe. Irradiation facilities must be available to determine fuel operational limits and the behaviour of these fuels under off-normal conditions. The current generation of research reactors are nearing their end-of-life and a concerted international effort may be needed to ensure new research reactors become available in a timely fashion. Currently there are plans in both Canada and France for large new research reactors which may provide timely replacements for current reactors. In addition, major refurbishment programmes such as that at BR-2 in Belgium will keep some reactors available. The dedicated safety research reactors (NSRR, CABRI) should also be maintained to conduct in-pile experiments under rapid power transient conditions.

Since fuel behaviour analysis codes for normal operating conditions are used to define the initial conditions for accident analysis, they should be treated as safety analysis tools. This means that they must be verified and validated in the same manner as that for other safety codes and in turn this means having access to data from commercially irradiated fuel. The capability to continue to develop these codes for both high burnup fuel and for advanced or improved fuels must be maintained. An action in this respect has been undertaken in the framework of the activities of the NEA Nuclear Science Committee (NSC) through the establishment of an International Database (IFPE) of fuel performance parameters under normal operating conditions allowing the improvement of predictive fuel behaviour models. In addition, we believe that these models should themselves be subject to more intensive collaboration along the lines already followed in the thermal-hydraulics area. The continued development of the codes may also require further research on the physical properties of both fuel and cladding.

Hot cell facilities and Post-Irradiation Examination expertise are required to support any in-pile experiments, together with implementation of up-to-date apparatus such as for thermal property measurements, element analysis, etc. Skilled personnel trained in hot-cell operation and nuclear materials science are needed, along with people skilled in in-pile instrumentation.

Recent developments in sophisticated poolside examination facilities should also be included in the forward thinking since they offer a quick and low cost source of information which could be useful for code validation.

3.1.3 Summary and Recommendations

i) Facilities potentially interesting for international collaboration

The main area of concern identified as short-term needs is the establishment of a technical database on both UO_2 and MOX fuel behaviour at high burnup, particularly establishing the safety criteria for possible transients in current LWRs. In addition, specific requirements may emerge since the MOX fuel behaviour is dependent on fabrication procedure and Pu content.

As for long-term needs, design and utilisation of fuels will keep evolving for better economy, different fuel cycles or responding to different core design and operation of future reactors. In particular, the use of inert-matrix fuels, which may well be a workable technology, could require significant R&D effort.

The needs for fuel irradiation and in-pile testing for unusual events and for simulated accident conditions are not expected to decrease in the foreseeable future. Therefore, in order to meet these needs, research reactors such as CABRI, NSRR, HBWR and BR-2, and the hot cell facilities with the capabilities necessary for the maintenance and utilisation of the research reactors, should be maintained.

ii) Cost information for potential international facilities

For very large facilities mentioned above, although the basic funding for keeping the facility may have to depend on national needs and support, international support by direct funding, by providing experts or by other means of co-operation for specific experimental programmes is now becoming more important. Multilateral international collaboration through the NEA should be strengthened to ensure the effective use of the limited facilities and expertise. Activities to identify the potential for common research subjects, to assign or arrange the tasks for the facilities so identified, and, to co-ordinate any necessary sharing of funding, will promote co-operation and provide detailed justification for the proposed work.

iii) Specific recommendations

Proprietary information on fuel design and utilisation often becomes a big obstacle to international co-operation. This often causes it to be restricted to bilateral and exclusive arrangements. More efforts need to be focused towards realising a clear definition of the common tasks, disclosure and easy access to the information, and active and general exchange of the output from the contributing organisations.

iv) Other possible forms of international collaboration

Some Member countries have been using MOX fuel in their reactors for a significant operational time period and have subsequently generated operational data. The NEA should try to develop international co-operative activities with vendors, utilities and research organisations to allow access to relevant information.

Table 3.1. **Fuel research needs**

Research need	Relevant programme	Comments
High burnup/MOX fuel performance	IAEAs FUMEX MICROMOX (EU)	Considerable data has been obtained from fuel irradiated in commercial power stations.
Steady state	OECD Halden (UO2 and MOX fuel) BR-2 (Belgium) (MOX fuel) NRU (Canada) (MOX; natural and enriched CANDU fuel) OSIRIS (France)	
Slow power transient	OECD Halden BR-2 (Belgium) JMTR (Japan) R2 (Sweden) OSIRIS (France)	
RIA[1]	NSRR (Japan) CABRI (France)	Further test at higher burnups and of irradiated MOX fuels also needed.
Fuel materials irradiation	HFR (Netherlands) NRU (Canada) ATR (USA) OECD Halden JMTR (Japan) OSIRIS (France) BR-2 (Belgium)	
LOCA	BTF in NRU (Canada) ANL hot cell test (USA) EdF hot cell test (France) JAERI Hot cell test (Japan)	BTF experimental programme completed and facility closed.
New fuel designs; operational limits	NRU (Canada) BR-2 (Belgium) OECD Halden	
New fuel designs; transients		Same as high burnup and MOX

1. This could include rod-ejection, ATWS or other RIA.

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Template for Chapter 3.1 – Fuel

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 3.1: Fuel	**Future needs and challenges**	**Safety significance**	**Sufficient knowledge available?**	**FAP essential?**	**Action needed?**
Steady state fuel behaviour	• Performance of higher burnup fuel.	M	No	Yes	No
	• Performance of higher burnup MOX fuel.	M	No	Yes	No
	• Performance of new designs of fuel.	M	No	Yes	No
Transient behaviour of fuel	• RIA studies on higher burnup UO_2 fuel.	H	No	Yes	No
	• RIA studies on higher burnup MOX fuel.	H	No	Yes	No
	• RIA studies on new designs of fuel.	H[2]	No	Yes	No
	• LOCA studies on higher burnup UO_2 fuel.	H	No	Yes	No
	• LOCA studies on new designs of fuel.	H	No	Yes	No

2. YES if the new design introduces new uncertainties on the safety-relevant transient behaviour.

3.2 Reactor Physics

3.2.1 Background

Neither of the documents produced so far by SESAR makes extended reference to any needs in the general field of reactor physics. There is passing reference to the requirement for a capability for coupled reactor physics and thermal-hydraulic analysis of plant transients, for criticality analysis, and for shielding analysis. No particular outstanding needs were identified, particularly for existing reactor systems. Several facilities have been dismantled during the last decade. Current facilities appear, however, to be adequate world-wide which could respond to the needs of existing systems or for proposed advanced systems, although the number of active programmes of experiments in zero energy reactors is small. Phasing out of additional facilities would lead to a shortage of expertise and lack of independent measurement and review capabilities.

For many years within the NEA, matters concerning reactor physics fell under the NEA Committee on Reactor Physics (NEACRP) and the closely related matters concerning nuclear data fell under the NEA Nuclear Data Committee (NEANDC) and the NEA Data Bank. In recent times all these activities have been combined, with some new ones, under the NEA Nuclear Science Committee (NSC). The NSC has been concerned with loss of expertise and facilities for some time, and has been seeking to promote international collaboration to preserve them. This has been a particular concern for facilities designed to measure differential nuclear data, but has also extended to those used for measuring integral data of significance for various reactor types, as well as those for obtaining data relevant to criticality safety studies.

This matter is treated in different working parties sponsored by NSC:

WPMA: Working Party on Nuclear Data Measurements.

WPEC: Working Party on Nuclear Data Evaluation Co-operation.

WPPR: Working Party on Physics of Plutonium Recycling and Innovative Fuel Cycles.

WPNCS: Working Party on Nuclear Criticality Safety.

Reference should be made to the various documents produced by the NSC on these matters for a fuller picture of the situation [1,2]. Some idea of the range of near term programmes can be obtained by considering activities that are ongoing and sponsored by the NSC. In the specific area of reactor physics these include:

- a working party on physics issues in plutonium recycling and innovative fuel cycles;
- reactor physics benchmark studies on “Core Transients in PWRs”, “BWR Stability-Comparison with Operational Data” and “Coupled 3-D Neutronics with thermal-hydraulics (Main Steam Line Break)”;
- a benchmark on “Power Distribution Within Assemblies”;

- reactor diagnostics related benchmarks on "Loose Parts Monitoring (noise analysis for detecting changes in the core)" and "Neural Networks"; and

- 3-D pressure vessel dosimetry benchmark based on VENUS-3 data.

In the area of criticality studies the NEA Nuclear Science Committee has established a Working Party on Nuclear Criticality Safety covering a wide-ranging set of issues. For example the International Criticality Safety Benchmark Project Working Group was formed to identify and evaluate a comprehensive set of benchmark critical experiments data, to verify the data, review them, compile them into a standard form and to reinterpret them with standard criticality safety neutronics codes. Both experimental data and modelling data are integrated into a benchmark handbook (ICSBEP) [3]. This handbook contains at the time of the writing of this report experimental data for about 2000 critical configurations. Experiments from sub-critical experiments are also being added. Another task force has started to investigate more definitive, quantitative processes for demonstrating an experimental need relative to a particular application or application area. Methodologies to help identify/justify experimental needs are being analysed and their applicability verified. The data from the handbook will be used for establishing how much of the parameter space is covered by these experiments.

The objectives of this task force include:

a) Compile needs for high-priority experiments, update and reaffirm on a regular basis.

b) Promote the exchange of measurement objectives and experimental planning on a national and international basis.

c) Encourage the development of bilateral and multilateral international programmes.

d) Facilitate co-operative and comparative studies through the sharing of technology and materials.

e) Provide technical reviews of facility capabilities and experimental techniques.

The quantitative methodologies will allow to focus on high priority needs for which no experimental data or poor quality data is available. Through international sharing of facilities, coverage of the high-priority needs with the limited number of available facilities would be ensured.

In the area of shielding a data base of experimental results (SINBAD) is being assembled, again for validation of codes. It should be noted that the CSNI and NSC are collaborating on a review of the adequacy of methods available in reactor radiation dosimetry and radiation induced degradation of reactor components followed by a blind benchmark based on experimental data aimed at verifying the claimed precision attained.

An important area in which reactor physics contributes to safety analyses of transients, at least before the reactor is shut down, is in the provision of reactivity coefficients. These include, in particular, the fuel temperature and coolant density and temperature coefficients, as well as the parameters that define the kinetic response of the core. These are generally well understood for existing reactor types and established fuel cycles. However, the desire to use MOX fuel as an increasingly large fraction of the core loading, and the consideration of using multiple recycling of MOX fuel with the corresponding increase in the fraction of higher Pu isotopes and the required higher fraction of Pu in the fuel, significantly alters the reactivity coefficients and the kinetics parameters. This is being studied in the benchmarks sponsored by the NSC working party on the physics of Pu recycling and innovative fuel cycles and is implemented through specific benchmark exercises. Experimental programmes specifically related to MOX use in current LWRs are conducted in EOLE in France and in the VENUS critical facility in Belgium. Furthermore the NSC set up a task force which will deal with the status and trends of reactor physics fuel performance, and fuel cycle issues related to the disposition of weapon-grade plutonium as mixed oxide fuel.

Another area of safety where reactor physics results play an important part is transient and accident analysis codes. Safety analyses are increasingly being made with 3-D and accident analysis codes; for instance studies on high burnup fuel effects (see 3.1.2). Accurate neutron kinetics calculations are important in analysing Reactivity Insertion Accidents, boron dilution and ATWS type accidents. They are also important in accidents which include the possibility of recriticality after reactor trip (e.g. steam line break). In 3-D analysis as much reactor physics data of actual fuel and core conditions is needed as in fuel management calculations. Therefore, it is important to ensure that accurate reactor physics data under accident relevant conditions, and the methods to evaluate it, are available not only to the fuel vendors and utilities but also to the safety authorities and their research organisations.

The incorporation of full three-dimensional (3-D) modelling of the reactor core into system transient codes allows “best estimate” simulations of interactions between reactor core behaviour and plant dynamics. The progress in computer technology makes development of such coupled computer code systems feasible. Considerable efforts have been made in different countries in this direction. At the NSC Bureau meeting of December 1996 as well as at the CSNI meeting of January 1997 a new OECD Benchmark on a Main Steam Line Break (MSLB) in a Pressurised Water Reactor was approved. This plant transient benchmark is intended:

- to verify the three-dimensional capability of system codes to analyse complex transients with coupled core-plant interactions;
- to fully test the neutronics/thermal-hydraulic coupling;
- to evaluate discrepancies between predictions of coupled codes in best-estimate transient simulations.

A summary of the research needs and the programmes addressing them are given in Table 3.2.1.

3.2.2 Needs and Challenges

3.2.2.1 Short-term needs

Capabilities for existing reactor systems appear to be generally adequate. Needs for advanced systems are being addressed for relatively near-term advances, such as actinide burning and full MOX cores in current LWRs and PHWRs, and extended burnups in PHWRs using slightly enriched uranium fuel. However, in the presence of high burnup fuel, the role of plutonium requires further examination. The needs for safety analysis have not yet been fully defined, but are expected to become clearer as plans for such systems are clarified.

The facilities that are used to provide purely reactor physics data for code validation are the various zero energy reactors available in the world. Those include: VENUS in Belgium, ZED-2 in Canada, EOLE in France, DCA and HCA in Japan and PROTEUS in Switzerland. There are also a number of such facilities in the CEEC which could be used.

Up to now, the programmes conducted in the various zero power reactors for reactor physics code validation are largely based on the requirements arising from national policies with respect to the adopted fuel cycle. However, as mentioned in Section 3.2.1, the desire to use MOX fuel in existing reactors, in even larger fractions than already achieved up to now in those countries which have adopted the Pu recycling option is becoming more general. Safety related aspects such as void coefficient, control rod worth, and kinetic parameters become of general concern within Pu recycling policies. Computer codes for accident analysis are applied to determine safety related consequences of changes in nuclear physics parameters. As the results rely strongly on code capabilities, activities of benchmarking these computer codes should be continued. Collaborations endorsed by CSNI and the NSC, specifically the Working Party on the Physics of Pu recycling and innovative fuel cycles (WPPR) and the Working Party on Nuclear Criticality Safety (WPNCS) could be beneficial for the entire nuclear community.

New issues in criticality safety continue to emerge as spent fuel storage facilities reach the saturation point, fuel enrichments and burnups increase and new types of plutonium-carrying fuels are being developed. New fuel cycles, handling of surplus fissile materials from the weapons programmes and its possible use for civil energy applications make new demands. The new challenges related to the manipulation, transportation and storage of fuel demand further work to improve models predicting behaviour through new experiments, especially where there is a lack of data in the present databases.

With the declining use of and actual loss of experimental facilities and personnel world-wide, the basis for criticality safety limits has been shifting from physical measurements to reliance on complex computer code calculations. Validation efforts for these codes must include some physical measurements, as this is the tie back to reality. Establishments operating criticality safety facilities from the OECD area and outside have been struggling to prevent their permanent closure. Only a very few facilities are still operational today.

3.2.2.2 Long-term Needs

To ensure that the capability to make integral measurements required for computer code validation for advanced reactor systems is maintained, some of the facilities mentioned in Table 3.2.1 must be kept available along with personnel having the skills and knowledge to use them properly. It would be desirable to define international programmes utilising selected facilities involving experimenters from a mixture of interested countries. Unfortunately, militating against this is the trend to regard the data generated from such programmes as being proprietary to the reactor vendor or designer who has paid for the measurement. The more “immediate” the aims of the experimental research are, (e.g. the validation of power distribution calculations as against integral tests for minor actinide data, for example) the stronger is this aspect. Dealing with this fact is one of the main challenges to any strategy for maintaining facilities and expertise through international collaboration.

3.2.3 Summary and Recommendations

Several facilities have been dismantled during the last decade. Current facilities appear, however, to be adequate world-wide which should be able to respond to the needs of existing systems or for proposed advanced systems. However, phasing out additional facilities would lead to a shortage of expertise and lack of independent measurement and review capabilities. The NSC promotes international collaboration through its different working parties[3] to preserve in particular facilities designed to measure differential nuclear data (such as the GELINA Facility, IRMM, Geel, Belgium and the ORELA Facility, ORNL, Oak Ridge, US) as well as those used for measuring integral data of significance for various reactor types and those for obtaining data relevant to criticality studies.

Template 3.2.2 summarises the recommendation for the reactor physics issues. It is felt that no immediate action is needed assuming that the existing FAPs are maintained. In the longer term the status of the facilities should be monitored and, if there is a danger of losing the capability to measure operational or basic nuclear data, action should be recommended.

The efforts of the NSC and its different working parties to promote and establish international collaborations in the general field of reactor physics should be pursued. Support of CSNI is in that respect important.

REFERENCES

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2. "Experts Meeting on Experimental Needs in Criticality Safety", (Annex 1, The Need for International Co-operation in Providing Nuclear Criticality Experiments), NEA/NSC/DOC (95) 17.

3. International Handbook on Evaluated Criticality Safety Benchmark Experiments.

4. Plutonium Systems. II. Highly Enriched Uranium Systems. III. Intermediate and Mixed Enrichment Uranium Systems. IV. Low Enriched Uranium Systems. V. Uranium-233 Systems, VI. Mixed Plutonium Uranium Systems, VII. Special Isotope Systems.

5. Dae Y. Chung (US DoE), J. Blair Briggs, Lori Scott and June Williams (INEL).

6. NEA/NSC/DOC(95)3 (edition September 1998).

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3. WPMA: Working Party on Nuclear Data Measurements.
   WPEC: Working Party on Nuclear Data Evaluation Co-operation.
   WPPR: Working Party on Physics of Plutonium Recycling and Innovative Fuel Cycles.
   WPNCS: Working Party on Nuclear Criticality Safety.

Table 3.2.1. **Reactor physics research needs**

Research need	Relevant programme	Comments
Pu-recycling		
Reactivity coefficients	EOLE (France)	
Kinetic parameters	VENUS (Belgium)	
Power distribution and reactivity effects in assemblies	ZED2 (Canada)	Unique CANDU facility
	PROTEUS (Switzerland)	
Reactor diagnostics	DCA and HCA (Japan)	
	LR-0 (Czech Republic)	VVER
Criticality safety	VALDUC facility (France)	
	EOLE and MINERVE (France)	
	NUCEF and TCA (JAERI, Japan)	
	KUCA (Kyoto, Japan)	
	DCA (JNC, Oazai, Japan)	
	SPH (IPPE, Obninsk, Russia)	
	FKBN-2M and FKBN-M (Russia)	
	RBC-I, ARGUS, IIN (Russia)	
	LACEF, SFSX, ZPPR (USA)	
Differential nuclear data	GELINA (Belgium)	
	ORELA (USA)	

Table 3.2.2. **Template for reactor physics**

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 3.2: Reactor Physics	**Future needs and challenges**	**Safety significance**	**Sufficient knowledge available?**	**FAP essential?**	**Action needed?**
Basic operational data	• Established fuel cycles	M	Yes	No	No
	• Higher Fuel Burnup	M	No	Yes	No
	• Increased MOX utilisation	M	No	Yes	No
	• Multiple Pu recycling	M	No	Yes	No
Basic operational data	• Established fuel cycles	M	Yes	No	No
	• Higher Fuel Burnup	M	No	Yes	No
	• Increased MOX utilisation	M	No	Yes	No
	• Multiple Pu-recycling	M	No	Yes	No

Chapter 4

SEVERE ACCIDENTS

This chapter is divided into three main topic areas: (1) in-vessel phenomena, (2) ex-vessel phenomena and (3) fission products. Each main topic area follows the general format of the other chapters in that there is a background section, a needs and challenges section and a summary and recommendations section. This separation into main topic areas was adopted because of the enormous diversity of subject matter in this area. There are many topics where there is overlap and it is important that the synergies and interfaces are kept in mind.

4.1 In-vessel Phenomena

4.1.1 Background

The high priority research needs discussed in the SESAR/CAF report addressed the following in-vessel phenomena:

Core Degradation and Melt Progression:

- reduce uncertainties in high temperature physical properties of materials;
- modelling steam convection through a damaged core; and
- quantity of molten corium transported to the lower head.

Fuel Coolant Interactions (FCI):

- develop analytic capability to quantify the complete spectrum of FCI events (in-vessel and ex-vessel) and consequences, including:
  - better definition of what constitutes a potentially explosive mixture;
  - better understanding of propagation and energetics;
  - prediction of the generated pulse and load estimates; and
  - understanding of chemical augmentation of FCI.

Debris Interaction with Lower Reactor-Vessel (RV) Head:

- under what conditions might the lower RV head and penetrations fail;
- what is the likely failure mode, location and timing; and

- what conditions of internal and/or external cooling of the RV might prevent failure.

At the time the SESAR/CAF report was written there was general agreement that the existing research facilities and programmes in the areas of fuel-coolant interaction (FCI) and debris interaction with the lower reactor vessel head were sufficient to address the open issues. A consensus was not reached on the need for additional research on core degradation and melt progression. Some countries were of the view that sufficient experimental data and analytical tools exist and that sensitivity studies can be used to bound the various scenarios, while others felt additional work was warranted to reduce uncertainties. Therefore, there was no consensus on the adequacy of the current facilities and programmes in this area. However, in the intervening time period, some facilities have been closed or are scheduled to be closed in the near term. A current status in each area is provided below, followed by a reassessment of facility and programme needs.[1]

4.1.1.1 Core Degradation and Melt Progression – Current Status

The only facilities currently in operation conducting core melt progression experiments for PWRs are the QUENCH facility in Germany, PHEBUS in France and CODEX in Hungary. QUENCH is to investigate the early phase of core melt progression, with emphasis on reflooding and the resulting fragmentation and H_2 generation. It is planned to operate in the near term (~5 years). PHEBUS has three tests remaining, primarily directed toward fission product transport; however, some information on early and late phase melt progression will be obtained, although at a small scale. Work on irradiated fuel is undertaken in PHEBUS. The continued conduct of core melt progression experiments will complement analytical code development and assessment and may be useful in addressing any concerns related to the changing operational characteristics of operating plants (e.g. higher burnup fuels, higher power density).

Studies on the disassembly and collapse of a PHWR core are being addressed in the Core Disassembly Test Facility (CDTF) at AECL. This new facility (a one-fifth scale, multi-channel facility containing 12 simulated fuel bundles per channel) was established because information from analogous LWR programmes are of limited application to CANDU due to the significant differences in reactor design. Information obtained from these experiments, coupled with the associated model development, is providing a good understanding of core degradation and melt progression in a CANDU.

4.1.1.2 Fuel/Coolant Interactions (FCI) – Current Status

Currently there are several facilities in operation investigating FCI. Small-scale facilities (using simulant materials) exist in the US (University of California-Santa Barbara – MAGICO and SIGMA), Japan (ALPHA), Korea (CONVEX), Sweden (MIRA), Germany (Premix), COTELS (Kazakhstan, sponsored by Japan) and France (MICRONIS and TREPAM). Each of these facilities, with the possible exception of ALPHA, is projected to continue operation in the near term or be available, if needed. The FARO and KROTOS were shut down in 1999. The loss of FARO will eliminate large scale experimental data using prototypic material as well as useful data on melt

1. As with all the topics covered by this report, this chapter should be seen as a snapshot of the situation at the time of writing. In the severe accident area, this means that results becoming available from the European Commission's Fourth Framework Programme are not discussed in detail. A recent conference gives a very comprehensive review of this work. (FISA 99. EC Research in Reactor Safety, 29 November – 1 December 1999, European Commission).

quenching and spreading. KROTOS is expected to be relocated to France, where it will again be available for use.

In Canada, a new programme is being established to investigate in-vessel fuel/water interactions in CANDU reactors for accident scenarios (e.g. flow blockage in a fuel channel) that can lead to molten fuel being forcibly ejected from fuel channels into the surrounding D_2O-filled calandria vessel. A new facility, the Molten Fuel Moderator Interaction (MFMI) Facility is currently being designed for this purpose at AECL's Chalk River Laboratories. The experimental programme will investigate the energetics associated with the ejection of up to 25 kg of prototypic molten CANDU corium into the moderator. This programme is expected to be completed by the end of 2005.

4.1.1.3 Debris Interaction with Lower RV Head – Current Status

Currently, work in this area is being conducted in the following places:

- RASPLAV(Russia) – OECD co-operative project to investigate load to the RV lower head and chemical interaction between the RV steel and molten core debris, using prototypic materials.
- COPO2 (Finland) – to investigate heat load to the RV using water as a simulant.
- SNL-LHF (USA) – OECD co-operative project to investigate the mechanical behaviour failure modes of the RV lower head under high temperature pressurised conditions.
- BALI (France) – to investigate the heat load to the RV using water as a simulant.
- ALPHA (Japan – to investigate in-vessel debris coolability.
- RIT FOREVER (Sweden) – small scale using prototypic materials for ex-vessel cooling strategies.
- SONATA (Korea) – investigating interactions with the lower head. Currently using simulant materials.
- RUPTHER (France) – investigating mechanical behaviour (creep rupture) under accident conditions.

These projects are scheduled to complete their planned series of tests.

4.1.2 Needs and Challenges

Each of the areas described in 4.1.1 was assessed with respect to where work or expertise in the area would be called upon in the future (i.e. future needs and challenges), what is the safety significance of the area and is sufficient knowledge already available to meet the future needs and challenges. The results of this assessment are discussed below.

4.1.2.1 *Core Degradation and Melt Progression – Programme Needs*

Core degradation and melt progression defines the initial conditions for severe accident analyses, including consequences. Knowledge in this area is important to determining the amount of hydrogen released to the containment early in an accident, the state of the core and controls/blades at the time of reflooding, the amount of molten material that relocates to the lower RV head in a PWR (which is key to determining whether or not it will fail and, if so, when and in what manner), and to the calandria vessel in a PHWR, the potential for FCI and the timing of fission product release to the containment. Knowledge in this area will continue to be of importance in developing new LWR and PHWR designs, developing accident management programmes and communicating information to the public and decision makers during an emergency. However, experimental data and analytical tools are available to analyse and bound the phenomena associated with core degradation and melt progression, with the possible exception of the potential for FCI or for recriticality (especially in a BWR) if reflooding the RV occurs after control blade melting and for PHWRs where data is lacking. Continued operation of the QUENCH facility and PHEBUS will provide additional data for code assessment and maintain facilities and expertise for the foreseeable future (~5 years) capable of addressing early phase and limited late phase core degradation. The introduction of new materials and conditions (e.g. MOX fuel or higher burnup) may require some modification to analytical methods and the database, but large-scale facilities and programmes are not expected to be needed. Given that analysis can be used to bound late phase scenarios for most events of interest (and high risk), the need for a large scale experimental programme or additional capability for melt progression is not of high priority for PWRs.

4.1.2.2 *Fuel/Coolant Interactions (FCI) – Programme Needs*

In-vessel fuel/coolant interactions can challenge RV integrity, if they are large enough. Knowledge sufficient to predict their occurrence and energy release is not mature and analytical tools based on FCI fundamental physics need further development. However, there is general consensus that as long as the primary system is pressurised, large-scale in-vessel FCI is very unlikely and additional research on this topic is not warranted. For low-pressure conditions there remains the potential for energetic FCI. Therefore, the fundamentals of mixing and triggering FCI still need investigation to develop adequate models to predict FCI during unpressurised conditions (in-vessel and ex-vessel) and this is an area where additional experimental work is warranted to characterise under what conditions and to what extent FCI will occur, including under what RV re-flood conditions. Also, confirmatory testing using prototypic materials is highly desirable to completely assess the analytical tools and resolve the remaining issues.

Work is continuing on the development of analytical tools and small-scale FCI experiments. This work is directed toward understanding the fundamentals of FCI. As discussed above, additional work on FCI under low-pressure conditions is warranted since FCI is an important consideration in RV integrity and accident management (e.g. should the reactor cavity be flooded before or after vessel failure) and in developing new designs. With the loss of the FARO facility at the end of 1999, large scale experimental facilities, data and expertise in handling large amounts of prototypic materials needed to address FCI will be lost. Therefore, there is a need to consider whether or not additional confirmatory experiments using prototypic materials under low-pressure conditions are needed. This would also help maintain expertise in handling and understanding the effects of prototypic materials.

For PHWR reactors, in-vessel fuel/water interactions are being studied for accident sequences that lead to molten fuel being forcibly ejected from fuel channels into the surrounding water-filled calandria vessel. A prototypic experimental programme is being developed to investigate relevant phenomena.

As noted in Section 4.1.1.2, Canada is embarking on an extensive five-year programme in the MFMI facility at AECL-Chalk River Laboratories to address the issue of the energetics and effects of the interaction of prototypic CANDU corium with the moderator in PHWRs. The experimental programme will investigate relevant phenomena and be used to develop appropriate tools for the analysis of such events.

4.1.2.3 Debris Interaction with the Lower RV Head – Programme Needs

Retention of molten core debris in the RV is a desirable goal of accident management for LWRs, as is retention within the calandria vessel for PHWRs. If this can be done, then ex-vessel phenomena such as core-concrete interactions, direct containment heating, contact of molten debris with the containment wall and ex-vessel FCI can be avoided. The issues remaining in this area are associated with defining the conditions under which RV integrity for LWRs and calandria vessel integrity for PHWRs can be maintained (from either in-vessel or ex-vessel cooling) and, if the RV or calandria fails, the most likely timing and failure mode.

New reactor designs can be engineered to protect the lower RV head through cavity flooding or other design features. However, recent experimental results from the RASPLAV programme indicate that the potential for and effects of stratification of the molten core debris needs further understanding to ensure such features are fully capable of protecting the RV lower head for LWRs and calandria vessel for PHWRs. Also, currently operating LWR reactors may not have the capability to flood the reactor cavity or, if they do, the effects of thermal insulation on the RV outer surface on ex-vessel cooling may need to be assessed. In addition, understanding of the effectiveness of water in-vessel to provide sufficient cooling to protect the RV is important to accident management. Current facilities and programmes will address, on a small scale, relevant issues; however, large scale experimental work is also needed. Remaining issues involve determining the extent and nature of stratification in a molten pool of core debris, its effect on the heat transfer to the RV steel, and the conditions under which water in-vessel can cool the molten core debris. Accordingly, it is prudent to keep facilities and programmes to investigate these areas.

4.1.3 Summary and Recommendations

The attached table summarises the safety significance and sufficiency of knowledge discussed in Section 4.1.2. This table serves as a template that defines the baseline from which to judge whether or not a facility and/or programme is needed to resolve the issue and, based upon current facility and programme status, whether or not action by CSNI is recommended. Given the current state of knowledge and facility and programme status, the following recommendations are made:

Core degradation and melt progression – the state of knowledge for LWRs is sufficient to address all the phenomena associated with high priority issues. In addition, work at the QUENCH facility and PHEBUS will provide experimental data for the foreseeable future to assess analytical tools. For PHWRs, additional information on CANDU core disassembly and degradation is required, but new programmes and facilities have been established by AECL to address CANDU-specific issues. Therefore, no action by CSNI is recommended at this time.

Fuel/coolant interaction – additional model development and experimental data are needed to predict the occurrence and energy from FCI under low-pressure conditions. Although some facilities and programmes remain in progress, and a new large-scale programme in support of molten fuel-moderator interactions in PHWRs is being established in Canada, the loss of FARO nevertheless eliminates large-scale experimental data using LWR prototypic materials.

Recommendation 4.1.1 CSNI should consider having a specialist meeting to review the specific modelling and experimental needs to complete development and validation of FCI analytical tools and recommend a programme, in the absence of FARO and KROTOS, to meet the needs. This programme should acknowledge and build upon ongoing work and make recommendations for CSNI action to fill in any gaps. Recommendations should be provided in time for CSNI consideration at their annual meeting in 2000. In addition, some consideration should be given to maintaining FCI data.

Debris interaction with lower RV head – additional work to understand corium stratification in the lower RV head and the cooling effects of in-vessel and ex-vessel water are needed for the assessment of new and existing designs. This information is important to assessing the success of accident management strategies. For PHWRs, additional information on CANDU core disassembly and degradation is required, but new programmes and facilities have been established by AECL to address CANDU-specific issues.

Recommendation 4.1.2 CSNI should consider having a specialist meeting to review specific modelling and experimental needs and recommend a programme to meet these needs. This programme should acknowledge and build upon ongoing work, including the RASPLAV follow on work, and make additional recommendations for CSNI action to fill in any gaps. Recommendations should be provided in time for CSNI consideration at their annual meeting in 2000.

Template for Chapter 4.1 – In-vessel Phenomena

Criteria for inclusion:
Overriding Criteria
- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria
- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 4.1	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?
Core degradation and melt progression	• Accident Management	H	Yes – LWRs; No – PHWRs[2]	No[2]
	• Analysis of new designs	L	Yes – LWRs; No – PHWRs[2]	No[2]
	• Higher fuel burnup or MOX	L	No	No
FCI under HP conditions	• Analysis of new designs	H	Yes	No
	• Analysis of PHWR designs	H	No[3]	No[3]
FCI under LP conditions	• Accident Management	H	No	Yes Recommendation 4.1.1
	• Analysis of new designs	H	No	Yes Recommendation 4.1.1
Debris interaction with Lower RV Head	• Accident Management	H	No[4]	Yes Recommendation 4.1.2
	• Analysis of new designs	H	Yes[5]	No

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2. Additional work needed for PHWRs; programme and facility are established in Canada to address need.
3. Additional work needed for PHWRs; programme and facility are being established in Canada to address need.
4. NO is for currently operating plants.
5. YES if cavity flooding is ensured.

4.2 Ex-vessel Phenomena and Containment Integrity

4.2.1 Background

The need to provide more stringent safety requirements also in the very unlikely case that the accidental events propagate outside the reactor pressure vessel, maintaining the containment integrity as long as possible and minimising the off-site radioactive releases, has received a growing world-wide consensus.

In the existing plants, this objective is achieved by improving the traditional defence in depth strategy, with the implementation of new mitigation systems and the improvement of operating procedures against the consequences of severe accidents.

For the future reactors, new measures can be developed at the design stage in order to strengthen the protection of the environment and assure, even after a severe accident involving core melt with RPV failure, that there is no need of evacuation of the population and no long-term land contamination outside the immediate vicinity of the plant.

The following phenomena have been identified as having the potential to threaten the integrity of the containment in an ex-vessel accident sequence:

- direct containment heating;
- core-concrete interaction and FCI resulting from debris cooling action; and
- hydrogen combustion.

All these phenomena, independent of their relative importance and of the likelihood of the challenge, have been studied in a large number of national and international research programmes. The capability of the containment to withstand severe accident loadings has also been a closely related area of research.

In the SESAR/CAF document the research priorities and needs were examined for each identified area, discussing the ability of current research programmes to address these priorities and evidencing where additional effort was needed.

These needs are reviewed and updated in the present document, providing further recommendations in those areas where research capability and facilities are likely to be under threat.

4.2.2 Direct containment heating (DCH)

4.2.2.1 Needs and Challenges

The DCH problem was recently reviewed by a Working Group of the CSNI and a State-of-the-art report (SOAR) was published providing extensive information on experiments, modelling and resolution approaches in different countries.

The US has placed particular emphasis on the resolution of this issue. A very comprehensive programme, sponsored by NRC, provided a good understanding of the key physical processes with integral scaled experiments and established a basic methodology to address the probability of the containment failure in different types of plants.

The resolution of the DCH issue has been achieved for all PWR types. Although the DCH is not perceived as a high risk issue in BWR plants, due to a reliable automatic depressurisation system, specific studies and experiments could be necessary to reach a conclusive statement on a specific plant design.

There is a consensus that in the future reactors, the risk to the containment from the DCH can be eliminated by the provision in the design of a reliable depressurisation system and a suitable reactor cavity able to prevent dispersion. CEA and FZK, in co-operation with NRC, have conducted experiments in the SURTSEY facility at SNL, in support to the European Pressurised Reactor (EPR) design. The research activity was oriented to analyse the phenomena in case of a failure occurring with low driving pressure, following the intentional depressurisation action, in order to determine threshold of failure pressure for no significant dispersal and optimise reactor cavity design for retention capability. Follow-up experiments could include different RPV failure sizes and different paths from the cavity to the containment dome.

4.2.2.2 Short term

These arguments, involving the resolution of specific DCH issues remaining open for operating reactors and new evaluation tests for melt ejection and dispersal phenomena in future reactors, show that research capabilities are still requested and active in this area. Co-operative agreements offer the possibility of maintaining and utilising the expertise and the experimental facilities developed in the past.

No specific recommendation seems necessary in this area, in the short term.

4.2.2.3 Longer term

There is a general consensus that the resolution of the DCH issue from the risk perspective will be achieved in the short term. Nevertheless, the impact of vessel lower head failure and melt ejection phenomena on containment remains design-specific and supplementary work could be necessary for future reactor designs, mainly when a deterministic approach to the resolution of severe accidents will be required.

The possibility of maintaining part of the existing experimental capability with adequate adaptations should be anticipated. Experts in the CSNI Working Group could briefly discuss in the near future if some specific demand of integral and separate effect tests can be expected in the long term.

4.2.3 Core-concrete interaction, melt spreading and debris coolability

4.2.3.1 Needs and Challenges

Stabilising the molten corium in the reactor cavity is one relevant aspect of the severe accident management. MCCI has been investigated in many experiments, which have contributed to the understanding of the main processes and provided a comprehensive database for the validation of codes in this area.

On the other side, a clear understanding of the coolability of core debris and of the conditions leading to an energetic steam explosion have not been achieved yet. Furthermore, the generation of hydrogen during the melt cooling process requires additional research.

The OECD Specialist Meeting on Fuel/Coolant Interaction, held on May 1997, JAERI's Tokai Establishment, Japan, identified several areas where further investigation is needed to understand FCI phenomena (in-vessel and ex-vessel) and to improve the modelling.

The out-of vessel melt coolability is receiving increasing attention in the advanced reactor studies, where different concepts of ex-vessel retention devices are studied with the design objective to collect and retain the corium in the long term and provide an efficient heat removal mechanism.

Several research programmes are under way aimed to address these studies, in the framework of co-operative programmes and within projects supported by the European Commission, which offers a good possibility of integrating the existing capabilities and facilities (Table 4.2.1).

More specifically, the extension to ex-vessel corium coolability of the fuel/coolant interaction studies made originally for in-vessel steam explosions needs further consideration of different conditions, such as:

- quenching of corium in shallow pool geometry;
- water sub-cooling;
- low pressure;
- metal-rich melt composition;
- water flooding in shallow molten pool geometry; and
- water injection below a layer of melt.

4.2.3.2 Short term

It is worthwhile noting that some facilities, although having proven experimental capabilities and in spite of the interest demonstrated by the large international participation in the programmes, will not be available anymore in the short-term period. The FARO facility was closed at the end of 1999 and the way in which the closure was managed left no possibility for interested partners to negotiate further programme extension. FARO was dedicated to the investigation of the interaction process of a large mass of corium with water (MFCI), under realistic melt composition and prototypical in-vessel and ex-vessel accident conditions, and to the evaluation of corium spreading phenomena. No other large-scale facility remains available in the area of MFCI studies. A question mark has been put on the future of MACE. Although with some limitation in scale, MACE provided unique results for melt cooling from the top in the long term, with sustained heat of the molten pool. The possibility of operating this facility in a co-operative OECD project should be considered on the basis of a host country proposal.

The continuing operation of other facilities, the complementarity of their programmes and the maintenance of a co-operative effort are the necessary requisites for the resolution of the short term needs. It is expected that the CSNI Working Group will monitor all these conditions.

4.2.3.3 *Longer term*

Considering the specific interest for this subject in the advanced reactor projects, it will be necessary to maintain the effort over the short-term period, in order to achieve valuable results for the resolution of the safety issues.

Experiments in this area require overcoming many technical difficulties arising from high temperature, aggressive materials, and simulation of sustained heat. To save the investments made, it is recommended that those facilities where more representative results have been reached (prototypic materials, large scale, etc.) be maintained in the future, through co-operative participation and with the support of international organisations.

Considering the natural co-operative environment of the European research programmes, the FARO facility might have been favourably considered as a candidate to maintain capabilities in this area. Other facilities may encounter difficult conditions, considering national policies and budgetary constraints. Consideration should also be given to a proposal for the RASPLAV continuation programme, including ex-vessel core catcher studies. The CSNI Working Group should discuss these points and provide information on complementarity of the facilities and their actual perspectives of long term operation, in order to anticipate possible actions.

Table 4.2.1. **Corium spreading and coolability research topics**

Research Topic	**Relevant Programme**	**Comments**
Molten Corium-Concrete Inter-action Molten Corium-Ceramic interaction	PERCOLA (France) MACE (USA) VULCANO (France) CIRMAT (Russia)	Experimental work on concrete practically finished. Existing codes (as WECHSL, CORCON) considered as satisfactory.
Melt spreading and catching devices studies	CORINE, VULCANO (France) KAJETS, KAPOOL, KATS, CARLA (Germany) RIT Exp. (1D-2D)(Sweden) CSC, COMAS, HTCM, THMO Projects (EC)	An active research area in Europe due to the design options in EPR.
Coolability	MACE (USA) COMET (Germany) DECOBI & POMECO (Sweden) COTELS (Kazakhstan, sponsored by Japan) CSC Project (EC)	The future of MACE is in question. COMET is a design specific experiment for water injection below a layer of melt.
Ex-vessel Steam Explosions		Extrapolation of in-vessel steam explosions results to ex-vessel may require further work.

4.2.4 Hydrogen transport and combustion

4.2.4.1 Needs and Challenges

The energetics associated with hydrogen combustion has been identified as one of the major threats to the containment integrity and a significant research effort is maintained in this area. On one side, the need for implementing new mitigation measures and accident management procedures in the operating plants and, on the other side, the interest in the new reactor design to take into account the hydrogen issue at the design stage, converge in demanding further research activities, in order to cover the following main aspects:

- the assessment of mitigation techniques with special devices able to remove hydrogen from the containment; and
- the improvement and validation of methods for the analysis of hydrogen mixing and combustion phenomena.

An OECD Workshop on the implementation of hydrogen mitigation techniques was held in 1996. It provided an updated view of the subject identifying specific needs in different areas. With respect to mitigation measures, there is a general trend towards implementation of passive auto-catalytic recombiners (PARs), possibly supplemented by other measures (igniters, post-accident dilution). Work is still underway for the qualification of the PARs, in order to assess their availability (possibility of fouling and poisoning) during the accident and their potential to cause detonation.

A CSNI Working Group has completed a "SOAR on Containment Thermohydraulic and Hydrogen Distribution", whose objectives are to assess the current capabilities of the codes to make predictions with respect to flow, temperature distribution and gas concentration distribution inside the containment, to address strengths and weaknesses of analytical methods predicting the effectiveness of chosen mitigation techniques and to take into consideration accident management actions with their possible feedback on processes in containment.

A "SOAR on Flame Acceleration and Deflagration-to-Detonation in Nuclear Safety" has been completed by the same Working Group.[6] Priority needs in the area of prediction tools for different modes of combustion have been identified, as:

- criteria to assess flammability and DDT limits, validated on an extended database;
- a scaling methodology to extrapolate from test results to reactor conditions; and
- validated computational tools for the analysis of loadings on the containment structure.

4.2.4.2 Short term

Experimental programmes on PARs are conducted in an active co-operative framework, using experimental facilities having complementary characteristics (Table 4.2.2). The interest in the qualification of various industrial products provides an additional support to the research work and valuable results are expected in the short-term period.

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6. This document has been published with the reference NEA/CSNI/R(2000)7.

Further development and validation work is necessary for 3-D codes for gas mixing in the containment, including other phenomena as condensation and effect of igniters and recombiners.

A quite large number of experiments providing information needed to validate thermal-hydraulic codes, flame acceleration models and DDT criteria are underway (see Table 4.2.2). Good co-operation exists through information exchange agreement and in the framework of specific projects promoted by the EC in the framework of the 4th FWP.

Table 4.2.2. **Hydrogen transport and combustion research topics**

Research topic	Relevant programme	Comments
H2 Mitigation	SNL Surtsey PAR test (USA) LSVCTF Tests (Canada) H2PAR (France) BMC Zx Tests (Germany) PHEBUS (France) HYMI Project (EC) CTF (Canada)	Qualification of PARs Qualification of PARs Accidental ignition Tests autocatalytic recombiners. Deliberate ignition
Mixing and distribution	MISTRA,TOSQAN (France) VICTORIA model containment (Finland) LSGMF (Canada) PANDA (Switzerland) VOASM, EUCOFA Projects (EC)	Validation on large scale 3-D separate effects Large scale validation Multi-compartment and 3-D separate effects
Standing flames	Diffusion Flame Facility(Canada)	stability and thermal load
Deflagration	LSVCTF (Canada) RUT (Russia) UNI-Pisa LVIEW tests (Italy) TU-München PuFlaG, MuSCET tests (Germany) VOASM, HDC Projects (EC) CTF (Canada)	• vented deflagration • jet igniting • flame acceleration • internal structures • equipment survivability
DDT	BNL HTCF (USA) Hyjet Calthec (USA) RUT facility (Russia) FZK Tube, RWTH/SWL (Germany) H2-DDT Project (EC)	• criteria for DDT • limits for DDT • prediction of DDT

4.2.4.3 Longer term

Expertise and large facilities in the domain of gas dispersion, combustion and detonation are widespread outside the nuclear industry and it is not expected that even in the nuclear domain there will be any decline in activity that would put at risk the availability of basic competence to meet emerging safety needs.

As regards hydrogen control technology (recombiners, igniters and inerting) there are several R&D and testing facilities and there appears to be sufficient commercial interest to ensure that adequate research capability will be maintained.

Even if, as there is common consensus, the hydrogen issue will be resolved, from the risk perspective, in the near term, further work is certainly needed where, as in some EC Member states, a different, more deterministic approach is being adopted.

In accordance with recent safety requirements set up by the major European Utilities for the next generation of the LWR nuclear power plant, which include the adoption in the design of prevention and mitigation measures against severe accidents, the assessment of the containment integrity against the most relevant phenomena, including the hydrogen combustion event, plays a key role in safety evaluation. Related safety research will become increasingly important; experiments at different levels (separate-effect tests, coupled-effect tests and integral tests) will be needed to support further development and validation of computer codes in those specific conditions expected for future reactor designs.

The discussion of the type of containment-related test facilities, inevitably needed for the future, was going on in Europe inside the EUCOFA project of the 4th FWP. It is worthwhile noticing that large facilities such as the HDR containment, used as an integral type test facility, and the Battelle Model Containment (BMC) were taken out of service.

The research-related interest for a large European facility extends from hydrogen studies to thermal-hydraulics (convection, stratification, multi-compartment effects), aerosols behaviour (distribution, depletion, pool-scrubbing effects) and iodine transport.

If consensus is reached, the assembly of a large European facility will cover a period of several years and its operation will fall in the long term.

The CSNI Working Group should monitor the follow-up discussion in the EUCOFA project.

4.2.5 Containment integrity

4.2.5.1 Needs and Challenges

The capability of both steel and concrete containments to withstand quasi-static load under pressures representative of those that could arise in severe accidents has been studied in the past in a large number of experiments in scale models. Currently, NRC and NUPEC are engaged in a joint effort: a model representative of a prestressed concrete PWR containment is under construction to be tested to high pressure in 2000. Representatives of many countries plan to participate in the evaluation of the experiments.

Moreover, a new experimental programme on a mock-up containment of EPR type (MAEVA, 1/3 scale, 16 M diameter) has been started by EDF and IPSN. The goal is to study the integrity of a prestressed concrete containment (without liner) in terms of mechanical strength and leaktightness, using a more representative simulation of the thermal and hydraulic phenomena (presence of steam/air mixture) that appear in the containment, with loading conditions extending to those of severe accidents.

The MAEVA project, for the mock-up construction and testing, was supported, as part of the 4th FWP and with the participation of many international partners, by the R&D project CESA, which provides the analytical support to the experiments.

To assess the effectiveness of the decay heat removal and assure containment integrity, many large scale thermal-hydraulic experiments have been performed to identify phenomena involved in the containment atmosphere for design basis and severe accidents. These studies are closely coupled with source term analysis and hydrogen issues. The experiments were mainly designed to validate computer codes for which the complexity has increased progressively with the demands of reactor safety analysis.

For future reactors of all water-cooled designs, the use of passive safety features for containment cooling is presently being considered. Experimental programmes are going on, all aimed at demonstrating their operational performance under various conditions and producing a database for validation and development of advanced multi-dimensional computer codes.

For the VVER-440/213, the need of additional experimental and analytical studies to assess the capability of the “bubbler condenser” pressure suppression structure to maintain containment integrity and limit the radioactive release to the environment has been confirmed by the study of an OECD Support Group. These activities have been followed up by work undertaken in the framework of the EC TACIS/PHARE assistance programme.

4.2.5.2 Short term

Considering the safety significance of this subject, a research effort is adequately maintained, through co-operative programmes and EC projects. No specific recommendation seems necessary in this area, in the short term.

Ongoing programmes are primarily studying the containment in its “new” condition. There is a clear need for studies oriented towards ageing effects on the whole containment structure (concrete, penetration seals, etc.). An OECD Workshop was held in October 1998 concentrating on assessing the ability of current Finite Element methods to predict loss of capacity of concrete structures due to degradation. The outcome of this Workshop[7] has provided recommendations for experimental programmes necessary to develop confidence in the analytical predictions. The relevant CSNI Working Group has made specific proposals in this field.

4.2.5.3 Longer term

For future reactors, the containment system will have a key role to play in demonstrating that an improved level of safety has been achieved and innovative design solutions and auxiliary systems

7. Workshop on F.E Analysis of Degraded Concrete Structures, Brookhaven National Laboratory, October 1998.

which will be adopted in future containments will need a verification process if they are to be licensed. The ongoing discussion on the realisation of a large European facility for containment related studies demonstrates the large interest in this area.

Experimental facilities and capabilities need to be maintained for confirmatory research in the following supplementary areas:

- characterisation of structural materials under severe accident conditions, particularly the constitutive laws of concrete as a function of temperature, age and loading;
- performance and reliability tests of advanced containment cooling systems;
- development of codes able to assess mechanical and thermal loading for different scenarios, including the use of mitigation systems for containment cooling; and
- the effects of corrosion.

Table 4.2.3. **Containment integrity research topics**

Research topic	Relevant programme	Comments
Containment integrity analysis	MAEVA EPR Type Containment Tests (France) PCCV NRC/NUPEC (USA) CESA Project (EC)	Margin to failure and leakage behaviour studies.
Material Structural Behaviour	High performance concrete constitutive laws at high temperature (EDF France) Performance of concrete at elevated temperature. UKNSR Programme. Concrete cracking under dynamic loads (CEA France)	Constitutive laws to be implemented in structural codes. Advanced modelling methods for concrete structures including; Tensile fracture energy release rate, behaviour under multi-axial loading. Ageing and corrosion effects need to be considered.
Containment cooling	PASCO (FZK) (Germany) MISTRA (CEA) (France) DRAGON/AIDA, PANDA (Switzerland) VVER Bubble Condenser studies (EC TACIS/PHARE) DABASCO, EUCOFA Projects (EC) LSGMF (Canada)	Operational performance of passive containment cooling systems (long-term containment integrity) Testing of passive containment concepts for CANDU
Ageing		Programme to be proposed by CSNI Working Group
Cable penetration integrity	PCCV NRC/NUPEC(USA) ALPHA JAERI (Japan)	An important contribution to containment leakage pathways

4.2.6 Summary and Recommendations

In the area of ex-vessel severe accidents, where the resolution process of safety issues as corium coolability, hydrogen control and containment integrity is still in progress, research remains particularly active, characterised by a high level of co-operation and association. This clearly depends not only on the fact that experiments are technologically complex and expensive and national budgets are facing progressive reductions, but also on the possibility of having access to facilities with unique characteristics, and, more generally, on the interest of extending the consensus on the validation approach to safety arguments.

The European Community, supporting through the Nuclear Safety Programmes many projects in this area, provided further opportunities for integrating research capabilities and for taking benefit of complementarities of experimental facilities. The situation is slightly changing in the 5th Framework Programme, where severe accident research has been de-emphasised among the objectives of the area "Operational Safety of Existing Reactors". In this perspective, because of the lack of financial support, there is risk of losing, in the short term, other capabilities and facilities in this area.

The template for determining essential facility and programme needs and need for CSNI action illustrates this situation. In a large number of safety significant areas, research needs are adequately satisfied by existing facilities and programmes. Only in two cases the risk of losing unique facilities, putting in danger the resolution of safety significant issues, convinced the SESAR/FAP group to express the following recommendations requiring action to be taken by the CSNI:

Recommendation 4.2.1: Corium Debris Coolability: On the basis of a host country proposal, the MACE facility should be considered for a co-operative OECD project.

Recommendation 4.2.2: FCI (see also *Recommendation 4.1.1*): The CSNI should approve a Specialist Meeting with the objective of evaluating the impact of the loss of a unique facility like FARO on the resolution of key safety issues in the ex-vessel area (as well as in in-vessel). Actions should be recommended to safeguard other remaining facilities and to preserve experimental and analytical capabilities on FCI for the future needs including the feasibility of setting up a Centre of Excellence in this field.

Recommendation 4.2.3: Consideration should also be given to a proposal for the RASPLAV continuation programme, including ex-vessel core catcher studies.

For the facilities going to a final closure, it is finally recommended that a complete database of (at least) the most qualified experiments is maintained for further code validation processes, and that the most relevant technological solutions adopted for achieving the required simulation for facility instrumentation and for facility operation remain documented.

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9. "The status of the Bubbler Condenser Containment System for the Reactors of the VVER-440/213 Type" – Report NEA/CSNI/R(98)13.

10. "State-of-the-Art Report on Flame Acceleration and Deflagration-to-Detonation in Nuclear Safety", Report NEA/CSNI/R(2000)7.

Template for Chapter 4.2 – Ex-vessel phenomena and containment integrity

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 4.2: Research topics	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	Action needed?
DCH	• Accident Management	H	Yes[8]	No	No
	• Analysis of new designs	H	No	Yes	
Corium-Concrete Interaction	• Accident Management	H	Yes	No	No
Melt Spreading and Coolability	• Accident Management	H	No	Yes	Yes (see Recommendation 4.2.1)
	• Analysis of new designs	H	No	Yes	
Ex-Vessel Steam Explosion (See Also Ch. 4.1: FCI Under LP Conditions)	• Accident Management	H	No	Yes	Yes (see Recommendation 4.2.2)
	• Analysis of new design	H	No	Yes	
Hydrogen Mixing and Distribution	• Accident Management	H	No	Yes	No
	• Analysis of new designs	H	No	Yes	

8. For the issue resolution on a risk based approach.

Chapter 4.2: Research topics	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	Action needed?
Hydrogen Combustion, DDT	• Accident Management	H	No	Yes	No
	• Analysis of new designs	H	No	Yes	
Hydrogen Mitigation	• Accident Management	H	Yes[9]	No	No
	• Analysis of new designs	H	No	Yes	
Containment Integrity	• Accident Management	H	Yes[9]	No	No
	• Analysis of new designs	H	No	Yes	
Containment Cooling	• Accident Management	H	Yes[9]	No	No
	• Analysis of new design	H	No	Yes	
Ageing, Penetration Leak-tightness	• Accident Management	H	No	Yes	No
	• Analysis of new designs	H	No	Yes	

9. For the implementation of mitigation measures in operating plants.

4.3 Fission Products

4.3.1 Background

The ultimate safety implication of this work is to assess and to limit the release of fission products to the environment and therefore to human beings. This is achieved through the development of an understanding of the various fission product release, transport, deposition and resuspension phenomena, which facilitates the improvement of the reliability of source term evaluation. From a regulatory viewpoint, however, knowledge of fission product behaviour in the containment is of prime importance since this has implications for containment and plant design and the effectiveness of mitigation features and accident management. A good understanding is therefore required of the timing, quantity, identity and physical and chemical form of the released fission products, as well as their transport and retention in the primary coolant system and containment, and the effects of fission product removal and other mitigation features.

The development of models incorporating the results of separate effects studies has led to the assumption that there is a good understanding of the general behaviour of fission products, from their release from the fuel to their behaviour in the containment. Large-scale integral experiments, such as the PHEBUS-FP programme, are essential to confirm this overall view and to highlight remaining uncertainties.

The current level of research in this area is thought to be appropriate. The priority issues are:

- to complete the PHEBUS-FP experimental programme;
- to develop an understanding of the mechanisms and quantification for the release of low volatility fission products from the fuel, including MOX;
- to assess late in-vessel fission product release; and
- to quantify the effects of specific elements (e.g. Ag and Cs), pH, radiation, surfaces and organic impurities on iodine volatility.

A number of other issues on which further discussion is required have also been identified:

- release of low-volatility fission products from core-concrete interactions, although the results of the MACE tests now indicate that this is of low priority; release of structural material may also promote faster aerosols deposition;
- fission product retention, resuspension and revaporisation in the circuits, with some attention being given to containment by-pass sequences;
- pH level to be maintained in containment water pools for optimised iodine control; and
- effects of hydrogen burns on fission product chemistry in the containment, including fission products interaction with surfaces.

It should be noted that a number of related studies have also been conducted in recent years to assess the needs for and status of fission product research. Fission product work has been identified which is needed to help probabilistic safety analyses to define the consequences of severe reactor accidents, avoid inappropriate conservatism and support severe accident management studies and crisis tools development. Detailed recommendations have been specified in a number of reports, notably in the 1995 Consensus Document on Safety of the European LWR [2], the CSNIS State-of-the-art report on Fission Product Release and Transport [3] and the CSNI Workshop on Iodine Chemistry for Reactor Safety (1996) organised at PSI in Switzerland.

4.3.2 Needs and Challenges

- The experimental programmes which are currently underway or just completed in the fission product area are described below [3-6].

- It should be noted that the following sections only cover current or very recent programmes. While it is recognised that earlier facilities and programmes have produced valuable data, discussion of these is outside the scope of this work.

4.3.2.1 Separate Effects Studies

Fission Product Release from Fuel

Out-of-pile studies of fission product release from fuel are ongoing at CEA/IPSN (France), AEA Technology (UK), AECL (Canada), JRC (TUI, Karlsruhe), BCI (USA) and JAERI (Japan). In all of these programmes, small quantities of irradiated fuel are heated to temperatures in excess of 1 600°C and the kinetics and/or speciation of the release is monitored by γ-spectrometry or mass spectrometry. The main parameters of these studies are the fuel burnup and the composition of the atmosphere. In contrast to earlier work in inert or reducing conditions, current programmes are tending to move towards more technically demanding oxidising atmospheres. Other variables which are being studied in individual national programmes are the effects of Ag/Cd/In control rod material and boric acid (AEA Technology, CEA/IPSN), heating power (AECL), Zircalloy cladding (AEA Technology, AECL) and fuel grain size (JAERI). AECL has also performed kinetic studies of fuel oxidation in steam up to 2030°C and in air environments up to 2080°C. AEA Technology has investigated fuel oxidation in CO/CO_2 atmospheres up to 1800°C.

Key facilities for the study of fission product release from irradiated fuel exist at CEA/DRN (the VERCORS facility at Grenoble). It should be noted that this latter programme has recently been extended to fuel liquefaction temperatures and investigation of the release of transuranics from UO_2 and MOX. Key facilities also exist at AECL (Chalk River). All these facilities consist of dedicated in-cell furnaces, capable of heating fuel samples to at least 2000°C, and associated instrumentation: γ- and mass spectrometry at AEA Technology to acquire kinetics and speciation data, γ-spectrometry and aerosol impactors in VERCORS, oxygen sensors, γ-spectrometry and aerosol collectors/impactors at AECL. A new high temperature and high-pressure fission product release facility (VEGA) is under construction at JAERI, with tests planned up to 3000°C and 1 MPa.

Reactor Materials Release

The main current programmes in this area are the CEA/IPSN EMAIC programme on release kinetics from Ag/Cd/In control rods, the release of elements from the same control rod material at temperatures below 1500°C at JAERI, and the Harrier facility at AEA Technology Winfrith developed for non-active simulant studies of late-phase release from molten pools at up to 3000°C. Experiments with simulant fission products in molten pools above 3000°C have also been conducted by LSK (St. Petersburg) as part of a 4th Framework Programme project. Other studies in the past have included the release of Sn from Zircalloy, and the behaviour of boric acid at degraded core temperatures. It is also worth noting that there is work in progress in Finland (STUK and VTT) on boron carbide oxidation at temperatures below 1100°C. In general, the facilities used for this type of work are similar to those used for fission product release studies or vapour phase thermodynamics (see below).

High Temperature and Vapour Phase Thermodynamics

The ability to measure fundamental data for incorporation into databases and modelling codes is important for nuclear safety research since many of the most important species are specific to this area and the required data often do not exist elsewhere. These capabilities, therefore, are essential for current and future reactor and safety needs.

Facilities exist at AEA Technology, ECN Petten, JRC (TUI Karlsruhe), BCI Columbus and AECL (Chalk River). In addition to the measurement of fundamental data, these facilities are used in support of fission product and reactor material release, vapour-surface interaction, revaporization and molten core-concrete interaction studies.

Vapour-Surface and Vapour-Aerosol Interactions

Work in these areas is currently underway at AEA Technology and VTT. The absorption of simulant fission products compounds onto metal surfaces has been studied in the DEVAP (CEA/DRN – IPSN) programme. The VTT and AEA Technology programmes have all been designed to study the interactions of vapour with aerosols at high temperatures. These latter two programmes make use of key capabilities in the iodine chemistry area and, in AEA's case, of specialised measurement capabilities. The VERCORS HT experimental programme (CEA/IPSN), producing both FP release and high temperature transport data, is also making a valuable contribution in this area. Prior to these, programmes have been completed at Siemens and AEA Technology (Falcon facility) to study the interaction of volatile iodine species onto surfaces and aerosols respectively.

Resuspension and revaporization

A collaborative programme has recently taken place on fission product revaporization, involving experimental studies at AEA Technology, JRC/TUI and VTT. This work provides support to the PHEBUS-FP interpretation, and involves the use of hot-cell analytical capabilities at AEA Technology.

A programme to study the behaviour of deposited fission products under re-heat and re-wet conditions, using specimens from in-cell tests, is ongoing at AECL (Chalk River). The STORM (EC/JRC) facility is no longer used for aerosol studies.

Aerosol Nucleation and Transport

Studies of aerosol nucleation have been completed at AEA Technology (Falcon facility) and VTT. The objective of these programmes was to study the nucleation and growth of aerosols from metal vapours descending a thermal gradient.

The TRANSAT and TUBA facilities (CEA/IPSN) have studied aerosol transport in pipes; however the programmes are now completed. Experiments on aerosol deposition in pipes have been conducted in the WAVE and WIND (JAERI) facilities. A new WIND programme to study the fission product aerosol deposition and revaporization, and the integrity of the piping due to decay heating from deposited fission products is now underway at JAERI (see for example [8]). Experiments are also underway at AECL to investigate and characterise aerosol behaviour in components of a CANDU primary heat transport system. Phenomena controlling the transport of fission products in piping continues to be of strong interest because of their importance in bypassing sequences. In addition, the Aerosol Transport for Steam Generators (ARTIST) Programme is being conducted at PSI Switzerland as part of an EC funded project.

Aerosol Physics in Containment

A collaborative programme involving experimental studies at CEA/IPSN, VTT and IVO is performed on aerosol physics in containment. These studies of aerosol behaviour in condensing steam conditions took into account the hygroscopic nature of fission product aerosols. These experiments were performed in the PITEAS, AHMED and VICTORIA facilities. Other facilities included VANAM and KAEVER in Germany, and the JAERI ALPHA facility. Of these, the AHMED, VANAM and KAEVER facilities are now closed. Work is underway on aerosol trapping in penetrations (AEA Technology) and filtered venting (VTT). The EDF CIVAUX facility is also designed to study aerosol and vapour leakage across large-scale containment structures. Tests on fission product trapping in containment leakage paths and fission product removal by containment spray have been conducted at NUPEC. Detailed tests are in progress at CEA/IPSN in the CARAIDAS facility to support the modelling of the removal of aerosols by sprays. An extensive database is available from the LWR Aerosol Containment Experiments (LACE) and Advanced Containment Experiment (ACE) studies. DRAGON (PSI Switzerland) is addressing aerosol physics in passive containments. Facilities at AECL's Whiteshell Laboratories are establishing aerosol behaviour in flashing and steaming jets as providing an important boundary condition for aerosol physics in containment.

Pool Scrubbing

A programme of pool scrubbing experiments to determine decontamination factors of soluble and insoluble aerosols under two-phase and jet-flow conditions has been completed at CIEMAT in the PECA facility. Experiments on aerosol removal in churn turbulent conditions are currently underway at AEA Technology (Heron facility), while the POSEIDON facility (PSI) is being used to study aerosol and vapour removal. High-pressure (7 MPa) and high temperature (350°C) pool scrubbing experiments have been completed at JAERI in the EPSI facility.

Iodine Chemistry

Extensive experimental programmes on aqueous iodine chemistry are underway at AECL, AEA Technology, CEA/IPSN and Siemens. Small programmes exist in Japan (JAERI, Hitachi), Finland (STUK), Latvia and Switzerland (PSI). The objective of the work is to understand and quantify the processes leading to iodine volatility from water pools in the containment, and the effects of surfaces and impurities on iodine volatility and speciation. The fundamental chemistry of iodine solutions under irradiation is now reasonably well understood and the emphasis of the work is moving towards clarifying the effects of other reactor materials such as silver aerosol and paints. Recent programmes include small-scale studies on iodine volatility at high pH (AEA Technology, AECL), iodine interactions with silver (Siemens, IPSN), organic paints (AECL, Siemens, CEA/IPSN) and metal surfaces such as stainless and zinc primer paint (AECL, Siemens), and the formation and stability of organic iodides at various temperatures and in the presence of radiation (AECL). AECL has also conducted a wide-ranging series of integrated experiments in the medium-scale Radioiodine Test Facility (RTF) for its own national programme as well as in support of the internationally-sponsored Advanced Containment Experiments (ACE) and PHEBUS FP (i.e. PHEBUS-RTF tests) programmes. The most recent of these, the PHEBUS-RTF programme, was completed in 1997.

A collaborative programme has been completed on the effects of silver on iodine volatility and a new programme is in progress on iodine organic chemistry, involving small scale irradiated experimental studies which is being performed at AEA Technology, SIEMENS, STUK, VTT and CEA/IPSN.

Significant facilities in this area are the medium scale CAIMAN (CEA/IPSN) facilities for the study of iodine behaviour and volatility under irradiation and the Co-60 irradiation sources for small scale experiments at AEA Technology, CEA/IPSN and AECL.

Another study will be undertaken for the containment atmosphere behaviour. Under radiation some oxidants (ozone and nitrous oxides) are generated, thus may influence the chemistry of both silver aerosols and volatile iodine. EPICUR facility (CEA/IPSN) will be used.

4.3.2.2 Integral Studies

PHEBUS-FP

PHEBUS-FP [6] is the major large-scale in-pile programme in this area. The objective of the programme is to provide data for code validation on a wide range of severe accident processes:

- fuel degradation;
- release of fission product and reactor materials such as B_4C;
- aerosol and vapour transport through the primary circuit; and
- aerosol behaviour and iodine chemistry in the containment [6].

The extensive post-test analysis programme associated with PHEBUS-FP tests involves active handling and analytical facilities at CEA/Grenoble and Cadarache, AEA Technology and JRC/TUI.

Data acquired from the first two PHEBUS tests emphasise the importance of fuel degradation mechanisms, where new data have been acquired, and lead to increased efforts devoted to the study of silver-iodine interactions in the containment.

Blowdown Test Facility

A series of in-pile loss of coolant and loss of emergency core cooling experiments have been performed in the Blowdown Test Facility (BTF) in the NRU reactor at AECL (Chalk River) [7]. These experiments used single elements or trefoils of fuel rods. Fuel behaviour, fuel degradation, fission product release, aerosol and vapour transport and fuel oxidation were measured during the experiments, in order to provide well qualified data for code validation. The most recent BTF experiment was completed in late 1997 and efforts are now underway to complete the documentation of this experimental programme. No additional BTF experiments are currently planned and the NRU reactor in which the BTF is located will likely be shutdown by 2005. However, the Canadian Neutron Facility (CNF), which is expected to replace NRU later this decade, has been designed to allow the addition of a severe fuel damage and fission product release and transport facility, should there be a need to do so.

4.3.3 Summary and Recommendations

The following table 4.3 summarises the safety significance of the information discussed in Section 4.3.

Table 4.3. **Fission product research**

Research need	Relevant programme	Comments
Effects of specific elements on iodine volatility	Iodine chemistry (AECL, AEA, Siemens, PSI, CEA) PHEBUS-FP	Recent programme to address effects of Ag in the presence of other core components. Current programme to study iodine interactions with organic paints, organic impurities and steel surfaces. Also, quantify pH and radiation effects on iodine chemistry. PHEBUS information on representative materials.
Release of low volatility fission products	Fission product release PHEBUS-FP VERCORS RT (France) VEGA (Japan) In-cell experiments (AECL)	No programme on effect of quenching in the presence of boron carbide

Research need	Relevant programme	Comments
Late in-vessel fission product release	Fission product release PHEBUS-FP VEGA (Japan)	High temperature thermo-dynamics. Fission product release. Release from debris bed and pools. Air effects.
Low volatility FP release from core-concrete interaction	ACE (USA)	Good database available from ACE programme. Questionable need for further work.
Aerosol Physics in containment	PITEAS (France) AHMED (Finland) VICTORIA (Finland) VANAM (Germany) KAEVER (Germany) ALPHA (Japan) CARAIDAS (France) MAEVA/CESA(France) DRAGON/AIDA (Switzerland) LSGMF (Canada) Flashing Jet Facility (Canada)	aerosol condensing environments FP interaction with sprays leakage deposition during condensation in tubes
FP retention, resuspension and revaporization in the primary circuit	Current programme on revaporization of deposited material from PHEBUS-FPT1 pipework and resuspension. AECL (Canada) studies on fission product retention in circuit and end fittings. WIND (JAERI Japan) particularly for revaporization	
pH level for water pools	No specific programme on pH control but existing capability in iodine volatility prediction	Iodine Chemistry
Effects of hydrogen burns on FP chemistry in the containment	Previous ACE programme (conducted at AECL, AEA and ORNL). Current NUPEC work examining effects of hydrogen burns on containment performance; data-base on chemistry effects covered by Sandia and AEA Technology experiments.	

It can be seen from the table that, in general, facilities and capabilities are available to meet the research needs identified in Section 4.3.1. Clearly the PHEBUS–FP facility and programme represents the major activity in the area, with significant contributions being provided by other programmes and facilities world-wide. Available results have already led to the shifting of iodine research priorities. While there are no major programmes addressing low volatility fission product release from core-concrete interactions, there is a good database from the ACE tests and this area is not considered a priority when compared with other issues (e.g. late-phase release). Similarly, the impact of hydrogen burns on fission product chemistry is considered of secondary importance compared with their effect on containment performance. Continuing efforts are being made to understand and model fission product behaviour in shutdown accidents, the impact of high fuel burnup on FP release and transport, and the short-term and long-term removal paths for aerosols in the containment. Attention should nevertheless be given to the risks associated with the closure of several important Canadian facilities: both the Blowdown Test Facility (BTF) and the Radioidone Test Facility (RTF) have recently been closed, and the NRU reactor will likely be shutdown by 2005.

For future power plants, there is a clear trend towards a reduction in the possible releases in the case of accidents. The demonstration of this, even if it is made easier by special design features, requires an improved knowledge of all the significant phenomena involved in the source term determination, and the continuation of the research and expertise.

Many of the capabilities and facilities described above are unique to the nuclear industry. In the long term, PHEBUS-FP will be the only remaining in-pile facility and would be extremely expensive and time consuming to replace. Some of the other facilities, e.g. for out of pile release studies, are smaller and less expensive. However, they provide an ability to meet emerging safety issues such as air ingress and the impact of using higher burnup and MOX fuel and for low power and shutdown conditions. Furthermore, the technological environment of specialised skills and equipment within which the smaller facilities exist can provide considerable flexibility allowing a rapid response to changes in research requirements.

To examine some important release phenomena such as the second and third items of the table more deeply than possible with present programmes, some specific installations or modifications of existing facilities will be needed, the VERCORS and VEGA facility are of special importance in order to answer to this need.

Perhaps even more important than the facilities themselves is the expertise residing in this area. Such expertise involves a variety of disciplines (e.g. reactor physics, materials science, mathematical modelling, chemistry and aerosol science) which has been built up over many years and is unique to the nuclear industry.

The fission product domain has benefited greatly in the past from international collaboration. This has been especially true of the experimental programmes. Some examples of this are:

- Marviken V aerosols test;
- LOFT programme;
- LWR Aerosol Containment Experiments (LACE); and
- Advanced Containment Experiments (ACE).

This tradition of collaboration continues with such international work associated with the PHEBUS-FP experiments and the associated analytical projects. As may be seen from Section 4.3.2, other smaller facilities which address different aspects of the problem are in operation or are being built in many countries. This gives some optimism that there is no **global** danger of losing competence or expertise in this area over the next few (i.e. 2) years. However, some national programmes and facilities in the areas of iodine chemistry and fission product behaviour have been closed and there are clear indications that others are under threat of closure; capabilities and facilities have already been lost in, for example, Canada, the USA and the UK, and further erosion of capabilities is expected over the next few years. It is important that further loss of expertise in this important area of reactor safety be averted. Efforts should be made to establish opportunities for technical specialists from those countries where programmes are declining to continue to contribute via active programmes established elsewhere. Furthermore, to ensure that key expertise is retained for the industry as a whole, Centres of Excellence should be established in the areas of fission product behaviour and iodine chemistry, based on the facilities and expertise currently resident in national programmes.

Specific recommendations

Considerable expertise in fuel and fission product release and transport have been developed world-wide over the past many years in support of individual national programmes. However, as these programmes are being wound down, facilities are being shut down and expertise is being lost from these important components of reactor safety. Particularly noteworthy examples are the termination or significant reduction of national research programmes in iodine chemistry and fission product release and transport, and the prospect that, left un-addressed, these programmes will decline even further. These are very complicated areas of reactor safety, which will require many years of hands-on experience before any level of expertise can be demonstrated. Although programmes such as PHEBUS play an important role in helping to retain expertise in these areas, additional initiatives are required to ensure that expertise residing in many national programmes is available to assist industry in the future. The SESAR/FAP group thus recommends the following:

Recommendation: 4.3.1 The CSNI should encourage the establishment of a Centre(s) of Excellence in Iodine Chemistry and Fission Product Behaviour. This Centre of Excellence should be structured so as to facilitate maximum participation of OECD countries interested in maintaining core expertise in these fields. The Centre should thus be linked to existing major programmes on iodine chemistry and fission product behaviour, but be closely linked via modern methods of telecommunication with experts from participating countries and organisations with the aim of executing a collaborative programme of research, development and modelling. As a starting point to this initiative, the CSNI should request proposals from members interested in hosting this Centre of Excellence.

Overall, it is important to consolidate the whole area by establishing a Centre of Excellence in fission product behaviour and iodine chemistry.

Other possible forms of international collaborations

In this area of long-term expertise, it is important that a benchmark activity be established in order to interpret the extensive database already acquired and to assess its applicability to reactor safety cases. Such an activity is also considered essential to maintain the knowledge of the teams in countries where facilities are threatened with decommissioning. In addition, this would contribute to the longer-term aim of achieving rationalisation and harmonisation of codes for use in PSA level 2. Possible themes for such benchmarks are iodine behaviour, aerosol removal phenomena, or the estimation of the source term under realistic accident conditions. Coupling with core degradation mechanisms should also be studied, namely for fission product releases and their interaction with control rod and structural material emission from the core.

References to Section 4.3

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4. E. Della Loggia (Compiler), "National and Community Research Policies and Programme on Reactor Safety", EUR Report 15618 EN, 1994.

5. OECD, "Brief Informal Reports on Recent National/International Activities in the Area of Confinement of Accidental Radioactive Releases in OECD Countries" (March-September 1995), supplement to NEA/SEN/SIN/WG4(95)8, 1995.

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9. Proceedings of the OECD CSNI Workshop on the Chemistry of Iodine in Reactor Safety (PSI, Würenlingen, June 1996), Report NEA/CSNI/R(96)6.

Template for Chapter 4.3 – Fission products

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 4.3	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	Action needed?
Fission products release	• release of volatile products	H	H	Y[10]	Yes
	• release of low volatility FPs and transuranides				
	• late phase release				
	• release from MCCI				
Reactor materials release	• structural materials	M[11]	N	Y	N[12]
	• control rod materials				

10. Priority should be given to low volatility and late phase release, where there are currently the major uncertainties.

11. Mainly in conjunction with iodine chemistry.

12. With the assumption that current programmes will be finalised; specific actions may be necessary for B_4C or new control materials, if any.

Chapter 4.3	Future needs and challenges	Safety significance	Sufficient knowledge available?	FAP essential?	Action needed?
Transport in piping	• nucleation	L	N	Y	Y
	• vapour/surface and vapour/aerosols interactions				
	• deposition/resuspension				
	• effect of complex structures				
Transport in the containment	• aerosol growth and deposition	H	N	Y[13]	N
	• phoretic effects				
	• sprays				
	• interaction with combustion and/or FCI				
Iodine chemistry	• release at the break	H	N	Y	Y
	• chemistry in the sump				
	• interaction with containment surfaces				
Retention mechanisms	• aerosols: filters, scrubbing	H	N	Y[14]	N
	• iodine				
Integral studies	• source term prediction	H	N	Y[15]	N
	• coupling of phenomena				
	• influence of AM measures				

13. There is a necessity to maintain the capability to interpret lessons learned from integral experiments like PHEBUS.

14. Experiments would be plant and device specific.

15. Taking into account existing programmes, priority has to be given to interpretation and synthesis work, in a co-operative frame.

Chapter 5

HUMAN FACTORS

5.1 Background

Screening of operating events at nuclear power plants indicates that human factors have contributed in 50 to 80% of the cases. Human factors are a vast and diverse research area requiring an interdisciplinary approach combining ergonomic, psychological and social expertise. To anchor the research into plant operation and to secure the availability of real data, plant operators and maintenance staff have to be involved. Human factors expert resources are relatively scarce and establishing good communication between human factors and technical experts and managers requires special attention.

In human factors research, cultural aspects and differences in national practices are of great importance and must be taken into account in international collaboration. Nevertheless and despite of differences in national emphasis and specific features of the nuclear field, most problems have a generic character, and there are good opportunities for international collaboration as well as for links with the related programmes in other industries, in particular aviation, off-shore activities and the chemical industry.

Generally, it is felt that a fairly good theoretical framework for applied human factor studies exists, with some reservations on management and organisational performance [1]. The major problems are the availability of relevant data (primarily possessed by utilities), the sensitive character of many issues and the difficulties in transferring the know-how.

5.2 Needs and challenges

As a response to the recommendations of the SESAR/CAF, CSNI Working Groups called in summer 1997 a group of senior experts to discuss possible strategies for research in the area of human performance. The group produced a report, Research Strategies for Human Performance [2], summarised below under items A to C. The report characterised important research topics for the medium and long term and expressed some priorities for further work by NEA.

A. Data collection for support of understanding and modelling human performance

Understanding human behaviour in risky situations and activities, in stressful conditions and in plant disturbance situations helps to avoid errors under such situations and to develop models for PSA. In particular, data is needed to support development, testing and applications of cognitive models and to study errors of commission.

B. Design of control rooms, man-machine interfaces and operator aids

Analogue equipment is being replaced by digital in new plants and increasingly also in existing plants in the modernisation and life-extension projects. Thus, there is a need to develop methods to assess, compare and optimise the man-machine interface associated with totally new or hybrid control rooms [3]. Also, criteria are needed for evaluating the adequacy of the safety design of the new or hybrid control rooms. The use of computer based operator aids, including development and verification of computerised procedures, is an evolving area [4]. The effects of these aids on operator performance have to be evaluated.

C. Evaluation of human performance safety issues

Organisational practices: A wealth of information has been produced on the influences of organisation and management on human performance and plant safety. Identification, assessment and quantification of organisational factors are needed to make the information useful to practitioners in operational experience feedback, PSA and regulation [5].

Low-power and shutdown operations: Generic and plant specific improvement of control rooms, plant information systems and procedures. See also [6].

Decommissioning: Specific measures may be necessary to counteract the negative effects on staff motivation and availability after plant closure has been announced or is anticipated.

Evaluation of operating experience: Improvement in the rigor and detail of event reporting and analyses when human errors and organisational deficiencies are important as a root cause. A guideline for crews on how to behave under circumstances and events unanticipated by designers.

Validation of existing techniques for Human Reliability Analysis (HRA): Verification and validation of skill and rule based human reliability analysis methods against operating experience and simulator data in order to reduce uncertainties in applications.

Safety culture: Development of indicators that help to characterise safe organisations. More recently, deregulation and privatisation of the electric utility sector has led to additional human factors challenges, whose impact on safety needs to be monitored. These include reduced staffing levels, additional schedular pressures, more use of overtime and changes in management.

Current programmes

Within the CSNI, two Working Groups have important human factors related activities. One of them maintains an overall view of activities world-wide. It has recently published a thorough study on the Role of Simulators in Operator Training [7], including identification of future research needs. Another major contribution is the report Improving Reporting and Coding of Human and Organisational Factors in Event Reports [8].

Understanding human capabilities and limitations in controlling complex processes requires extensive gathering and in-depth analysis of data on human performance. Human Reliability Analysis (HRA) is currently an important item in the agenda of the second Working Group. In 1997 the group completed a comprehensive study titled Critical Operator Actions – Human Reliability Modelling and Data Issues [9]. A new HRA task addresses Errors of Commission. For practical applications human performance models have to be developed and validated. Models have already shown some promise for group performance, e.g. in staffing studies.

The report Research Strategies for Human Performance [2] proposed several immediate actions for NEA, in particular further work by the two CSNI Working Groups on Cognitive Models and Errors of Commission and workshops and subsequent SOARs on Organisational Practices and Man-Machine Interfaces. It also proposed co-ordinated NEA activities in the areas of shutdown operations and decommissioning and stressed the importance of networking activities and combined efforts by the two Working Groups. As an immediate result of the recommendations, work is underway on a SOAR on Identification and Assessment of Organisational Factors. An interim SOAR has already been published [10]. Also, work on a SOAR on Man-Machine Interface has been started and a workshop on Approaches for the Integration of Human Factors into the Upgrading and Refurbishment of Control Rooms was organised in August 1999.

The IAEA's programme emphasises the publication of guides and technical reports on the impacts of new technologies for man-machine interfaces and I&C, and support to training and advancing safety culture. IAEA programmes in support of safe plant operation, OSART and ASSET missions, produce information that is potentially valuable as a source for human factor studies.

The Incident Reporting System (IRS), operated jointly by NEA and IAEA, currently contains some 2800 reports and is a useful source of information for studies on human factor issues. The value for human factor studies will increase in the future as a result of improved coding of human and organisational factors in the reports. The IRS collects only events which are truly significant for safety and therefore includes only a fraction of events occurring in nuclear power plants; hence it only allows qualitative analyses. Collaboration with the rather similar system of the utilities' WANO organisation is important.

Many nuclear regulatory bodies and their technical support organisations, e.g. USNRC, SKI in Sweden and GRS in Germany, have substantial human factors expertise in-house or access to human factors expertise (e.g. National Laboratories). The work is mostly directed to the support of regulatory activities related to training and PSA with research emphasis on human/system interface and digital I&C and, particularly in Sweden, on organisation and management issues. Specifically, the USNRC has developed guidelines for the review of advanced control rooms (NUREG – 0700, Rev. 1) and hybrid control rooms (NUREG/CR – 6637).

Halden's Man-Machine Laboratory, HAMMLAB [11], is being further developed into a world-leading centre for experimental studies in the control room area. It provides a flexible, reconfigurable environment and infrastructure for carrying out human factor research for diverse processes and serves as a test bed for development, test and demonstrations of advanced man-machine interfaces and operator support tools. It is possible to produce experimentally controlled realistic and cognitively demanding simulated scenarios. A specific Human Error Analysis Project has been underway since 1994.

In Japan, JAERI has completed a human factors experimental facility with a full scope reactor simulator (high fidelity model for a 2-loop PWR, actually the MUTSU ship reactor system), flexible CRT-based man-machine interface and extensive functions for data collection and analysis. It is used for investigations of operator cognitive behaviour under accident situations and interface design concept evaluation. NUPEC also completed an experimental facility for part task simulation of commercial nuclear power plants of PWR type. With its fully CRT-based man-machine interface, it serves as a test bed for demonstration of operator capability to cope with transients and accidents including cognitively demanding situations, even provocative commission errors. The utility TEPCO commissioned in 1995 a large soft-panel based simulator for man-machine research, including equipment for psychological measurement [12]. In Korea, KAERI has built a comprehensive Integral Test Facility (ITF), containing a full-scale PWR simulator and a sophisticated experiment data analysis system.

In the USA, the Nuclear Engineering Department of the North Carolina State University has created a good collaborative relation with the Human Factors faculty and operates the Scaled PWR facility (SPWRF) for research and training. The facility combines a 1:9 scale working model of a PWR (electrically heated fuel rod simulators, Freon as coolant, most major components in the primary and secondary circuits) with a realistic though limited control room.

The utilities have collected large amounts of human performance data from simulator exercises and they also have the most comprehensive databases on operational events. There are efficient techniques available for collecting raw data from simulators (e.g. EPRI’s OPERAS/CREDIT system), and also from plant maintenance activities (e.g. the LOTI system at the Finnish Loviisa plant). EPRI has a Research Initiative for Human Performance Management of Nuclear Power under way, which includes database development and analysis activities as well as development of indicators and practical tools for use at the plants. Data collection and analysis programmes in France (EdF) and Hungary also deserve to be mentioned. On the whole, unfortunately, the availability of material from the industry for R&D purposes is limited and the results of industry’s own activities are not widely published.

For R&D purposes, advanced engineering simulators and plant analysers offer broad availability at low cost and are particularly well suited for the study of new control room concepts and operator support tools [13]. Japan has prepared for the CSNI Working Groups a new task in this area.

As the amount of potentially interesting operating experience data is huge, there is a need to develop efficient methods and tools to help in selecting the incidents of true safety interest and to carry out various analyses using the data. The Common Cause Data Exchange project (ICDE) under the CSNI Working Groups could serve as a model to develop systematic ways for data related co-operation.

In the development of new control room concepts, man-machine interfaces and operator support tools, there are extensive industrial R&D activities as the modernisation and life extension of the existing plant population promises substantial markets in the foreseeable future. These activities are closely related to the introduction of modern digital I&C systems. Modernisation of control rooms and I&C systems will present major challenges for training, maintenance of digital systems, PSA and devising ways to avoid negative impacts on human performance.

In all human factor research, intensified networking of research groups is needed since comparisons and benchmarking (good operational practices and regulatory approaches) are the major ways of working, especially in the area of management and organisation. As there are severe limitations in the open publication of results, such as the findings from plant peer reviews offered by WANO, mechanisms need to be devised to allow confidential discussions between experts. Utilities should be strongly encouraged to take initiative themselves and results from other relevant industries should be incorporated too. As regards benchmarking of regulatory approaches, IAEA offers a service called IAEA International Regulatory Review Team (IRRT). Several reviews of East-European regulatory organisations have been carried out and the first western review was carried out in the end of 1998.

Language and cultural differences limit the world-wide usefulness of the facilities for certain human factors studies; facilities in Japan and Korea serve primarily as national or regional centres.

5.3 Summary and recommendations

Table 5.1 gives a summary of human factor research needs and examples of relevant current programmes.

Table 5.1. **Summary of human factor research needs and examples of relevant programmes using the categorisation and reflecting the priorities in [2] for further work by NEA**

Research needs	Relevant programmes	Comments
Basic knowledge and data: • Gathering of data • Development of human performance models • Applications to studies of errors of commission	HAMMLAB (Norway) JAERI, NUPEC & TEPCO facilities (Japan) ITF (Korea) SPWRF (USA) High-fidelity simulators (industry)	• Many facilities are new or are being upgraded • Many research groups can make use of the experiments • Exchange of data requires special attention
Design and validation of control rooms, man-machine interfaces and operator aids	• Above mentioned HF • Experimental facilities • Simulators and plant analysers	• Strong efforts are under way by the industry • Management of change at existing plants is of much current interest • There is a strong coupling with I&C issues
Evaluation of human performance safety issues: • Organisational practices • Maintenance activities • Shutdown operations • Decommissioning (staffing issues) • Evaluation of operating experience • HRA validation • Safety culture	• National programmes • The ORFA and ISA-NEW • concerted actions of EU NFS-2 programme • The Nordic NKS/SOS • Programme	• Promotion of co-operation and networking, including other industries • Coupling with CNRA, IAEA and WANO activities • Promotion of human factor training for larger amount of specialists • Monitor effects of utility restructuring due to deregulation and privatisation

The facility situation for collecting basic data on human performance is currently improving. In particular, the Halden Project can continue to provide a focus for international research and technical networking in this area and also its role as a centre for education could be extended. The Halden Board of Management recently provided views on the long-term direction of the project [14]. In particular, the new multifunction-multipurpose HAMMLAB 2000 man-machine laboratory in Halden, to be commissioned by the year 2000, will extend the scope of potential studies to all operational and abnormal states and emergency conditions and provide a near-real-life testing environment. It has been estimated that the initial investment cost of the HAMMLAB 2000 laboratory (with a flexible control room, three different plant simulators and buildings) is close to 3 million USD. The expected annual operating cost is some 2 million USD, of which the salaries of the research staff (human factors specialists) is more than half.

Through close co-operation with the Halden Project even small national programmes may be viable. Project members can also use the software and know-how from HAMMLAB to build their own facilities for specific applications. Indeed, due to cultural differences, man-machine research laboratories are needed in several places. Good networking and benchmarking activities between the laboratories are highly useful.

The simulator facilities of utilities and their support organisations, and those of plant and simulator vendors, are an essential resource. Such facilities include EPRI Simulator and Training Centres in Kansas City and Houston, EdF facilities in Paris and Lyon, CRIEPI facilities in Japan and the KSG simulator training centre at Essen in Germany.

On the basis of the attached systematic decision-making table, the template, Table 5.2, specific recommendations are given.

Table 5.2. **Template for Chapter 5 – Human Factors**

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 5	**Future needs and challenges**	**Safety significance**	**Sufficient knowledge available?**	**FAP essential?**	**Action needed?**
Basic data collection and Modelling human performance	• Selection and training of personnel • AM procedures • PSA	H	No	Yes	Yes (See recommendation 5.1)
Man-machine interface	• Modernisation of existing & Design of new control rooms • Design of operator support tools	H	No	Yes	No
Evaluation of human performance safety issues	• Operational safety, accident prevention • Deregulation	H	No	Yes	Yes (See recommendation 5.2)

Recommendation 5.1: Collection of basic data and development of human performance models:

- Promote efficient international use of the new unique research facilities, such as Halden's HAMMLAB.

- Facilitate availability of operating experience and simulator data from the utilities. Develop a Common Human Reliability Database.

- Use the data for the development of human performance models and for definition of criteria for personnel selection and training.

It is essential that there is strong international participation in the planning and execution of the experiments at the unique facilities as well as in the analysis of the results. Much of the data analysis of experiments on the major facilities can be carried out by expert teams from several organisations – periodically attached to the facility teams as necessary.

There is a need for a stronger role of the utility industry. Operational experience and simulator data, mostly available only to utilities, is the most relevant source of information for in-depth studies by human factor experts. In the future, much more basic data on human performance in better-defined format should be collected and exchanged. Real operational experience data is vital to validate data from training simulators and research facilities. Credible work with operating organisations requires that the research team combines behavioural science and technical expertise and people that are skilled in working closely with plant operational, maintenance, engineering and management staff. In certain experiments at research facilities, operating staff from commercial plants must be made available. It would be an obvious benefit if industry representatives could participate more in the relevant Working Groups of the NEA.

Data from high fidelity, well-equipped simulator facilities are needed to provide qualitative and quantitative data for the development of human performance models, for both individuals and groups, particularly for use in PSA.

Human performance data is also necessary for setting criteria for personnel selection and for defining training programmes. Educational background, experience and personal characteristics are all important.

Recommendation 5.2: Specific actions by and co-operation between CSNI Working Groups to enhance research planning and networking:

- Monitor and update human performance research strategy.

- Reinforce systematic networking and benchmarking activities, in particular in the area of management and organisational performance.

- Watch the potential effects of utility restructuring due to deregulation and privatisation.

- Encourage joint activities with other industries.

It has to be understood that progress in many human factors research projects takes time. Interaction of small groups at various institutes is essential. Transfer of the results to the end users requires in most cases close interaction between the research and user groups.

Reinforced systematic networking activities are needed particularly in management and organisational performance research to initiate and manage comparison and benchmarking activities. As much of the information needed and also results of the research are confidential, these networks should serve as fora for confidential in-depth discussions between researchers as well as between researchers and the users of the results.

It is important to identify the potential negative effects of the extensive restructuring of nuclear utilities in many countries. Outsourcing may result in losing company memory, overview and core competencies. Increasing competition may threaten the sharing of good practices. Seeking efficiency easily overloads key people. Also, positive impacts are possible and should be utilised to the extent possible.

Since most human factor issues have a common generic basis, joint projects and sharing of facilities and experts with other industries having similar problems is of high value.

Longer-term considerations:

The experimental facilities being completed or planned will provide many new opportunities and there are a large number of training simulators, which will continue in operation and can deliver data for information and research in this field. However, to ensure expertise applicable to the nuclear industry, Halden should be maintained as a Centre of Excellence.

In some countries, the availability of human factor experts is very limited. It would help these countries to support relevant groups at universities and to provide the universities with access to problems of true practical interest for nuclear industry and regulation. Some human factor training should be offered to a larger number of technical specialists, e.g. in the context of experience feedback. European Commission and IAEA could organise relevant courses.

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Chapter 6

PLANT CONTROL AND MONITORING

6.1 Background

The SESAR/CAF report presented a list of research needs that had been identified in the earlier SESAR/FU report. This chapter expands on the list presented in the SESAR/CAF report for Instrumentation and Control (I&C) used in plant safety, control and monitoring systems. In addition, some general research needs that span all of these systems are reviewed. Both the requirements for new systems in new plants, and upgrades and maintenance for existing plants are covered in this chapter. It is also important to recognise that although the topic of Human Factors is dealt with separately in Chapter 5 of this report, the latter topic and Plant Control and monitoring are strongly interlinked, and should be considered together. Chapter 5 addresses human performance issues associated with the operation of nuclear stations, whereas Chapter 6 focuses on technological aspects of software and hardware used in the I&C systems of current and future reactor designs.

For general I&C technology, the nuclear industry is a very small, yet demanding, component of the world market. As a result, within I&C manufacturers there are many high quality teams and facilities available world-wide capable of addressing most design and manufacturing problems. Even in the area of nucleonics, which is the only I&C area dominated by the nuclear industry, there is no sign of a decline in the capability of OECD Member countries to carry out high quality research and development. Nevertheless, there are particular aspects of nuclear I&C development where joint research programmes within OECD countries may be beneficial.

Industrial I&C systems have been characterised by rapid changes in the underlying technology used to implement many systems, typically a new generation of hardware and software becoming available every two years, with the use of computer-based systems becoming steadily more widespread. The primary reason for the increase in computer-based systems is the potential performance enhancements of an overall system made possible by this technology. The main issue facing suppliers and regulators of nuclear I&C systems is how, using modern technology, to design, develop, verify, validate and maintain I&C systems to the levels of safety, reliability and performance required by the nuclear industry.

Research is also needed on methods to assess the reliability, performance and integrity of safety-critical I&C systems, and the effectiveness of these methods. Uncertainty about what methods are effective and suitable is leading to significant additional costs and schedule risk for suppliers and customers, with no clear indication that new methods are superior to previous simpler techniques. Given that many computer-based upgrades for nuclear plants have already been designed in many countries, somewhat higher priority should be given to assessment versus development of new safety-critical I&C systems.

The I&C areas within a nuclear plant can be roughly divided into the following three categories:

- Safety systems, primarily responsible for mitigating against consequences of failure of other plant systems.

- Control systems, primarily responsible for maintaining the operating state of the plant; and

- Monitoring systems, primarily responsible for collecting, logging and presenting current or past data on the status of the plant systems.

Furthermore, these systems can be categorised in terms of their origin or pedigree for use in nuclear applications:

- In-house development, and

- Commercial off-the-shelf (COTS) electronic systems.

- And combinations of the above items.

In each case, qualification of the process and product is required for nuclear application. In-house development can be controlled and documented, whereas off-the-shelf products need to have their processes audited and qualified before their products can be used in nuclear applications.

6.2 Needs and Challenges

This section reviews individually the research needs for safety, control and monitoring systems, concentrating upon those aspects that are not likely to be covered by internal corporate research and development programmes. In some cases, there are common research requirements, spanning all three areas.

Safety Systems

A standard fundamental target for safety-critical I&C systems in nuclear plants is that they comply with adequate standards for safety and reliability, given the current state of the art for designing such systems. Since techniques for developing reliable systems with high safety integrity are continuously evolving and improving for a wide variety of industries (e.g. aircraft avionics and telecommunications) the nuclear industry should monitor, and if appropriate, should adopt some of these techniques to take advantage of developments in related industries. There are, however, some specific aspects of application to nuclear plants that must be integrated with these new techniques. Thus, there is a strong need for research programmes that seek to evaluate and incorporate new methods for developing and implementing high safety integrity systems for the nuclear industry. These programmes should examine:

- advanced techniques (including formal methods, hardware description languages, etc.) for specifying hardware and software requirements;

- advanced methods (including automated theorem provers, advanced simulation tools, etc.) for verifying consistency and correctness of requirements; and

- new technologies for implementing hardware systems, such as Application Specific Integrated Circuits and Field Programmable Gate Arrays, etc.

Some of these techniques are already being used on projects throughout the world. There is a need to critically assess the effectiveness of these techniques to determine what improvements could be achieved for future projects.

Ongoing research is also required for the difficult problem of assessing safety-critical systems, including their safety performance, reliability, robustness, fault behaviour, coverage and integrity, for both new systems and operating systems to assure that the original requirements are being met. For new systems, improved verification and validation techniques are required to demonstrate that the reliability requirements can be met. For existing systems, methods are required that demonstrate that there has been no degradation from the original performance, and that digital obsolescence is not an issue.

Continuing research is also needed into the requirements, design and performance of I&C equipment for safety-critical applications, especially those instruments that monitor plant parameters during and after severe accidents, where correct instrumentation performance in potentially harsh environments is essential. There is also a need to ensure that ageing phenomena are well understood for I&C equipment used in safety-critical systems, including reactor core instrumentation.

Control Systems

The technology and design of plant control systems is generally well known and, with a couple of exceptions, based on standard industrial techniques and algorithms. The main difference from most industrial applications is the need for a system with significant safety integrity (although not as high as safety systems), very high reliability and availability, and much greater complexity than safety systems. While in safety systems high reliability is primarily required for safety reasons, in plant control systems high reliability is needed equally for both safety and economic reasons.

The main research needs for new or replacement plant control systems are guidlines on appropriate levels of verification and development of suitable techniques to verify that the design goals for safety and reliability have been achieved under anticipated plant demands. The greater complexity of some plant control systems, compared to safety systems, means that some of the techniques developed for verifying safety systems are impractical or unsuitable. As well, the technology used to implement plant control systems is changing rapidly, often requiring the development of different verification techniques for newer systems.

There is also a long-term need to monitor and review the performance of existing plant control systems to look for systematic faults that indicate aspects which of the plant control system design are not being adequately covered by present development techniques. Unexpected coupling between sub-systems assumed to be independent, or the influence of human factors in the development process are examples of the kind of faults that might be found through this analysis.

There are some new developments in control theory, such as neural nets and fuzzy logic that need to be evaluated for their suitability in systems requiring high safety integrity; these developments are also applicable to safety systems and monitoring systems.

Monitoring Systems

Like plant control systems, the basic technology required for plant monitoring systems is well known and under active development by many companies world-wide. The main area requiring continuing research is how to manage and analyse data acquired in plant monitoring systems in the most effective manner to support plant maintenance and surveillance programmes so that unplanned outages and plant equipment failures can be reduced. Development of practical guidelines for plant maintenance and surveillance programmes would be very useful, as would a review of the effectiveness of current approaches to plant surveillance.

There are a few areas where further technical work on instrumentation is required. These areas include further development work on instrumentation for specific nuclear plant applications, such as monitoring water chemistry, and sensor ageing phenomena with the associated impact on sensor performance and for in-core instrumentation; and smart sensor technology that can be qualified for nuclear applications.

6.2.1 Near-term needs

Safety Systems

Modern techniques to develop safety-critical systems are increasingly dependent upon software-based tools. The requirements for developing, verifying, and qualifying these tools for both software and hardware development, need to be reviewed.

Control Systems

The development of more effective guidance on appropriate safety and reliability verification techniques, and the creation of a database of significant faults and associated root cause analysis for significant events in existing plant control systems should have high priority.

Ambiguity and uncertainty in the verification requirements for plant control development projects increase the cost and likelihood of schedule slippage. Excessive verification requirements can make important control system upgrades uneconomic, yet inadequate verification can result in unacceptable safety risks. Performing a systematic assessment is another option.

The creation of a plant control system fault database, based on events reported in the IAEA/NEA IRS database, is already being considered by a Working Group of the CSNI. No programme is in place, however, to review systematically the events for significant trends.

Monitoring Systems

With many nuclear plants now well into their original design lifetime, many utilities are looking to extend the plant operation for longer periods of time, provided this life extension can be done safely and economically. Effective plant surveillance is one of the key tools to assuring that the plant continues to operate within the original design basis. Consequently, development of guidelines for effective plant maintenance and surveillance programmes should have high priority. High priority should also be given to the development of monitoring systems for severe accident parameters to assist with accident-management procedures. A review of international processes for the qualification of monitoring systems used in the support of on-line control room decisions and a formal process for the introduction of changes to existing control rooms should be performed to provide new approaches to ensure that operators are using qualified information for safety-related decisions. With the increase in the use of automated information for operational use, advanced display systems, international processes for failure modes in software-based systems with human operators should be researched to identify recommended approaches to ensure "system" errors (Humans and automation) can be mitigated or eliminated without consequence and with a high level of confidence.

Nuclear plant vendors are increasingly adopting/adapting commercial non-nuclear products and systems for nuclear applications. This trend offers greater potential for cost reduction and expanded benefits, while increasing the potential risk of unknown problems associated with poorly controlled product engineering. The requirements for categorising and qualifying third party software products should be researched, and standard processes for qualifying vendors and products should be identified.

6.2.2 Longer-term Needs

Safety Systems

Formal techniques are now being used frequently for the requirements specification of safety-critical software for nuclear plants. There is a similar trend in some other industries to use formal techniques to give hardware or system requirement specifications that can be analysed mathematically for consistency and completeness; an assessment of the tools used by these industries may indicate that modelling languages and simulation-based methods apply to the nuclear industry, as well. In the long term, appropriate variations of these techniques should be developed to support complete specification of both hardware and software for a system in the same broad mathematical framework, supported by a set of software tools to permit formal development and verification of the system, including the analysis of timing and sequencing problems. This work should also include automated and formal techniques to analyse designs for sensitivity to single failure modes and to perform Failure Modes and Effects Analysis (FMEA).

Modern high-integrity development techniques are leading to the production of software that can meet requirements with a high degree of confidence. However, studies have shown most of the remaining faults in such systems are effectively errors in the original requirements. A research programme to investigate formal techniques used to verify the correctness of specifications against high-level safety requirements should be created. This work could also include the incorporation of the complete safety system and control system dynamics into design basis accident safety analysis codes to better understand the potential coupling between the control and safety systems.

Control Systems

Over the long term there is a need to improve models and simulations of plant control system behaviour for off-normal and accident conditions.

6.3 Summary and Recommendations

The I&C requirements for the nuclear industry can be divided into three areas: safety systems, control systems and monitoring systems. Based on the discussion in the previous section, the future needs and challenges for these areas, along with the assessments leading to any required actions, can be summarised as in Table 6.1.

For safety systems, there are four key basic needs. First, the effects of ageing phenomena on I&C equipment need to be addressed. While this need has a high safety significance and a programme is required to address our knowledge deficit, no action is required of the CSNI. Member countries should, however, maintain ongoing monitoring programmes for trends in ageing related failures and problems. Secondly, adequate performance of safety-critical I&C systems should be ensured under accident conditions. This need also has a high safety significance, but there is sufficient knowledge available, and therefore, no programme is required. Thirdly, the overall reliability of safety-critical I&C systems should be enhanced. While our knowledge is not entirely sufficient to secure this enhancement, it has only medium safety significance, so again no formal international programme is required. Lastly, support is required for jurisdictions pursuing new reactor designs. This too has a high safety significance, and there should be additional R&D, but no specific action is required of the CSNI, as developments for improved designs should be pursued by the jurisdictions and corporations involved.

For control systems, there are three basic needs or challenges. Similar to safety systems, adequate performance of control systems should be ensured under accident conditions, the overall reliability of control systems should be enhanced, and support is required for jurisdictions pursuing new designs. For all three, the safety significance is not as high because the existing systems should perform adequately. Nevertheless, Member countries should maintain ongoing monitoring programmes for trends in ageing related failures and problems, and jurisdictions pursuing improved reactor designs will require an active R&D programme.

For monitoring systems, there are also three basic needs. First, monitoring systems should be developed for severe accident parameters to support accident-management procedures. Such systems would have a high safety significance, and our current knowledge is insufficient, making a programme essential. No action is required of the CSNI, however, because a large integrated programme is not required to develop monitoring systems for severe accident parameters. The last two needs, to support life extension of components and to provide for future designs, are of low safety significance and, therefore, do not require specific OECD-sponsored programmes.

Having concluded that there are no specific programmes needed to address future needs and challenges for nuclear I&C, and there are no significant facility requirements, there are nonetheless opportunities for co-operative programmes that should be pursued in the short term. In particular, the nuclear industry can take good advantage of the active research and development on I&C technology for many non-nuclear industries, particularly those that require high safety integrity and reliability (e.g. aircraft avionics and telecommunications). It is therefore recommended that CSNI Working Groups monitor developments in non-nuclear industries on behalf of Member countries, and encourage collaboration between research groups in Member countries.

6.3.1 Specific Recommendations

In the near-term, the following recommendations are proposed:

Recommendation 6.1: CSNI Working Groups should monitor developments in non-nuclear industries (e.g. aerospace, telecommunications, advanced software methods) in the area of instrumentation and control, co-ordinate collaboration between nuclear and non-nuclear industries in this area, and communicate advances in associated technology to Member countries. They should also encourage increased collaboration between research groups in Member countries working on nuclear plant safety, control and monitoring systems. This collaboration should be widely based, but with a focus on advancing reliability, ensuring performance in hostile environments, and addressing ageing phenomena and the issue of digital obsolescence. Collaboration is also encouraged in evaluating the safety assessment methods used in other industries having probable application to the nuclear industry.

Recommendation 6.2: CSNI Working Groups should monitor developments in Member countries for the qualification of non-nuclear software-related products for nuclear applications. They should also encourage increased collaboration between research groups in Member countries working on software qualification and related safety-system requirements.

Recommendation 6.3: CSNI Working Groups should monitor developments in Member countries on the processes used to introduce software-based monitoring and decision-aid tools into control rooms, including approaches to failure modes and effects on human operators.

Recommendation 6.4: The I&C activities of the Halden Project should be continued as a means of fostering collaboration on research related to nuclear I&C systems, and as a means of ensuring the continued development of nuclear I&C experts to address future issues.

In the longer term, the nuclear industry requires an assured supply of qualified personnel capable of understanding and addressing I&C issues that are unique to nuclear reactors. This can be achieved in part by assuring the continuation of the I&C activities of the Halden Project.

Template for Chapter 6 – Plant control and monitoring

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 6: Plant control and monitoring	**Future needs and challenges**	**Safety significance**	**Sufficient knowledge available?**	**FAP essential?**	**Action needed?**
Safety Systems	• Address ageing phenomena	H	No	Yes	No[1]
	• Ensure adequate performance under accidents	H	Yes	No	No
	• Enhance overall reliability	M	No	No	No[1]
	• Provide for improved designs	H	No	Yes	No[2]
Control Systems	• Ensure adequate performance under accidents	H	Yes	No	No
	• Enhance overall reliability	H	Yes	No	No[1]
	• Provide for improved designs	H	No	No	No[2]
Monitoring Systems	• Develop monitoring for severe accident parameters	H	No	Yes	No[3]
	• Support life extension of components	L	Yes	No	No[2]
	• Provide for improved designs	L	No	No	No[2]

1. Member countries should maintain an ongoing monitoring programme for ageing and trends in failures or problems.
2. Jurisdictions looking at life extension and/or improved designs will need an active programme.
3. A large integrated programme is not required to develop monitoring systems for severe accident parameters.

Chapter 7

INTEGRITY OF EQUIPMENT AND STRUCTURES

7.1 Background

Only a limited investment is being made today and in the near future for new nuclear power plants. Thus ageing and ageing degradation of key components is the major concern for structural integrity of components and structures. Managing the safety aspects of nuclear power plant (NPP) ageing requires the implementation of effective programmes for the timely detection and mitigation of ageing degradation of plant systems, structures and components (SCC) important to safety, so as to ensure their integrity and functional capability throughout plant service life. General guidance on NPP activities relevant to the management of ageing (maintenance, testing, examination and inspection of SCC) is given in the International Atomic Energy Agency (IAEA) Nuclear Safety Standards (NUSS) Code on the Safety of Nuclear Power Plants: Operation [1] and associated Safety Guides on in-service inspection [2], maintenance [3] and surveillance [4]. By far the largest activity in this area is associated with the integrity of components of the primary pressure boundary, including those in the reactor core itself (pressure vessel for LWR [12] or pressure tubes for PHWR) and the associated piping. It also includes the integrity of the secondary side components in order to deal with concerns over loss of heat sinks. Main work on ageing is oriented towards the extension of the operating lifetime of the nuclear power plants. The technical feasibility of this extension is well established and documented, but the regulatory process is under study in most of the countries.

The research seeks to determine in both qualitative and quantitative terms, the way in which the reactor environment and operating conditions degrade the strength and integrity of SSC over their operational lifetime. Integrity of equipment and structures has to address:

- the state of the material of the components or structures which may be affected by material composition, manufacturing processes and operational parameters;

- the loads which result, from out of normal and transient operation, incidents, and accidents including seismic events, and have to be combined with the initial state of stress and with the environmental conditions;

- the state of defects which is determined by the manufacturing practice and environmental attack of damage mechanisms; and

- the safety margins which have to be built in with respect to the sensitivity of the examination and testing methods applied.

This chapter focuses on major metallic components under operational conditions. Some aspects are given for concrete components and cable ageing.

There are several relevant organisations working on integrity issues. The regulatory staff of the countries, the research institutions, the utilities and the owners' groups, and international organisations are developing rules and guidelines to guarantee the useful life of the plants, which has to be achieved with the required safety. Collaborative programmes have been taking place at international level.

The activities of the IAEA on the safety aspects of NPP ageing management include symposiums on ageing, publications, training programmes, technical co-operation projects, safety advisory groups and co-ordinate research programmes [5]. The activities on NPP life management are co-ordinated by the International Working Group on Life Management of Nuclear Power Plants (IWG-LMNPP). Radiation embrittlement of reactor pressure vessel steels and thermal degradation, corrosion, fatigue have been major subjects of concern. In the monitoring field, non-destructive examination techniques and fracture strength are areas included in the IWG-LMNPP. The IAEA has supported Co-ordinated Research Programmes (CRP) on "Assuring structural integrity of reactor pressure vessel" and on the "Management of ageing of in-containment I&C cables".

The OECD Nuclear Energy Agency (NEA) has an Expert Group on Nuclear Power Plant Life Management (PLIM) with the objectives of clarifying essential elements of plant life decisions and reducing the associated safety, technical and economic uncertainties. In addition, the NEA has some working groups with competence in ageing effects. A CSNI Working Group has generated several documents on ageing effects on safety related components, but passive components studies (RPV, piping, etc.) are of competence of another Working Group, supported by three sub-groups (metal, concrete, and seismic SG) to broaden the scope and emphasise ageing concerns. The main areas of activity of the metal sub-group are structural integrity, plant ageing and service life management and NDE. Current activities include a piping fatigue crack growth benchmark that continues the plate bending fatigue benchmark. The sub-group on the ageing of concrete structures plans to continue an effort that concentrates on evaluating the uncertainties associated with the current inspection practices and the response of a structure to accident or severe environmental loads as a result of degradation. The aim is to develop tools to obtain time-dependent fragility of degraded structures so that the staff can readily ascertain the impact on risk and safety and quickly establish a threshold for regulatory concern.

In the European Union research co-operation is organised in European networks. Specialists exchange their new results and discuss about ageing strategies. The management of the networks initiates specialists meetings, co-operative programmes with in-kind contributions from the partners, concerted actions and projects within the EU framework programmes.

- AMES (Ageing Materials Evaluation and Studies) is co-ordinated by the JRC-IAM (Petten) and relies on irradiation and annealing data available by the members or to be generated.
- ENIQ (European Network for Inspection Qualification) is an effective way of harmonising inspection requirements with the objective of being equally open to any inspection technology.
- NESC (Network for Evaluating Steel Components) connects the NDT methods and material aspects with the structural integrity assessment (co-ordinated by JRC-IAM, Petten)
- EURIS (European Network of Risk-Informed In-Service Inspection) deals with the analysis of different approaches to the estimation of failure probabilities of components and of the failure consequences, collecting operator feedback.

In the Eastern European countries major ageing issues of safety significance are the RPV embrittlement of the VVER plants and the IGSCC in the piping of the RBMK plants. The Russian reactor annealing technology has been evaluated and demonstrated [6]. Additional monitoring systems for vertical displacement of reactor body, lost part and vibration monitoring systems were installed in most NPPs of the Eastern European Countries. Some plants have installed leakage monitoring systems, fulfilling the requirements of LBB concept.

In the United States some committees were created in order to collaborate in the field of ageing between industry, research and regulatory organisations. The Electric Power Research Institute (EPRI) is performing the technical, regulatory and economical programme for nuclear plant Life Cycle Management (LCM). The LCM programme is directed toward providing utilities with the data, information and tools needed to make decisions concerning long term nuclear plant operation. The LCM programme is closely connected to the Plant Lifetime Improvement programme (PLIM) and the Commercial Operating Light Water Reactor programmes (COLWR) co-ordinated by the US Department of Energy (DOE). The DOE co-operative effort with the nuclear industry is to develop technology to manage the effects of material degradation that have an impact on plant safety or can significantly improve plant performance or economics and to establish and demonstrate the license renewal process. As a result of the Nuclear Regulatory Commission (NRC) activities related to license renewal, Environmental Qualification (EQ) of electric equipment was identified as an area that required further review. A major concern related to EQ was whether the requirements for older plants were adequate to support license renewal.

In Japan the activities related to NPP ageing issues have been organised and implemented by the Ministry of International Trade and Industry (MITI) and the Nuclear Safety Commission (NSC). The Japan Atomic Energy Research Institute (JAERI) has been conducting research authorised by the NSC. The main subject of JAERI activity has been RPV ageing. MITI instructed the Japan Power Engineering and Inspection Corporation (JAPEIC) to co-ordinate a PLEX Technology Development Programme [7]. In addition to the PLEX programme, JAPEIC has also been involved in other projects related to NPP ageing, i.e. advanced NDE technology, structural assessment of flawed structures, and inspection/maintenance technology assessments.

7.2 Needs and challenges

In the CNRA report on “New Future Nuclear Regulatory Challenges” the corresponding regulatory challenges are summarised as follows [10]:

- to have an adequate knowledge of the design basis of the plant;
- to have a correct picture of the actual state of the plant, through periodic tests, in-service inspection and feedback of operating experience, in order to repair or replace aged components and maintain the design basis; and
- to define the analyses needed to support life extensions and demonstrate that the plant will still operate within its design basis.

Consequently the current research objectives are defined as follows:

- to develop qualified methods for detection, surveillance and control of degradation, including methods for early detection;

- to gain a better physical and microstructural knowledge of ageing related damage mechanisms;

- to develop predictive models based on the “true” damage mechanism of the microstructure to extrapolate behaviour of components and structures; and

- to provide the scientific basis for possible mitigation methods.

7.2.1 Short-term needs

Irradiation embrittlement and thermal ageing

The major unsolved issue in assessing irradiation embrittlement is the lack of general consensus to understand irradiation mechanisms. The copper and phosphorus contents of the steels are considered to be the main factors responsible for neutron embrittlement. Recent analysis of data indicates that copper and nickel have a synergistic effect. The parameter to describe the embrittlement level should be generally agreed and qualified. The effect of thermal annealing on the recovery of embrittlement is a matter that cannot be explained from the mechanistic point of view. The embrittlement rate of thermally annealed reactor pressure vessel steel seems to differ from that before the annealing.

The ageing management programme activities that address radiation embrittlement and mitigation methods should include the following:

- Radiation embrittlement databases/trend curves to predict the degree of radiation embrittlement.

- RPV materials radiation surveillance programme.

- Thermal annealing.

- Post-annealing surveillance programme.

- Re-embrittlement rate after thermal annealing.

Current projects within the EU Framework Programme FP-5 deal with embrittlement issues. The work proposed in FRAME will focus on the development of a method that allows direct measurement of fracture toughness. Within RETROSPEC work will be done to improve the evaluation of the neutron doses induced in reactor structural materials in those cases where no or unreliable data from surveillance specimens are available (for example the older generation of VVER-440 type reactors). The objective of PISA is to better understand the role of phosphorus in the embrittlement process of RPV steels.

Experimental irradiation is undoubtedly the research theme in which international co-operation is most effective. The research tools needed for the irradiation of existing materials or prospective materials are NRU (Canada), OSIRIS (France), FNR (USA), BR2 (Belgium), Halden (Norway), HFR (The Netherlands), LVR-15 (Czech Republic), and HANARO (Republic of Korea) experimental reactors.

Research needs

- Clarify the links existing between microstructural changes in the materials and the changes in their mechanical properties.

- Consider non-destructive examination techniques to determine material properties of un-irradiated, irradiated and re-irradiated RPV steels.

- Reduce uncertainties in fluence calculation results.

- Study microstructural changes in accelerated irradiation tests versus normal irradiation.

- Use of RPV material of decommissioned reactors for research purposes.

Fatigue (HCF, LCF, THF)

Fatigue initiation curves indicate how many stress cycles it takes to initiate fatigue cracks in components. There is a need for a better understanding of the material response to complex i.e. variable amplitude and frequency loading. Micro-mechanical damage models are required to characterise the influence of the different parameters. Furthermore, the environment can significantly influence fatigue crack initiation. Thermal fatigue due to stratification and the mixing of hot and cold water is a recurring phenomenon, but with a relatively low frequency. It may be that these phenomena are not included in the design conditions. An original way of coping with thermal fatigue issues is to rely on on-line fatigue monitoring devices.

Research Needs

- Analysis and extension of databases to characterise and consider the influence of environment and variable loading on fatigue behaviour.

- Investigations to characterise material specific microstructural changes due to fatigue.

- Correlation of material specific microstructural changes, like arrangement of dis-locations, formation of martensite, formation of microcracks, to physically based values like lifetime of positrons, magnetic/acoustic Barkhausen signal, eddy current signal, etc.

- Development of advanced models for understanding material response during fatigue.

Environmentally assisted cracking EAC (SCC, IASCC, CF)

There are a number of plausible mechanisms or physical processes that can account for stress corrosion cracking (SCC). No single mechanism is adequate to describe SCC in the variety of materials in which it has been observed. Intergranular stress corrosion cracking (IGSCC) observed in the austenitic piping of some BWRs is a consequence of welding causing heat affected zones leading to sensitised regions. Irradiation assisted stress corrosion cracking (IASCC) is restricted to core internals affecting stainless steel and Ni-base alloys. Whereas the principle factors are known and a mechanistic understanding exists, the interaction of the individual parameters and their specific influence on IASCC is not well cleared. Crack growth data are lacking, but are now being generated in some ongoing programmes. The preferred cracking in the HAZ of components is not satisfactorily explained. The mechanism of IASCC under PWR conditions is not clear and is suspected that it may differ from that of BWR cracking. The mechanisms of corrosion fatigue (CF) and strain induced corrosion cracking (SICC) are mainly connected to start-up/shut-down and feedwater on-off transients. Zirconium-alloy pressure tubes in CANDU reactors can be susceptible to crack growth through delayed hydride cracking (DHC). For DHC to occur there must be a crack starter, enough hydrogen to precipitate brittle hydrides, and a high enough stress. The mechanism has been well characterised and its onset is effectively prevented by ensuring low concentration of hydrogen.

Examples for current research activities in this field are two projects under the EU Framework Programme FP-5 dealing with irradiation assisted cracking of austenitic steels. Within the framework of INTERWELD the radiation induced damages that promote cracking in the heat affected zones of PWR and BWR core internals components will be studied looking at parameters such as neutron fluence/irradiation conditions, residual stresses, microstructural and microchemical conditions. Further work will be performed in PRIS to produce materials data for irradiated stainless steels of LWR internals as a function of fluence up to 70 dpa.

Research needs

- Extension of databases to characterise the SCC susceptibility of different RPV steels under static, dynamic and cyclic loading under water chemistry conditions representative for current BWR operating practice.

- Further development of physical based models for the prediction of SCC in the different construction materials in LWR service environment.

- Development of a database on crack growth of irradiated materials under BWR as well as PWR conditions.

- Experiments to clarify the role of alloy composition on the radiation induced segregation (RIS), in particular of interstitial atoms.

- Investigations in the interaction of irradiation hardening and irradiation creep with residual stresses and its effect on IASCC.

Inspection and monitoring

During in-service inspection of NPP, flaws in the structures have to be reliably detected and also characterised. To be able to determine the criticality of flaws, the non-destructive testing methods must be able to not only assess the size, but also the position, type and orientation of the flaw. Up to now, there is a need to improve the detection and sizing of SCC and fatigue cracks, especially in austenitic and dissimilar welds. The cumulative plant operation data and experience gained from inspection are able to indicate the potential flaw areas. In these areas there is a need to apply versatile combination techniques and extensive post-analysis of inspection data. For example in the EU project SPIQNAR (FP-5), attention will be devoted to improve the inspection performance of ultrasonic inspection aimed at detecting and sizing possibly present cracks in structural components.

A new challenge is the development of diagnostic systems for the monitoring of material degradation. The aim is to develop techniques for the early detection of microstructural changes due to embrittlement, fatigue and EAC. Under the EU Framework Programme FP-4 the concerted action

AMES-NDT was dealing with this issue. Some of the NDT techniques applied such as thermal power measurements, non-linear harmonics analysis, and positron annihilation measurements showed promising results. Under FP-5, this work will be continued within the project GRETE. As already mentioned before, some of the major corrosion problems for LWRs are IASCC damage to core internals and IGSCC damage to piping welds. An important parameter for EAC is the corrosion potential, which should be measured as an on-line monitoring technique inside the reactor. The development of relevant reference electrodes is the objective of the EU FP-5 project LIRES.

Research needs

- Implementation of the validation practice of various NDT methods.
- Advanced techniques for detecting and sizing of EAC and fatigue cracks in welds.
- Development of new NDE techniques for the detection of microstructural changes due to ageing phenomena (embrittlement, SCC, IASCC, fatigue) and continuous monitoring of the degree of ageing. The applications will involve combinations of various techniques, novel data analysis methods and on-line monitoring systems.

Assessment of material degradation and component integrity

The integrity assessment is generally done by using codes such as ASME, RCC-M, RSE-M, KTA, etc. but it is necessary to determine the actual values of the different parameters as they may be different from those used at the design stage of the component. This is especially true for the mechanical properties of materials. It is also necessary to have available well-suited methods that allow demonstrating the structural integrity of reactor components.

The direct measurement of the actual mechanical properties requires significant quantities of the representative material available. The fracture toughness of RPV steels in the ductile-to-brittle transition remains one of the principal concerns in guaranteeing the structural integrity of nuclear reactors. The transferability of fracture toughness value obtained from small size specimens to real component situations (shallow cracks, wall thickness) is an open question. Statistic methods are currently being introduced into the transition temperature characterisation. The objective is to replace imprecise correlation between empirical impact test methods and universal fracture toughness lower-bound curves with direct use of material specific fracture mechanics data. Master Curve Approach is a crucial method for such an improvement.

The use of damage models (local and non-local damage mechanics, microstructural modelling) helps to assess the real safety margins of the power plants and to reduce unnecessarily large conservatism. Examples for current research activities are the EU FP-5 projects VOCALIST and LISSAC. Within VOCALIST the defect assessment techniques will be further improved to better predict safety margins, in particular with respect to the constraint effect (i.e. the pattern of crack-tip stresses and strains causing plastic flow and fracture). Size effects for fracture and strains will be investigated within the framework of the project LISSAC.

Research needs

- The usability of fracture mechanics based material values in structural analyses with respect to constraint must be investigated analytically and experimentally with special emphasis on small test specimens (miniature fracture toughness testing).

- Investigations on size effects in deformation and failure of various materials and structures including theoretical modelling, local approach modelling (GTN models, etc.), non-local damage modelling on the base of the gradient theory of plasticity, small size specimen testing, large scale testing.

- Re-evaluation of the thermal shock experiment results (FALSIRE, NESC Spinning Cylinder tests) based on the Master Curve concept or based on local approach models (GTN model, etc.)

Concrete structures

Instances of concrete degradation have been observed in containment and other structures, and there is a long history of occurrences of degradation over time in industrial and transportation structures. The nuclear industry has adopted regulatory and codified methods for predicting the loss of prestress in concrete containments from international and national standards that are not necessarily specific to nuclear design. The application of the different methods to a specific case is likely to lead to significant differences in the predicted losses.

The theoretical and experimental research has established the importance of understanding how chemical, mechanical and thermal factors influence the short-term and long-term behaviour of prestressed concrete. In particular, they have differentiated between creep, drying shrinkage and relaxation of prestressing steel and identified the interdependency of these phenomena. However, research has, as yet, failed to formulate a universal and reliable model for predicting both short- and long-term loss of prestress in actual prestressed concrete structures.

Comprehensive and regular monitoring of the behaviour of containments and pressure vessels at operational plants assist our understanding of the cause of loss of prestress. Containments around the world include instruments to measure: anchorage loads, concrete strain, structural geometry, concrete temperature, and surface cracking.

Research needs

- Early detection of concrete degradation by means of new non-destructive methods and associated criteria.

- Predictive modelling of the progression of concrete degradation and influence on plant integrity and safety.

- Validation of methods for mitigation and repair.

Cable material

The main parameters causing age-related degradation of cable material are temperature, radiation dose rate and total dose, oxygen, moisture, mechanical stress, ozone, and contaminating chemicals. For many of the polymers of interest, oxidation is the dominant ageing mechanism and is initiated both thermally and by irradiation. In PVC, which was widely used in cables in older NPPs, the loss of plasticiser from thermal ageing is an important degradation mechanism. Both oxidation and plasticiser loss can result in the embrittlement of cable material, increasing the probability of cracking of the insulation under mechanical stresses. Recent indications of broken PVC insulation in older NPP illustrate the safety-related and economical significance of the ageing of cables.

Research needs

- Further development of condition monitoring methods.
- Methods for assessment of residual life.
- Comparison of lifetime prediction models with real-time ageing.
- Realistic failure criteria.

7.2.3 Longer-term needs

Even when current issues are resolved, core expertise will need to be retained to address future problems that may occur during the operation of stations. While the mechanical behaviour of materials is a field that extends beyond the industry, special expertise in nuclear materials and effects of the nuclear environment, especially irradiation effects must be maintained. A long-term programme studying degradation mechanisms is a vehicle for retaining expertise.

Mechanical testing of irradiated material is a special skill which needs to be maintained. This could be accomplished through continued surveillance programmes involving the testing of components removed from reactors. Clearly, hot cells will need to be maintained for this programme at a time when cells are being shut down in many countries. Hot cells are an essential requirement if a country is to function in the nuclear industry and it could be risky to rely too much on a foreign supplier of hot cell services given the urgent nature of some examinations. Opportunities to examine materials that have seen a long service in reactor will occur as reactors are decommissioned. International co-operation is the route through which many nations could share in the expense of expanding our knowledge of ageing processes by testing such materials.

To support the development of new materials for reactors, especially advanced designs, access to a reactor for test irradiation is important. These do not necessarily have to be in every country as specimens can be sent abroad for irradiation.

Advanced analytical instruments need to be used for microchemical and microstructural material characterisation, such as Atom Probe Field Ion Microscopes (APFIM), Field Emission Gun Scanning Transmission Electron Microscopes (FEGSTEM), Auger-, ESCA- and SIMS- equipment, SAXS at synchrotron light sources, SANS at neutron sources. The development of advanced diagnostic systems for the monitoring of material degradation is a chance and challenge for multidiscipline and international co-operation.

There is a long-term need to do more to qualify electrical cables for full design life and any license renewal periods, or at least to determine the useable life.

7.3 Summary and recommendations

The optimal ageing management of reactor materials requires substantial knowledge on degradation phenomena and evaluation techniques. There exists a substantial quantity of data and information on ageing, but the information is on different organisations, different countries and diverse formats. It is necessary to collect the data and to homogenise it.

7.3.1 Facilities potentially interesting for international collaboration

Most facilities for materials testing, including standard and medium-scale testing machines, hot cells and EAC rigs, exist in the Member countries and there is no need to identify a central facility that might be operated co-operatively. However there are very few facilities for testing very large specimens and the issue of predicting failure of large structures from small size specimens is not yet fully resolved. If the maintenance of such large testing facilities is too great a burden for any one country, then interested parties should consider ways of operating one facility for the benefit of all countries through international partnerships. CSNI Working Groups have been a focus for such activities in the past. Table 7.1 shows large facilities available for international co-operation.

7.3.2 Cost information for potential international facilities

Typical annual costs for operating and maintenance of large structural integrity facilities range from 8 million USD/a for the material test reactor, 800 000 USD/a for the EAC rig in the material test reactor, 920 000 USD/a for the hot cell with analytical tools for PIE, and 525 000 USD/a for the hot cell with mechanical test equipment.

7.3.3 Specific recommendations

For critical needs in the area of structural integrity assessment (ageing of NPP components with safety significance) information and knowledge should be available. It is proposed to better co-ordinate the activities and experiences in qualified data banks; prevention based systems, international co-operating groups. Therefore, the CSNI Working Groups should define possibilities of such integrating structures and propose a plan of actions.

Recommendation 7.1: Define a plan of action (CSNI Working Groups) for the development of an international data bank and Centre of Excellence for materials ageing data and analytical tool, for example to better define safety margins, with the aim of attracting industry and government agency participation.

In the past, single ageing processes have been investigated predominantly with regard to safety, e.g. neutron and thermal embrittlement, environmentally assisted cracking and fatigue phenomena. The measures taken to cope with all the ageing mechanisms regarding continuous monitoring, inspection, surveillance and maintenance were mainly related to collect information using empirical approaches. From today's point of view, structural integrity must be optimised by the implementation of advanced tools and methods like qualified methods for ISI and the monitoring of degradation, predictive micro-mechanical models, more-dimensional phase diagrams (temperature, pressure, concentration, time), and prevention based systems and risk informed methods.

Recommendation 7.2: Focus CSNI and CNRA activities on the synergistic interaction between all the ageing mechanisms and its influence on continuous monitoring, inspection, surveillance and maintenance, initiate a strategic debate over the need to co-ordinate and develop an overall international research and training programme in materials ageing issues relevant to the nuclear industry, especially those with safety implications.

Recommendation 7.3: Continue and start benchmarks (CSNI Working Groups) to investigate the impact of boundary conditions, like thermal-hydraulics, temperature variations, radiation, chemistry, for different operational and accident situations on the integrity of aged components.

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Table 7.1**. Large facilities available for international co-operation**

Research need	Facility (Country)	Specification
Material test reactors[1]	NRU (Canada) OSIRIS (France) FNR (USA) NRI Rez (Czech Rep.) JMTR (Japan) BR2 (Belgium) Halden (Norway) HFR (The Netherlands) LVR-15 (Czech Republic)	nearing end of life
Hot cells[2]	ANL (USA) ORNL (USA) AEA T (UK) Sellafield (UK) CEA (France) EDF (France) ITU Karlsruhe (JRC) JAERI (Japan) NRG (The Netherlands) NRI (Czech Republic) AEKI (Hungary) PSI (Switzerland) CIEMAT (Spain) SCK/CEN (Belgium) VTT (Finland) AECL (Canada)	

1. A new materials testing reactor is planned in France (Joules Horowitz) which will be available later.
2. These only include large facilities with mechanical testing capability.

Table 7.1. **Large facilities available for international co-operation (continued)**

Research need	Facility (Country)	Specification
Mechanical testing	NIST (USA)	large test facility for piping
	MPA Stuttgart (Germany)	100 MN universal test rig
	EDF (France)	bimetallic weld testing
	ISI Ispra (JRC)	large Hopkinson impact facility
	Skoda (Czech Rep.)	80 MN universal test rig
		large test facility for vessels
	SGTI (USA)	leak and burst testing of steam generator tubes
Inspection qualification[3]	Swedish Qualification Centre	
	PAKS NDE Centre (Hungary)	
	Tecnatom (ES)	
	VTT (FIN)	
	AEA T (GB)	
	NRI (Czech Republic)	
	Pacific Northwest NL (USA)	
	EPRI (USA)	
	IAM Petten (JRC)	
	MPA Stuttgart (Germany)	
	Prometey (Russia)	
	Skoda (Czech. Republic)	

3. Large test sections for qualification are available at all of these laboratories.

Table 7.1**. Large facilities available for international co-operation (continued)**

Research need	Facility (Country)	Specification
EAC laboratories[4]	MHI Takasago (Japan) NRU AECL (Canada)[5]	rig in material test reactor
	Studsvik (Sweden)	
	EPRI (USA)	
	GE (USA)	
	VTT (FIN)	rig in material test reactor
	Siemens KWU (Germany)	4 rigs in material test reactor
	MPA Stuttgart (Germany)	
	IAM Petten (JRC)	
	CIEMAT (Spain)	
	NRI (Czech Republic)	
	PSI (Switzerland)	
Cable testing	CIEMAT (Spain) JAERI (Japan) SNL (USA) OPG (Canada) Siemens KWU(Germany) NRI (Czech Republic)	gamma irradiation facility (Co-60) LOCA testing of aged cables

4. All laboratories simulate either LWR or BWR environments (but not the Canadian specific facility at NRU).

5. For the Canadian-specific environment.

Table 7.2**. Template for Chapter 7 – Integrity of equipment and structures**

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 7	Needs and challenges	Safety significance	Sufficient knowledge available ?	Fap essential ?	Need for CSNI action ?
Embrittlement (irradiation and thermal induced)	• better description of embrittlement status (using the fracture toughness) • influence of impurities (Cu, Ni, P) • monitoring technique	H M H	Yes No No	Yes Yes Yes	Yes [3)] No No
Fatigue (HCF,LCF,THF)	• database for THF and CF • microstructural models • monitoring technique	M M M	No No No	Yes No No	Yes [1),2)] Yes [3)] No
Environmentally Assisted Cracking (SCC,CF,IASCC)	• database for SCC and IASCC • role of alloy composition on the radiation induced segregation • physical based models	H M M	No No No	Yes Yes No	Yes [1),2)] No Yes [2)]
Inspection and Monitoring	• implementation of the validation practice of various NDT methods • detection and sizing of EAC and fatigue cracks • new techniques for monitoring of embrittlement, EAC and fatigue	H M M	Yes Yes No	Yes Yes No	Yes [2)] Yes [3)] Yes [3)]
Assessment of Integrity	• more realistic computation • micromechanical models for size effects for fracture and failure	H M	Yes No	No Yes	Yes [2),3)] No

Chapter 8

SEISMIC BEHAVIOUR OF STRUCTURES

8.1 Background

For the seismic evaluation of structures in nuclear power plants, very extensive work is required in areas such as definition of design ground motion, wave propagation analysis, structural response analysis, equipment response analysis and the resultant stress evaluation. Topics and research needs in the seismic area have not been previously discussed in Senior Group meetings [1], but have been addressed in the Task Group on Seismic Behaviour of Structures [2] which supports one of the CSNI Working Groups. In the Task Group, the following five seismic areas have been given priority.

8.1.1 Piping analysis and design

Extensive research on the seismic response of piping systems has been performed. A result of those studies is that current seismic design practice requires excessive support structures and it is therefore believed that these piping systems have a large seismic margin. In order to evaluate the margin, elasto-plastic analysis is required to simulate piping behaviour under large deformation, but an effective and practical way of performing this kind of analysis, based on test data, is still a research problem. Therefore it is necessary to study more effective and rationalised methods of elasto-plastic analysis for the seismic response of piping systems.

Topics in this area include:

- damping factors for design analysis;
- response characteristics under multi-direction input motions;
- elasto-plastic analysis methods; and
- ageing effect evaluation of piping structures.

For example, there is a PWR primary loop piping test by NUPEC and USNRC [3] concerning elasto-plastic piping tests using a large-scale model. A special working group organised by ASME has been studying allowable stress re-evaluation to cope with elasto-plastic behaviour, and there is also discussion with USNRC [4]. The same kind of study for piping allowable stress rationalisation is also being implemented by the Joint Study between Japanese Utility Companies and Vendors. In addition, NUPEC has started a test project to evaluate elasto-plastic behaviour of the piping system.

8.1.2 Engineering characterisation of seismic input

The procedure to define seismic ground motion is one of the most difficult in the seismic design of nuclear power plants because it involves many uncertainties. Several records from very large earthquakes were recently acquired. In particular, data from the 1994 Northridge Earthquake and the 1997 Hyogo-ken Nanbu Earthquake has shown that ground motion in the epi-center region seems likely to become a very important issue.

One of the most important issues in the area is the effect of the high-frequency component in earthquake motion and how it should be reflected in response analysis, design and the re-evaluation of design input motion. Large velocity and displacement, due to near field earthquakes, especially in the low frequency region, is also a very important problem for specific site conditions.

Seismotectonic conditions may differ widely among countries, and the definition of the design ground motion will be strongly affected by local circumstances. It is therefore important to exchange information and knowledge in this area.

8.1.3 Ageing effects

Degradation due to ageing is mainly handled from the maintenance viewpoint in most countries. From the seismic point of view, there will be two aspects. One is the effect on structural integrity; the other is the effect on the functional capability of systems and components, such as electrical equipment and seismic snubbers, for example. Both of these are closely linked to the approach to plant maintenance. However, because there is little research in this area, the database of ageing effects is insufficient. It is therefore important to understand the effects of ageing degradation and its potential impact on seismic behaviour. It is therefore important to collect data relating to ageing degradation now and analyse its effect on seismic capability.

This issue is also important in relation to maintenance, life extension, and the re-evaluation of existing nuclear power plants.

8.1.4 Validation of analysis methods

It is widely recognised in the world that there is a need to establish a benchmark library to be available for seismic analysis and simplified analysis verification of structures and large equipment. A benchmark for non-linear analysis is especially necessary. SSWISP held in Yokohama in 1997 [5] effectively demonstrated the needs in this area. These kinds of activities should be progressed for large equipment, soil-structure and structure-structure interaction analysis.

8.1.5 Re-evaluation of existing facilities

There is world-wide interest in this area and proper and efficient performance of such assessments may have very important safety and economic considerations.

Important knowledge can be gained by studying the lessons learned from past seismic PSAs and the safety margin analysis from plant re-evaluations. A world-wide database of the results of such studies would be very useful.

8.2. Needs and challenges

The state of the art and research facilities in each area are discussed in this section.

8.2.1 Piping analysis and design

A lot of research is currently being carried out to upgrade the presently available conservative design methods or practices from the safety and economic viewpoint. For example, USNRC/EPRI/ASME are implementing a study concerning the ultimate strength of piping. In this study, piping elbow elements with a short strait pipe are excited on a sliding raft to study the behaviour of each element under higher stress levels in order to determine allowable stress. In Japan, a similar study is under implementation jointly between electric power companies and manufacturers using shaking tables with less than 100-ton capacity. NUPEC has just begun a study to verify this issue by using a large-scale model on its 1 000-ton shaking table. The shaking test will be performed in 2002.

In France, CEADRN is implementing piping system tests using the shaking table of 100-ton capacity.

8.2.2 Engineering characterisation of seismic input

In general, seismic input motions are defined as a standard response spectrum, site by site, taking into account historical earthquakes, the distribution of active faults and seismotectonics in order to define the maximum earthquake near the site. As accumulation of strong motion data such as the1994 Northridge Earthquake and 1995 Hyogo-ken Nanbu Earthquake became available, the vertical motion and frequency content of near field earthquakes, i.e. earthquake characteristics in the fault region, become more important.

Specialists will have to exchange the implications of this kind of information on the generation of the design spectrum including the development of design aspects based on a probabilistic design hazard analysis.

An OECD workshop on the Engineering Characteristics of Seismic Input was held in November 1999.

8.2.3 Ageing effects

There are some differences between countries on how to deal with this issue.

One position is that because of very thorough annual and even daily inspections and maintenance, ageing does not affect seismic capability. Therefore ageing is not a critical issue for these countries. The other position is that ageing will have some effect on seismic capability, because it can affect concrete and other structures and so should be examined. For these countries information exchange should be done actively.

For both positions, it is important to exchange experiences and data because events of this kind are very rare and there does not exist a satisfactory database.

An OECD workshop on degraded concrete structures was held in November 1998. Its outcome should lead to expert advise on future requirements and actions.

8.2.4 Validation of analysis methods

Many research programmes are being carried out with experimental work to develop and validate seismic analysis. In particular, extensive programmes have been done in the piping area. For example, medium size piping tests have been performed by CEA [6], elasto-plastic behaviour test of pipe elements by EPRI/BNL and actual size piping system tests by NUPEC [7], to mention just a few of the many tests which have been carried out.

In the area of soil-structure interaction, fewer experiments have been performed. Typical of tests in the area are a field test in Hualien and a field test by NUPEC. In the latter test, large-scale concrete models were used to verify and improve soil-structure analysis codes. As part of the series of NUPEC soil-structure tests [5], structure-structure interaction tests and small-scale tests by shaking table are also under implementation.

The results of these tests will provide a good database for an International Standard Problem.

Studies of data on the seismic base or component isolation are also good candidates for exchange. Such studies have been implemented by EPRI, ANL and CRIEPI. A component base isolation study is currently being carried out by JAERI. This type of information also has important possibilities for international data exchange.

8.2.5 Re-evaluation of existing facilities

There are some differences in attitude among countries according to their seismic design history. In countries where seismic design was introduced at the initial phase, this issue does not cause severe problems, but in those countries where it was not, or where they had adopted small accelerations, this is potentially a big problem.

Re-evaluations of the plants in question have been completed and knowledge from these studies will be useful. We recommend that they be used to form a database.

8.2.6 Short-term needs

Shaking tables[1] of tens of tons capacity exist in many countries for nuclear safety research. However, tables of hundreds of tons are much rarer and only exist in a limited number of countries, specifically Japan, U.S., Russia and France.

The five hundred to one thousand ton class is very limited indeed and only two tables exist, both of them are in Japan. One is the five hundred-ton table of the National Research Institute for Earth Science and Disaster Prevention; the other is the one thousand-ton table of NUPEC [8]. The former is a one axis, and the latter a bi-axial table. (*Refer to Table 8.1*)

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1. We note that large scale "reaction walls" also exist and are used for stiffness measurements and more detailed structural response experiments, but are not treated further here.

It is necessary to validate analysis methods and codes applicable in the plastic region for the rationalisation of piping seismic design and evaluation of existing plant. For this purpose, real size tests are very effective and necessary.

Shaking tables of less than several hundred tons capacity are more readily available and this class of table is utilised for various industrial purposes. It is felt, therefore that there is no present need to consider their future availability. On the other hand, the Tadotsu one thousand ton Table is the only one that can handle large size specimens. In addition it has very good high frequency characteristics (up to 30 Hz). This table is adequate for the tests which require two directions of input motion (one vertical, the other horizontal). This table has an extensive experience of large-scale tests [8,9] of nuclear equipment including international co-operative work. However, its use has been decreasing in recent years. Because there is a possibility that this table cannot be retained in the future, it is recommended to raise its operating rate by international collaboration.

8.2.7 Longer-term needs

Although the design and construction of new reactor types which might need novel mechanical or seismic testing has been at a very low level in recent years, in the future it will still be necessary to continue large size shaking tests *as shown in Table 8.2*. The relevant test facilities are recommended to maintain their capability and prepare for future needs. To sustain this, it is recommended that an international test programme be planned which takes a long-range view of the future needs.

8.3 Summary and recommendations

8.3.1 Facilities potentially interesting for international collaboration

The main area of concern is the shaking test facilities. These are normally general-purpose facilities and are used for various types of industrial equipment and structures. There are very few facilities exclusively used for nuclear testing.

Shaking tables with less than one hundred tons capacity are widely available in the world as shown in the attached table. From the name of the organisations owning those tables it is apparent that they are used for general purposes. Because of this, there seems to be no need to worry about keeping up these test facilities in future. On the other hand, regarding the shaking table class with a capacity of a thousand tons, only one table, the Tadotsu Shaking Table [8], exists in the world, and it is used exclusively for nuclear equipment and structures.

With respect to shaking test needs in future, full size tests are still necessary for seismic margin tests, degraded structure tests for testing structures and components to failure. It is imperative to keep this kind of capability to test full-scale and large size specimens.

8.3.2 Cost information for potential facilities

The very large-scale shaking table mentioned above is mainly used for nuclear equipment tests and only to a lesser extent by other industries. Although the basic funding for keeping the facility may have to depend on national needs and resources, international support by direct use of the facilities or by other means of co-operation for specific experimental programmes is now becoming more important. Activities to identify the potential for common research subjects and promotion of co-operative work to use the large shaking table on identified common issues are now necessary and worthwhile.

8.3.3 Specific recommendations

For seismic design, there are differences among countries according to their earthquake environments. Existence or not of new plant projects also affects countries' attitudes on the needs for seismic research. According to local circumstances, each country has different interests and this is reflected in the priorities given to such topics as: re-evaluation of existing power plants, research into methods for economic improvements, fragility evaluation for seismic PSA, and so on.

In order to identify common interests, it is necessary to encourage international information exchange and collaboration, and to improve international capability of seismic design through workshops, ISPs and sharing of earthquake observation data. It is also important to promote seismic test programmes requiring large-scale tests by selecting from these common issues projects which would be attractive as international collaborative research exercises. Public acceptance is also an important factor.

8.3.4 Other possible forms of international collaboration

Because the seismic capability of equipment differs according to their design conditions, seismic evaluation results are not always directly adaptable for use in other countries. By clarifying these differences, it is recommended that efforts be made to utilise test data and test information from different countries as a common database. It is important that this type of collaboration is given a try. In this case, a "virtual" Centre of Excellence would seem to be the most appropriate form of collaboration.

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Table 8.1**. Some existing shaking tables in the world**

Country	Organisation	Table size (m)	Capacity (ton)	Excitation direction	Max. acceleration (Gal)	Max. disp. (cm)	Note
Japan	Nuclear Power Engineering Co.	15 x 15	1,000	X Z	2,660 - 1,330	20 - 10	
Japan	National Research Institute for Earth Science and Disaster Prevention	15 x 14.5	500	X	1,960 - -	22 - -	
Japan	Public Work Research Institute, Ministry of Construction	8 x 8	Rated 100 Max. 300	X Y Z	1,960 980	60 60 30	Hybrid Vibration Sys.
Japan	Railway Technical Research Institute	12 x 8	200	X	400 - -	5 - -	
Japan	Mitsubishi Heavy Industries	6 x 6	100	X Y Z	1,960 1,960 1,960	5 30 4	At 60 ton load
Japan	Central Research Institute of Electric Power Industry	6 x 6.5	100	X	980 - -	5 - -	
France	CEN Saclay	6 x 6	100	X Y	980 980 -	12.5 12.5 -	
Russia	Hydroproject Research	6 x 4	100	X	1,470 - -	5 - -	
Russia	AtomenergoExport	6 x 6	50	X Y Z	1,170 1,170 980	10 10 5	
Iran	Arya Hehr(Univ. Tehran)	5 x 5	50	X	590	5 - -	
U.S.A.	University of California	6 x 6	45	X Z	660 - 215	15.2 - 5.1	
Yugoslavia	University of Kiril & Metodij	5 x 5	40	X Z	660 - 390	15.2 - 5	
Italy	Istituto Sperimentale Modelli e Strutture	4 x 4	30	X Y Z		10 10 10	
U.S.A.	Wile Laboratories	6.1 x 5.5	27	X Z	1,960 - 1.960	7.6 - 7.6	
China	Water Conservancy and Hydro-electric Power Research Institute	5 x 5	20	X Y Z	980 980 780	4 4 3	
Mexico	National University of Mexico	4.5 x 2.5	20	X	1,170 - -	5 - -	
Iran	University of Pahlavi	4 x 4	20	X	1,080 - -	5 - -	
China	Tongji University	4 x 4	15	X Y	1,170 780 -	10 - -	
Greece	National Technical University	4 x 4	15	X Y Z	1,470 1,470 2,650	1 1 1	

There are many shaking tables less than 100-ton payload in Japan in addition to those listed.

Table 8.2**. Template for Chapter 8 – Seismic behaviour of structures**

Criteria for inclusion:

Overriding Criteria

- Is there a major short-term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Technical area	Future needs and challenges	Safety significance	Sufficient knowledge available	FAP essential	CSNI action needed
1) ***Seismic Analysis & Design of Piping System*** • Response Behaviour in Elasto-Plastic Strain Area and destructive stage condition	• Conducting destructive tests for piping elements and prototype piping system, and evaluating/confirming safety margin. Proposal of new rationalised allowable stress criteria.	H (1)	No(2)	Yes	No(10)
2) ***Ageing and Seismic Capability*** • Strength and Capability of Degraded Equipment and Piping	• Verification of capability of aged equipment and piping for life extension of NPP by real size tests	H(3)	No(4)	Yes	No(10)

Table 8.2. **Template for Chapter 8 – Seismic behaviour of structures (continued)**

Technical area	Future needs and challenges	Safety significance	Sufficient knowledge available	FAP essential	CSNI action needed
3) ***Validation of Analysis Method***					
• Elasto-Plastic Behaviour of Concrete Structure	• Verification of elasto-plastic behaviour of reactor building under multidirectional forces for improvement of reliability by large scale model test	H(5)	No(5)	Yes	No(10)
• Behaviour of Isolated Structure	• Verification of seismic behaviour of base isolated NPP by large scale model	H(6)	No(7)	Yes	No(10)
• Seismic data	• Generation of collaboration on seismic data	H	No	No	Yes
4) ***Others***					
• Ultimate Capability of Equipment (Except piping system)	• Confirmation of ultimate strength by real size tests for quantitative evaluation of safety margin and rationalisation of design	H(8)	No(9)	Yes	No(10)

(1) Making new rational allowable stress criteria through clarifying destructive limit.
(2) No effective and economical method to evaluate a large number of piping. There are several piping element tests but no prototype piping system test.
(3) Confirmation of seismic capability of aged facilities is necessary to get not only engineering data but also public acceptance.
(4) Ageing effect is mainly evaluated by analysis but few seismic tests.
(5) Several one dimensional tests but no multidirectional test.
(6) Verification of effectiveness of base isolated system.
(7) No actual size system test although simulation technology exists.
(8) Needs from seismic PSA and existing NPP capability evaluation.
(9) Very few destructive test and fragility test data unde very high acceleration.
(10) However, it is necessary to monitor large shake table availability.

Chapter 9

RISK ASSESSMENT

9.1 Background

Risk assessment is an important subject for both regulators and utilities and they will need to maintain and develop the necessary expertise. As a technical discipline, risk assessment utilises the input of many areas of research, especially when levels 2 and 3 are included. However, it does not of itself require specific facilities, although a capability in the methods is certainly a requirement for regulators and utilities. This chapter, therefore, concentrates upon the capability aspects. It is recognised that, to a large extent, Member countries will maintain their own expertise, both to produce and review PSAs according to the importance given to the methodology in the particular regulatory framework. There are nevertheless areas of common interest, such as collaboration on data exchange, criteria and standards that we highlight below. In addition there is a clear need for Member countries to keep abreast of current developments for example in "living PSA" and, more recently, "risk informed" methods both for maintenance and for regulation. It is these aspects which are the focus of the chapter, and which fall most naturally into SESAR/FAP's terms of reference.

9.2 Needs and challenges

As PSA information is used more extensively by utilities and regulators, to both improve safety and optimise plant operation, it is increasingly important that the PSA tools and data be accepted and their strengths and limitations understood. In this regard, the challenges include establishing criteria on PSA quality, understanding any deficiencies in the PSA and ensuring that decisions appropriately compensate for such deficiencies. These difficulties could include, for example:

- not using plant specific data;
- not modelling all important systems or initiating events;
- not modelling important contributions such as human actions or common cause failures;
- not modelling important modes of operation such as shut down conditions.

Accordingly, the following research areas have been prioritised in the SESAR/FAP report.

Level-1 PSA

- Efforts will continue on data collection and analysis schemes as well as up-date methodology development.

- Methods for human reliability and common cause failures are areas to be further developed. An example is the USNRC's ATHEANA programme on human reliability modelling.

- Uncertainties caused by modelling and assumptions should be more systematically studied and quantified.

- Improved methods of prediction of failure mode and frequency of structures and components need to be developed for new reactor designs using passive safety systems.

- Improved methods for analysis of digital and/or programmable software based control systems are needed.

- Techniques for modelling ageing effects of equipment should be developed, evaluating plant operational experience in order to identify the various ageing factors.

- User-friendly Living PSA and Risk Management (RM) tools should be further developed in order to reflect the rapid changes which the industry is undergoing.

- In some countries the development of "risk informed" performance based assessment methods is seen as being of high importance.

- A database corresponding to shutdown periods should be established by carrying out systematic analysis of plant shutdown records.

Extended PSA Models, e.g. *external events and internal fires*

- Models for fire propagation and fire countermeasures require validation and refined physical models for key phenomena. Databases for fire safety analysis need strengthening. The increased potential for fires during shutdowns is of special importance.

- Continuous efforts to reduce uncertainties in extended PSA models are needed by standardising seismic hazard curves and techniques for fragility analysis.

- Level-2 PSA and Level-3 PSA

- Uncertainties caused by modelling and assumptions should be more systematically studied and quantified.

- Results of safety research on severe accidents should be incorporated into Level-2 PSA.

- The possibility of extending living PSA and RM to the scope of Level-2 PSA should be considered.

- Further discussion is needed how Level-3 PSA should be developed to make it useful for safety related decision making and to the choice of probability based acceptance criteria.

9.2.1 Level-1 PSA

It is the normal pattern in each Member country that utilities provide the analysis results of Level-1 PSA in response to the regulatory requirement or voluntary activities and that the regulatory authority reviews them [1,2].

For this purpose, most countries adhere to the spirit of the IAEA Level-1 PSA guidelines. In some countries, methods used are chosen on a case by case basis e.g. IPE and NUREG-1150. Shutdown and Low Power PSA continue to be regarded as important items.

Efforts are being made to maintain and update data collection in relation to plant specific data. Workshops on the exchange of experience of reliability have been held in Toronto in May 1995 [3] and reliability data collection for Living PSA in Budapest, April 1998. In some countries such as France, Finland and Sweden, data collection is performed systematically under the auspices of the national authority. In the UK, Periodic Safety Reviews have led to increased efforts in plant specific data collection and analysis.

Common cause failure is incorporated into the PSAs but the associated quantification of such contributions does have large uncertainties. Due to the scarceness of relevant plant-specific data a combination of information originating from different sources is necessary. It is hoped that the International Common Cause Failure Data Exchange (ICDE) will improve matters in the future.

Parameter uncertainties may be incorporated into PSA models and propagated by Monte Carlo simulation or if feasible, analytically. Uncertainties caused by modelling assumptions are rarely treated – the use of structured expert judgement can be helpful in clarifying hidden assumptions, inconsistencies and modelling shortcomings.

Root cause analysis and precursor studies are being further developed to overcome some of their inadequacies in the areas of human factors and initiating events.

Concerning ageing, data for components are being accumulated through plant specific data collection. However, as the study of the effect that ageing has on structures and passive components in general has only just started, the method to incorporate ageing effects into PSA technology is also in its early stages.

In contrast with existing reactors, the PSA of future plants will begin early in the plant development, particularly in the design certification stages. Significant changes in the existing modelling will be necessary to cope with the new design features [4].

Digital instrumentation and control is being introduced everywhere and must be included in PSA evaluation. Methodology is under development for the quantification of the reliability and failure modes of programmed systems including the problem of software reliability

The technology of living PSA [5] is quickly gaining applications as a means to reduce human and capital resources without increasing risk. Various countries are now considering developments and applications of living PSA in relation to regulation and some countries have already started partial applications. Four TÜV workshops on Living PSA development and applications and various other seminars organised by the IAEA since 1988 have demonstrated increasing interest in this approach. The concept of RM has various interpretations and must still be clarified in some countries, but a common denominator is the objective of providing assistance on safety and reliability decisions to the plant operators. The number of RM tools is growing and it can be expected that the use of RM and the development of *ad hoc* tools are likely to continue. However, experience gained from the use of living PSA shows that implementation has been most successful when it was deemed a requirement by regulatory bodies. In the long term the motivation of the utilities to support such programmes is an essential factor for their success. Such motivation is realised when PSA can be used to optimise their plant operations.

A number of Member countries have now conducted shutdown PSAs and a peer review of the existing studies has been undertaken to compare the different methodologies. [12]

9.2.2 External events and fires

In most countries, the analysis of internal fires and flooding are included in the Level-1 PSA.

Several computer codes exist for modelling fire growth and spreading, toxic emissions and dispersion and fire/water spray interaction. The IAEA has underway a fire safety project which assembles fire experience and produces guidelines for fire hazard analysis and inspection of fire protection and fire fighting techniques [6,7,8,9]. There is also a CSNI Group for fire PSA, which aims to address fire growth and spread.

Concerning seismicity, development activities on fragility analysis and probabilistic hazard analysis are being performed by several countries including the USA, Japan and Switzerland [10]. In the UK, research into seismic fragilities is underway. **[See chapter 8 on Seismic Assessment]**

9.2.3 Level-2 PSA and Level-3 PSA

Fewer Member countries use Level-2 PSA rather than Level-1 PSA in relation to regulation. However, the number of countries applying Level-2 PSA in relation to regulations is expected to increase in the near future for reasons of general safety, for containment evaluation, and sometimes as a matter of public policy [11].

Level-2 PSA is being used more frequently for the risk assessment associated with severe accidents, but its generalisation in the Member countries depends on the position of the regulatory bodies. It can be expected that a Level-2 PSA will be required for each plant in the near or medium term. It must also be pointed out that new reactor designs like the EPR already include the results of Level-2 PSA insights.

Level-2 PSA studies are essentially based on the results provided by systems codes (e.g. MELCOR, MAAP, ASTEC and THALES) and the uncertainties in those results cannot be greatly reduced without a better understanding of severe accident phenomena. The experiments performed as part of the PHEBUS-FP programme should significantly improve the knowledge in this field. Accident management strategies will help to mitigate the consequences of severe accidents and such strategies also require assessment and PSA.

Despite continuing effort uncertainties still remain large for several phenomena. The use of expert judgement is common in such cases, and it is expected that increasing attention will be paid to its formalisation and to the quantification of associated uncertainties.

Only a few countries have experience in Level-3 PSA. More discussion is needed on the applicability of it in safety-related decisions, especially in the areas of cost and or risk analysis, site evaluation analysis and emergency preparedness.

9.2.4 Short-term needs

To establish the database for Level-1 PSA, reliability data collection of safety system components based on operational experience should be systematically and continuously promoted under the auspices of national authorities. The IRS might pave the way in this field. Standardised analysis methods should be adopted.

Efforts in the modelling of common cause failures (CCF) and the development of related databases will continue. A CSNI Working Group is sponsoring the International Common Cause Failure Data Exchange (ICDE) which aims to collect CCF data from Member countries in a consistent format. Other groups provide technical input to the ICDE programme.

Human reliability analysis methods applicable to plants, together with the continuation of the current fundamental human factor development programme, should be developed so as to be available for practical use. For this purpose, in addition to data from operating experiences, a project using a simulator such as in the Halden Project should be maintained, as described in Chapter 5. Root cause analysis can be helpful in the areas of both common cause failure and human reliability.

The treatment of low power and shutdown situations has emerged as one of the key areas for future work, as well as the treatment of digital and programmable control systems.

Fire propagation analysis and evaluation methods for mitigation/restoration by fire fighting activities require further efforts. Fire simulation codes need refined models for certain key phenomena, such as heat release and flame spreading on cables and equipment. This is part of the work of the CSNI Group for fire PSA mentioned earlier.

Additional efforts are needed to take into account the plant specific features in connection with seismic design as described in Chapter 8.

Results of R&D on severe accidents, such as containment bypass and exceeding the design limits of the containment vessel, are expected and they should be incorporated into Level-2 PSA analysis when available.

9.2.5 Longer-term needs

It is expected that Level-1 PSA technology will be improved through practical use in application to, or in relation with regulations.

Databases should not only be maintained but also improved based on operating experience.

Expertise for the development of Level-2 PSA technology based on the results of severe accident R&D should be maintained. n addition, methods for eliciting expert judgement and for quantifying uncertainties require continuing attention.

International consensus is desirable for the application of Level-3 PSA in areas such as site evaluation and emergency preparedness, and hence for the formulation of a consistent opinion on the necessity to maintain expertise in Level-3 PSA.

In the case of Risk Assessment maintaining the capability and the facilities is understood to mean maintaining the total expertise of PSA. Both utilities and regulators need to maintain their competence to play their roles correctly.

Utilities should perform Level-1 PSA analyses in response to regulatory requirements or voluntarily, and are expected to extend their application of PSA techniques in operational management in response to regulations regarding performance and risk-informed management.

Their co-operation is indispensable for reliability data collection as such data can only be obtained from operating experience. Except where commercial needs for confidentiality are strong the resulting data should be made available for research.

It is the minimum requirement of a regulatory authority to establish criteria on PSA quality and to maintain the capability of reviewing the results of utilities analyses and evaluation of operating data. To perform their duties with sufficient impartiality the regulators should not depend on the utilities for their PSA expertise. Mechanisms for supporting PSA – related research and for incorporating its results in the regulatory process will need continuing support on a national basis.

At the international level, the OECD/NEA contributes to information exchange and to establishing international consensus in the field of risk assessment through the various task force activities of the CSNI Working Group. This group contributes to the development of PSA in the Member countries. To date this has been done by producing reports on the state-of-the-art on topics which are important in carrying out PSA. At present, consideration is being given to carrying out activities aimed at developing PSA methods. This is the case for the current task on human errors of commission. Consideration is also being given to carrying benchmark type studies on particular aspects of PSA. Also relevant in this respect is COOPRA that is a series of agreements between the USNRC and the governmental organisations from a number of countries to review and share PSA based research. One way COOPRA may develop is by establishing a central database of research reports. The majority of COOPRA participants see the principal way forward as providing their own work in kind, although joint funding of research projects is also possible.

9.3 Summary and recommendations

Risk assessment is an important subject for both regulators and utilities and they will need to maintain the necessary expertise. The template outlines the challenges which it classifies as of either high or medium safety significance. Further knowledge is necessary and in all areas facilities and/or expertise is essential. With one exception, which is in the area of data collection and analysis, no proactive action is required. Each Member country should maintain technical capability in this area.

Co-operative efforts to develop methods, standards, and criteria and share experience should be pursued. Of particular note here are the CSNI Working Group activities (e.g. on fire PSA) and the recent COOPRA collaborations. Pooling and sharing of data and the analysis of data are also encouraged. In this regard, the ICDE programme for common cause failure and the continuation of the reliability data workshops, are valuable. There are no facility needs associated with this area.

It is recommended that the CSNI Working Group on Risk Assessment be charged with developing methods and standards criteria, and to ensure the sharing of experience is encouraged, including strengths and limitations of using PSA.

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Template for Chapter 9 – Risk assessment

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 9: Risk assessment	Future needs and challenges	Safety significance?	Is sufficient knowledge available?	Is a facility essential?	Is CSNI action needed?
Data collection and analysis	• Collection and update of data • Making the best use of data based insights • Validation of assumption • analysis of plant shut down record	H	No	Yes (expertise only)	Yes
Living PSA and risk management	• User friendly living PSA • Risk management tools	H	No	Yes (expertise only)	No

Chapter 9: **Risk assessment**	**Future needs and challenges**	**Safety significance?**	**Is sufficient knowledge available?**	**Is a facility essential?**	**Is CSNI action needed?**
Plant ageing	• Modelling ageing effects • Evaluating plant operational experience to identify ageing factors	M	No	Yes (expertise only)	No
Extended PSA models	• External events • FIRE • Physical phenomena • Seismic hazard curves and fragility analysis	H	No	Yes (expertise only)	No
Level 2 and 3 PSA	• Incorporation of research on severe accidents into level 2 PSA • Extending living PSA and RM to Level 2 PSA • Level 3 PSA for safety related decision-making.	Medium	No	Yes (expertise only)	No

Chapter 10

FIRE ASSESSMENT

10.1 Background

Fires can be significant contributors to nuclear power plant risk. The most important events are those involving the occurrences of relatively infrequent fires whose location and severity are such that critical sets of plant equipment are likely to be damaged by such a fire. Serious fire-induced challenges to reactor core cooling are not common events but have occurred. However, the general conclusions regarding the potential magnitude and character of nuclear power plant fire risk appear to be consistent with empirical evidence. Moreover, fire hazards and associated phenomenological research are common to both nuclear and non-nuclear industries. Therefore, most countries do not have experimental facilities that are dedicated only to the nuclear industry.

10.2 Needs and challenges

The nature of research priorities is that they are not necessarily nuclear specific.

- NPPs often rely on fire segregation/separation rules which have evolved through usage and there is a need to underpin these rules and determine their limits of applicability.

- Methods for determining fire growth, spread and interaction with adjacent compartments need to be further developed and validated by experiment.

- Where appropriate, fire databases need to be maintained and improved. This covers a number of areas including: NPP fire frequency, fire detection and suppression system reliabilities.

- Further work is needed on establishing the fire (and smoke) vulnerability of NPP equipment, inc. cables, electro-mechanical devices, i.e. relays, and electronic equipment. Smoke generation and transport issues need further work both in terms of simulation (e.g. CFD) and experimental validation.

- Adequacy of fire barriers, including penetration seals – data to support claims for both active and passive barriers is needed.

- Risk significance of Minor Control Room (MCR), cable spreading room and switchgear room fires also requires further investigation.

10.2.1 Short-term needs

Fire safety is a nuclear regulatory issue in most Member countries and it is normal practice that the utilities provide analysis to demonstrate adequate levels of safety. As in other areas of nuclear safety, there is a need to develop and increase knowledge so that fire safety issues are better understood. This will allow for a more realistic view of the impact of fire on NPP safety to be taken and for any improvements to be better targeted. In this respect, there is a growing trend towards probabilistic treatment of fires within NPP PSAs. While such probabilistic treatment does offer benefits with regard to fire safety, it relies on the same science and engineering as the traditional deterministic approach and thus has the same sort of research requirements. The PSA based approach allows a prioritisation of fire (and other) risk issues, it is not a substitute for fundamental fire research.

In most countries, analysis of internal fires is included in the Level-1 PSA. Several computer codes exist for modelling fire growth and spreading, toxic emissions and dispersion and fire/water spray interaction. The IAEA has underway a fire safety project which assembles operational experience in fire safety assessments of nuclear power plants and also produces guidelines for fire hazard analysis, the inspection of fire protection and fire fighting techniques, refer to IAEA Safety Practices and Safety Guides within the NUSS programme. There is also a CSNI Group for fire PSA (see also Chapter 9), which aims to address fire growth and spread.

10.2.2 Fire research facilities

Experimental and computer modelling fire facilities are available in centres in many countries. Some of the large fire facilities have evolved from needs in the nuclear sector and probably the two most well known are at Cadarache in France and RUT in Russia. Information suggests that where most other countries have extensive facilities and expertise; in general these meet the needs of a number of industries including nuclear.

10.2.3 Additional effort needed

Most countries have their own nuclear fire research programmes, often dealing with the same issues. In many cases there are benefits to be gained from co-operation and collaboration and consideration should be given to the establishment of an expert group by the CSNI to explore how this might be brought about and to look at the requirements for code comparisons in this area.

10.2.4 Longer-term needs

Fire safety research requires both expertise and facilities. In many instances the facilities will be the same as those needed to support general fire safety research and fire safety research in other industries. Nevertheless it would seem worthwhile to compile a list of facilities and their capabilities and review this from time to time.

In the case of fire safety, maintaining the capability means maintaining the total expertise of fire analysis/assessment and the research facilities which are needed to validate models and assist in the development of standards. Both utilities and regulators need to maintain their competence to play their roles correctly.

Utilities should continue to perform fire analyses and assessment in response to regulatory requirements or voluntarily, and are expected to extend their application of new fire modelling and analysis techniques.

The co-operation of utilities is indispensable for fire detection and suppression system reliability data collection, as such data can only be obtained from operating experience. Except where commercial needs for confidentiality are strong the resulting data should be made available for research.

It is the minimum requirement of a regulatory authority to maintain the capability of reviewing the results of utilities fire analyses and evaluation of operating data. To perform their duties with sufficient impartiality the regulators should not depend on the utilities for their fire safety expertise. Mechanisms for supporting Fire-related research and for incorporating its results in the regulatory process will need continuing support on a national basis.

At the International level, the OECD/NEA can contribute to information exchange and to establishing international consensus in the field of fire safety through the setting up of a task force to consider all aspects of fire safety and co-ordinate international fire data exchange arrangements.

10.3 Summary and recommendations

A summary of the fire assessment activity described in the template indicates that the research challenges in this area are quite significant but need better co-ordinating and are all of high or medium safety significance. Further knowledge is required for all topics and in some a research facility will continue to be essential. However, additional requirements in other industrial sectors should ensure that facilities and expertise continue to be maintained.

It is recommended that CSNI Working Groups be given the task of defining the experimental research needs based on fire PSA and in particular consider the possible uses of the RUT facility.

Template for Chapter 10 – Fire risk assessment

Criteria for inclusion:

Overriding Criteria

- Is there a major short term safety issue, and is it relevant in the overall context of future needs?
- Is the FAP currently under threat?

Additional Criteria

- Is the project risk-important?
- Is the technical area still open?
- Is the FAP unique to the nuclear industry?
- Is the FAP flexible?

Chapter 10: Fire risk assessment	**Future needs and challenges**	**Safety significance**	**Is sufficient knowledge available?**	**Is a facility essential[1]?**	**Is CSNI action needed?**
Fire segregation/ separation rules	• Underpin rules which have evolved through usage • Determine limits of applicability	Medium	No	Yes	No
Fire growth and spread	• Development and validation of methods • Interaction with adjacent compartments	High	No	Yes	No
Fire databases to be maintained and improved	• NPP fire frequency • Fire detection and suppression system	High	No	No	No

1. Although facilities are required for these tasks, they are the same facilities as for conventional fire safety and no “nuclear specific” facility needs are required.

Chapter 10: Fire risk assessment	**Future needs and challenges**	**Safety significance**	**Is sufficient knowledge available?**	**Is a facility essential[2]?**	**Is CSNI action needed?**
Fire (and smoke) vulnerability of NPP equipment	• Establish vulnerability of NPP equipment • Smoke generation and transport issues • Adequacy of fire barriers	High	No	Yes	No
Risk significance of fire location	Minor Control Room • Cable spreading room • Switchgear room	Medium	No	No	No

2. Although facilities are required for these tasks, they are the same facilities as for conventional fire safety and no “nuclear specific” facility needs are required.

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1998-2000

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