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Nuclear Safety Research in OECD Countries

Support Facilities for Existing and Advanced Reactors (SFEAR)

NEA No. 6158

NUCLEAR ENERGY AGENCY
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FOREWORD

A Senior Group of Experts on Nuclear Safety Research (SESAR) was established by the NEA Committee on the Safety of Nuclear Installations (CSNI) to assess the need for and strategy of maintaining key research facilities. This activity is a follow-on to a similar activity conducted by the CSNI in the late 1990s which led to a number of actions by the CSNI to establish co-operative research projects directed at developing information relevant to safety issues on operating water reactors, while at the same time preserving key facilities and programmes. A report on this activity was issued by the NEA in 2001 entitled *Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk* (SESAR/FAP). In response to the recommendations expressed in that report, the CSNI has undertaken initiatives, notably in the thermal-hydraulics and severe accident areas. These initiatives mainly consisted of initiating and carrying out internationally funded OECD projects on relevant safety issues, centred on the capabilities of key facilities identified in the SESAR-FAP report. These projects are ongoing and constitute a means for effectively maintaining basic technical infrastructure through international co-operation.

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EXECUTIVE SUMMARY

At its December 2002 meeting, the CSNI approved the establishment of a senior group of experts on nuclear safety research (SESAR) to assess the need for and strategy of maintaining key research facilities. This activity is a follow-on to a similar activity conducted by the CSNI in the late 1990s which led to a number of actions by the CSNI to establish co-operative research projects directed at developing information relevant to safety issues on operating LWRs and PHWRs, while at the same time preserving key facilities and programmes. A report on this activity was issued by the NEA in 2001 entitled: *Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk* (SESAR/FAP). In response to the recommendations expressed in that report, the CSNI has undertaken initiatives, notably in the thermal-hydraulics and severe accident areas. These initiatives mainly consisted of initiating and carrying out internationally funded OECD co-operative projects on relevant safety issues, centred on the capabilities of key facilities identified in the SESAR/FAP report. Four such projects (involving five facilities) were initiated and are ongoing, constituting a means for effectively maintaining basic technical infrastructure through international co-operation.

Since publication of the SESAR/FAP report, research facilities have continued to be shut down worldwide. In fact, of the facilities listed in the SESAR/FAP report in the areas of thermal-hydraulics, fuel, reactor physics, severe accidents and integrity of equipment and structures (i.e. those areas most unique to the nuclear power industry), approximately 35% have been shut down in the past five years. Accordingly, loss of critical research infrastructure (i.e. facilities, capabilities and expertise) remains a concern and is a major factor in conducting the current study. However, it should be recognised that the SESAR/FAP effort led to CSNI actions that preserved five key facilities during the 2000-2006 time period. These are discussed in Chapter 4.

The activity described in this report builds upon and updates the SESAR/FAP work, but also expands its scope to cover advanced LWRs (ALWRs), VVERs, advanced PHWRs (APHWRs) and high-temperature gas-cooled reactors (HTGRs). Accordingly, the title of this activity is SESAR Support Facilities for Existing and Advanced Reactors (SESAR/SFEAR). The need to maintain databases of experimental data is also recognised as an important issue, but is not treated in this report, since preservation of data is being addressed separately by the NEA.

The focus of this report is on the safety issues, research needs and supporting research facilities associated with currently operating water-cooled reactors in NEA member countries. These reactors include pressurised water reactors (PWRs), boiling water reactors (BWRs), pressurised heavy water reactors (PHWRs) and Russian-designed VVERs. For these reactors, the main purpose of this report is to:

- Summarise the currently identified safety issues, whose resolution depends upon additional research work.
- Provide the current status of those research facilities unique to the nuclear industry that support resolution of the safety issues.

- Where such facilities represent a substantial investment of resources and are in danger of premature closure, recommend actions the CSNI could take in the short term to help maintain them.

- Provide recommendations on long-term nuclear safety research facility infrastructure needs and preservation.

In addition, where research facilities do not exist, but may be useful to address currently identified safety issues, these areas are identified. The report also provides information on safety issues and research needs not unique to the nuclear industry and on safety issues and research needs associated with HTGRs. This information is presented for completeness and for use by designers, operators and researchers in planning and conducting future work.

The issues addressed in this report are those associated with nuclear reactor safety (excluding spent fuel storage) and are organised into the following technical areas:

a) Those unique to the nuclear industry:
   - Thermal-hydraulics.
   - Fuel.
   - Reactor physics.
   - Severe accidents.
   - Integrity of equipment and structures.

b) Those not unique to the nuclear industry:
   - Human and organisational factors.
   - Plant control and monitoring.
   - Seismic effects.
   - Fire assessment.

c) Those unique to HTGRs.

The report is organised into four chapters. Chapter 1 is the introduction describing the scope, purpose and approach used in assessing the safety issues and facilities. Chapter 2 provides a short overview of each of the reactor types covered in this report and their associated safety issues. Chapter 3 contains sections on each of the technical areas. Each section contains a description of the safety issues and research facilities associated with that technical area. For areas unique to the nuclear industry (item (a) above), information is provided in tabular form showing:

- The safety relevance of each issue.
- The state of knowledge on each issue.
- The relevance of each facility to address the safety issue.

This information is then used to calculate a relative numerical ranking for each facility in each technical area (as described in the beginning of Chapter 3) which is one factor used in developing recommendations for CSNI consideration.

Chapter 4 then provides conclusions and recommendations for CSNI consideration, based upon the information in Chapter 3. The conclusions and recommendations only pertain to those technical areas unique to the nuclear industry [item (a) above].

In general, in developing recommendations for CSNI consideration, the group focused on those facilities that have unique capabilities, would be very expensive to replace and have high relevance to the resolution of current safety issues (as indicated by their relative numerical ranking), as well as the potential to be highly relevant in support of the resolution of ALWR and APHWR safety issues.

Accordingly, such facilities represent an infrastructure of substantial resource investment and, if lost, it is unlikely that such facilities would be replaced due to the reality of cut backs in nuclear safety research funding over the past few years. Due to the cost of operating such facilities, co-operative efforts would most likely be needed to maintain them in the longer term.

The conclusions and recommendations in Chapter 4 are organised into general, short-term and long-term items as follows:

(a) The following are the general conclusions and recommendations (i.e. they pertain to both the short term and long term).

- CSNI efforts aimed at facility preservation should focus on large facilities, whose loss would mean the loss of unique capability as well as the loss of substantial investment that in the current climate of tight resources would not likely be replaced. Such preservation also includes maintaining the expertise, knowledge, capabilities and personnel essential to infrastructure preservation. In this regard, it should be noted that due to previous CSNI efforts, several large facilities (i.e. PANDA, PKL, MACE, ROSA) have been kept active over the past five years, thus helping the current SFEAR effort. However, many large, expensive and unique facilities are projected to close over the next 1-5 years. Examples include thermal-hydraulic and severe accident facilities. In addition, many of the test reactors are old and will reach their end of life without substantial refurbishment. The loss of such facilities would severely detract from the nuclear safety research infrastructure. Additional discussion on a strategy for long-term facility preservation is discussed in item c) below.

- The NEA Nuclear Science Committee (NSC) should take the lead to monitor the status of and make recommendations for actions to preserve key facilities in the reactor physics area. The facilities and information in this report in the reactor physics area represent the SESAR group's views on the safety issues and facilities important to nuclear safety research and are for NSC use in carrying out this responsibility.

- To help stimulate industry interest in facility and infrastructure preservation, it is recommended that both the CSNI and the CNRA take steps to encourage industry co-operation by emphasizing: 1) the responsibility of industry to develop sufficient data to support their applications, 2) the benefits of co-operative research, and 3) the value of preserving critical research infrastructure.

- Hot cells and autoclaves are essential to nuclear safety research. However, due to the large number of hot cells and autoclaves, it is impractical for the CSNI to monitor their status. Accordingly, each country should monitor the status of these facilities and bring to the CSNI's attention any concerns regarding loss of critical infrastructure.

- Certain safety issues have no large-scale facilities identified for the conduct of relevant research. The appropriate CSNI working groups should evaluate whether or not large-scale facilities are needed to support resolution of these issues. The issues that fall in this category are:

  – ECCS strainer clogging (thermal-hydraulic issue #6).
  – 3-D core flow distribution (thermal-hydraulic issue #12).
  – Long-term behaviour of concrete structures (structural integrity issue #7).
  – Flow-induced vibrations (structural integrity issue #9).

(b) The following recommendations are directed toward those actions that the CSNI could take in the short term (2006-2007) to prevent the loss of key facilities in imminent danger of closure.

- In the thermal-hydraulics area, both existing large integral BWR thermal-hydraulic test facilities (PANDA and PUMA) are in danger of being closed in the next 1-2 years. These facilities are unique and expensive, and at least one should be maintained to be available for supporting research related to current or future BWR safety issues. Accordingly, preservation of one integral BWR thermal-hydraulic test facility (either PANDA or PUMA) is considered essential for preserving a BWR thermal-hydraulic research infrastructure. SESAR is of the view that PANDA is the preferred facility for preservation due to its scale, replacement cost and versatility (i.e. it is useful in the severe accident as well as the thermal-hydraulics area). Accordingly, CSNI action is recommended in the short term to support a co-operative research programme in PANDA. It should be noted that CSNI actions resulting from the SESAR/FAP report played a major role in the preservation of PANDA over the past five years.

- In the severe accident area, most facilities supporting the resolution of the following safety issues for BWRs, PWRs, VVERs and ALWRs are in danger in the short term:

  – Pre-core melt conditions.
  – Combustible gas control.
  – Coolability of over-heated cores.

- Based upon a review of the facilities in short-term danger, their importance to the resolution of the above safety issues and long-term infrastructure preservation, the group concluded that the following should be preserved due to their replacement cost, high relative ranking and versatility:

  – PHEBUS.
  – QUENCH.
  – MISTRA.

Each of these is discussed further below.

- PHEBUS is a unique facility representing a substantial financial investment, capabilities and expertise. Due to the high cost of its operation and the long time frame necessary to plan and conduct experiments, it is not considered practical to propose that the CSNI organise a co-operative research programme in PHEBUS. Accordingly, it is recommended that PHEBUS be treated as a special case, with the French authorities taking the lead to propose and organise a future research programme using PHEBUS. In this regard it should be noted that a PHEBUS Expert Group has been organised to assess future experimental programmes in PHEBUS in the areas of LOCA (fuel response to LOCAs) and of severe accidents (fuel degradation and fission product release and transport). Both separate affects and integral tests are included in the assessment. The recommendations from this group should provide valuable input for justifying and planning future programmes in PHEBUS.

- The QUENCH facility has been used extensively in the past to investigate pre-core melt conditions in LWRs. Although it is a unique facility in near-term danger, the group has concluded that any effort to preserve it for the long term should be dependent upon identifying a future experimental programme that can provide useful information beyond what has already been done in QUENCH. Accordingly, it is recommended that QUENCH be treated as a special case, with the German authorities taking the lead to propose a future research programme in QUENCH capable of generating useful new information. In this regard, it is also recommended that the appropriate CSNI Working Group (WGAMA) be requested to consider future uses for QUENCH and provide a recommendation to the CSNI that can be factored into deliberations on the future of QUENCH.

- The MISTRA facility has the capability for conducting experiments on combustible gas mixing and transport in multi-compartmental configurations with detailed instrumentation and helium as a simulant for hydrogen gas (H_2). As such, it can measure 3-D effects useful for assessing 3-D analytical tools. MISTRA complements the THAI facility which uses H_2 and can conduct experiments on H_2 combustion and aerosol distribution. THAI is not in near-term danger and is recommended for long-term preservation (see below). Accordingly, it is recommended that the CSNI take action to reserve the MISTRA facility (so as to maintain the complementary infrastructure and expertise) by organising and conducting a co-operative research programme in MISTRA.

- In the other technical areas (fuels, and integrity of equipment and structures), no short-term CSNI actions are recommended.

- It should be recognised that implementation of the above recommendations are dependent upon interest and commitment of the "host countries" to provide sufficient resources to attract participation of other interested parties and the ability to propose experimental programmes relevant to resolution of the issues and of interest to member countries.

(c) In the longer term (beyond 2007), it is recommended that the CSNI adopt a strategy for the preservation of a research facility infrastructure, based upon preserving unique, versatile and hard-to-replace facilities. The number and nature of these facilities should be based upon supporting currently operating LWRs and PHWRs and the licensing of future ALWRs and APHWRs. The strategy should include consideration of short- and long-term priorities, cost of preservation (e.g. would the cost of preservation detract substantially from other programmes/facilities) and contingency plans in case of facility loss.

In this regard, many of the factors used in the report to arrive at conclusions and recommendations could be useful in developing a long-term strategy for assessing and initiating future co-operative research projects.

These factors include:

- Facility operating and replacement cost,
- The ability to define a useful experimental programme,
- Long-term resource implication and priorities,
- Industry participation,
- Host country long-term plans and commitment.

Additional discussion on each of these factors is provided in Chapter 4.

In addition, critical research capabilities and expertise are defined qualitatively in the OECD/NEA *Collective Statement Concerning Nuclear Safety Research* (2004). Using this statement and the safety issues contained in Section 3.1 of this publication, a table of critical research facility infrastructure needs was developed along with a list, by reactor type, of existing facilities that could fulfil those needs. These are shown herein as Tables ES-1 and ES-2, respectively. The facilities listed in Table ES-2 are those considered unique, hard-to-replace and identified as having high relative importance in their technical area, as discussed in Chapter 3 of this report. Accordingly, based upon Tables ES-1 and ES-2, it is recommended that the CSNI focus on these facilities in developing a strategy for long-term infrastructure preservation. The CSNI should monitor the status of these facilities in the longer term with a goal of taking action, as appropriate, to ensure that critical facilities are available for each reactor type to meet the critical research infrastructure needs. In addition, for investigating safety issues associated with new reactors and technologies, the CSNI should take an active role in encouraging and organising co-operative research efforts. This will also contribute to infrastructure preservation. Similar to the short-term recommendations above, host country interest will be an important factor in determining which facilities to preserve.

Table ES-1. **Critical research facility infrastructure**

Technical expertise needed	**Facility capability needs**	**Important factors for facilities**
Thermal-hydraulics: modelling and analysis.	Large-scale integral test facilities for each reactor type.	Scale, temperature, pressure and instrumentation capability the key factors. Also, the completeness of the facility with respect to factors such as: auxiliary systems, number of loops and instrumentation capability are important.
Fuels: performance and phenomena.	Test reactor for steady state and reactivity insertion testing. Hot cells for PIE and simulated LOCA testing.	Ability to achieve representative values of energy deposition in transient tests. Ability to achieve linear heat rating, burn-up and adequate in-core instrumentation for steady state testing. Ability to do experiments with MOX and high burn-up fuel. Hot cells for full length and pin segment PIE.
Reactor physics: modelling, cross-sections, parameters and analysis.	Critical facilities for measuring physics parameters and performing benchmark experiments.	Ability to do experiments with MOX and high burn-up fuel.
Severe accidents: phenomena, accident progression, modelling and analysis.	Facilities for testing: • Integral SA phenomena. • In-vessel phenomena. • Ex-vessel phenomena. • Containment atmosphere mixing/combustion. • Accident management strategies.	Use of prototypic materials and large scale are important.
Integrity of equipment and structures: materials behaviour, structural design.	Test reactors for irradiating material samples under controlled conditions. Hot cells for examining large and small irradiated material samples. Autoclaves for materials testing.	Ability to achieve fluence and other prototypic conditions (e.g. temperature, simulate impurities, stress, etc.). Hot cells and autoclaves for ex-reactor testing of irradiated materials.

Table ES-2. **Critical facilities to be monitored in the long term**

Technical area	BWR	PWR	VVER	PHWR/ APHWR	ALWR
Thermal-hydraulics	PANDA[1]	LSTF/ROSA PKL ATLAS	PSB-VVER PACTEL	RD-14-M	LSTF/ROSA PKL PANDA[1] ATLAS
Fuels[2]	Halden NSRR PHEBUS	Halden NSRR CABRI PHEBUS	Halden MIR CABRI PHEBUS	Halden NRU	Halden NSRR CABRI PHEBUS
Reactor physics*	Proteus Venus	Proteus Venus	Proteus LR-O	Proteus ZED-2 Venus	Proteus Venus
Severe accidents	*Integral testing* PHEBUS[1]	PHEBUS[1]	PHEBUS[1]	PHEBUS[1]	PHEBUS[1]
	In-vessel phenomena – QUENCH[1] – VERDON – KROTOS	– QUENCH[1] – VERDON – KROTOS	– QUENCH[1] – VERDON – KROTOS	– Fuel Channel Safety Facility – VERDON – KROTOS	– QUENCH[1] – VERDON – KROTOS
	Ex-vessel phenomena – MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]
	Containment mixing/combustion – PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]
	Accident management Uses data generated in the resolution of other severe accident and thermal-hydraulic issues. No unique facility needs.				
Integrity of equipment and structures[3]	Halden JMTR LVR-15 ATR	Halden JMTR LVR-15 ATR	Halden JMTR LVR-15	Halden NRU LVR-15	Halden JMTR LVR-15 ATR

* (Included for completeness; NSC to monitor status).

Notes:
1. Assumes actions will be taken in the short term to preserve these facilities.
2. Assumes ongoing effort to initiate a co-operative research programme will be successful.
3. Due to the large number of hot cells and autoclaves, each country should monitor the status and identify concerns.
4. Experimental programme under discussion in the CSNI/SERENA programme.

Chapter 1

INTRODUCTION

At its December 2002 meeting, the CSNI approved the establishment of a senior group of experts on nuclear safety research (SESAR) to assess the need for and strategy of maintaining key research facilities. This activity is a follow-on to a similar activity conducted by CSNI in the late 1990s and which led to a number of actions by CSNI to establish co-operative research projects directed at developing information relevant to safety issues on operating LWRs and PHWRs, while at the same time preserving key facilities and programmes. A report on this activity was issued by NEA in 2000 titled *Senior Group of Experts for Nuclear Safety Research: Facilities and Programmes* (SESAR/FAP).

The activity described in this report builds upon and updates the SESAR/FAP work, but also expands its scope to cover advanced LWRs (ALWRs), VVERs and high-temperature gas-cooled reactors (HTGRs). Accordingly, the title of this activity is SESAR: Support Facilities for Existing and Advanced Reactors (SESAR/SFEAR). The need to maintain data bases of experimental data is also recognised as an important issue, but is not treated in this report, since preservation of data is being addressed separately by the NEA. The SESAR/SFEAR work can be considered a follow-on to the 2000 NEA SESAR/FAP report, which was highly successful in stimulating co-operative research projects as well the preservation of important facilities.

Specifically, SESAR/FAP led to a number of recommendations, many of which resulted in CSNI organising and initiating co-operative research projects to preserve key facilities that were in near-term danger and to improve the state of knowledge in selected areas. The actions taken by CSNI as a result of the SESAR/FAP recommendations have also impacted the current SFEAR study by keeping active a number of facilities that would otherwise have been lost or in need of short-term CSNI action. A summary of the SESAR/FAP recommendations related to facilities, the resulting CSNI action and the impact on the SFEAR report is contained in Table 1-1.

However, despite the efforts of CSNI, the overall situation with respect to loss of research facilities and infrastructure has continued to deteriorate since issuance of the SESAR/FAP report in 2002. In the six years since the SESAR/FAP report was issued, approximately 35% of the research facilities listed therein have been closed. While it is appropriate that some facilities and programmes be closed, such a closure rate points to the need to monitor the situation closely so as to be able to take action if key facilities are in danger. Accordingly, the SFEAR effort represents a look at the current situation, including the previous SESAR/FAP recommendations and resulting actions, and makes recommendations related to short-term and long-term actions which should be taken to preserve the key safety research facility infrastructure in member countries and Russia. The scope and approach used in the SFEAR programme are discussed below.

1.1 Purpose

For operating reactors, the purpose of the SESAR/SFEAR activity is to (1) summarise the currently identified safety issues, whose resolution depends upon additional research work, 2) provide the current status of those research facilities unique to the nuclear industry that support resolution of the safety issues, 3) where such facilities represent a substantial investment of resources and are in danger of premature closure, recommend actions CSNI could take to help maintain them and, 4) provide recommendations on long-term nuclear safety research facility infrastructure needs and a strategy for its preservation. In addition, where research facilities do not exist, but would be useful to address currently identified safety issues, facility needs are identified. For future reactors, the purpose of the SESAR/SFEAR activity is to identify safety issues, safety research and facility needs for use by designers, operators and researchers.

This activity is considered vital to achieving one of the main goals for CSNI, as described in its operating plan, which is to help maintain the critical infrastructure and expertise to ensure the continued safety of nuclear power. Maintaining such infrastructure and expertise is necessary to understand and address safety issues and ensure the safe operation of existing and future NPPs. The existence of interesting international research projects is also a factor in being able to attract and train new people into the nuclear safety field. This is an incentive for countries to serve as the host country for co-operative research projects involving their facilities.

Finally, in 2004 the NEA issued a publication entitled *Collective Statement Concerning Nuclear Safety Research*. This publication emphasised the role research plays in support of the efficient and effective regulation of nuclear power plants. In this publication, research capabilities and expertise which should be maintained as part of the overall regulatory infrastructure were identified. Accordingly, the SESAR/SFEAR work is also directed toward helping to maintain this infrastructure.

1.2 Scope

The scope of this activity is limited to safety issues and facilities associated with nuclear reactor design, construction and operation (spent fuel storage is not included in the scope of this activity) in OECD member countries and Russia and covers the following technical areas:

- Thermal-hydraulics.
- Fuel.
- Reactor physics.
- Severe accidents.
- Integrity of equipment and structures.
- Human and organisational factors.
- Plant control and monitoring.
- Seismic effects.
- Fire assessment.
- High temperature gas reactor (HTGRs) safety issues.

Of the technical areas, the first five address phenomena, safety issues and facilities unique to the nuclear industry. The next four technical areas address phenomena, safety issues and facilities that are relevant to the nuclear industry but are also relevant to other industries and, thus, research and facilities may be supported by others. Finally, the last technical area (HTGR safety issues) is related to phenomena and safety issues for a technology that has yet to see widespread use and, thus, much research remains to be done by the nuclear community to support its deployment. It should also be noted that the technical areas contained in this report are the same as those contained in the SESAR: FAP report, except for HTGR issues, which have been added; and risk assessment, which has been deleted since there are no facilities associated with this area. The safety issues identified in this report are those where additional research is needed to support their resolution or to reduce uncertainties, thus supporting more realistic treatment of the issue.

The reactor designs to be assessed in each of the technical areas are:

- Currently operating PWRs, BWRs, VVERs and PHWRs in member countries.
- Future designs that could be deployed in the next 5-10 years. These future designs include advanced LWRs, PHWRs, and HTGRs.

Due to their long-term schedule and preliminary nature of their designs, it was decided not to cover designs where deployment is beyond 10 years (e.g. Generation IV) in this report.

It should also be recognised that research in some of the technical areas listed above does not require large facilities, but rather large-scale coordinated programmes.

1.3 Approach

In each of the technical areas above, the SESAR/SFEAR participants developed two sets of recommendations: (1) those for the technical areas unique to the nuclear industry that are applicable to currently operating and future LWRs and PHWRs and (2) safety research and facility needs for future HTGRs. For those technical areas not unique to the nuclear industry, no recommendations are provided. Rather, the safety issue and facility information is provided for information only. For item (1) above, operating and future plants, the members identified the safety issues still needing research work and the major facilities currently performing or available to perform research directly relevant to these issues. For each of the five technical areas unique to the nuclear industry, the following information related to each safety issue and facility was generated:

- Safety issues:
  - Safety relevance.
  - State of knowledge.
  - Applicability (to which reactor types does the safety issue apply).
  - Importance of relevant facilities to resolve the issue.
- Facilities:
  - Applicability (to which reactor types does it apply).
  - Annual operational cost.
  - Replacement cost.
  - Capability and versatility of facility (including quality of instrumentation).
  - Planned duration of facility operation.

This information is displayed in the form of two tables for each technical area, as illustrated at the end of this chapter (Tables 1-2 and 1-3). Also shown are the definitions for the terms used in the tables.

To ensure that the SFEAR activity focused on only those major facilities large enough to warrant CSNI consideration (i.e., those that would need a co-operative effort to finance), the following guidelines were established regarding facility size considered in this study:

- To be considered in the SESAR/SFEAR activity, the facility would need to have at least an annual operating cost of >P$1 000 000 per year or at least a replacement value of >$ 2 000 000.

In addition, for a viable co-operative research project, there must also be a host country commitment for a programme and resources consistent with CSNI guidelines.

Based upon the safety issue and facility information, the group then developed a relative ranking for each of the facilities identified as relevant to one or more of the safety issues in those technical areas unique to the nuclear industry. The relative ranking process is described in the introduction to Chapter 3. Based upon this relative ranking, the importance, versatility and critical nature of the facility the group developed recommendations for CSNI consideration. These are described in Chapter 4.

For item (2) above, HTGRs, the group identified the safety issues, and then developed high level recommendations regarding the safety research needs and facility needs that should be considered to address the issues. The group's recommendations are not for CSNI action, but rather for HTGR plant designers and regulators to consider in planning future programmes.

1.4 Co-ordination

The issues, conclusions and recommendations contained in this report represent the personal views of the SESAR:SFEAR participants, but not necessarily the views in their organisations. However, in assembling the information contained in the report the participants had the benefit of input from the CSNI Working Groups, Special Expert Groups and CSNI members. In addition, input was obtained from the NEA Nuclear Science Committee on the reactor physics section and other external organisations (CNRA, IAEA and others) were offered an opportunity to comment on a draft of the report.

1.5 Organisation of report

The report has been organised to provide first, a short overview (Chapter 2) of the reactor designs and their safety issues addressed within the scope of this report, followed by a chapter (Chapter 3) addressing each of the technical areas listed above. Chapter 3 is organised into three sections pertaining to those technical areas unique to the nuclear industry, those not unique and HTGR unique issues. The introduction to Chapter 3 describes in more detail the organisation and purpose of each of these three sections. Chapter 4 then provides the group's conclusions and recommendations regarding critical facilities unique to nuclear safety research in danger of being lost that deserve and need international support and possible actions for CSNI consideration.

The conclusion and recommendations are based upon several factors. These include:

- The importance of the facility to resolving the safety issues, based upon the relative ranking of a facility within a given technical area using the ranking approach described in the introduction to Chapter 3.
- The versatility of the facility, including quality of instrumentation.
- The importance of the facility to maintaining a minimum infrastructure of safety research capability, (i.e. uniqueness and replacement cost).

The conclusions and recommendations are divided into near term (CSNI action needed in the next 1-2 years) and long term (CSNI should monitor the status).

However, any action by CSNI would be contingent upon the willingness of the host country of the facility to contribute substantially to its continued operation in accordance with NEA/CSNI guidelines (i.e., CSNI does not intend to serve as host country for facility preservation).

Table 1-1. **Impact of SESAR/FAP facility recommendations on SESAR/SFEAR**

SESAR/FAP recommendation	Resulting CSNI action	Impact on SFEAR
1) Maintain the PANDA, PKL and SPES facilities in the thermal-hydraulic area (these facilities were in near-term danger of closure).	Initiated the SETH programme utilising the PANDA and PKL facilities (no host country support for SPES).	• PANDA maintained through 2005. Currently in near-term danger and addressed in the SFEAR study. • PKL active and not in near-term danger.
2) Monitor and maintain key thermal-hydraulic facilities in the long term. T/H facilities should be maintained in North America, Europe and Asia.	Facility status monitored. Initiated programme utilising the ROSA facility when it was in danger of being shut down.	ROSA is active and not in near-term danger. Other T/H facilities continue to be monitored.
3) Maintain the RASPLAV and MACE facilities in the severe accident area (these facilities were in near-term danger of closure.	• Initiated the MASCA programme as a follow-on to RASPLAV to maintain facilities. • Initiated the MCCI programme utilising the MACE facility.	• MASCA is active and not in near-term danger. • MCCI is active and therefore the MACE facility is not in near-term danger.
4) Develop centre of excellence on fuel-coolant interaction (FCI) in consideration of potential loss of the FARO and KROTOS facilities.	Initiated the SERENA programme (group of experts to discuss status of FCI and future experimental needs). FARO shutdown. KROTOS kept in standby.	SERENA programme has recommended an experimental programme be conducted in KROTOS and thus may impact preservation of KROTOS facility. CSNI expert group to review SERENA recommendation.
5) Develop centre of excellence (COE) on iodine chemistry and fission product behaviour.	Proposal for COE currently under evaluation.	No additional CSNI action needed.

Table 1-2. **Issues versus facilities**

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue	Facility		
				Name	Importance of facility to resolution of the issue?	Versability

Table 1-3. **Facilities**

Facility name	Applicability (type of reactor)	Cost/Yr. * operation	Replacement* cost	Issues covered	Capability	Planned duration of operation	Relative ranking

Notes:
- High, Medium, Low:
- Operational Cost = Low is < \$1.0 million/yr.; Medium is \$1-2 million/yr.; High is > \$2 million/yr.
- Replacement Cost = Low is < \$2.0 million; Medium is \$2-10 million; High is > \$10 million

Definitions for terms used in tables

A) Issues versus facilities table

Applicability of issue

To what types of reactor designs does the issue apply?

Safety relevance of issue

HIGH	is directly related to establishing the safety parameters affecting the successful performance of critical safety functions (e.g., reactor shutdown, decay heat removal, containment of fission products, etc.), ensuring key assumptions in the safety analysis are valid and/or determining the scope and significance of potential new issues.
MEDIUM	is related to reducing uncertainties regarding known safety issues to support realistic analysis and issue resolution.
LOW	is not related to critical safety functions or issues.

State of knowledge on issue

HIGH	no additional information beyond that obtained from existing research programmes is required to resolve the issue.
MEDIUM	additional information beyond that from existing research programmes is useful to reduce uncertainties and support realistic issue resolution, but the issue can be resolved conservatively without such additional information.
LOW	additional information is necessary to understand the phenomena and/or scope of the issue to allow its resolution.

Importance of facility to resolution of the issue

HIGH	Can directly contribute to issue resolution.
MEDIUM	Can help reduce uncertainties.
LOW	Can get the information in other ways.

Versatility of facility

Would the facility likely be used in the future (i.e., can it be used for other issues, for future plants, etc.) with reasonable modifications?

B) Facilities table

Capability

What types of experiments at what scale can be done?

Current planned duration of operation

What is the current status of the facility? When may the facility likely be shut down without CSNI action?

Relative ranking

Based on process described in Chapter 3.

Chapter 2

OVERVIEW OF REACTOR DESIGNS AND SAFETY ISSUES

This section contains short overviews of each of the major reactor types included in the scope of this report. These overviews are for the purpose of familiarising the reader with the basic characteristics of the designs as well as the major safety features and issues. The reactor designs included in this section are:

- Boiling water reactors (BWRs).
- Pressurised water reactors (PWRs).
- VVER reactors.
- Pressurised heavy water reactors (PHWRs) and advanced PHWRs.
- Advanced light water reactors (ALWRs).
- High-temperature gas-cooled reactors (HTGRs).

2.1 Boiling water reactors

Introduction

The direct cycle boiling water reactor (BWR) nuclear system is a steam generation and steam utilisation system consisting of a nuclear core located inside a reactor pressure vessel and a conventional turbine-generator and feedwater supply system. Associated with the nuclear core are auxiliary systems to accommodate the operational and safety requirements and necessary controls and instrumentation. Water is circulated through the reactor core, producing saturated steam which is separated from the recirculating water, dried in the top of the vessel, and directed to the steam turbine-generator. The turbine employs a conventional regenerative cycle with condenser deaeration and condensate demineralisation.

The steam produced by the nuclear core is radioactive. The radioactivity is primarily due to -16, a very short-lived isotope (7 seconds half-life) so that the radioactivity of the steam from the reactor vessel is mostly present during power generation. However, other radioactive material (e.g., from fuel cladding failures) can also be entrained in the steam increasing its radioactivity and primary coolant system contamination.

BWR core designs and containment designs can vary depending upon the age of the design and product type. However, they all have certain common characteristics which include:

- The almost universal use of recirculation inside the reactor vessel (using either jet pumps or centrifugal pumps) to increase water flow through the core (and thus control boiling and power level).
- The use of a pressure suppression type containment, whereby steam released from the reactor coolant system would be condensed by being directed to a pool of water (called the pressure suppression pool) thus allowing a smaller containment building.

- The use of control rod drives that enter through the bottom of the reactor vessel, thus allowing for installation of steam dryer and moisture separator equipment above the core outlet. Such a control rod location also allows removal of the upper reactor vessel head for refueling without disturbing the control rod drives.
- Automatic depressurisation capability for the reactor coolant system to allow water injection at low pressure.
- No use of boron in the primary coolant in normal operation.

Described below are the main design features and safety issues associated with BWRs.

BWR design features

The nuclear core consists of fuel assemblies and control rods contained within the reactor pressure vessel and cooled by a recirculating water system. The recirculating water system consists of the feedwater flow and flow internal to the reactor vessel (recirculation flow) which is the result of pumps which increase flow internal to the reactor vessel. These pumps generate about two-thirds of the flow within the reactor vessel. The pumps can be mechanical or jet pumps internal to the reactor vessel. BWR power level is maintained or adjusted by positioning control rods up and down within the core. The BWR core power level is further adjustable by changing the recirculation flow rate through the core without changing control rod position. This BWR feature helps achieve load-following capability for the BWR.

The BWR employs bottom-entry control rods and bottom-mounted control rod drives which allow refuelling without removal of control rods and drives, and allow drive testing with an open vessel prior to initial fuel loading or at each refuelling operation.

BWRs operate at constant steam pressure (approx. 70 bars or 1 000 psi) at the corresponding saturation temperature. They employ moisture separators and steam dryers to enhance the quality of the steam entering the turbine.

The BWR reactor is housed in a reactor building structure. The reactor building structure includes two main structures, the shield building and the containment system. The containment portion of the reactor building is divided into two main compartments called the drywell and wetwell. Components located within the drywell include, but are not limited to, the reactor vessel, the reactor water recirculation system, the main steam lines, main steam line safety/relief valves and discharge piping, control rod drives and piping, piping and valves associated with reactor vessel, nuclear system instrumentation, and heating and ventilation. Components located outside the drywell, but inside the containment vessel include, but are not limited to, the control rod drive hydraulic modules, standby liquid control system components, reactor water cleanup system heat exchangers, auxiliary system piping, refuelling bridge, polar crane, nuclear systems instrumentation heating and ventilating, and the pressure suppression pool.

The containment is a steel leakage barrier which prevents significant fission product release to the outer shield building in the event of an accident. The containment, including all penetrations and welded attachments, acts as an independent structural component within the reactor building for the maximum temperature and pressure conditions that can occur as the result of a LOCA, and accommodates reactor blowdown through the safety/relief valve discharge piping to the suppression pool.

The suppression pool is toroidal pool of demineralised water located in the wetwell drywell and within the containment boundary. The suppression pool provides (a) a means to condense any steam released in the drywell area during a LOCA; (b) a heat sink for the reactor core isolation cooling system during hot standby operation until the decay heat can be removed by the residual heat removal (RHR) heat exchangers; (c) a heat sink for venting the nuclear system safety/relief valves; (d) a source of water for emergency core cooling, and (e) a source of water to the containment spray system.

Surrounding the containment is the shield building. The shield building is a cylindrical shell of reinforced concrete. It completely encloses the reference free-standing containment. The primary function of the shield building, the "secondary fission product barrier", is to further limit nuclear radiation to the environment in the event of an accident involving the release of fission products. The structure also protects the containment from adverse atmospheric conditions and external threats, such as missiles.

The annulus between the shield building and the containment provides a plenum for the collection and filtration of fission product leakage from the containment that may occur following a design basis accident. The annulus is normally kept at a negative pressure relative to atmospheric pressure so any leakage through the shield building or containment is into this space. Under accident conditions, the ventilation exhaust from this space is automatically diverted through the filtered standby gas treatment system before release to the environment.

A number of safety systems are provided on BWRs to respond to loss of coolant, loss of power and reactivity insertion events. These consist of:

- A fast acting SCRAM system.
- A standby liquid control system for emergency boron injection into the core in the event of a failure to SCRAM.
- High and low pressure coolant injection systems in the event of a LOCA.
- Primary coolant system depressurisation capability.
- Backup power supplies.

BWR safety issues

Design, operation and research associated with BWRs since the mid-1950s has generated information used to address many safety issues. These issues include:

- Anticipated transient without scram.
- Recirculation pipe cracking.
- Severe accident concerns:
  - MK-I containment shell melt through.
  - MK-I containment vent.
  - Combustible gas control.
  - In-vessel melt retention.
  - Ex-vessel core debris coolability.
  - Source term.
  - Accident management.
- Station blackout.
- Core spray distribution.
- Suppression pool dynamics.
- Stability.

However, these and other issues have undergone re-evaluation due to the development of new designs, operating experience feedback (lessons learned) or industry initiatives to extend plant lifetime, raise power levels, increase fuel burn-up levels or increase operating cycle length. These issues include:

- Plant ageing.
- Power uprates.
- High burn-up fuel.
- Materials cracking/corrosion.
- Installation of digital I&C.
- ECCS sump strainer clogging.
- Stability.
- Impact on human actions and reliability.
- Flow-induced vibration.
- Cracking of reactor internals.

These issues have been or are being resolved through generic or plant-specific reviews with significant input from safety research programmes. Some of the above issues are unique to BWRs and others are shared with PWRs.

Although the basic safety issues are the same, research to address the above issues may be different due to the differences in design and operation between BWRs and PWRs. Accordingly, in assessing facility and programme needs, these differences need to be considered.

Recently, more emphasis has been placed on risk-informed regulation in some countries and in more detailed assessments of accident scenarios and accident management actions. This has led to new issues needing research, such as:

- Redefinition of large break LOCA.
- Break location and orientation.

2.2 Pressurised water reactors

Introduction

The pressurised water reactor (PWR) consists of a high pressure reactor vessel (about 140 bar or 2 200 psi) with anywhere from 2 to 4 coolant loops. Each coolant loop has a reactor coolant pump and a steam generator where heat is transferred from the reactor coolant to generate steam in separate secondary loops. The secondary loops carry steam from the steam generators to the turbine and also pump feedwater back into the steam generators. The steam generators produce saturated steam at approximately 70 bar or 1 000 psi.

PWR core and plant designs can vary depending upon the product type (2, 3 or 4 loop plant). However, they all have certain common characteristics, which include:

- The use of large dry containment buildings, although there are some PWRs with sub-atmospheric containment buildings, ice-condenser pressure suppression containment systems, or double wall designs with a partial vacuum in the annulus.
- The use of soluble boron in the reactor coolant to help control reactivity and achieve cold shutdown.
- The use of control rod drives that enter through the top reactor pressure vessel (RPV) head.
- The use of an electrically heated pressuriser to maintain reactor coolant pressure.

Described below are the main design features and safety issues associated with PWRs.

PWR design features

The reactor core is of the multi-region type. All fuel assemblies are mechanically identical, although the design (e.g., grids, nozzles) and the fuel enrichment is not the same in all the assemblies. In the typical initial core loading, three fuel enrichments are generally used. Fuel assemblies with the highest enrichments are placed in the core periphery, or outer region and the two groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, one part of the fuel is discharged (generally one fourth to one third) and fresh fuel is loaded into the outer region of the core. The remaining fuel is arranged in the central part of the core in such a manner as to achieve optimum power distribution.

High pressure water circulates through the reactor core to remove the heat generated in the fuel. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generators. There it gives up its heat to the secondary coolant (feedwater) to generate steam for the turbine generator. The primary coolant loop is closed when the water is pumped back to the reactor vessel. The entire reactor coolant system is composed of leak tight components to ensure that all radioactivity is confined to the system.

The reactor system containment building is usually a reinforced concrete or steel shell pressure vessel. The contained volume and design pressure of the vessel are sufficient to withstand and contain the contents of the reactor coolant system in the unlikely event of a loss-of-coolant accident or a main steam line break. The containment building houses the reactor and reactor coolant system including the steam generators, reactor coolant pumps, pressuriser, piping, and the safety injection equipment. The reactor coolant system is arranged with the reactor vessel adjacent to and below the fuel transfer canal to permit complete underwater fuel handling. The fuel storage building located immediately adjacent to the containment building has underwater fuel storage facilities which are connected to the containment refuelling canal by a fuel transfer tube and a mechanised fuel transfer dolly.

The reactivity of the reactor is controlled by the temperature coefficient of reactivity; by control rod cluster motion, which is required to follow load transients and for startup and shutdown; and by a soluble neutron absorber, boron, in the form of boric acid which is inserted during cold shutdown, partially removed at startup, and adjusted in concentration during core lifetime to compensate for such effects as fuel consumption and accumulation of fission product poisons which determine the core reactivity tend to slow the nuclear chain reaction

Rod cluster control (RCC) assemblies are used for reactor control and consist of clusters of cylindrical absorber rods. The absorber rods move within guide tubes in certain fuel assemblies. Above the core, each cluster of absorber rods is attached to a spider connector and drive shaft, which is raised and lowered by a drive mechanism mounted on the reactor vessel head. Downward trip of the RCC is by gravity.

A number of safety systems are provided on PWRs to respond to loss of coolant, loss of electric power and reactivity initiated events. These consist of:

- A fast acting SCRAM system.
- Boron injection capability.
- High pressure and low pressure coolant injection systems.
- An auxiliary feedwater system for decay heat removal through the steam generators.
- Back-up power supplies.

- In-containment spray systems to condense steam and scrub fission products from the containment atmosphere (although not all PWRs have this feature).
- Containment cooling systems.

PWR safety issues

Design, operation and research associated with PWRs have generated information to address many safety issues, which include:

- Anticipated transient without scram.
- Loss of coolant accidents.
- Loss of feedwater transients.
- Steam line breaks.
- Station blackout.
- Severe accident concerns:
  - Direct containment heating due to RPV bottom head failure.
  - Accident management.
  - In-vessel melt retention.
  - Ex-vessel core debris coolability.
  - In and ex-vessel FCI.
  - Core debris criticality under reflood conditions.
  - Combustible gas control.
  - Source term.

However, these and other PWR safety issues have undergone reevaluation for example, in view of industry initiatives to raise power levels, increase fuel burn-up levels, increase operating cycle lengths or extend plant lifetime. These issues include:

- Reactivity initiated events (e.g. boron dilution).
- Plant ageing.
- Power uprates.
- High burn-up fuel.
- Materials cracking corrosion.
- Installation of digital I and C.
- ECCS sump strainer clogging.
- Impact on human actions and reliability.
- Steam generator tube rupture.

Some of these issues are shared with BWRs. In addition, similar to BWRs, emphasis on risk-informed regulation in some countries and more detailed assessments of accident scenerios and accident management actions has led to new issues needing research, such as:

- Redefinition of large break LOCA.
- Break location and orientation.

2.3 VVER reactors

Introduction

Nuclear power plants with VVER reactors of Soviet origin are presently operated in seven countries. Four of these are NEA/OECD countries (Czech republic, Finland, Hungary and Slovakia) where the NPP with VVER-440/213 and VVER-1000/320 plants are in operation. VVER reactors are classified as a specific type of PWR reactor. The review of their design and safety features shows that the main concept of these reactors is similar to PWR units designed at the same time in the other countries.

VVER design features

In the design safety philosophy of the early VVER-440/230 preventive features dominated over mitigating actions, which led to certain inherent safety features, such as low power density, large coolant volumes and cracking resistance of the primary circuit, as well as low impact of equipment failures due to large number of primary loops with isolation valves. However, there were also a number of deficiencies that are related to the differences in engineering design solutions, shortcomings in engineered safety features such as insufficient emergency core cooling system, missing containment, quality of manufacturing and reliability of equipment. One weld of the reactor pressure vessel also proved to be prone to radiation embrittlement. Many of these positive and negative features were inherited by the next VVER generations, but the design organisations started to improve safety in line with the Western safety standards. Back-fitting of the existing reactors has been intensive during the recent decade and the process is not much different from that which is going on in plants built to earlier safety standards all over the world.

VVER safety issues

As a general statement, it may be concluded that the majority of safety issues of VVERs are the same or very similar to those of other PWRs. This includes severe accident issues.

The safety of NPPs with VVER reactors was reviewed in the framework of the Extrabudgetary Programme (EBP) of the IAEA on the safety of VVER and RBMK NPPs during the period 1990-1998. The programme addressed the safety issues and ranked them according to their safety significance into four categories in the areas of reactor core, component integrity, systems, instrumentation and control, electrical power supply, containment, internal and external hazards, accident analysis, operating procedures, management, plant operation, radiation protection, training and emergency planning.

As concerns safety ranking of the more modern reactor type VVER-440/213, no safety issues of the highest category were identified. The safety issues with high safety concern, included such issues as insufficient qualification of equipment for anticipated ambient and seismic conditions, seismic safety in general, strength of some structural elements of the bubbler condenser, deficiencies of in service inspection of reactor coolant system, ECCS clogging under LOCA, layout of the emergency feedwater system, fire protection and possible multiple failures of safety related systems in high energy pipe breaks at certain locations. These safety issues have since been resolved at all the plants via back-fitting programmes.

The VVER-1000 concept may be considered much closer to the other PWRs. Its power is higher and safety margins smaller than in the VVER-440 concepts, but in its safety philosophy the defense in depth has been taken into account from the beginning. In the IAEA safety review for the standard series VVER-1000/320, no safety issues of the highest category were identified. Safety issues for the standard series included qualification of equipment, control rod insertion reliability, reactor pressure vessel embrittlement and monitoring, non-destructive testing, steam generator collector integrity, steam and feedwater piping integrity, steam generator safety and relief valves qualification for water flow, reactor vessel head leak monitoring, emergency battery discharge time and fire prevention. For the moment these safety issues have been resolved.

The majority of safety issues at all the VVER plants have been identified as deviations from current standards and practices, which have evolved since these NPPs were designed. In all countries the extensive back-fitting and upgrading programmes have been performed which have resolved a great majority of remaining safety issues.

The particular problem of VVERs is that their safety analysis was not validated via experimental facilities. For this reason the experimental validation of computer codes and system behaviour is of a great importance. As an example, the successful experimental validation of the bubble-condenser system of VVER-440/213 was completed and experimental validation of thermal-hydraulic computer codes for VVER-1000/320 at the PSB-VVER facility is underway. During recent years the safety analysis methodology has also been improved considerably via international co-operation in benchmarking and validation exercises.

In the future it may be expected that the lifetime of the current VVERs will be extended. This will lead to having to address many of the same issues as PWRs, such as:

- Materials behaviour.
- Increased inspections.
- Component replacement or refurbishment.

Among other things, this most probably will include increased use of digital automation, which may bring about new safety issues, but most probably these will not be specific to VVERS.

2.4 Pressurised heavy water reactors (PHWRs) and advanced PHWRs

Introduction

PHWRs are unique for containing the nuclear fuel and coolant in an array of horizontal fuel channels, rather than a pressure vessel. Beyond the headers that distribute coolant to individual feeders for the fuel channels, the remainder of the reactor coolant system is similar to a PWR with reactor coolant pumps, steam generators, etc. To moderate the reactor, the fuel channels are surrounded by low-pressure heavy water in a cylindrical calandria vessel. This moderator water is kept below 100^{o}C, and serves as a back-up heat sink in the event that primary and emergency cooling are lost. In addition, the calandria vessel is contained within a shield tank or vault, filled with light water to serve as a biological shield that provides an additional heat sink in a severe core damage accident.

PHWR design features

The fuel for a PHWR consists of a 0.5 m long bundle of fuel elements (28, 37 or 43, depending on the reactor), with typically 12 bundles in a fuel channel. Similar to LWRs the fuel is UO_2, usually natural uranium with some newer fuel designs considering using slight enrichment. The central element in some fuel designs can contain a small amount of burnable neutron poison (Dy) to reduce void reactivity. The fuel cladding, end caps, appendages, and end plates are all made from Zircaloy.

PHWRs can be refueled on-power through the use of fuelling machines that connect to the ends of a fuel channel. Once connected, the short fuel bundles can be repositioned in the channel to optimise fuel utilisation, and used fuel is replaced with fresh fuel. Because a PHWR can be refuelled on-power, there is not much excess reactivity in the core, and there is no requirement to poison the coolant to reduce reactivity with the introduction of fresh fuel. Therefore, reactivity excursions such as boron-dilution are not a concern. On the other hand, attention is paid to ensuring adequate protection against fuel handling accidents.

The current generation of PHWRs use natural uranium fuel and heavy water coolant, leading to a positive void coefficient. This is accommodated by employing two independent fast-acting shutdown systems. The first is spring-assisted shut-off rods that drive down between the fuel channels in the moderator. The second is liquid poison (Gd) injection into the moderator. Both shutdown systems put neutron absorbing material directly into the low pressure moderator, and are therefore not subject to high pressure, nor jamming due to fuel damage, in the event of an accident.

Similar to all water-cooled reactors, PHWRs use an emergency core cooling system (ECCS) to provide back up cooling in a LOCA. There are typically three modes of operation: high pressure injection, intermediate/low pressure injection, long-term recovery and recirculation. Emergency core cooling is accompanied by venting steam from the secondary side to "crash cool" the steam generators and reduce the primary side pressure below the ECCS injection pressure.

There are currently three major types of containment, single unit containment, multiple unit containment (incorporating a common vacuum building), and a double containment system used in Indian PHWRs.

The single-unit containment consists of a cylindrical, pre-stressed, post-tensioned concrete building with a concrete dome. The building has an epoxy lining to reduce leakage. Short-term pressure rises are mitigated with a dousing system, while local air coolers are used to provide long-term pressure control and heat removal. Hydrogen igniters prevent build-up of hydrogen to explosive levels.

In a multi-unit vacuum containment, 4 or 8 reactors, each with its own individual containment, are connected to a vacuum building by large-scale ducting. In the event of a LOCA, self-actuating valves connect the vacuum building to the ducting. Effluent is then drawn from the reactor building to the vacuum building, reducing the pressure. Dousing is used to condense steam in the vacuum building, and to wash out soluble fission products. In the longer term, an emergency filtered air discharge system is used to control pressure, while filtering out fission products.

Current Indian PHWRs use a double concrete containment. The inner containment is a cylinder and dome of prestressed concrete, with an epoxy lining for leak tightness. The outer containment is a cylinder and dome of reinforced concrete. The intervening space is maintained at a negative pressure with a purging arrangement. A suppression pool between drywell and wetwell volumes in containment is used to limit peak pressures. The suppression pool also provides a source of long-term low-pressure emergency core cooling. Local air coolers also provide pressure control and heat removal, and there is a filtered system for controlled gas discharge in the longer term.

PHWR safety issues

For operating PHWR reactors, the main residual safety issues revolve around improved understanding of phenomena, and reduced uncertainties in safety code predictions. Information has been generated to address the following generic safety issues:

- Limited core damage accident (damage is contained within fuel channels).
- Combustible gas control.
- Core cooling in the absence of forced flow.
- Pressure tube failure with consequential loss of moderator.
- Void reactivity uncertainty allowance.
- Moderator subcooling requirements.

- ECCS sump strainer clogging.
- Flow distribution in headers during LOCAs.

Issues that continue to be evaluated, particularly with feedback from operating experience, with initiatives to extend plant lifetime and with development of new designs, include:

- Plant ageing.
- Materials cracking and corrosion.
- Molten fuel moderator interaction.
- Channel voiding during a LOCA.
- Flow distributions between channels and header effects.
- Severe accidents:
  - Core disassembly.
  - Source term.

Many severe accident issues and phenomena are similar to those for LWRs and thus much of the LWR severe accident research and strategies for issue resolution can be applied to PHWRs.

Advanced PHWR (APHWR) designs are also being developed to improve the safety and economics of the PHWR. Key features being explored in APHWR designs include:

- The use of light water coolant in the primary coolant system with heavy water remaining as the moderator in the calandria.
- A more compact fuel channel arrangement to optimise neutron moderation and help eliminate any positive coolant void coefficient.
- The use of slightly enriched u instead of natural uranium.

Such features affect reactor physics parameters and, thus, the response of the plant under certain accident situations. Accordingly, it is important to understand these affects in assessing plant safety.

NEA has published the results of a workshop held in February 2002 *Advanced Nuclear Reactor Safety Issues and Research Needs*. It also contains information which may be of use to researchers, designers and regulators in planning and conducting future work on APHWRs.

2.5 Advanced LWRs (ALWRs)

Introduction

ALWRs designs constitute improvements of current generation pressurised water reactors (PWRs) and boiling water reactors (BWRs). For the purpose of the SFEAR study, the ALWR designs considered are those being developed for deployment in the next 5-10 years. One or more of the following ALWR design features are likely to be employed on ALWR designs:

- Longer design life (up to 60 years.).
- Advanced materials more resistant to corrosion and cracking.
- Advanced fuel designs.
- Longer operating cycles.
- The use of passive safety systems for emergency core and containment cooling and for decay heat removal.

- More automated controls and safety systems, including the use of digital technology.
- Less reliance on operator action (and fewer staff).

Although LWRs with other design features (e.g., prestressed concrete reactor vessel, thorium fuel) have been considered in the past, it is assumed that the most likely LWRs will employ steel RPVs, use UO_2, or possibly MOX fuel and operate at conditions similar to present day LWRs.

ALWR safety issues

Key safety issues associated with ALWRs are not substantially different from current LWR plants, but the design features to cope with them need to be assessed.

Of particular interest will be:

- Advanced fuel transient performance.
- Advanced materials performance.
- Passive safety feature performance.
- Use of digital technology and the impact on human performance.

NEA has published the results of a workshop held in February 2002 *Advanced Nuclear Reactor Safety Issues and Research Needs*. It also contains information which may be of use to researchers, designers and regulators in planning and conducting future work on ALWRs.

2.6 High-temperature gas-cooled reactors (HTGRs)

Introduction

The origins of commercial gas-cooled reactors are found in the graphite-moderated carbon dioxide cooled "Magnox" reactors developed in the early 1950s in the United Kingdom and France. The high-temperature aspect, which is the HTGR concept, dates in the United States from the latter 1950s when the design of the fully ceramic core and the use of the helium gas for cooling were pioneered by the General Atomics. This development effort resulted in the 40 MWe Peach Bottom 1 HTGR, and the 330 MWe Fort St. Vrain HTGR, which adopted the block-type core. Also in the late 1950s, the Germany began designing the pebble-bed type of HTGR. Two HTGRs were constructed in the Germany, the experimental 15 MWe AVR and the 300 MWe THTR300. All HTGRs mentioned above have been decommissioned.

During the 1980s, the modular high temperature gas cooled reactor (HTGR) concept was developed, primarily in Germany and the United States of America. This concept utilised characteristics of HTGR technology to arrive at a design wherein safety issues were addressed through the inherent response characteristics of the system. These initial modular HTGR designs were primarily directed toward electricity generation using a steam turbine.

The 1990s were witness to the initiation of plant designs that incorporated the reactor directly coupled to a gas turbine power conversion system for the production of electricity. This design replaced the steam cycle components with fewer gas turbine cycle components, and with an attendant benefit of increasing net plant electrical efficiency from approximately 40% into the range of 45 to 50%. The elimination of the steam cycle reduces the potential for large water ingress events. Investigation of the modular HTGRs as the high temperature heat source for industrial co-generation and non-electric applications (to realise products including hydrogen and synthesis fuels as well as the production of electricity) is also taking place.

HTGR design features

HTGRs utilise a single phase gas coolant (usually helium) and graphite as a moderator. They may range from small modular designs (100 MWe) to large central station plants. They can utilise steel reactor pressure vessels (RPVs) or prestressed concrete reactor vessels (PCRVs) with a steel liner. Their fuel can be in the shape of tennis ball size pebbles (to support on-line refueling) or be embedded in stationary graphite blocks. However, in either case the basic component of the fuel is a small particle of UO_2, UCO or UC coated with multiple layers of graphite and silicon carbide, which are intended to retain the fission products. The current state-of-the-art in fuel coating is called the TRISO process which involves three layers of coatings. HTGRs normally operate at core exit temperatures in the range of 900°C to 1 100°C and claim that the coated fuel particles maintain their integrity up to temperatures of 1 600°C.

HTGR safety is centreed around ensuring that fuel quality is sufficient to maintain fuel integrity at high temperatures. HTGRs also have a long response time for core heat-up due to the large heat capacity of the graphite moderator. HTGRs may be designed with steam generators (usually in the primary coolant system) or may utilise an in-line helium turbine (also usually in the primary system). In addition to the above, future HTGRs are likely to contain one or more of the following features:

- Passive reactor shutdown capability.
- Passive decay heat removal.
- Confinement building in lieu of a conventional low leakage containment building.
- Modular design.
- High temperature materials.
- Automated control and safety systems with less reliance on operator action.

HTGR safety issues

Key safety issues associated with HTGRs include:

- Ensuring fuel quality over the life of the plant (this may involve more regulatory attention being given to fuel fabrication controls, QA and sampling).
- Performance qualification and in-service testing of passive safety features.
- Potential for and the effect of air and/or water ingress into the primary system on:
  - Reactivity.
  - Fuel integrity.
  - Graphite oxidation.
- Qualification and acceptance of graphite structures (used in the RPV).
- Fission product transport and release.
- Staffing, in view of long response times, passive features and automatic systems.
- On-line refueling of pebble-bed HTGRs.
- High-temperature material performance (metallic materials): creep fatigue data; environmental characteristics; and in-service inspection and surveillance plan and techniques.
- Nuclear-grade graphite behaviour (including ceramic materials such as c/c composite): measurements of changes in physical properties induced by thermal, radiation and chemical exposures; oxidation measurements in the event of an air-ingress accident; and in-service inspection plans and techniques.

- Fuel performance: irradiation testing of fuel simulating steady state, reactivity insertion, and slow heat-up during transients, including fission product release data.
- Analytical programmes and tools.
- Thermo-fluid dynamics codes as well as severe accident analysis codes; data for code validation and assessment; and development of probabilistic risk assessment models and approaches, considering the new and different equipment that will be used.
- Analytical models for events such as, air and water ingress, fission products release in an air environment, fuel behaviour under reactivity insertion accidents.
- Containment performance: evaluation of containment versus confinement option for all accident scenarios, radiological source terms, and emergency planning.

NEA has published the results of a workshop held in February 2002 *Advanced Nuclear Reactor Safety Issues and Research Needs*. It also contains information which may be of use to researchers, designers and regulators in planning and conducting future work on HTGRs.

Chapter 3

TECHNICAL ISSUES AND ASSOCIATED FACILITIES

This chapter describes the safety research issues currently being investigated or identified as needing investigation to support the continued operation of current plants and/or the development of future plants. This chapter is organised into three sections as follows.

The section on issues and facilities unique to the nuclear industry addresses those technical areas where experimental data specific to the nuclear industry may be essential for addressing the safety issues. The technical areas addressed in this section are:

- Thermal-hydraulics.
- Fuel.
- Reactor physics.
- Severe accidents.
- Integrity of equipment and structures.

For each technical area, the currently identified safety research issues associated with that area are listed, along with the reactor types to which the issue applies. The reactor types addressed in this sub-section include BWRs, PWRs, VVERs, PHWRs, ALWRs and APHWRs. HTGR issues corresponding to the above technical areas are addressed in Section 3.3.

In each technical area in Section 3.1, there are three tables. The first table (Table 1) is a list of the safety issues that currently may require some degree of additional research to improve the state of knowledge and support issue resolution. The second table (Table 2) lists the safety issues from Table 1 versus the facilities currently available that are conducting or have the potential to conduct research relevant to each issue. In Table 2, the group’s views on the safety importance and state of knowledge associated with each issue is shown, using the format and definitions for High, Medium and Low shown in Chapter 1. Also shown on Table 2 is information related to the ability of each facility to produce information relevant to the issue, also using the High, Medium and Low definitions listed at the end of Chapter 1.

Finally, Table 2 also provides a brief description of the facility’s capability related to conducting research on the issue. Table 3 provides a list of each of the facilities in Table 2 with information on their cost of operation, replacement cost, planned duration of operation and additional description. Also shown on Table 3 is a numerical score that represents the relative importance of the facility to be able to do research related to the issues in described Table 1. This numerical score was developed by assigning point values to each of the factors listed in Table 2, as shown below:

	Safety relevance	**State of knowledge**	**Facility importance**
High	1.0	0.3	1.0
Medium	0.6	0.6	0.6
Low	0.3	1.0	0.3

For each issue, a numerical score is calculated for each facility listed as applicable to that issue by multiplying above the numerical scores that correspond to the high, medium or low designations listed in Table 2. The scores for each facility are summed over all the issues in a given technical area to which that facility applies. It is these summations that are shown in Table 3. These numerical rankings then provide a relative indication of the importance of each facility in the given technical area (the higher the score, the more important, on a relative basis, is the facility to that technical area). It is important to note, however, that the numerical rankings are for the purpose of relative comparisons of facilities within a given technical area. Numerical rankings should not be compared between technical areas.

The technical areas addressed in the section on issues and facilities not unique to the nuclear industry are:

- Human and organisational factors.
- Plant control and monitoring.
- Seismic behaviour of structures.
- Fire assessment.

For each of these technical areas the currently identified safety research issues associated with that area are listed, along with the reactor types to which the issue applies. The reactor types addressed in this section include BWRs, PWRs, VVERs, ALWRs, APHWRs and HTGRs.

However, data needs to resolve these issues can be met using facilities of low cost, not in danger or not needing international support, not unique to the nuclear industry or where no facilities are needed. Accordingly, this section does not make recommendations for CSNI considerations, but only lists facilities for information, as appropriate.

The section on HTGR unique issues addresses the safety research issues unique to HTGRs. These issues are in the following technical areas:

- Thermal-hydraulics.
- Fuel.
- Reactor physics.
- Severe accidents.
- Integrity of equipment and structures.

For each safety research issue listed in this section an assessment is made regarding the research needed to address each issue and high level recommendations are made regarding facility needs. No recommendations for CSNI consideration are made in this section. Rather, the recommendations regarding HTGRs are for use by HTGR designers and regulators in long-term planning.

Section 3.2 also contains safety research issues applicable to HTGRs, where those issues apply to other reactor types as well. Accordingly, to get a complete list of safety research issues applicable to HTGRs, Section 3.2 must also be reviewed along with Section 3.3.

3.1 Issues and facilities unique to the nuclear industry

Thermal-hydraulics

Introduction

Thermal hydraulics became one of the main nuclear safety disciplines when postulated accidents like the Loss-of-Coolant Accident (LOCA) and other thermal-hydraulic transients were identified as the dominant safety concern for LWRs. As full-scale experimentation was not feasible in most situations, significant computational developments had to be undertaken to be able to properly simulate such transients, as needed for the safety case of these reactors. Numerous national and international experimental programmes provided the data necessary for understanding the phenomena and simulating them.

The CSNI has always considered with great attention the issue of thermal-hydraulic code validation as well as the experimental database needed for such validation. An overview of the large number of separate-effect test programmes that have been carried out in the past is given in /2/. The results from these programmes provide a sound basis for model validation of traditional system codes, whereas they are insufficient for multi-dimensional codes.

In recent years, the CSNI has taken initiatives to support safety relevant thermal-hydraulic facilities that were in danger of closure. This was done through the establishment of international projects addressing issues of broad international interest and centred on the technical capabilities of selected facilities. These projects are still ongoing and include SETH, PSB-VVVER and ROSA.

The SETH Project focuses on the capabilities of the PKL and PANDA facilities, which were recommended for international consideration.[1] The PKL experiments addressed the issue of potential boron dilution accidents in PWR reactors. They are being continued under a new PKL project. The PANDA experiments are to provide data on containment three-dimensional gas flow and distribution that are important for code prediction capability improvements, accident management and design of mitigating measures. In relation to VVER reactors the international PSB-VVER Project was started, with the objective to provide unique experimental data needed for the validation of thermal-hydraulic codes used for the safety assessment of VVER-1000 reactors. Following a JAERI proposal, the CSNI also recommended to make all necessary steps in order to establish an international experimental project to be conducted in the Japanese ROSA facility.

Scope

As the scenarios of primary concern shifted from the Large Break LOCA to small breaks and other incidents (e.g. boron dilution), the thermal-hydraulic research effort shifted accordingly to cover the more complex phenomena associated with this category of accidents. Improved computational tools were also developed to properly handle these. Although reactivity-related accidents and transients were of course considered from the beginning of the deployment of LWRs, increasing emphasis has been put on the accidents having a neutronic origin or strong neutronic aspect. Accordingly, it has been realised that multidimensional, coupled thermal-hydraulic/neutronic computations were needed to reduce the conservatism of the earlier simpler analyses and/or to simulate properly some complex situations. Although many of the existing facilities are not sufficiently instrumented to be used to validate finely detailed analysis tools (e.g. CFD codes), they are included in this section for completeness.

The new concerns that regulators faced after the Chernobyl accident, more generally related to the understanding and simulation of situations and phenomena in reactors designed in the former Eastern block countries provided additional needs for research and development. The emergence of advanced LWRs having passive safety systems opened another new area of relatively new phenomena and situations that had to be addressed. Power upgrading of existing reactors may also require refinement and further validation of existing analytical tools and more extensive experimental data bases. The thermal-hydraulic safety issues that could benefit from additional research are listed in Table 3.1.1-1.

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1. NEA (2001), *Nuclear Safety Research in OECD Ccountries. Major Facilities and Programmes at Risk,* OECD, Paris.

Description

In spite of these continuing developments, often conducted internationally, a number of issues still require some attention. New issues will certainly also arise in relation to the design and safety analysis of future reactor systems. For example, as current plants continue make operational changes (e.g. power uprates) analysis will be needed to assess changes in safety margins and plant response to off-normal conditions. Also, the increasing use of risk-informed regulation will require better tools and data. The following sections summarise the issues of current and near-term interest in the thermal-hydraulics area.

- LWRs. The CSNI carried out a review of current LWR system behaviour conducted through large integral test programmes.[2] This publication includes a systematic selection of openly available data for code validation. Thermal-hydraulic data is routinely needed to assess analytical tools, particularly when new issues or designs are to be analysed. The current emphasis on risk-informed regulation in some countries, additional interest in more detailed analysis of certain types of accidents and new plant designs provide incentive for maintaining experimental capability to validate analytical tools.

- VVERs. The CSNI also carried out a review of the level of validation of thermal-hydraulic codes applied to the analysis of VVER reactors.[3] The aim was to supplement the review done on integral and separate-effect test facilities, including special features of VVER reactor systems with respect to large and small break LOCA.

- PHWRs. For PHWR reactors, experimental programmes have been carried out in specialised facilities (e.g. large-scale header facility, cold water injection facility) using full-size components such as headers, fuel channels and end fittings. The facility RD-14M has been used for a comprehensive programme on emergency core cooling system effectiveness, natural circulation and shutdown cooling using a full-height, multi-channel test simulation of a CANDU reactor cooling system.

Table 3.1.1-1. **Current thermal-hydraulic safety issues and safety related phenomena**

Issues and relevant reactors	**Description**
1) Boron dilution: PWR, VVER, ALWR	Events where boiling and condensation can occur in the primary cooling system have the potential to form areas where non-borated water can collect. If primary coolant pumps are started, or the restart of natural circulation occurs in combination with other unfavourable circumstances, it was postulated that there wais the potential for a slug of non-borated water to enter the core and cause a criticality concern. This issue involves understanding both the thermal-hydraulic response as well as the reactor kinetics, with emphasis on the 3-D aspects of boron transport and mixing in the RPV. Although resolved in some countries (e.g., U.S.), this issue remains open in others.

2. NEA (1996), *CSNI Integral Test Facility for the Assessment of Thermal-hydraulic Codes for LWR LOCA and Transients*, NEA/CSNI/R(96)17, OECD, Paris.
3. NEA (1993), *CSNI Separate Effect Test Matrix for Thermal-hydraulic Code Validation*, NEA/CSNI/R(93)14, OECD, Paris.

Table 3.1.1-1. **Current thermal-hydraulic safety issues and safety related phenomena** (Cont'd)

Issues and relevant reactors	Description
2) Passive safety system performance: ALWR, APHWR	Some future designs propose the use of passive systems or features for emergency core cooling, decay heat removal and/or containment cooling. Although some experimental work has been done on passive systems in the past, the applicability of this work to future designs will be dependent upon the design and proposed operating conditions. Accordingly, the capability to experimentally investigate passive safety system performance should be maintained and used to validate analytical tools and confirm system performance, including reliability and performance under a range of conditions that could result from aging, environmental conditions, etc.
3) Non-pipe breaks: BWR, PWR, VVER ALWR, PHWR and APHWR	Breaks in plant components can affect plant response to the event, including actuation of engineered safety systems. Additional data is needed to confirm analysis of such conditions, considering location and orientation of the break.
4) Steam generator tube rupture: PWR, VVER, ALWR, PHWR, APHWR	Breaks in SG tubes have the potential to remove coolant inventory from the reactor coolant system and bypass the containment, thus affecting the coolant inventory available for recirculation and the ability to provide sufficient coolant to makeup the loss. The response of the plant to various numbers and locations of SG tube ruptures requires experimental data to confirm analysis methods.
5) Stability and power oscillations: BWR, ALWR	Plant owners are raising power levels at some plants in order to improve economics. The impact of higher power levels on BWR stability and BWR power oscillations, particularly under ATWS conditions, needs to be understood. Also, for future BWR designs, this issue may need further research to confirm predicted behaviour.
6) ECCS strainer clogging: PWR, BWR, VVER, PHWR	Performance of the ECCS in the recirculation mode is dependent upon the ability of the containment sump strainers to remove debris, without plugging to the point of cutting off recirculation flow. This includes consideration of solid debris as well as chemical effects which can cause gelatinous material.
7) Pressure tube reactor system T-H: PHWR, APHWR	In a PHWR, coolant is distributed to the fuel channels via individual feeder pipes from a distribution header. The performance of the PHWR primary coolant and ECCS system needs to be assessed for a wide variety of possible LOCA scenarios.

Table 3.1.1-1. **Current thermal-hydraulic safety issues and safety related phenomena** (Cont'd)

Issues and relevant reactors	Description
8) Two phase natural circulation: PWR, BWR, VVER, PHWR, ALWR, APHWR	Two phase natural circulations may be expected in PWRs during LOCA accident conditions or during an ATWS. The flow rate may be oscillatory under some conditions and depends on the circuit geometry. In BWRs with natural circulation, the oscillations in two-phase natural circulation may lead to power oscillations. Under low power conditions in BWRs, non-uniform conditions may occur in the reactor vessel. Circulation characteristics at low power have large uncertainty. In the VVER-400 the hot leg loop seals may actuate oscillations. The phenomena may be studied by using plant specific test facilities. Natural circulation can be effective under pressurised or depressurised conditions and can be used for decay heat removal or power operation. Uncertainties associated with reliability, predictability, stability and effectiveness of two-phase natural circulation heat removal need to be better understood, modelled and verified, including the effects of non-condensable gas, to support improved analysis.
9) Thermal stratification: BWR, ALWR	Thermal stratification in large components (e.g., pipes, tanks, vessels) can cause structural stresses and/or affect the performance of T/H systems (e.g., cause density gradients that could inhibit passive system performance). Thermal stratification is more prevalent and more a concern in systems that employ natural circulation water injection and/or heat removal.
10) Thermal cycling: PWR, BWR, VVER, PHWR, ALWR and APHWR	In piping systems, where two or more flow paths meet, there may be the potential for fluids of different temperatures to cause localised cycling of the temperature of surrounding components (e.g., fittings, etc.) due to the turbulent mixing of the streams of different temperature fluids. Such thermal cycling of materials can cause alternating thermal stresses and ultimately thermal fatigue failure of the material. Identification of such locations, non-destructive examination of crack status, and understanding the mixing phenomena is important to developing corrective measures and to ensuring system integrity.
11) Moderator thermal-hydraulics: PHWR, APHWR	The moderator in a PHWR serves as a back-up heat sink for accidents involving loss of primary and emergency cooling. The three-dimensional flow regime in a horizontal tube bank needs to be assessed.
12) 3-D core flow distributions: BWR, PWR, VVER and ALWR	The trend towards higher power densities means that in open cores, DNB in the maximum loaded fuel bundle and single rod may be prevented only due to the cross flow connections inside the core. For the simulation multidimensional analysis tools are needed. For model development, multidimensional core void and temperature experiments are needed.

Table 3.1.1-1. **Current thermal-hydraulic safety issues and safety related phenomena** (Cont'd)

Issues and relevant reactors	Description
13) Flow distribution, mixing and stratification in cold legs and downcomer: PWR and ALWR	Data from separate effects mixing experiments (e.g., Creare) and integral system tests (e.g. UPTF, LSTF, APEX, and LOFT) show pronounced thermal stratification of ECC flow in the cold leg, but that the downcomer is well mixed. However, more detailed data would help define the mixing processes and provide information to benchmark CFD codes. It should be noted that systems codes do not represent fluid-fluid mixing and stratification phenomena.
14) Accidents initiated during shutdown: BWR, PWR, VVER, ALWR, PHWR and APHWR	Most work to date in the thermal-hydraulic area has been associated with analysing accidents initiated from full power conditions. However, during shutdown conditions, plant configurations, temperature, pressure, system operability and coolant inventories may be substantially different than they are at full power. Accordingly, accidents that occur will likely have a much different progression, timing and potential consequences than those at full power. For example, loss of residual heat removal (RHR) under shutdown conditions, i.e. during mid-loop operation, has occurred several times world-wide and still plays an important role in risk studies for PWRs. When the RCS is open at the time of RHR failure, the decay heat cannot be transferred to the steam generators and other measures become necessary in order to prevent or compensate for the loss of inventory that finally would lead to core damage. To ensure analytical capabilities are adequate, experimental confirmation of system behaviour will likely be required.

Table 3.1.1-2. **Issues versus facilities** (Thermal-hydraulics)

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue
1) Boron dilution	PWR, VVER, ALWR	Medium	High (except for 3-D effects which are medium)
2) Passive safety system performance	ALWR, APHWR	High	Medium
3) Non-pipe breaks	BWR, PWR, VVER, PHWR, ALWR	Medium	Low
4) S.G. tube rupture	PWR, VVER, ALWR, PHWR	High	High
5) Stability and power oscillations	PWR, BWR, VVER	High	Medium
6) ECCS strainer clogging	PWR, BWR, VVER	High	Medium
7) Pressure tube reactor T/H	PHWR, APHWR	High	Medium

Table 3.1.1-2. **Issues versus facilities** (Thermal-hydraulics)

Facility		
Name	**Importance of facility to resolution of the issue?**	**Versatility**
LSTF/ROSA	High	Large-scale PWR systems simulation.
PSB-VVER	High	Large-scale VVER systems simulation.
PKL	High	Large-scale PWR systems simulation, with on-line boron concentration measurement capability.
PACTEL	Medium	Can simulate non-borated slug formation in boiler condenser mode or during S.G. tube ruptures.
APEX	Medium	¼ scale passive PWR systems.
ATLAS	Medium	Medium scale PWR Systems simulation for both DVI and CLI.
LSTF/ROSA	High	Large-scale active and passive PWR systems simulation. Full height, full pressure.
PSB-VVER	High	Large-scale active and passive VVER systems simulation.
PKL	Medium	Large-scale PWR system simulation. Can simulate gravity feed from accumulators.
RD-14M	High	PHWR full scale primary circuit.
PACTEL	High	Active and passive systems simulation.
APEX	Medium	¼ scale, passive PWR systems simulation from intermediate press.
PUMA	Medium	¼ scale, passive BWR systems simulation from intermediate pressure.
PANDA	High	Large-scale passive systems.
ATLAS	High	Active and passive PWR system simulation for various injection modes.
LSTF/ROSA	High	PWR systems simulation, including non-pipe breaks.
PSB-VVER	High	VVER systems simulation, including non-pipe breaks.
PKL	High	Can simulate non-pipe breaks with asymmetric loop behaviour among PKL's 4 loops.
PACTEL	Medium	Active and passive systems simulation.
ATLAS	High	Active and passive PWR system simulation for various injection modes.
RD-14M	High	Can simulate header breaks.
LSTF/ROSA	High	PWR systems simulation.
PSB-VVER	High	VVER systems simulation.
SKODA-VVS	Medium	PWR simulation.
PKL	High	Simulation with all relevant primary and secondary systems.
RD-14M	High	PHWR primary circuit.
PACTEL	Medium	Can simulate S.G. tube rupture in horizontal S.G.
APEX	High	¼ scale facility.
ATLAS	High	PWR systems simulation.
PUMA	Medium	BWR systems with neutronic simulation.
THYNC	Medium	Can simulate coupled neutronics.
No large-scale facilities identified		
RD-14M	High	PHWR full-scale primary circuit.

Table 3.1.1-2. **Issues versus facilities** (Thermal-hydraulics) (Cont'd)

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue
8) Two-phase national circulation	PWR, BWR, VVER, PHWR, ALWR, APHWR	High	Medium
9) Thermal stratification	BWR, ALWR	Low	Medium
10) Thermal cycling	PWR, BWR, VVER, PHWR, ALWR, APHWR	Low	Medium
11) Moderator T/H	PHWR, APHWR	Medium	High
12) 3-D core flow distributions	PWR, BWR, VVER, ALWR	Medium	Medium
13) Downcomer flow distribution	PWR, ALWR	Low	Medium
14) Accidents initiated during shutdown	BWR, PWR, VVER, ALWR, PHWR, APHWR	High	Medium

Table 3.1.1-2. **Issues versus facilities** (Thermal-hydraulics) (Cont'd)

Facility		
Name	**Importance of facility to resolution of the issue?**	**Versatility**
LSTF/ROSA	Medium	Active and passive PWR systems simulation.
PSB-VVER	Medium	Active and passive VVER systems simulation.
PANDA	Medium	RPV of PANDA has a large-scale 3-D-RPV and a natural circulation 2-phase loop.
RD-14M	Medium	PHWR primary circuit.
PKL	High	PWR systems simulation with 4 identical loops for assessing asymmetric effects.
PACTEL	Medium	Active and passive system simulation with horizontal S.G.
PUMA	Medium	¼ scale.
APEX	Medium	¼ scale.
ATLAS	Medium	Active and passive systems. Small scale.
SKODA-VVS	Medium	PWR simulation.
LSTF/ROSA	Medium	Active and passive PWR systems simulation.
PSB-VVER	Medium	Active and passive VVER systems simulation.
PUMA	Medium	¼ scale.
APEX	Medium	¼ scale.
PANDA	High	Large vessels of PANDA have been used for thermal stratification studies.
PACTEL	Medium	Can simulate thermal stratification in tanks and pools.
ATLAS	Medium	PWR system simulation.
PKL	Medium	Active and passive PWR systems simulation.
PANDA	Medium	Large vessels of PANDA can be used for detailed investigations of mixing.
PACTEL	Medium	Can simulate thermal cycling at joints.
MTF	High	Moderator T/H flow distr.
No large-scale facilities identified		
LSTF/ROSA	Medium	Large scale, full height, full pressure.
PKL	Medium	Large scale, full height.
ATLAS	Medium	Annular downcomer.
MIDAS	Medium	Annular downcomer.
LSTF/ROSA	High	Can simulate open PWR primary system at large scale.
PKL	High	Can simulate open PWR primary system at large scale, including mid-loop operation.
ATLAS	Low	Small volume scale.
PSB-VVER	High	Large scale.
PANDA	High	Large scale.
PUMA	Medium	¼ scale.
APEX	Medium	¼ scale.
RD-14M	High	Large scale.
PACTEL	High	Large scale.

Table 3.1.1-3 **Facilities in the thermal-hydraulics area**

Facility name	Applicability (type of reactor)	Cost/year, operation	Replacement cost	Issues covered
LSTF/ROSA (Japan)*	PWR, ALWR	High	High	1,2,3,4,8,9, 13,14
PSB-VVER (Russia)	VVER, ALWR	Medium	High	1,2,3,4,8,9,14
THYNC (Japan)	BWR	Low	High	5
PKL (Germany)	PWR	Medium	High	1,2,3,4,8,9,13,14
APEX (USA)	PWR, ALWR	Low	High	1,2,4,8,9,14
PUMA (USA)	BWR, ALWR	Low	Medium	2,5,8,9,14
PANDA (Switzerland)	ALWR, BWR, PWR	Medium	High	2,8,9,10,14
RD-14M (Canada)	PHWR, APHWR	Medium	High	2,3,4,7,8,14
MTF (Canada)	PHWR, APHWR	Low	Medium	11
SKODA-VVS (Czech Republic)	VVER, BWR, PWR	Low	High	4,8
ATLAS (Korea)	PWR, ALWR	Medium	High	1,2,3,4,8,9,13,14
MiDAS (Korea)	PWR, ALWR	Low	Medium	13
PACTEL (Finland)	VVER	Medium	Medium	1,2,3,4,8,9,1014

Notes:

Specify range : Low, Medium, High.
Operation cost : Low is < 1.0 MUS$/y; Medium is 1.0-2 MUS$/y; High is >2 MUS$/y.
Replacement cost : Low is < 2 MUS$; Medium is 2-10 MUS$; High is >10 MUS$.
* Also discussed in Severe Accidents – Section 3.1.4.

Table 3.1.1-3 **Facilities in the thermal-hydraulics area**

Capability	Planned duration of operation	Relative ranking
Full height full pressure-2 loop PWR. Active and passive system simulation capability (1/48 volume and power scale).	Through – 2009	2.9
Full height, full pressure VVER facility. 1/300 volume and power scale. Active and passive systems.	Through 2010	2.8
Coupled, simulated neutronics and T/H simulation.	Indefinite	0.4
Scaled PWR Facility. Full-height, 40 bar pressure. 1/145 volume scale, 4 identical loops and secondary systems.	Through 2006	2.9
¼ height, reduced pressure, 2-loop PWR, including passive ECCS.	Through 2006	1.6
¼ linear scale – low pressure-BWR, including passive ECCS and DHR systems.	To be put in standby in 2007	1.6
Full height, very large interconnected volumes, low pressure, large steam generation capability, air and helium supply systems.	Until the end of the SETH Project in 2006	1.9
Full height, full pressure.	Through 2010	3.1
¼ scale calandria vessel and loop.	Through 2008	0.2
Large Water loop for T/H experiments.	New programme (indefinite)	0.5
½ height, 1/1444 vol. scale, full pressure and temperature facility for APR-1400. Includes annular downcomer, direct vessel injection, and cold leg injection. Can simulate most accidents and transients.	Through 2015	2.4
1/5 linear scale, 1 Mpa-pressure and 300 C super heated steam test facility for APR1400 with DVI; separate effect test facility, steady-state ECC bypass for DVI injection mode during LBLOCA late reflood phase.	Through 2010	0.1
Full height VVER-440 scale facility (1:305 volumetric scale), high pressure, 1/5 scaled full power. Facility to be modified to simulate western PWR designs.	Indefinite	2.4

Notes:

Specify range : Low, Medium, High.
Operation cost : Low is <1.0 MUS$/y; Medium is 1.0-2 MUS$/y; High is >2 MUS$/y.
Replacement cost : Low is <2 MUS$; Medium is 2-10 MUS$; High is >10 MUS$.
* Also discussed in Severe Accidents - Section 3.1.4.

3.1.2 Nuclear fuel

Introduction

The fuel of a nuclear reactor is tied directly to the economic performance, investment protection and safety of the nuclear power plant (NPP). Fuel that performs well means fewer plant shutdowns, less radioactive contamination, less radiation exposure to operations and maintenance personnel and less potential for releases of radioactive material offsite. The role of fuel performance in NPP safety can vary depending upon plant design and technology. Plants are generally designed to prevent fuel damage for events that are expected to occur one or more times during the life of the plant and for more rare events varying degrees of fuel damage may be allowed.

The largest quantity of radioactive material in the plant is contained in the fuel in the form of fission products and higher actinides. These build up over the lifetime of the fuel and, to varying degrees, have the potential to be released from the fuel in the event of fuel damage. As such, the amount, timing and nature of fuel damage during accident conditions determine the amount, timing and nature of fission products available for release in the plant and potentially outside the plant. These fission products and actinides have the potential to be released from the reactor core into the reactor coolant system, containment and ultimately the environment and affect the design and qualification of plant safety systems, site suitability and emergency preparedness.

Scope

The scope of this chapter includes safety issues associated with the performance of LWR (including VVER), ALWR, and PHWR fuel. The fuel type is primarily UO_2 for LWRs, ALWRs and PHWRs, although mixed oxide (PuO_2 and UO_2) fuel is of interest in some countries. Burnable poison (such as Gd_2O_3) may also be present in the fuel to compensate for excess reactivity at the beginning of life. Cladding materials are made from various zirconium based alloys and it is the cladding performance, under accident conditions, which is of primary interest in assessing fuel performance. The safety issues of interest that could benefit from additional research are those associated with understanding and establishing fuel damage limits necessary for licensing purposes, as well as understanding fuel behaviour over a range of design and beyond design basis conditions. These safety issues are listed in Table 3.1.2-1. Programmes, facilities and analytical tools necessary to understand and resolve the outstanding safety issues are the subject of the remainder of this chapter.

Description

Fuel designs vary by reactor type and technology. Described below for each reactor type is summary information regarding fuel.

- *LWRs (including VVERs)*. For LWRs, the uranium enrichment ranges from two to nearly five percent, although in some countries LWR recycle mixed oxide fuel is being used in portions of the core. Burnable poison (such as Gd_2O_3) may also be present in the fuel to compensate for excess reactivity at the beginning of life. The UO_2 pellets (or mixed oxide pellets) are housed in zirconium alloy tubes (called cladding) 12 to 14 feet long. These tubes are then assembled into tube bundles of varying sizes which are then inserted into the reactor core in a vertical orientation. The fuel design (U enrichment, cladding material, service condition, etc.) determines how long the fuel can stay in the reactor and still be able to perform satisfactorily under steady state and accident conditions. For currently operating LWRs, the accidents that pose the greatest potential for fuel damage are large reactivity insertions, loss of coolant events and events that cause dry out or departure from nucleate boiling on the cladding. The behaviour of the fuel under these conditions will be affected by the fuel burn-up level (which affects internal fuel pin pressure), fuel cladding condition (e.g. corrosion, oxidation and embrittlement) and location in the core. For economic reasons, increases in burn-up levels are being considered in many countries. To support higher burn-ups, new cladding materials are being developed that exhibit reduced oxidation characteristics. The safety issues associated with current LWR fuel are all related to deciding where to establish safety limits based upon how fuel performance under accident conditions changes with changes in burn-up, cladding material and service condition.

- *ALWRs.* For ALWRs, it is expected that high burn-up levels will be desired for economic and possibly security reasons (e.g., proliferation resistance). To support such fuel designs burnable positions and additional advances in cladding materials will likely be needed.

Ensuring these new fuel designs achieve an acceptable level of safety will require testing and analysis to confirm fuel performance, validate analysis tools and establish safety limits. However, such testing may only require analyses and unirradiated clad testing, or may require more extensive testing, depending upon the fuel design and the types of accidents that may be postulated.

- *PHWRs.* PHWR fuel has many of the same basic characteristics as LWR fuel, although it is always UO_2 composition. The major fuel design differences are the shorter fuel pin length (1.5 feet versus 12-14 feet), smaller pin bundles, low enrichment or the use of natural uranium and horizontal orientation in pressure tube core geometry. Generally, PHWR fuel is designed for lower burn-ups than LWR fuel; however, the safety issues and performance concerns are essentially the same as for LWR fuel. However, PHWRs operate using an on-line refuelling system and thus the fuel handling (and potential refuelling accidents) of PHWR fuel is very different than that for LWRs.

Table 3.1.2-1. **Current nuclear fuel issues**

Issues and relevant reactors	Description
1) Response to LOCAs: PWR, BWR, VVER, PHWR, ALWR, APHWR	As fuel designs change to achieve higher burn-up (e.g. through the use of new cladding materials), or to utilise MOX fuel, the response under LOCA conditions needs to be investigated to support development of appropriate criteria that ensure coolable geometry is maintained during and after design basis LOCA events. Experimental data is needed, consistent with the design basis, and perhaps beyond the design basis LOCA. Also, small break LOCAs can lead to clad embrittlement. Some experimental data on small break LOCAs are useful to confirm fuel clad condition after such events and such experiments are underway.
2) Response to reactivity insertion accidents: PWR, BWR, VVER, PHWR, ALWR, APHWR	As fuel designs change to achieve higher burn-up (e.g. through the use of new cladding materials) or to utilise MOX fuel, the response to reactivity insertion accidents needs to be investigated. Failure modes and criteria associated with new cladding materials and MOX fuel are currently being investigated through experimental programmes, consistent with design basis and beyond design basis events.
3) Response to power oscillation events: BWR, ALWR	As fuel designs change to achieve higher burn-up (e.g. through the use of new cladding material) the response to power oscillation events (such as could occur due to an ATWS or a loss of stability) needs to be determined. Experimental data is needed to establish failure modes and limits; however, conducting experiments simulating these conditions is difficult.
4) Fuel performance under steady state conditions: PWR, BWR, VVER, PHWR, ALWR, APHWR	Fuel performance under steady state conditions can affect coolant circulating activity which impacts the dose to operating personnel. The performance of new and changes in fuel and clad properties during operation can also affect the ability of the fuel to withstand design basis accidents. Therefore, the performance of new fuel under steady state conditions is important to ensuring the safety of operating personnel and to understanding and predicting fuel performance during design basis accidents (i.e. the condition of the fuel and cladding resulting from steady state operation represents the initial conditions for transients).
5) New materials: (fuel property database) PWR, BWR, VVER, ALWR	Improving plant performance in both existing and future plants will likely include improving fuel performance with respect to higher burn-up and power levels. These improvements will require new cladding materials, and the use of burnable poisons. The performance of these new materials and poisons will need verification to ensure their safety and to establish a fuel property database for safety analyses.

Table 3.1.2-2. **Issues versus facilities** (Nuclear fuel)

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue
1) Response to LOCAs	PWR, BWR, VVER, PHWR, ALWR, APHWR	High	Low
2) Response to RIAs	PWR, BWR, VVER, PHWR, ALWR, APHWR	Medium	Medium
3) Response to power oscillations	BWR, ALWR	High	Low
4) S.S. fuel performance	PWR, BWR, VVER, ALWR, PHWR, APHWR	Medium	Medium
5) New materials (fuel properties data base)	PWR, BWR, VVER, ALWR	Medium	Medium

Table 3.1.2-2. **Issues versus facilities** (Nuclear fuel)

Facility		
Name	**Importance of facility to resolution of the issue?**	**Versatility**
ANL-Hot Cells	High	Can be used for other irradiated material tests.
Halden + hot cells	Medium	Integral (single rod) test.
NRU and CRL-hot cells	Medium	Can be modified to simulate fuel heatup.
MIR + hot cells	High	In-reactor LOCA tests.
RIAR-hot cells	Medium	Simulated LOCA tests.
Phebus	High	Capability for in reactor testing.
Cabri	High	Capability for in reactor testing.
Cabri	High	Can run a range of RIA tests.
NSRR	High	Can run short pulse RIA tests.
BIGR	High	Can run broad pulse tests.
Halden	High	Capability for in reactor testing.
NSRR	High	Capability for in reactor testing.
Cabri	High	Capability for in reactor testing.
Halden	High	Full range of instrumentation capability.
NRU	High	Can test full size PHWR fuel bundles.
ATR	High	Can be used for LWR and PHWR fuel testing.
JMTR	High	
HANARO	High	
BR-2	High	
HFR	High	
Osiris	High	
Halden	High	Full range of instrumentation capability.
ATR	High	Can be used for LWR and PHWR fuel testing.
JMTR	High	
HANARO	High	
BR-2	High	
HFR	High	
Osiris	High	
NRU	High	Can test full size PHWR fuel bundles.

Table 3.1.2-3. **Facilities in the fuel area**

Facility name	Applicability (type of reactor)	Operation cost/year	Replacement cost
ANL hot cells* (USA)	PWR,BWR, VVER, PHWR, ALWR	High	High
Phebùs and CEA Hot Cells (LECASTAR, PELECI) (France)	PWR, BWR, VVER, ALWR	High	High
Cabri + CEA hot cells (LECASTA, PELECI) (France)	PWR, VVER, ALWR	High	High
NSRR + hot cells (Japan)	PWR, BWR, ALWR	High	High
Halden Reactor + hot cells (Norway)	PWR, BWR, VVER, PHWR, ALWR	High	High
NRU and Chalk River Lab Hot Cells (Canada)	LWR, PHWR, APHWR	High	High
ATR (USA)	PWR, BWR, ALWR	High	High
JMTR (Japan)	PWR, BWR, ALWR	High	High
HANARO (Korea)	PWR, BWR, VVER, PHWR, ALWR	High	High
BR-2 + Hot Cells (Belgium)	PWR, BWR, VVER, ALWR	High	High
HFR (Netherlands)	PWR, BWR, VVER, ALWR	High	High
BIGR (Russia)	PWR, VVER, ALWR	High	High
Osiris + CEA hot cells (LECASTAR, PELECI) (France)	PWR, BWR, VVER, ALWR	High	High
MIR + hot cells (Russia)	PWR, VVER, ALWR	High	High
RIAR-hot cells (Russia)	PWR, VVER, ALWR	High	High

Notes

* Hot cells also discussed under Integrity of Equipment and Structures Section.
Specify range : Low, Medium, High.
Operation cost : Low is <1.0 MUS$/y; Medium is 1.0-2 MUS$/y; High is >2 MUS$/y.
Replacement cost : Low is <2 MUS$; Medium is 2-10 MUS$; High is >10 MUS$.

Table 3.1.2-3. **Facilities in the fuel area**

Issues covered	Capability	Planned duration of operation	Relative ranking
1	Conducting simulated LOCA tests, on irradiated clad. Cannot handle full length fuel assemblies.	Through 2008.	1.0
1	Capable of conducting 25 pin bundle in-reactor LOCA tests LECASTAR hot cells can conduct simulated LOCA tests.	In danger of shutdown in 2007.	1.0
1, 2, 3	Conducting in reactor RIA tests. Good in-core instrumentation capability.	Indefinite.	2.4
2, 3	Pulse reactor for conducting in reactor RIA experiments.	Through 2014.	0.7
1,3,4,5	19 MWt test reactor mostly for conducting in reactor S.S. tests. Good in-core instrumentation capability.	Indefinite.	2.3
1,4,5	135 MWt S.S. test reactor.	Through 2010.	1.3
4,5	250 MWt S.S. test reactor.	Indefinite.	0.7
4,5	50 MWt S.S. test reactor with power ramp capability.	Through 2008, Future beyond	0.7
4,5	30 MWt test reactor.	Indefinite.	0.7
4,5	60 MWt test reactor.	Through 2016.	0.7
4,5	45 MWt test reactor.	Through 2015.	0.7
2	In-reactor reactivity pulse testing.	Indefinite.	0.4
4,5	70 MWt test reactor.	Through 2014.	0.72
1	Can conduct DBA and beyond DBA - LOCA tests.	Indefinite.	1.0
1	Can conduct simulated LOCA tests in hot cells.	Indefinite.	0.6

3.1.3 Reactor physics

Introduction

Reactor physics issues are becoming increasingly important as plants continue to seek improved performance through power uprates, higher burn-up fuel, longer operating cycles, etc. Core configurations are becoming increasingly heterogeneous in composition and distribution of power generation. This makes prediction of core behaviour and of safety parameters, such as reactivity coefficients, that dictate transient behaviour more difficult. Experimental validation of neutronics methods is needed. In addition, the use of advanced computational methods (e.g. 3-D neutronics) to better refine safety analyses and safety margins has emphasised the need for more detailed reactor physics data and experimental confirmation of analytical methods. Also, thermal-hydraulic and neutronic codes are being coupled to address issues such as boron dilution and ATWS and to analyse PHWR pressure tube reactors.

It is recognised that the NEA Nuclear Science Committee (NSC) has activities in the reactor physics area and is also concerned about the status of key facilities. Accordingly, this section has been written in co-operation with the NSC. However, to ensure an integrated approach to the preservation of critical facilities in this area, the NSC will take the lead to monitor facility status and recommend appropriate actions for consideration by the NSC and CSNI.

Scope

The scope of work in reactor physics covers the current and future needs of nuclear power plants such as pressurised water reactors and VVER reactors, boiling water reactors, gas cooled (thermal) reactors, light water and heavy water moderated reactors, gas cooled (fast) reactors, liquid metal fast reactors, and molten salt reactors. The latter three are outside the scope of this report, yet experiments and research on these systems contribute to a wider and more comprehensive validation of the models and computer codes used and to their further development.

Reactor physics data (cross sections, neutron spectra, reactivity coefficients, etc.) and facilities to measure data for code assessment are the areas of interest in this section. These issues and facilities are shown in Tables 3.1.3-1 and 3.1.3-3, respectively.

Table 3.1.3-1. **Current reactor physics issues**

Issues and relevant reactors	Description
1) MOX fuel data: PWR, BWR, VVER, ALWR, PHWR	Reactor physics data to support the use of MOX fuel in current and future reactors is essential to ensure safe operation. Such data includes cross-sections, and their uncertainties, delayed neutron generation, power distributions, decay heat production and power, temperature and void coefficients. This issue relates to the use of weapons grade Pu in LWRs, PHWRs and VVERs as well as the use of PWR recycle Pu. Advanced fuel cycles involving Pu and other actinides are also being studied for use in LWRs and PHWRs.
2) High burn-up fuel data: PWR, BWR, VVER, ALWR, APHWR	Reactor physics data to support the use of high burn-up fuel in current and future reactors is essential to ensure safe operation. Such data includes cross-sections, and their uncertainties, delayed neutron generation, power distribution and power, temperature and void coefficients.
3) Coolant void coefficient: PHWR, APHWR	LOCA conditions can cause voiding in some PHWR coolant channels. This voiding may lead to positive reactivity input prior to reactor shutdown. The timing and degree of voiding (and the subsequent reactivity effect) needs to be understood and included in PHWR safety analysis.
4) Neutron flux and spectra: PWR, BWR, VVER, ALWR, PHWR, APHWR	The neutron flux and spectra on the reactor pressure vessel (RPV) internal structures and the RPV wall are critical to determining material embrittlement, component lifetime and the potential for RPV failure due to pressurised thermal shock. Such data is especially critical to plants seeking extended lifetime or those being designed for long lifetimes. This issue also applies to the ageing of pressure tubes in PHWRs.
5) Shielding: PWR, BWR, VVER, ALWR, PHWR, APHWR	The ability of materials inside the reactor vessel to shield key components from irradiation induced damage is key to understanding their lifetime and ability to withstand transients. Also, protecting operating personnel and predicting the environment in which equipment must function depends upon predicting shielding performance.
6) Moderator coefficients: PHWR, APHWR	The coolant and moderator in PHWRs are separate. Thus the impact on reactivity of changes in the heavy-water moderator temperature, density, and poison concentration must be included in the safety analysis.

Description

Work in the area of reactor physics is of particular importance for the continued development of nuclear power. Key areas include:

- Reactor core and fuel-cycle physics issues at very high burn-up.

- Minor actinides recycling in LWRs.
- Physics related to plutonium management in the medium term (before GEN-IV systems are deployed).
- Effects of radiation on reactor internals and the reactor vessel at high fluence from current operation and extended plant lifetime.

Support facilities for providing data required for resolving these issues continue to be essential. Integral data collected from past experiments carried out on now dismantled or still existing facilities are not sufficient to cover the need of the evolutionary and next generation power systems. Specific new experiments are required, many of which can be covered by existing facilities, provided they are maintained and refurbished.

The experimental facilities, research reactors and tests in power reactors need to cover the measurement of the following parameters in critical and sub-critical configurations:

- Neutron multiplication and K-effective.
- Buckling and extrapolation length.
- Spectral characteristics.
- Reactivity effects.
- Reactivity coefficients.
- Kinetics measurements.
- Reaction-rate distributions.
- Power distributions.
- Nuclide composition.
- Shielding.

Computational models and codes have to cover core physics, coupled neutronics/thermal-hydraulics, radiation shielding, criticality safety, physics of the fuel cycle, materials activation, decay heating, and energy deposition. The necessary basis in integral experimental data for model development and validation must be available, maintained and expanded to meet requirements from advanced reactors.

The Nuclear Science Committee together with the OECD/NEA Data Bank, in collaboration with the member countries and other specialised institutions have developed data bases with evaluated and qualified experimental data shared internationally in addition to a large set of computer codes covering the different needs in nuclear applications modelling. The databases cover:

- Basic nuclear and chemical thermodynamics data.
- Radiation shielding a dosimetry experiments (SINBAD).
- Criticality experiments (ICSBE).
- Reactor core and lattice experiments (IRPhe).
- Data from coupled neutronics/thermal-hydraulics experiments and reactor operation.
- Fuel behaviour experiments (IFPE).

Table 3.1.3-2. **Issues versus facilities** (Reactor physics)

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue
1) MOX fuel data	PWR, BWR, PHWR, APHWR, VVER, ALWR	High	Medium
2) High burn-up fuel data	PWR, BWR, VVER, ALWR, APHWR	High	Medium
3) Coolant void coefficient	PHWR, APHWR, PWR BWR, ALWR	High	Medium
4) Neutron flux and spectra	PWR, BWR, VVER, ALWR, PHWR, APHWR	High	Medium
5) Shielding	PWR, BWR, VVER, PHWR, ALWR, APHWR	Medium	Medium
6) Moderator coefficients	PHWR, APHWR	Medium	Medium

Table 3.1.3-3. **Facilities in the reactor physics area**

Facility name	Applicability (Type of reactor)	Cost/Year operation	Replacement cost M US$
MINERVE (France)	PWR, BWR, ALWR	Medium	High
VENUS (Belgium)	PWR, BWR, ALWR, PHWR, APHWR	High	High
ZED-2 (Canada)	PHWR, APHWR	Low	High
PROTEUS (Switzerland)	PWR, BWR, VVER, PHWR, ALWR, APHWR	Medium	High
LR-0 (CZECH Republic)	VVER (issue 4). All (issue 5)	Medium	High
TCA (Japan)	PWR, BWR, ALWR	Low	High
KUCA (Japan)	PWR, BWR, ALWR	Low	High

Notes:
Specify range : Low, Medium, High.
Operation cost : Low is <1.0 MUS$/y; Medium is 1.0 MUS$/y; High is > 2 MU$/y.
Replacement cost : Low is <2 MUS$; Medium is 2-10 MUS$; High is > 10 MUS$.

Table 3.1.3-2. **Issues versus facilities** (Reactor physics)

Facility		
Name	**Importance of facility to resolution of the issue?**	**Versatility**
ZED-2	High	Critical assembly.
TCA	Medium	Reactor physics experiments.
PROTEUS	High	Large test volume.
VENUS	High	Zero power reactor.
MINERVE	Medium	Limited.
VENUS	High	Extensive.
PROTEUS	High	Extensive.
ZED-2	High	Critical assembly.
TCA	High	Extensive.
KUCA	High	Extensive.
PROTEUS	High	Extensive.
MINERVE	Medium	Limited.
VENUS	High	Zero power reactors.
LR-0	High	Extensive.
LR-0	High	Can mock-up various shielding configurations.
ZED-2	High	Critical assembly.
TCA	Medium	Zero power research reactor.
KUCA	Medium	Zero power research reactor.
MINERVE	Low	Critical assembly.
PROTEUS	High	Extensive.
VENUS	High	Extensive.

Table 3.1.3-3. **Facilities in the reactor physics area**

Issues covered	Capability	Planned duration of operation	Relative priority
2,3,6	Criticality safety.	Indefinite	0.8
1,2,3,6	Zero power critical reactor.	Indefinite	2.2
1,3,6	Critical assembly.	Through 2008	1.6
1,2,3	Critical facility: Large experimental volume (9 full length BWR fuel assemblies): Driven test zone with capability for fresh and irradiated UO_2 fuel can be used for strongly poisoned lattices at high void conditions and for non-LWR fuel.	Through 2011	2.2
4,5	Critical, VVER assembly with long experimental volume for neutron flux and reactivity measurements.	Through 2010	1.0
1,3,6	Reactor physics experiments, teaching and training.	Indefinite	1.2
3,6	Fundamental research and development, education and training.	Indefinite	0.8

Basic data needs, such as improved capture cross-sections of certain absorbers – hafnium, erbium and gadolinium, improved scattering cross-sections of oxygen, as well as improvement of yields of fission product isotopes in the fission of most heavy isotopes and decay schemes and energy yields of radioactive isotopes, are required. In general, higher than current resolution cross-section measurements, from thermal energies to several MeV are required for a number of important isotopes.

Such data will be useful in the evaluation of the accuracy of methods and codes through verification, validation and qualification studies and the measurements made in critical facilities, and irradiation measurements in reactors, play an essential role in the qualification studies. The interpretation of experiments is a driving force for the continuous improvement of computational methods and nuclear data.

Information on the activities and identified R&D needs by the NSC are provided.[4] An expert group addressing needs of research and test facilities in nuclear science has been set up and has held a first meeting in May 2005. The report that is being prepared covering this topic will provide further details on the issues discussed in the section.

3.1.4 Severe accidents

Introduction

Severe accidents (SA) are generally considered to be events beyond the traditional design basis of currently operating nuclear power plants. The prevention or mitigation of SAs is the largest contributor to reducing risk to the public from the operation of NPPs. SA scenarios involve an initiating transient, such as a loss-of-coolant accident, accompanied by the postulated failures of multiple safety systems, thus compromising the capability to shut down the reactor or maintain adequate cooling of the fuel, resulting in significant damage to the fuel (core melting), possibly leading to the release of significant amounts of radioactivity from the primary system into the containment. Under certain circumstances, the containment may also be postulated to fail or to be bypassed (e.g. through steam generator tube failure in a PWR), resulting in a major radioactive release to the environment. Although generally not considered during initial licensing, SAs have been assessed through specific plant reviews, generic analysis and the development of accident management programmes.

For many years important national and international programmes have been undertaken in the field of severe accidents and their results have been shared through international "networks." CSNI has played a major role in organising and administering co-operative research programmes in the area of severe accidents. These programmes include RASPLAV (conducted in Russia to assess the thermal load on the RPV lower head under core melt conditions), SNL-LHF (conducted in the U.S. to assess the mechanical behaviour of the RPV lower head under pressurised severe accident conditions), MCCI (conducted in the U.S. to assess ex-vessel molten core debris coolability), MASCA (conducted in Russia to measure the physical properties of molten core material) and SERENA (an analytical programme assessing the state of knowledge related to fuel-coolant interactions). In addition, CSNI sponsored an effort to assess accident management strategies and identify areas of consensus. The effort (called senior Group of Experts on Severe Accident Management – SESAM) published its report in the late 1990s. These programmes have contributed to the knowledge about severe accident phenomena, the resolution of questions related to severe accidents and the potential for accident management measures to successfully terminate or mitigate the accident progression. They have also

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4. NEA (2003), *Research and Development Needs for Current and Future Nuclear Energy Systems*, OECD, Paris.

served to maintain certain key facilities from premature shutdown. However, important issues remain and need to be studied to support the continued safe operation of nuclear power plants via severe accident management and/or reducing the potential for SA scenerios, as well as supporting the licensing of new LWR and PHWR designs.

Scope

The severe accident issues and phenomena that could benefit from additional research are related to reducing the remaining uncertainties in accident progression and mitigation and to understanding the safety implications caused by changes in plant design or operating characteristics (e.g. high burn-up fuel, MOX fuel).

The issues in this section are listed in Table 3.1.4-1 and arranged according to the phases of progression of a severe accident and the phenomena present in each of those phases, as follows:

- In-vessel phenomena (core heatup, clad/fuel melting and relocation, combustible gas generation, FCI).
- Ex-vessel phenomena (vessel failure, core-concrete interaction, DCH, FCI, combustible gas generation).
- Source term (quantity, chemical form, transport and timing of fission product release from the fuel, RCS and containment).
- Containment integrity (capability of containment to withstand severe accident conditions caused by combustible gas burning, decay heat, molten core attack).
- Accident management (actions that can be taken to terminate or mitigate the consequences of a severe accident).

Description

The prevention of severe (core damage) accidents and how to manage them if they do occur remains an important objective for the continued safe operation of LWR and PHWR nuclear power plants. Although in-vessel melt progression is fairly well understood, there remain significant uncertainties in predicting whether or not molten core material will remain in-vessel, the consequences of molten core material getting out of the reactor vessel (e.g., coolability, combustible gas, etc.), source term generation and the best accident management strategies for preserving RPV and containment integrity and reducing the amount of radioactive material available for release to the atmosphere.

Resolution of severe accident issues through prevention or mitigation is the goal of the remaining research. This can be accomplished by design changes, analysis showing the issue is of low safety significance or developing strategies to terminate or mitigate severe accidents prior to their resulting in the release of large quantities of radioactive material to the environment. To reduce uncertainties, current research should be conducted at sufficient scale to investigate the important phenomena and use real materials, whenever possible. PHWRs have similar severe accident issues as LWRs, however, the core melt progression in a pressure tube reactor presents additional challenges associated with propagation of pressure tube failure, fuel-coolant or fuel-moderator interaction and the potential to over pressurise the calandria and cause calandria and additional pressure tube rupture.

Table 3.1.4-1. Current severe accident issues

Table 3.1.4-2. Issues versus facilities (Severe accidents)

Table 3.1.4-3. Facilities in the severe accident area

Table 3.1.4-1. **Current severe accident issues**

Issues and relevant reactors	Description
A) In-vessel phenomena	
1) Pre-core melt conditions: PWR, BWR, VVER, PHWR, ALWR, APHWR	Understanding the conditions that can lead to core melt and the thermal-hydraulic conditions of the core prior to core melt are essential to understanding whether or not implementation of accident management strategies will be successful in preventing core melt (e.g. has flow blockage occurred?). Good knowledge of pre-core melt thermal-hydraulic conditions in the core will also help to refine accident management strategies so as to understand and be prepared for the outcome of actions taken by the operator. This issue is closely coupled with issue #17.
2) In-vessel melt progression: PWR, BWR, VVER, PHWR, ALWR, APHWR	The amount, composition, rate and timing of a core melt are important to determining the effectiveness of accident management measures and, the ability of the RPV or reactor calandria to maintain its integrity. The type of fuel (UO_2 or MOX), cladding material, burn-up and other factors which affect the composition of the melt, are also important in this determination. In-vessel melt progression includes relocation in the core and to the lower portion of the RPV and determines the heat load on the RPV during a core melt accident.
3) In-vessel fuel-coolant interaction: PWR, BWR, VVER, PHWR, ALWR, APHWR	Molten fuel contacting reactor coolant or moderator (PHWR, APHWR) may cause the rapid generation of steam and this is an important component of the load on the RPV or calandria.
4) Effect of air on core melt progression: PWR, BWR, VVER, ALWR	Core melt accidents where air is present in the RPV (such as during refuelling) could behave differently than those where no air is present. This could include the dynamics of the melt progression and the FP release.
5) Effect of high burn-up and MOX fuel: PWR, BWR, VVER, ALWR	The use of high burn-up or MOX fuel could change the dynamics of melt progression and fission product release. Data on these effects is needed to properly assess consequences and risk from accident sequences involving high burn-up or MOX fuel.
6) RPV pressure: PWR, VVER	Depressurising the primary coolant system is important during the in-vessel melt progression phase to reduce stress on the RPV and to facilitate water injection into the RPV. Accordingly, if the design does not have the capability to depressurise the primary system. It is important to understand the effect of high pressure on RPV and other RCS components' integrity and the subsequent effect on core melt progression. This is primarily an analysis issue.
7) Maintaining RPV integrity: PWR, BWR, VVER, PHWR, ALWR, APHWR	Maintaining the integrity of the RPV or reactor calandria vessel is important to terminating and confining a core melt accident, thus eliminating ex-vessel severe accident phenomena and their challenge to containment integrity. Cooling the RPV or reactor calandria both internally and/or externally are potential strategies for maintaining RPV integrity in the event of a core melt accident. However, higher core power densities will make it more difficult to maintain RPV integrity due to the higher heat flux on the RPV. Knowledge of RPV integrity as a function of heat flux is important in assessing the success of accident management strategies.
8) Pressure tube integrity: PHWR, APHWR	Maintaining the integrity of the pressure tubes in a pressure tube reactor is important for maintaining cooling of the fuel in the tube and preventing over-pressurisation and failure of the calandria due to high pressure water injection and/or molten fuel injection and FCI.

Table 3.1.4-1. **Current severe accident issues** (Cont'd)

Issues and relevant reactors	Description
B) Ex-vessel phenomena	
9) Ex-vessel melt progression and debris coolability: PWR, BWR, VVER, PHWR, ALWR, APHWR	The amount, rate, timing and spreading of molten core material released following RPV failure are important to determining the ability of the concrete basemat to maintain its integrity and the ability of an overlying pool of water or basemat cooling system to cool the debris and terminate the core-concrete reaction (i.e. ex-vessel melt coolability). Debris coolability can be affected by the amount of water overlying the core debris and the porosity of the debris or the strength of the crust formed on top of molten core debris. Obtaining the properties of the crust and underlying debris is important to understanding debris coolability and its uncertainties. In addition, high pressure melt ejection could result in molten core material being relocated to other parts of containment and a rapid pressure rise in containment due to the sudden release of steam and combustible gases from the RPV to containment.
10) Core-concrete interaction: PWR, BWR, VVER, ALWR, PHWR, APHWR	When molten core material leaves the reactor pressure vessel, it will likely come in contact with concrete. Depending upon the amount and depth of the molten core material and the composition of the concrete, various amounts of combustible and non-combustible gas will be released into the containment, thus raising its pressure. These gases can also be a source of additional energy if they ignite, thus causing additional pressure and temperature rise in containment. If not stopped, the core concrete interaction can potentially also penetrate the reactor containment basemat, thus failing containment. Understanding the rate and amount of gas generated from core-concrete interactions is important to understanding the potential for containment failure, the potential for success of mitigation strategies and, in the case of new plant designs, selecting materials and configurations to minimise core-concrete interactions.
11) Ex-vessel fuel coolant interaction: PWR, BWR, VVER, PHWR, ALWR, APHWR	Upon failure of the reactor pressure vessel, molten core material may fall or be ejected into water, if the reactor cavity has been partially or fully flooded. Such contact with water has the potential to cause rapid steam generation and, depending upon the amount, rate, fragmentation and mixing of the molten material, release a large amount of energy should be taken into account in assessing, which structural integrity of containment.
12) Combustible gas control: PWR, BWR, VVER, PHWR, ALWR, APHWR	Combustible gas (H_2 and CO) generated from metal-water reactions or core- concrete reactions in the in-vessel and ex-vessel phases of a core melt accident can ignite heat and/or pressurise containment, thus challenging containment integrity.

Table 3.1.4-1. **Current severe accident issues** (Cont'd)

Issues and relevant reactors	Description
C) Source term	
13) Fission product chemistry and release: PWR, BWR, VVER, PHWR, ALWR, APHWR	The amount, composition, chemistry, and timing of fission product release from the fuel through the reactor coolant system and into containment results in the source term available for release to the environment. This affects the onsite and offsite consequence analysis and protective actions which need to be planned for. The source term is also affected by the type of fuel (UO_2 or MOX) and burn-up level. In addition, the effectiveness of source term attenuation measures (e.g. sprays, water chemistry, filters) needs to be understood.
14) Post containment failure FP release to the environment: PWR, BWR, VVER, PHWR, APHWR, ALWR	Containment failure can lead to additional FP release due to revolitisation under depressurised conditions and/or due to air ingress. Understanding these phenomena is necessary for predicting the consequences and risk from accident sequences that fail and depressurise the containment.
D) Containment integrity	
15) Containment integrity: PWR, BWR, VVER, PHWR, APHWR, ALWR	Understanding the conditions which could lead to containment failure or bypass (e.g., pressure, temperature, and equipment failure) is important. Therefore, knowledge of the integrated effects of design basis and severe accident loads is necessary input to containment design. This issue provides input for structural analysis and containment failure modes testing (see Section 3.1.5 on integrity of equipment and structures). This is primarily an analysis issue.
16) Containment bypass-overheating and failing steam generator tubes: PWR, ALWR	During core heatup under severe accident conditions, significant amounts of heat are transferred by natural circulation of steam and hydrogen to the hot legs, surge line, and steam generators. This may cause one or more steam generator tubes to fail prior to failure of the hot leg, surge line, or vessel lower head. Understanding the behaviour of steam generator tubes is important to preventing containment by-pass scenarios. This is primarily an analysis issue.
E) Accident management	
17) Coolability of overheated core: PWR, BWR, VVER, PHWR, APHWR, ALWR	Accident management (which deals with the plant when core conditions are in excess of peak cladding temperature) has the potential to prevent as well as mitigate core melt accidents. In this regard, to develop and verify core melt prevention strategies, the symptoms, timing, and consequences of various strategies need to be understood to assess their effectiveness and enable the operator to anticipate the resultant consequences. This issue is closely coupled with issue #1. The effects of adding water to overheated cores still needs experimental verification.
18) Accident management strategies: PWR, BWR, VVER, PHWR, PHWR, ALWR	Once core melt begins, operator actions should be taken to terminate the accident as soon as possible. Therefore, it is important to understand the symptoms and implications of the various AM actions in order to develop the optimum AM strategies and actions. These actions can include depressurising the primary coolant system, injecting water, actuating containment sprays, etc. Specific AM actions depend upon the reactor design and accident scenarios. Analysing the potential for success and potential consequences of the various AM strategies/actions would rely on data developed in the resolution of many of the issues stated above. Relevant facilities are primarily the large integral thermal-hydraulic facilities discussed in Section 3.1.1.1.

Table 3.1.4-2. **Issues versus facilities** (Severe accidents)

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue
1) Pre-core melt conditions.	PWR, BWR, VVER, PHWR, ALWR, APHWR.	High	High
2) In-vessel melt progression.	PWR, BWR, VVER, PHWR, ALWR, APHWR.	High	Medium
3) In-vessel fuel coolant interaction.	PWR, BWR, VVER, PHWR, ALWR, APHWR.	Medium	Medium
4) Effect of air on core-melt progression.	PWR, BWR,VVER, ALWR	Low	Medium
5) Effect of high burn-up and MOX fuel.	PWR, BWR, VVER, ALWR.	Medium	Medium
7) Maintaining RPV integrity.	PWR, BWR, VVER, PHWR, ALWR, APHWR.	High	Medium
8) Pressure tube integrity.	PHWR, APHWR.	Medium	Medium
9) Ex-vessel melt progression and debris coolability.	PWR, BWR, VVER, ALWR, PHWR, APHWR.	High	Medium
10) Core-concrete interaction.	PWR, BWR, VVER, ALWR, PHWR, APHWR.	High	Medium
11) Ex-vessel fuel coolant interaction.	BWR, PWR, VVER, PHWR, ALWR, APHWR.	Medium	Medium

Table 3.1.4-2. **Issues versus facilities** (Severe accidents)

Facility		
Name	**Importance of facility to resolution of the issue?**	**Versatility**
PHEBUS	High	Can be used for various small bundle tests up to and beyond melting and tests on coolability of over-heated core.
Core disassembly test facility	High	Can be used to assess PHWR fuel bundle behaviour up to melting.
Fuel channel safety facility	High	Can be used to assess integrity of PHWR fuel channel.
QUENCH	High	Can be used to assess effectiveness of core-melt prevention strategies.
PHEBUS	High	Capable of small bundle in-core melt tests.
LIVE-FZK	Medium	Uses simulant material.
QUENCH	Medium	Can assess partially degraded core.
KROTOS	High	Capable of using prototypic materials.
MFMI	High	High and low pressure PHWR melt ejection.
TROI	Medium	Uses prototypic materials (20 kg).
PHEBUS	High	Capable of small bundle in-core melt tests with air.
VERDON	Medium	Can conduct hot cell experiment with irradiated fuel or air.
PHEBUS	High	Can conduct in-reactor experiments with high burn-up or MOX fuel.
VERDON (LECA-STAR)	Medium	Can conduct hot cell experiments with irradiated fuel.
MASCA/RASPLAV	High	Can obtain material properties using real materials.
LIVE-FZK	Medium	Uses simulant material.
Fuel ch. safety facility	High	Fuel channel thermal- mechanical behaviour.
MCCI	High	Can use prototypic materials.
VULCANO	High	Can use prototypic materials.
ARTEMIS	Medium	Uses simulant materials.
MCCI	High	Large-scale test (1 m^2) with real materials, simulated decay heat and with or w/o overlaying water cooling.
COMET-FZK	Medium	Uses simulant material.
VULCANO	High	Uses prototypic materials (oxide and metal).
ARTEMIS	Medium	Uses simulant material.
KROTOS	High	Can test with real materials.
MFMI	Medium	Facility for PHWR test configurations Uses simulant material.
TROI	Medium	Uses prototypic materials.

Table 3.1.4-2. Issues versus facilities (Severe accidents) (Cont'd)

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue
12) Combustible gas control	PWR, BWR, VVER, PHWR, ALWR, APHWR	High	Medium
13) Fission product release	PWR, BWR, VVER, PHWR, ALWR, APHWR	High	Medium
14) Post containment failure-FP	BWR, PWR, VVER, PHWR,	Medium	Low
15) Containment integrity	PWR, BWR, VVER, PHWR, ALWR, APHWR	High	Medium
16) Cont bypass overheating and failing S.G. tubes	PWR, ALWR	High	Medium
17) Coolabillity of overheated core	PWR, BWR, VVER, PHWR, APHWR, ALWR	High	High
18) Accident management strategies	PWR, BWR, VVER, PHWR, ALWR, APHWR	High	Medium

Table 3.1.4-2. Issues versus facilities (Severe accidents) (Cont'd)

Facility		
Name	**Importance of facility to resolution of the issue?**	**Versatility**
CTF	High	Separate effects combustion studies.
LSVCTF	High	Large-scale combustion studies.
PANDA	Medium	Large-scale-multi compartment capability for 3-D mixing.
MISTRA	Medium	Large-scale tests in multi-compartment configuration for mixing and distribution of H_2.
H_2-Tec	Medium	No compartments.
RUT	High	Large-scale tests in multi-compartment.
THAI	High	Addresses H_2, I and aerosols and their combined effects. Instrumentation can measure aerosol distribution and movement. Uses H_2.
TOSQAN	Low	Small scale, uses He as a simulant of H_2.
PHEBUS	High	Could be used for other tests.
VERDON	High	FP behaviour in containment.
ARTIST	High	Unique S.G. configuration: bundle, separator and dryer.
THAI	High	FP behaviour in containment.
EPICUR	Medium	I chemistry under SA conditions.
MAESTRO	High	Can test effect of B4C and air.
CHIP	Medium	I chemistry in RCS.
THAI	High	FP resuspension.
PANDA	Medium for current LWR. High for passive ALWRs	Large-scale tests in multi-comportment configurations for decay heat removal from containment.
MISTRA	Medium	Detailed 3-D instrumentation.
LSCF	High	1 700 m^3 containment T/H facility.
THAI	Medium	Capability for testing with H_2.
No facility identified. This is primarily an analysis issue		
RD-14M	High	Full height, full pressure PHWR T/H experiments.
QUENCH	High	Can test core melt prevention scenerios.
PHEBUS	High	Can simulate various SA conditions.
Uses data generated in the resolution of other SA and T/H issues. Facility needs primarily filled by other facilities, such as the large integral thermal-hydraulic facilities.		

Table 3.1.4-3. **Facilities in the severe accident area**

Facility name	Applicability (type of reactor)	Cost/year, operation	Replacement cost
KROTOS (France)	PWR, BWR, VVER, PHWR,	Low	High
MCCI (MACE) (USA)	PWR, BWR, VVER, PHWR, ALWR.	Medium	High
MASCA and RASPLAV (Russia)	PWR, BWR, VVER, ALWR, PHWR, APHWR.	Medium	High
ARTIST (Switzerland)	PWR, PHWR, ALWR, APHWR.	Medium	Medium
MISTRA (France)	PWR, BWR, VVER, ALWR.	Medium	High
RUT (Russia)	PWR, BWR, VVER ALWR, APHWR.	Low	High
PANDA (Switzerland)*	PWR, BWR, VVER, ALWR, APHWR.	Medium	High
TROI (Korea)	PWR, BWR, ALWR, VVER.	Medium	Medium
Hs-Tec (Germany)	BWR, PWR, ALWR, VVER.	Medium	Medium
CTF containment test facility (Canada)	ALL.	Low	Medium
LSVCTF large-scale vented combustion test facility (Canada)	ALL.	Medium	Medium
LSCF-large-scale containment facility (Canada)	ALL.	Low	Medium
Core disassembly test facility (Canada)	PHWR, APHWR.	Low	Medium
MFMI (molten fuel moderator interaction) test facility (Canada)	PHWR, APHWR.	Medium	Medium
RD-14M* (Canada)	PHWR, APHWR.	High	High
Fuel channel safety facility (Canada)	PHWR, APHWR.	Medium	Medium
VERDON (LECA-STAR)(France)	PWR, BWR, VVER, ALWR.	Medium	High
EPICUR (France)	PWR, BWR, VVER, ALWR.	Medium	Medium
CHIP (France)	PWR, BWR, VVER, ALWR.	Medium	Medium
MAESTRO (France)	PWR, BWR,VER, PHWR, ALWR, APHWR.	Medium	Medium
VULCANO (France)	PWR, BWR, VVER, ALWR.	Medium	High
QUENCH, FZK (Germany)	PWR, BWR, ALWR, VVER.	Medium	Medium
COMET-FZK (Germany)	PWR, BWR, ALWR, VVER.	Low	Medium
TOSQAN (France)	PWR, BWR, VVER PHWR, ALWR, APHWR.	Medium	Medium
ARTEMIS (France)	PWR, BWR, VVER PHWR, ALWR, APHWR.	Low	Medium

Table 3.1.4-3. **Facilities in the severe accident area**

Issues covered	Capability	Planned duration of operation	Relative ranking
3,11	FCI tests using prototypic materials.	Through 2009.	0.7
9,10	Large-scale ex-vessel core concrete interaction and cooling using prototypic materials.	Through 2009.	1.2
7	RASPLAV is a one-tenth scale (slab-geometry) facility to model RPV and molten core (using real materials). MASCA measures material properties with prototypic materials in small facilities.	Through 2010.	0.6
13	Retention of aerosols and fission products in a PWR steam generator.	Through 2007.	0.60
12,15	100 m^3 PWR containment. (0.1 linear scale) using He as a simulant. Flexible free and compartmented volumes. Capability for steam and gas injection. Spray system, 3-D instrumentation.	In danger of shutdown in 2007.	0.7
12	100-1 600 m^3 facility, with compartments for combustion experiment with or without steam	Shutdown in 2005.	0.6
12,15	Flexible large-scale facility for 3-D effects, multi-compartment containment behaviour, containment mixing and stratification studies and passive heat removal, Integral system behaviour. Has extensive 3-D instrumentation.	Until the end of the SETH project in 2006.	1.0
3,11	Can test ex-vessel SE with up to 20 kg of prototypic material.	Through 2015.	0.4
12,	Vessel for deflagration and detonation and recombiner testing. (100 bar capability).	Through 2012.	0.4
12	10 Mpa – Hydrogen combustion tests (6 m^3 and 10 m^3 connected vessels). Heated.	To be put in standby in 2007.	0.6
12	Hydrogen combustion large-scale vented facility (120 m^3) with heating, can be subdivided into 2-3 compartments.	To be put in standby in 2007.	0.6
15	1 625 m^3 facility for gas mixing and thermal hydraulics testing (no combustion testing). Uses He as a simulant.	Indefinite.	0.6
1	Facility for testing fuel bundle behaviour up to fuel melting.	Indefinite.	0.3
3,11	Facility to inject molten fuel simulant material into moderator fluid.	Through 2008.	0.4
17	Full height, full pressure PHWR T/H experiment.	2010.	0.3
1,8	Facility to test the integrity of fuel channels.	Indefinite.	0.7
4,5,13	FP release from irradiated fuel heated up to 3 000°K.	Indefinite.	0.9
13	Iodine chemistry under high radiation and high temperature.	Through 2009.	0.4
13	Source term: iodine chemistry in primary system.	Through 2010.	0.4
13	Source term under air oxidation, B4C degradation, fuel degradation.	Through 2010.	0.6
9,10	Core-concrete interaction test and debris coolability tests using prototypic materials.	Through 2007.	1.2
1,2,17	Reflood of over-heated core.	Through 2007.	1.0
10	Core-concrete interaction using simulant corium and decay heat simulation. Can add water.	Through 2006.	0.4
12	7 m^3 f facility to simulate H_2 (using He) mixing under SA conditions, including steam, sprays and aerosols.	Through 2008.	0.2
9,10	Debris coolability and core-concrete interaction using simulant material (molten salt).	Through 2008.	0.7

Table 3.1.4-3. **Facilities in the severe accident area** (Cont'd)

Facility name	Applicability (type of reactor)	Cost/year, operation	Replacement cost
PHEBUS (France)	PWR, BWR, VVER, PHWR, ALWR, APHWR.	High	High
THAI (Germany)	PWR, BWR, VVER, ALWR, APHWR.	Medium	Medium
LIVE-FZK (Germany)	PWR, VVER, BWR, ALWR.	Medium	Medium

Notes:
Specify range : Low, Medium, High.
Operation cost : Low is <1.0 MUS$/y; Medium is 1.0-2 MUS$/y; High is >2 MUS$/y.
Replacement cost : Low is <2 MUS$; Medium is 2-10 MUS$; High is >10 MUS$.
* Also discussed in thermal-hydraulic section.

Table 3.1.4-3. **Facilities in the severe accident area** (Cont'd)

Issues covered	Capability	Planned duration of operation	Relative ranking
1.2,4.5, 13,17	A 20 MWt pool type reactor, with a large central cell, connected to a Fission Products Bldg. (1/5000 scale containment) for conducting in-reactor fuel melt tests. Facililty is also capable of conducting other types of tests (e.g. thermal-hydraulic).	In danger of shutdown in 2007.	2.3
12,13,14, 15	60 m^3 facility can be used to test hydrogen burning, affects of containment sprays on mixing, iodine and aerosols. FP resuspension and recombiner performance.	Through 2009.	2.2
2,7	Test on formation of molten pools and relocation. 1/5 scale using simulant corium.	Through 2012.	0.7

3.1.5 Integrity of equipment and structures

Introduction

Many of the current problems with operating reactors are related to materials issues. While plants were initially designed for a 40 or more year lifetime, a number of unanticipated material problems have occurred. As plants continue to operate and seek to extend their lifetimes, and in some cases raise their power levels, issues related to component and structural integrity will continue to need to be investigated and solved. Accordingly ensuring the condition of equipment and structures is monitored and known becomes increasingly important. The ageing effects can include cracking, corrosion, erosion, cable insulation cracking, fatigue, embrittlement, etc., and can affect most plant equipment and structures. Identifying, monitoring and controlling the ageing effects are important to continued safe plant operation.

Scope

The issues addressed in this section are related to identifying the phenomena that are causing the problems, improving techniques for detecting and repairing problems and anticipating and preventing future problems before they become safety issues. The safety issues that could benefit from additional research are those that are associated with the ageing of existing plants and those that are associated with initiatives to improve plant performance or develop new designs and are listed in Table 3.1.5-1.

Description

The timely detection and mitigation of ageing degradation of plant systems, structures and components (SSC) are important to safety, so as to ensure their integrity and functional capability throughout plant service life. It means that managing the safety aspects of NPP ageing requires the implementation of effective programmes. General guidance for NPP activities which are relevant to the management of ageing, i.e. operation, inspection, testing, examination, maintenance and surveillance, are given in the frame of the International Atomic Energy Agency (IAEA) Nuclear Safety Standards (NUSSs) Code on the Safety of NPP. The largest activity in this area is related to the integrity of the primary pressure boundary, including reactor vessel and the associated piping. The integrity of the secondary side components is included also. The extension of the operating lifetime of NPPs is also important from the point of view of ageing.

Research programmes need to focus on the ways in which the reactor environment and operating conditions degrade the strength and integrity of equipment and structures over their operational lifetime. Therefore in the area of integrity of equipment and structures the research should address:

- The state of the material of the equipment or structures which may be affected by material composition, manufacturing processes or operational parameters.
- The loads imposed on the equipment and structures during operation (normal and transient operation, incidents, accidents and events) which are combined with the initial state of the stress and environmental conditions.
- The presence of defects which result from manufacturing practice and environmental attack.
- The safety margins available in the design.
- The sensitivity of the examination and testing methods applied.

Although the research needed to address the issues in Table 3.1.5-1 involves reactors, hot cells, autoclaves and, in some cases, other facilities, the focus of this section is on the reactors and other unique facilities needed. Due to the large number of hot cells and autoclaves and the fact that most member countries have those facilities, the SESAR group decided to not make specific recommendations on these facilities, but rather to recommend that each member country monitor the status of these facilities and bring to CSNI's attention any concerns regarding loss of critical infrastructure. Key hot cells and autoclaves are listed in Table 3.1.5-4.

Table 3.1.5-1. **Current integrity of equipment and structures issues**

A) Plant ageing issues/and relevant reactors	
1) Erosion/corrosion: PWR, BWR, VVER, PHWR	As plants age, environmental conditions can cause some materials to corrode or erode more than was originally anticipated in the design. This can lead to cracks, leaks or even large failures. The causes and corrective actions for these conditions needs to be understood and implemented to support continued operation and/or plant life extension. The corrosion can be both internal to the component or external due to leakage.
2) Embrittlement: PWR, BWR, VVER	Embrittlement of steels, particularly reactor pressure vessel steel, due to exposure to fast neutrons can reduce their ability to withstand thermal and mechanical stresses. The properties of embrittled materials and the ability to predict the amount of their embrittlement need confirmation by experimental data. Also, the effectiveness of any corrective actions needs confirmation.
3) Cracking and crack propagation: PWR, BWR, VVER PHWR	Cracking of materials and crack propagation (both steels and concrete) has caused problems at operating reactors. Cracking and crack propagation may be due to environmental conditions, fatigue or poor design. The causes and corrective action for cracking and crack propagation need experimental confirmation.
4) In-service inspection: PWR, BWR, VVER, PHWR	Inspection techniques to look for cracks, erosion or other ageing effects is important for early detection and correction, before they lead to a safety concern. Testing and validation of inspection techniques is essential.

Table 3.1.5-1. **Current integrity of equipment and structures issues** (Cont'd)

A) Plant ageing issues/and relevant reactors	
5) Cable insulation cracking: PWR, BWR, VVER, PHWR	Cable insulation (both power and I and C cables) can crack and become brittle over time. Environmental conditions affect the rate at which this happens. This can lead to shorts, fires or unexpected behaviour under conditions of high moisture (e.g., coolant leak, fire suppression). Detection and correction techniques need to be verified.
6) Pressure tube integrity: PHWR.	Corrosion and irradiation of PHWR pressure tubes during reactor operation can change material properties, in some cases making them more susceptible to failure. This could lead to random failures or common cause failures (e.g. due to seismic events), which in either case, could pressurise the calliandra and lead to more severe damage. Understanding the PHWR pressure tube condition, limits and failure modes is important to safety.
7) Long-term behaviour of concrete structures: PWR, BWR, VVER, PHWR	As plants age, concrete properties change and/or cracks develop. The safety implications of concrete ageing need to be understood to support the continued safe operation of existing plants and the review of requests to extend plant lifetime.
8) Containment integrity: PWR, BWR, VVER, PHWR	The conditions under which containments fail, and the timing and modes of failure, are important to understand to assess safety margins, consequences (i.e. FP release) and risk. Therefore, the structural analysis methods need experimental confirmation due to the complex nature of containment designs and penetrations. Currently, this issue is primarily an analysis issue using previous experimental data to assess analytical methods.
9) Flow induced vibrations: BWR, PWR	As current plants pursue power increases, the flow distributions, particularly in-vessel, and their contribution to mechanical loads and vibration of equipment needs to be understood. Predicting such flow distributions needs experimental data to validate analytical tools.
B) Performance improvement/new design issues	
10) New materials - existing plants: PWR, BWR, VVER, PHWR	To respond to materials problems (e.g. cracking, corrosion) on existing plants, new materials are being used in replacement components. The performance of these materials needs to be understood.
11) New materials: ALWR, APHWR	Future designs will try to use materials less susceptible to the problems occurring on existing plants. Qualification of these materials needs to be addressed.

Table 3.1.5-2. **Issues versus facilities** (Integrity of equipment and structures)

Issue	Applicability of issue	Safety relevance of issue	State of knowledge on issue
1) Erosion / Corrosion	PWR, BWR, VVER, PHWR	Medium	Medium
2) Embrittlement	PWR, BWR, VVER	High	Medium
3) Cracking and crack propagation	PWR, BWR, VVER, PHWR	Medium	Medium
4) In-service inspection	PWR, BWR, VVER, PHWR	Medium	Medium
5) Cable insulation cracking	PWR, BWR, VVER, PHWR	High	Low
6) Pressure tube integrity	PHWR	High	Medium
7) Long-term behaviour of concrete structures	PWR, BWR, VVER, PHWR	High	Medium
8) Containment integrity	PWR, BWR, VVER, PHWR	High	Medium
9) Flow induced vibrations	PWR, BWR	Low	Medium
10) New materials. Existing plants	PWR, BWR, VVER, PHWR	Medium	Medium
11) New materials. New designs	ALWR, APHWR	High	Low

Table 3.1.5-2. **Issues versus facilities** (Integrity of equipment and structures)

Facility		
Name	**Importance of facility to resolution of the issue?**	**Versatility**
Hot Cells*	Medium	There is a large number with varying capability.
Autoclaves*	High	There is a large number with varying capability.
LVR-15	Medium	Low power.
Halden	High	Many uses and versatile instrumentation.
ATR	High	High flux capability.
LVR-15	Medium	Low power.
JMTR	Medium	Medium power.
Hot Cells*	High	There is a large number with varying capability.
Halden	High	Many uses and versatile instrumentation.
ATR	Medium	High flux capability.
LVR-15	High	Various material testing loops (BWR, PWR).
JMTR	High	Medium power.
Hot Cells*	High	There is a large number with varying capability.
Autoclaves*	High	There is a large number with varying capability.
IQ	Medium	Corrosion measurement.
EPRI-N.C.	Medium	Many capabilities for ISI testing.
PANOZA	Medium	Gamma irradiation facility.
NRU	High	PHWR materials irradiation.
Hot Cells*	High	There is a large number with varying capabilities.
No applicable facilities identified		
Primarily an analysis issue		
No applicable facilities identified		
LVR-15	High	Various materials testing loops (BWR, PWR).
JMTR	High	Medium power.
Hot Cells*	High	There is a large number with varying capabilities.
Autoclaves*	High	There is a large number with varying capabilities.
SKODA	Medium	Mechanical testing.
Halden	High	Versatile uses and instrumentation.
NRU	High	High power capability.
ATR	High	High flux capability.
SKODA	Medium	Mechanical testing.
Autoclaves*	High	There is a large number with varying capabilities.
Hot Cells*	High	There is a large number with varying capabilities.
LVR-15	High	Has various material testing loops.
JMTR	High	Medium power.
Halden	High	Versatile uses and instrumentation.
NRU	High	High power capability.
ATR	High	High flux capability.

Note:

* In assessing issues versus hot cells and autoclaves, only a generic entry is included in Table 3.1.5-2, due to the large number of such facilities and the general similarity of their importance to issue resolution. Each country should monitor the status of their hot cells and autoclaves and bring to CSNI's attention any concerns regarding loss of critical infrastructure. Key hot cells and autoclaves are listed in Table 3.1.5-4 for information.

Table 3.1.5-3. **Facilities in the area of integrity of equipment and structures**

Facility name	Applicability (type of reactor)	Cost/year, operation	Replacement cost
Reactors			
NRU (Canada)	PHWR	High	High
Halden Reactor (Norway)	BWR, PWR, PHWR, ALWR	High	High
LVR-15 (Czech Rep.)	All types	High	High
JMTR (Japan)	PWR, BWR	High	High
ATR (USA)	PWR, BWR, ALWR	High	High
Other facilities			
Cobalt irradiation units PANOZA, PRAZDROJ (Czech. Rep.)	PWR, BWR, VVER	Low	Medium
SKODA (Czech. Rep.) ZZ800 Robertion Mechanical testing machine	BWR, PWR, VVER	Low	High
Inspection qualification (IQ) (Czech. Rep.)	VVER	Low	Medium
EPRI-North Carolina (USA)	PWR, BWR	Medium	High

Notes:
Specify range : Low, Medium, High.
Operation cost : Low is <1.0 MUS$/y; Medium is 1.0-2 MUS$/y; High is >2 MUS$/y.
Replacement cost : Low is <2 MUS$; Medium is 2-10 MUS$; High is >10 MUS$.

Table 3.1.5-4. **Key hot cells and autoclaves**

Hot cell facilities*	Cost/year, operation	Replacement cost
PSI-hot cells (Switzerland)	High	High
FHL (NDC) and NFD hot cells (Japan)	High	High
IMEF (Korea)	High	High
Chalk River Lab (Canada)	High	High
RIAR (KI) (Russia)	High	High
Argonne National * Laboratory (USA)	High	High
Oak Ridge National laboratory (USA)	High	High
Hot cells (Czech. Rep.)	High	High
Idaho National Lab. (USA)	High	High
LECI (France)	High	High
EC-JRC (Germany)	High	High
VTT – Hot Cell (Finland)	Medium	High
Autoclave facilities	**Cost/year, operation**	**Replacement cost**
PSI autoclaves (Switzerland)	Medium	High
Chalk River Lab. (Canada)	High	High
LECSI (France)	Medium	High
VTT Autoclaves (Finland)	Medium	High
Pacific Northwest National Laboratory (USA)	High	High

Note:
* Hot cells also discussed in fuel section (3.1.2)

Table 3.1.5-3. **Facilities in the area of integrity of equipment and structures**

Issues covered	Capability	Planned duration of operation	Relative ranking
6, 10, 11	135 MWt test reactor.	Through 2010	2.0
2,3,10,11	19 MWt test reactor capable of IASCC, corrosion / hydridling testing.	Indefinite	2.3
1,2,3,10,11	Materials test reactor (10 MWt).	Through 2018	2.3
2,3,10,11	Materials test reactor (50 MWt).	Through 2008 (under discussion to extend beyond 2008)	2.1
2,3,0,11	250 MWt test reactor.	Indefinite	2.2
5	Gamma irradiation facility, including cable ageing tests.	Indefinite	0.6
10,11	Mechanical testing laboratory.	Indefinite	0.8
4	Corrosion measurements.	Thru 2010	0.2
4	ISI qualification and training facility.	Indefinite	0.2

Table 3.1.5-4. **Key hot cells and autoclaves**

Issues covered	Capability
1,2,3,6,10,11	Hot cells with diagnostic equipment.
1,3,6,10	
1,3,6,10	
1,3,6,10,11	Hot cells with material and fracture testing capability.
1,3,6,10,11	
2,3,6,10,11	Hot cells for examination of small samples of irradiated materials.
2,3,6,10,11	Hot cells for examination of large samples of irradiated materials.
2,3,4,10,11	Examination of irradiated structural materials.
2,3,6,10,11	Can handle large items.
2,3,6,10,11	
2,3,6,10,11	
2,3,10,11	Fracture toughness radiation embrittlement, mechanical properties and microstructural characteristics.
Issues covered	**Issues covered**
1,10,11	Autoclaves for irradiated materials testing.
1,3,10,l11	Autoclaves, with corrosion loop.
1,3,10,11	Autoclaves for irradiated materials testing.
1,3,10,11	Corrosion, fatigue, stress corrosion cracking testing.
1,3,10,11	Examination of non-irradiated materials.

3.2 Issues and facilities not unique to the nuclear industry

3.2.1 Human and organisational factors

Introduction

The importance of the human and organisational factors in nuclear reactor safety has been recognised for a long time. These are issues that transcend engineering and involve social, psychological and other non-engineering factors and disciplines; for this reason, they are difficult to quantify and analyse. It is important, however, to address these issues in order to assess the root causes of human and organisational performance problems and attempt to remedy these by appropriate design, procedures, training, etc. It is also important to assess the degradation of safety related to such causes in order to establish the degree of severity of the problems and prioritise remedial actions.

Scope

Human and organisational issues apply to both currently operating and future plants. For currently operating plants, design solutions may be limited, whereas for future plants consideration of human factors early in the design process can contribute to good human performance. The human and organisational factors safety issues that could benefit from additional research are shown in Table 3.2.1-1.

Description

Currently operating plants

The assessment of operating experience can identify many of the root causes of human and organisational performance in operating plants. Solutions can then be identified. Also, upgrading I&C systems can lead to new human factors issues.

New reactor designs

New reactor systems may have different human interfaces, may respond in different ways to reactor operator actions. The continuing introduction of digital and computer controlled systems and advanced instrumentation also introduces some new human man-machine interface questions.

Organisational factors

Organisations, social systems and priorities and ways of operating within organisations are also evolving; an example is the liberalisation of the electricity markets in most countries. Globalisation of corporate activities and increased interactions with different cultures also introduce new organisational behaviour patterns, structures, etc. Such factors need to be considered and it is necessary to assess their impact on reactor safety.

Table 3.2.1-1. **Current human and organisational factors issues**

Issues and relevant	Description
1) Staffing ALWR, APHWR, HTGR	New designs are incorporating passive safety features and employing more automation. They are also being designed with longer response times. Accordingly, the role of the operator and the number of operational staff is changing. How to decide on correct staffing levels remains an issue. In addition the analysis of new tasks and the qualification of the staff are to be considered. Also, in both, new and existing plants, the effect of staffing cuts should be investigated. Many or most plants have cut staff as a result of market liberalisation and deregulation.
2) Human-machine interface: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	As more plants upgrade or introduce advanced instrumentation, issues related to how humans interface with the system must be addressed. This includes issues such as: • Role of the human vs. automation. • Navigation through software controlled displays. • Inputting commands. Testing of new human-machine interfaces will be useful. This should include verification and integrated system tests.
3) Organisational Influences: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	Organisational factors, market liberalisation, and deregulation can influence the performance of socio-technical systems and have negative effects on safety, factors such as communication, organisational learning, safety culture, etc., influence employee's knowledge and behaviour and have been fundamental factors in actual accidents. Although facilities are not relevant for this issue, understanding, monitoring and addressing these factors can improve safety. Accordingly, developing and maintaining expertise is of key concern. Objective measures of organisational performance would also be useful.
4) Human performance model: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	Developing reliable models of human performance will greatly enhance the accuracy of risk assessments and the ability to evaluate human related issues (e.g., procedures, training). Expertise, rather than facilities is the critical need.
5) Review of operating experience: PWR, BWR, VVER, PHWR	Many events at operating reactors have as their initiator or as an important element in the event human and/or organisational factor contributors. The review of operating experience to identify and correct those human and organisational contributors is key to maintaining/improving safety.

Table 3.2.1-2. **Facilities in the area of human and organisational factors**

Facility name	**Cost/year, operation**	**Replacement cost**	**Issues covered**	**Capability**	**Planned duration of operation**
Halden Reactor Project	High	High	1, 2	Ability to simulate control room environment and conduct with plant operators	Indefinite

Notes:
Specify range : Low, Medium, High.
Operation cost : Low is <1.0 MUS$/y; Medium is 1.0-2 MUS$/y; High is >2 MUS$/y.
Replacement cost : Low is <2 MUS$; Medium is 2-10 MUS$; High is >10 MUS$.

3.2.2 Plant control and monitoring

Introduction

All nuclear power reactors require plant control, and monitoring and protection systems (commonly referred to as instrumentation and control (I&C) systems), and there is a growing use of digital instrumentation and control I&C systems.

The issues associated with plant control and monitoring centre around ensuring that systems continue to perform reliably as they age, or are subjected to harsh conditions following an accident, and that replacement systems meet reliability goals. For the former, the challenges are to ensure that degradation mechanisms are understood and mitigated, and systems are appropriately qualified for post-accident operation. For the latter, in many cases it is neither possible nor desirable to replace existing systems with equipment of a similar vintage and capability. In that regard, the increasing use of digital I&C presents both opportunities and challenges. The primary opportunity is to replace systems with new equipment with enhanced functionality. The challenges are to ensure that the new systems perform with equal or better reliability. Demonstration of reliability requires consideration of hardware and software performance. A particular concern is that while enhanced functionality has benefits – for example the use of smart systems that have some assessment capability to improve operator response – it also has the drawback of increased complexity that makes reliability difficult to ensure.

Scope

The I&C areas within a nuclear plant can be divided into the following three categories (in order of decreasing safety significance and increasing functionality and complexity):

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

The plant control, and monitoring and protection system safety issues that could benefit from additional research are listed in Table 3.2.2-1.

Description

The challenges are to ensure that the new systems, primarily digital systems, perform with equal or better reliability. Demonstration of reliability requires consideration of hardware and software performance. A particular concern is that while enhanced functionality has benefits – for example the use of smart systems that have some assessment capability to improve operator response - it also has the drawback of increased complexity that makes reliability difficult to ensure.

- *LWRs (including VVERs)*. LWRs traditionally relied on analogue systems for plant safety and control. As these systems age, it is important to demonstrate that they continue to meet their original performance specifications, particularly when exposed to harsh environmental conditions. When the current systems reach the end of their design life, they are increasingly being replaced with digital systems. These digital systems must be shown to meet the same performance specifications, including qualification for expected environments. This will require demonstration of hardware and software reliability. In addition, increasing use is being made of advanced on-line monitoring and diagnostic systems to help manage the plant more reliably. The complex failure modes for these systems need to be investigated.

- *ALWRs*. The issues for LWRs are magnified for ALWRs. Advanced reactor designs will make increasing use of complex digital systems to lower costs, simplify operations and improve reliability, placing an increased burden on demonstrating that the digital systems are correctly deployed and will meet performance specifications. Advanced signalling techniques including multiplexing and the use of wireless technology will reduce cabling costs, while presenting challenges for assuring signal integrity.
- *PHWRs*. PHWRs started out using more digital systems for plant safety and control, but are now pretty much on par with LWRs, and therefore face the same issues.

Table 3.2.2-1. **Current plant control and monitoring issues**

Issues and relevant reactors	Description
1) Software quality and reliability: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	With the increasing use of digital instrumentation, control (I&C) and protection systems in currently operating plants, the extensive plans for complete control room retrofits using digital systems and the plans for their use in future plants, how to ensure the quality and reliability of the software used to perform safety functions is a growing concern. Software verification and validation methods, as well as, qualitative and qualitative software testing methods need to be assessed and their attributes and effectiveness established to aid in the review and regulation of software based systems important to safety.
2) Environmental qualification: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	The environment in which they operate (e.g. temperature, humidity, radiation, smoke, electro-magnetic/radio frequency interference, etc.) can affect the performance (reliability, failure rate, and failure mode) of digital I&C systems. Several standards developed by IEEE and IEC are used to guide qualification testing. For future plants with different environmental conditions new methods and tests will be needed to establish failure thresholds and modes of failure.
3) Digital system reliability: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	To understand the performance of digital systems, the integration of software, hardware and humans is needed. Although checks of the various system components individually are also required, they are not sufficient to confirm overall system performance and reliability. Facilities where such testing can be done in prototypical fashion are needed. Additionally methods are needed to support integration of digital system reliability models into current generation PSAs.
4) Wireless communication: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	The use of wireless communication for monitoring and control in nuclear power plant is expanding rapidly. The technology that supports the current generation of wireless applications were not designed for the challenging environments in nuclear plant that have the potential to disrupt signals. There are needs to do be research in order to understand possible effects before such communication is used for safety functions. Testing is needed to confirm design and performance. Additionally the security aspects of wireless communication needs to be explored.
5) On-line monitoring and advanced instruments: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	The use of advanced on-line monitoring and systems diagnostics in nuclear power plant instrumentation and control systems has added a higher level of complexity to the current generation instrumentation, control and protection systems, in addition to the complexity already added by the use of digital systems. Although these systems have the potential to reduce operator workload and increase system reliability, the new and complex failure modes for these systems needs to be investigated. Facilities where these systems can be tested and reviewed in a prototypical fashion are needed.

Table 3.2.2-2. **Facilities in the area of plant control and monitoring**

Facility name	Operational cost/year	Replacement cost
Sandia National Laboratory – Environmental qualification lab (USA)	Low	Medium
University of Virginia – Centre for Safety Critical Systems (USA)	Low	Medium
Ohio State University – INL Academic Centre of Excellence in Nuclear Instrumentation and Control and Safety Analysis (USA)	Low	Medium
Halden Reactor Laboratory – HAMMLAB and Software Engineering Laboratory (Norway)	Medium	Medium

Notes:
Specify range : Low, Medium, High.
Operation cost : Low is <1.0 MUS$/y; Medium is 1.0-2 MUS$/y; High is >2 MUS$/y.
Replacement cost : Low is <2 MUS$; Medium is 2-10 MUS$; High is >10 MUS$.

Table 3.2.2-2. **Facilities in the area of plant control and monitoring**

Issues addressed	Capabilities	Planned duration of operation
2, 5	Smoke temperature humidity and radiation testing	Currently in standby.
1, 3	Integrated systems testing. Simulates operating conditions, faults, instrumentation and equipment failures.	Indefinite. Has non-nuclear funding.
1,3,5,	Dynamic reliability analysis, advanced and high temperature sensor design and analysis	Indefinite.
1, 2, 5	Ability to simulate advanced control room and computer based control and diagnostic systems	Indefinite.

3.2.3 Seismic effects

Introduction

The seismic behaviour of components and structures has the potential to impact plant safety by simultaneously affecting all plant systems structures and components (i.e. common mode failure). To ensure plants are designed for such events, data to confirm seismic design and seismic safety evaluation methods need to be obtained.

Scope

Issues related to the seismic behaviour of components and structures are applicable to all reactor types, both currently operating and future plants. The magnitude of seismic events for which plants must be designed varies across member countries and with plant age, since seismic concerns are site specific and methods are evolving. For future plants (which are likely to be standard designs to be marketed worldwide), it is expected they will be designed to higher seismic standards to enable them to be sited in many member countries. Also, future designs may employ new features to improve plant seismic safety (e.g., below ground structures, seismic isolation devices) which will need experimental confirmation. The safety issues associated with the seismic behaviour of components and structures that could benefit from additional research are shown in Table 3.2.3-1.

Description

Data to validate seismic design methods is essential to ensuring plant seismic safety. The data needed is generally in one of the following two categories:

- Ground motion characteristics (e.g. frequency, magnitude, direction, etc.).
- Structural response to ground motion.

Seismic simulation facilities (i.e. shake tables and reaction walls) are needed to obtain data for the second category and are the subject of this section. Such facilities have the capability to simulate a wide variety of seismic motions (which are not specific to the nuclear industry) with scaled or full size equipment. Data generated from such simulations can be used to compare against analytical results or establish failure modes and thresholds, thus allowing quantification of margins to failure.

In July 2004, CSNI issued a report titled *Experimental Facilities for Earthquake Engineering Simulation Worldwide*, NEA/CSNI/R(2004)10, that summarised the capabilities of shake tables and reaction walls in member countries. Although the CSNI report concluded that none of the facilities were currently in danger of being closed, this section looks at the longer term prospects of facility value and use. Only large shake table (> 100 ton capacity) and large reaction wall (> 15 m tall) facilities are included, since these have the greatest versatility and capability.

Table 3.2.3-1. **Current seismic effects issues**

Issues and relevant reactors	Description
1) Confirmation of seismic design: APHWR, ALWR, HTGR	New plant designs are incorporating new safety features (e.g., passive ECCS, passive containment cooling) that need to be designed to withstand seismic events. Also, some designs may incorporate seismic isolation features to limit the transmission of ground motion to plant structures and equipment. In both cases, experimental confirmation of the design's ability to withstand seismic events and data to validate analytical tools will be necessary.
2) Below ground siting: ALWR, HTGR	Some future designs may locate all or some critical systems, structures and components below ground to protect them from external events. The response of below ground structures to seismic events needs experimental data to confirm analysis methods.
3) Continued safe operation: BWR, PWR, VVER, PHWR	As seismic events continue to occur and plants continue to age, data to confirm continued safe operation may be necessary. This data could be in the form of simulating the earthquake and the aged plant structure.
4) Seismic isolation devices: ALWR	Some future designs may incorporate seismic isolation devices into the design to reduce the seismic ground motion transferral to vital plant structures. Experimental confirmation of the performance of these devices should be obtained.

Table 3.2.3-2. **Facilities in the area of seismic effects**

Facility name	Cost/year, operation	Replacement cost	Issues covered	Capability	Planned duration of operation
Shake tables					
CEA TAMARIS Facility (France)	High	High	1,3,4	100 tons	Indefinite
NIED (Japan)	Low	High	1,3,4	500 tons	
NIED (Japan)	High	High	1,3,4	1 200 tons	
Public Works Research Facility (Japan)	Low	High	1,3,4	100 tons	
Reaction walls					
EC-JRC (Italy)	High	High	1,3	15 m	Indefinite
Building Research Facility (Japan)	Medium	High	1,3	25 m	

Notes:
* There are no facilities currently that can mock up below ground structures.
Specify range : Low, Medium, High.
Operational Cost : Low is <1.0 M US$/Y; Medium is 1.0-2 M US$/Y; High is >2 M US$/Y.
Replacement Cost : Low is <2 M US$; Medium is 2-10 M US$; High is >10 M US$.

3.2.4 Fire assessment

Introduction

Fires present a very demanding generic problem to plant safety, which has been demonstrated by some serious incidents of the past and by several plant specific safety analyses. The long experience of fire protection has resulted in well known codes and standards and good practices in design, construction and operation. Fire research has traditionally supported these goals by producing experimental results on active and passive fire prevention and mitigation. Theoretical modelling of fire is very demanding because of its multiple effects and because some key parameters are not well known. Modelling experience of fires has, however, gradually proceeded hand-in-hand with experimental work. In the past years zone models and multi-compartment zone models gained promising results. Zone models continue to be used and validated, due to their ease of use. During recent years, progress in computing power caused CFD models to become everyday tools for engineering applications. Zone models have been complemented by CFD codes, since CFD models are much more versatile, have no compartment size or configuration limitations, and allow prediction of all physical variables of fires if the fire size is known. Remaining problems include determination of fire size on solid fuels and development of reliable models for distributed fire loads like cables. Brute force methods like solving Navier-Stokes equations numerically are not within the foreseeable future. Promising theoretical and experimental results in micro-gravity and other aerospace related industries have indicated, that new analytical modelling for the flame spread is possible. The major efforts should be directed to work on this topic as a near-term goal to implement them into CFD codes. Particularly numerical fire simulation has prompted the development of deterministic and stochastic fire modelling. Comparisons of code predictions with relevant experiments, benchmarking, and other similar international comparisons of the codes during past years have given good reliance on the technology in general, and also selected the most useful numerical codes. The development of computational tools and accumulating experience is gradually enabling fire PSA on the same realistic level as in other branches of PSA.

However, additional experimental data is needed to assess the codes and new safety issues have arisen which may need experimental data to resolve.

Scope

The scope of this area includes fire safety issues related to plant design, fire analysis and quantitative fire risk assessment. The fire assessment safety issues that could benefit from additional research are shown in Table 3.2.4-1.

Description

Because of the generic nature of fire, nuclear power plant specific experimental facilities are not necessary. Most of the experimental work has been conducted using inexpensive small scale equipment.

Facilitated by the increase calculation power, the computational fire modelling is progressing fast. Much work in refining codes, models, computing algorithms and model validation is still needed until the methods are considered reliable enough for safety analysis. Additional experimental data is needed to fix crucial parameters of the modern fire models. The major problem is to calculate fire size, in connection with active air-solid interface of distributed fire loads like cables. Actions should be taken to test various proposed models of the emerging technology at all relevant scales, and implement the promising models in the best CFD codes.

Quantitative assessment of a fire scenario needs calculation using Monte Carlo techniques. Most of the needed deterministic fire models already exist, as well as some calculation platforms. Efforts of creating needed input data banks should be started soon.

While the computing and simulation models tools are, to some degree, able to utilise already available knowledge and tools, in practice several parallel development lines are needed; (a) data bases from the most safety relevant fire scenarios, (b) ignition and flame spread data measurements for the relevant materials, c) models of automatic fire protection, (d) and quantitative assessment of manual fire protection.

Table 3.2.4-1. **Current fire assessment issues**

Issues and relevant reactors	Description
1) Fire growth and propagation: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	Accurate modelling of fire growth and propagation is the key to determining the time and extent of equipment affected. While the best CFD simulation models are able to predict consequences for given fires rather satisfactorily common efforts should be taken to simulate most safety relevant typical scenarios as benchmarks, and make the results available in data bases as example and study material for plant specific work. Efforts should be taken to utilise the emerging technology of flame spread modelling on solids. Actions should be taken to test various proposed models at all relevant scales, and implement the promising models in the best CFD codes. Special efforts are needed to select the most suitable testing methods from existing or new concepts, which are needed to determine flame spread parameters for practical commercial products. For example for cables, none of the available methods are able to determine them at present. Benchmarking efforts are needed to transfer the technology from laboratories to industrial practice.
2) Hot shorts: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	Fires in cable trays can not only cause the loss of the cable but can also cause inadvertent signals in control cables affecting equipment. The likelihood and consequences of such “hot shorts” are not well understood or modelled in safety analysis. Experimental data is needed. There are some data already available for simple basic scenarios. Theoretical modelling is needed for assessing effects on systems performance.
3) Smoke propagation: PWR, BWR, VVER, PHWR, ALWR, APHWR, HTGR	The spread of smoke during a fire is generally not modelled, although most of the needed technology exists. Smoke can affect the operability of certain equipment and inhibit human fire fighting efforts by limiting access and visibility. However, this is not unique to the nuclear industry and steps should be taken to implement existing technology.
4) Equipment vulnerability: PWR, BWR, VVER, PHWR, ALWR, HTGR	When and how equipment fails under fire conditions is essential to fire assessments. This includes failures from heat, smoke, suppression system activation, shorts, etc. Experimental data will likely be needed to address this issue. There are some data and basic calculation models available on heat and smoke effects for some equipment. Establishing a data bank with benchmarking examples would be a good way to educate utilities to use that information. For Monte Carlo analyses establishing these data banks is mandatory.
5) High energy arcing faults: PWR, BWR, VVER, PHWR, ALWR, HTGR	Fires caused by arcing from high energy lines need to be modelled and included in risk assessments.

Table 3.2.4-2. **Facilities in the area of fire assessment**

Facility name	Cost/year, operation	Replacement cost	Issues covered	Capabilities	Planned duration of operation
GALAXIE (France)	High	High	1,3,4	Facility composed of compartments (from 1 to 680 m^3) and needs (up to 30 000 m^3/hr for fire intonsities up to 2 MW).	
Sandia Labs (USA)	Medium	Medium	1,3,4	Fire growth, propagation, effects of temperature and smoke on equipment.	In-stand by
Omega Point (USA)	Medium	Medium	2	Issues covered include spurious actuation of equipment.	Indefinite
U.S. National Institute of Standards and Technology (NIST, USA)	Medium	Medium	1,3	Issues covered include fire growth and propagation.	Indefinite
VTT (Finland)	Medium	Medium	1,4	Experimental/fire modelling	
DIVA (France)	Medium	Medium	1,3,4	Part of GALAXIE (5 rooms connected by a ventilation system) to investigate electrical cabinet fires, heat, and smoke propagation from room to room.	Through 2010

Notes:
Specify Range : Low, Medium, High.
Operational Cost : Low is < 1.0 MUS $/yr.: Medium is 1.0-2.0 MUS $/yr.: High is > 2 MUS $/yr.
Replacement Cost : Low is < 2.0 MUS $: Medium is 2-10 MUS $: High is > 10 MUS$.

3.3 HTGR unique safety issues and research needs

Introduction

In February 2002, CSNI hosted a workshop to discuss safety issues and research needs associated with advanced reactors. The results of this workshop were documented in an NEA publication entitled *Advanced Nuclear Reactor Safety Issues And Research Needs*, published in 2002. The scope of this workshop covered ALWRs, HTGR, and LMRs.

The SESAR/SFEAR group used the results of the CSNI workshop as a starting point in assessing ALWR, APHWR and HTGR safety issues and research needs. This was supplemented with more current information, when available, and reflects the views of the SESAR/SFEAR participants. The purpose of the assessment by the SESAR/SFEAR group is to identify those safety issues and research needs judged to be important to the safety and licensing of future HTGRs. The results of this assessment are for use by designers and regulators in planning and conducting programmes in support of future HTGRs. Due to the early stage of development of future HTGRs, no CSNI action is necessary or recommended at this time. It is also recognised that many safety issues associated with current plants and ALWRs/APHWRs also pertain to (e.g., human and organisational influences). These issues have been covered under current plant issues in Section 3.2 and are not repeated here.

Scope

This section addresses those issues unique to HTGRs. They fall in the same technical areas as covered in Section 3.1. The technical areas covered in this section are as follows:

- Thermal-hydraulics.

- Reactor fuel.
- Reactor physics.
- Severe accidents.
- Integrity of equipment and structures.

Description

Discussed below are the safety issues and research needs associated with each of the above technical areas. Where appropriate, facility needs are also discussed.

3.3.1 Thermal-hydraulics

A key safety attribute of HTGRs is their ability to remove decay heat, even in the event of loss of coolant. In addition, the large amount of graphite used as a moderator serves as a heat sink which provides for slow core heat-up during loss of coolant events. However, during normal and off-normal operation structural steel materials need to be protected from the high temperatures at which HTGR cores operate. Accordingly, the thermal-hydraulic area presents challenges for designers and regulators to ensure analytical tools can model the various modes of operation and accident conditions. This will require understanding and modelling of heat conduction, convention and radiation heat transfer and ensuring sufficient experimental data to validate the models. Discussed below are safety and research needs in the thermal-hydraulic area.

Data will be needed to evaluate the accuracy of codes and assess margins of safety. Test data can be obtained from facilities ranging in size and complexity from small-scaled component tests to scaled representations of the entire system. Past and ongoing HTGR research has been conducted at such reactor facilities as the AVR, Thorium Hochtemperaturreaktor (THTR) in Germany, the High-Temperature Engineering Test Reactor (HHTR) in Japan, and the 10-MWe High-Temperature Reactor (HTR-10) in China. These and other experimental programmes, such as the air-ingress tests done in the NACOK facility at FZ-Jülich and in a similar facility at Japan Atomic Energy Research Institute (JAERI), as well as the pebble-bed fluid-flow and heat-transfer tests performed in the SANA facility at FZ-Jülich, provide significant sources of measured T/Hs data. However, additional data is needed to investigate issues including pebble-bed hot spots inferred from the melt-wire test results at AVR, the incomplete mixing of reactor outlet helium and thermal stratification natural circulation under loss of forced circulation accidents, air and moisture ingress accidents with oxidation, and reactor cavity cooling.

Existing test reactors may be useful for obtaining some data for code assessment, particularly for steady state and some transient operation. However, scaled facilities testing may be needed to confirm plant performance under more severe accident conditions such as:

- Loss-of-coolant.
- Station blackout.
- ATWS.

Also, since power, decay heat and flux distribution are important to thermal-hydraulics calculations, coupled T/H and reactor physics codes may be necessary.

3.3.2 Reactor fuel

The safety of HTGRs is tied more closely to fuel performance than is the case for LWRs, ALWRs or PHWRs. The coated fuel particles are intended to retain fission products, even at high temperatures experienced during many accident conditions (e.g. up to 1600EC), and thus avoid conditions representative of widespread core damage. However, our current understanding of HTGR fuel safety limits and the factors that can affect them are not as well understood as those for LWR and PHWR fuel.

The safety claims of the HTGR design are inherent in the assumption of predicted performance of the TRISO-coated fuel particles (CFPs) under potential accident conditions. The HTGR fuel uses higher enrichment and operates at higher temperatures than conventional LWRs. The value of 1600EC is the proposed maximum permissible fuel temperature beyond which some degradation of the silicon carbide protective coating occurs. However, the integrity of the coated fuel particles is also dependent upon their manufacturing process and upon the environment within the reactor core.

Discussed below are the safety issues and research needs associated with HTGR fuel performance. Due to the importance of fuel performance to HTGR safety, the long lead time and the cost of fuel testing, this is one area where international collaboration should be strongly considered.

Virtually all of the past and ongoing worldwide irradiation testing research of HTGR fuel designs with TRISO CFPs include accelerated irradiations in MTRs. Although there subsequently was significant large-scale operating experience with these fuels in plants such as the AVR in Germany, accident simulation tests (i.e., fuel heatup test following irradiation) to qualify the fuel involved accelerated irradiations in MTRs. A well-established and thorough understanding of the mechanics and properties (e.g., creep) of CFP behaviour, failure, and FP release does not exist to allow one to conclude, with certainty, that fuel accident simulation tests following accelerated irradiations are conservative as compared to the rate of fuel irradiation in a power reactor. Accident simulation heatup tests, either after real time MTR fuel irradiations or after fuel irradiations in a power reactor, would be needed to resolve this issue.

Virtually all of the accident simulation tests for TRISO CFPs involved so called “ramp and hold” temperature increases. These typically consist of increasing fuel temperature at about 50EC/hr. up to a set temperature (e.g., 1600EC, 1700EC or 1800EC) and then holding the fuel at the set temperature for several hundred hours while FP release measurements are taken. The results of ramp-and-hold tests up to 1600EC, for qualified fuel, show that no additional CFP failures occur. However, in the FRG, there was at least one test in which the temperature was controlled to closely simulated the predicted accident heatup curve to 1600EC for a design-basis reactor coolant pressure boundary failure. For this test, CFP failures were observed to occur. Additional post-irradiation accident simulation tests that closely simulate the predicted temperature curve for a design-basis reactor coolant pressure boundary failure would be needed to determine if the traditional ramp-and-hold test accident simulation approach is conservative with respect to establishing CFP failure rates for postulated accidents.

Among the most limiting events that could challenge HTGR CFP integrity are those involving large-scale chemical attack, such as air intrusion following a large pipe break in the reactor coolant pressure boundary and moisture intrusion for a postulated heat exchanger tube failure with the reactor helium pressure falling below the heat exchanger tube pressure. Experiments on unirradiated HTGR fuel in air and water at HTGR accident temperatures have been conducted. These experiments have involved measurements of fuel oxidation due to air or moisture impurities in helium during fuel experimental irradiations. However, few experiments have been conducted on fully irradiated HTGR fuels that simulate the effects of large air or water ingress events. Additional post-irradiation accident simulation tests that closely simulate air or water intrusion events and take the fuel to the onset of CFP failures would be needed to fully assess the adverse effects of air and water corrosion on HTGR fuels and the margins to failure for such events.

Very limited testing has been conducted on fuels with TRISO CFPs to assess the capabilities and the margins to CFP failure for reactivity events involving a large energy deposition in the fuel over a very short time interval (less than 1 second). Some limited testing was conducted in Japan for a postulated control rod ejection accident in support of the HTTR licensing, this scenario was one of the limiting licensing basis events. Accordingly, in order to fully understand the margins to failure for reactivity events, fuel irradiation experiments involving such reactivity insertion events may be useful.

Only limited worldwide testing has been conducted on previously qualified FRG or U.S. HTGR CFP fuel for conditions that go beyond the maximum qualification operating temperature and maximum qualification fuel burn-up. In order to fully understand the margins to CFP failure and FP release for fuel operations beyond the maximum allowed operating temperature and design fuel burn-up limits, fuel experiments involving irradiation conditions beyond such limits would need to be conducted.

Accordingly, much experimental data will need to be generated to validate fuel performance. This will include:

- Data which explores the safety limits for fuel performance and FP release for conditions that are beyond the design basis for parameters important to fuel performance. These conditions involve fuel operating temperature, maximum fuel accident temperature, fuel oxidising environment, fuel burn-up, energy deposition and deposition rate in the fuel (due to reactivity accidents), beyond those that are expected to be examined by the fuel vendor or applicant. Such data will help established margins in fuel performance and establish appropriate operating and licensing limits.
- In-core hot spots. The results of melt-wire experiments conducted in the German AVR test reactor demonstrated the existence of unpredicted local hot spots under normal operating conditions in pebble bed cores. Such hot spots can be used to determine the maximum normal operating temperatures of the fuel. These hot spots may arise from a combination of higher local power density (e.g. due to moderation effects near the reflector wall or from chance clustering of lower burn-up pebbles), lower local bed porosity due to locally tight pebble packings, and reduced local helium flow due to the increase of helium viscosity with temperature. Whereas the slow evolution of loss-of-cooling heatup transients will tend to wash out any effects of pre-accident local flow starvation on subsequent peak fuel temperatures, the effects of higher local fission power densities will be retained throughout the heatup transient in the form of higher local decay heat powers. Therefore, data on the effect of decay-power hot spots, in particular, may be needed in evaluating the maximum fuel temperatures arising in pressurised or depressurised LOCAs.
- Physics of TRISO fuel irradiation in test reactors versus HTGRs. The extensive use of various test reactors for the irradiation testing of HTGR TRISO fuels raises questions about the non-prototypicality of the neutron energy spectra, accelerated fuel burn-up rates, and fuel temperature histories in the test reactors. Reactor-specific calculations of neutron fluxes and nuclide generation, depletion, and decay may need to be performed to provide a basis for analysing the sensitivity of computed fluences and fuel nuclide inventories to the neutronic differences between the test reactors and HTGRs. Of interest are the potential effects of such differences on TRISO fuel performance (i.e., FP retention) under normal and accident conditions. Such differences include the variations in irradiation temperature histories, burn-up rates, and neutron energy spectra that result in different neutron fluences, different rates of plutonium production and plutonium fission versus uranium fission, and, thus, different yields of important FPs. It is known, for example, that ^{236}U and ^{239}Pu give substantially different yields of various FPs that potentially affect TRISO fuel performance.
- Data on the effect of fuel fabrication parameters on fuel performance.

To obtain the experimental data needed to confirm fuel performance and validate fuel performance codes will require irradiation facilities and hot cells capable of steady state and transient testing. Prototypic irradiation and transient test conditions will need to be demonstrated. The effects of variations in fuel fabrication will also need to be tested to understand their impact on fuel performance. In this regard, test reactors capable of testing HTGR fuel at steady status and transient conditions are essential to generate data to establish fuel performance.

Existing test reactors, such as CABRI, NSRR and ATR will be import to maintain due to their ability to also test HTGR fuels.

3.3.3 Reactor physics

Since controlling fuel temperature is key to ensuring the safe operation of HTGRs (see Reactor Fuel below), reactor physics modelling and analysis is important in understanding power distributions and responses to off-normal events. Also, some future HTGR designs are attempting to incorporate passive shutdown capability (using Doppler feedback from ^{238}U as fuel temperatures increase) which will require a coupling of thermal-hydraulic and reactor physics analysis methods. Additionally, some reactivity insertion mechanisms are different in HTGRs than in LWRs (e.g. compaction of pebble fuel during a seismic event, water ingress acting as a neutron moderator) and these needs to be modelled and analysed.

All of the above will require good knowledge and data on cross sections, power distributions, reactivity coefficients and other reactor physics parameters. Discussed below are safety and research needs in the reactor physics area.

Nuclear analysis infrastructure development will be necessary to analyse reactor and fuel performance under a variety of normal and off normal conditions. Items needing research are:

- *Temperature coefficients of reactivity*. Validated analytical tools will be needed to confirm that the reactivity feedback effects from temperature changes in the fuel, moderator graphite, central graphite region, and outer reflector graphite are appropriately treated in safety analyses. Sensitivity analyses and validation against representative experiments and should be used to assess and account for computational uncertainties in the competing physical phenomena, including for example, the positive contributions to the fuel and moderator temperature coefficients associated with ^{135}Xe and bred fissile plutonium.
- *Reactivity control and shutdown absorbers*. The reactivity worths of in-reflector control and shutdown absorbers may be sensitive to tolerance in the radial positioning of the absorbers within the core. Analytical evaluations for reactivity control and hot and cold shutdown will need to account for absorber worth variations through burn-up cycles and the transition from initial core to equilibrium core loadings. Modelling uncertainties, and absorber worth variations caused by temperature changes in the core and reflector regions, xenon effects, variations or aberrations of pebble flow, and accidental moisture ingress will need to be understood to ensure valid predictions of absorber worth.
- *Moisture ingress reactivity*. Although the absence of high-pressure, high-inventory water circuits in closed Brayton cycle systems makes this issue less of a problem than in earlier steam cycle HTGRs, the effects of limited moisture ingress will nevertheless need to be evaluated for depressurised and pressurised accident conditions. Effects to be evaluated include the moisture reactivity (i.e. from adding hydrogenous moderator to the under moderated core), the effects of moisture on temperature coefficients (e.g. from spectral softening), shortened prompt-neutron lifetimes (i.e., faster thermalisation), and reduced worths of in-reflector absorbers (i.e. fewer neutrons migrating to the reflector).

- *Reactivity transients*. T/H-coupled spatial reactor kinetics analyses may be needed to assess axial xenon stability, as well as reactivity transients caused by credible events, such as overcooling, control rod ejection, rod bank withdrawal, shutdown system withdrawal or ejection, seismic pebble-bed compaction, and moisture ingress. Of particular importance is the need to identify any credible events that could produce a prompt supercritical reactivity pulse. Should any such prompt-pulse events be identified as credible, their estimated probabilities and maximum pulse intensities should be considered in establishing any related plans or requirements for pulsed accident testing and analysis of HTGR fuels. For loss-of-cooling passive shutdown events with failure of the active shutdown systems (i.e. anticipated transient without scram – ATWS), the delayed recriticality that occurs after many hours of xenon decay may also require spatial kinetics analysis models to account for the unique spatial power profiles and feedback effects caused by the higher local reactivity near the axial ends and periphery of the core where temperatures and xenon concentrations are lower.

To address the above, critical facilities capable of experimental confirmation of HTGR reactor physics analysis results are essential. Examples of such facilities are HTR-PROTEUS (Switzerland) and ASTRA (Russia).

3.3.4 Severe accidents

HTGR designers are attempting to reduce, as much as possible, the potential for and consequences of severe accidents. This is being done by incorporating passive safety features into the design (e.g., reactor shutdown, decay heat removal), by ensuring high quality fuel and by removing the potential for air and water ingress into the core. Likewise, understanding fission product release and transport mechanism is key to designing appropriate plant features to limit the release of radioactive material.

Discussed below are specific safety issues and research needs pertaining to HTGR severe accident behaviour, with emphasis on understanding the potential for the release of fission products (FP) from the fuel into the environment.

For HTGRs, both the types of sequences and the process by which FPs may be released from the fuel are different than current generation LWRs. In HTGRs, FPs may be released as a result of diffusion during normal operation, by rupture of coated fuel particles as a result of accidents, any by vaporisation during high-temperature degradation of the fuel.

The risk from HTGR operation is the risk from releases during normal operation and, from accidents involving rupture of coated fuel particles. Technical expertise and technical capability in the area of FP transport and behaviour during high-temperature fuel degradation is needed in order to assess the risk from HTGR operation. Because FPs released from the fuel are transported through the primary system and containment predominantly as aerosols, the offsite releases and offsite radiological consequences may be significantly reduced by FP deposition in the primary system and containment. Aerosol deposition occurs through a variety of mechanisms, such as gravitational settling, thermophoresis, and diffusiophoresis. Therefore, research activities should focus on FP transport and behaviour in the primary system and containment of other structural buildings.

HTGR designers will propose the accident source term to be used in their safety analysis based on models and methods that mechanistically predict FP release from the fuel. Should this be the case, it would differ from the traditional deterministic licensing approach to source term used by LWRs, which involves a pre-determined conservative upper bound for the accident source term. HTGR designers may also likely propose that HTGR plants utilise a non-leak-tight “confinement” structure, rather than a traditional leak-tight and pressure retaining containment structure. Accordingly, the safety analysis for modular HTGRs will largely hinge on the capability to confirm fuel FP release and associated uncertainties.

The qualification of HTGR fuels will be based on a wide range of technical areas and specific factors that are known to influence fuel performance, such as FP release and particle failure rates. The technical areas include fuel design; fuel manufacturing process, including process specifications; and statistical product specifications; design-specific core operating conditions; design-basis accident conditions, and postulated accident conditions beyond design basis. Key factors within the design-specific plant operating conditions that are known to affect fuel (particle) performance include fuel operating temperature, fuel burn-up, particle fast fluence, particle power, and fuel residence time in the core. The key factor affecting fuel particle performance during an accident (following the prior degrading effects of the operating conditions) is the peak particle temperature during the accident. Temperature increases can occur due to heatup events, which are caused by the loss of normal cooling, core power increases, or significant local reactivity insertion events. Other factors potentially affecting fuel (CFP) performance during accidents can include the effects of chemical attack (e.g., oxidation) on the fuel element and (possibly) the CFPs.

To predict CPF performance and a deterministic approach to the source term, capabilities in a number of interfacing technical areas will be needed. These include: (1) nuclear analysis for fuel burn-up, fast fluence (for particle coating behaviour), thermal fluence (for particle power and fuel kernel behaviour), and fuel particle power during reactivity events and (2) T/H analysis of normal operating core temperature distributions, accident core temperature distributions, and core temperature and flow distributions (for fuel oxidation during postulated air intrusion events). The FP release rates from the fuel during normal operation and postulated accidents are key inputs to the accident source term calculation. Accordingly, a range of significant fuel design, fuel manufacture, fuel quality, and fuel performance issues exist which will require research to resolve.

It should also be stated that understanding fission product release from the fuel will also require understanding the performance of other plant safety features, such as shutdown mechanisms, graphite behaviour, metallic structure behaviour, decay heat removal and FP transport. Accordingly there is an interrelationship among this section and other sections in this chapter.

3.3.5 Integrity of equipment and structures

During operation, various HTGR core internals, reactivity control elements and structural elements as well as system components will be exposed to higher temperatures than those in conventional LWRs. Issues that need further consideration would include a) applicability of the existing database of currently qualified high temperature materials, including the effect of various coolant impurity levels to the specific HTGR applications, b) the adequacy of procedures for evaluating material properties for HTGRs, and c) in-service inspection examination and surveillance plans and techniques.

There is also a need to establish a database related to the long-term performance and behaviour of graphite material under high temperature and irradiation levels expected during normal operation and accident conditions in HTGRs. This includes understanding the affect of the graphite manufacturing process and impurities on material properties and their changes due to temperature, irradiation, etc. Properties of interest are strength, irradiation creep, shrinkage, swelling, thermal conductivity, and fatigue. The issue of the loss of structural integrity of graphite material also needs careful consideration because it is one of the key issues which would impact the long-term performance of graphite structures, including the top- and bottom-reflector as well ad the end-of-life behaviour of all graphite structures. It is also important to understand graphite oxidation behaviour under accident conditions, such as air ingress.

Discussed below in more detail are the safety issues and research needs associated with graphite, metallic and concrete structures for HTGRs.

Graphite structures

In HTGRs, graphite acts as a moderator and reflector, as well as a major structural component, providing channels for the fuel and coolant gas, and control and shutdown rods, and acting as a thermal and neutron shield. Additionally, graphite components are employed as supports. Graphite also acts as a heat sink during reactor trip and transients. During reactor operation, many physical properties of graphite are significantly modified as a result of temperature, environment, and irradiation. Significant internal shrinkage, bowing, and stresses develop which may cause component failure, and/or loss of core geometry. Additionally, when graphite is irradiated to a very high radiation dose, swelling occurs, which also causes rapid reduction in strength, reducing the components structural integrity. In the event of an accident causing air ingress, subsequent graphite oxidation causes further changes in its physical and mechanical properties.

Some irradiation studies have been conducted on older graphites that are no longer available due to loss of raw materials supply and/or manufacturers. In addition, limited results are available at high levels of irradiation exposure. Thus, two key issues are the lack of data on irradiated properties of current graphites and the lack of data at higher doses of irradiation. Irradiated material properties are heavily dependent on the particular make-up of the graphite and the manufacturing process; therefore, at issue is whether the irradiated materials properties of the "old graphites" can be assumed to be the same as the "new graphites". Irradiation affects, and in many cases degrades, the physical and mechanical properties of the graphite. Important properties that change with irradiation are density, thermal conductivity, strength, and dimensions. Some of these changes are not linear with irradiation dose. Graphite strength initially increases with irradiation dose, then, at higher levels, it begins to decrease. With respect to dimensional changes, graphite initially begins to shrink with increasing dose, then, beyond turn-around, graphite beings to swell with increasing dose. During operation, thermal gradients and irradiation-induced dimensional and strength changes can result in significant component stresses, distortion, and bowing of components. These can lead to loss of structural integrity, loss of core geometry, and potential problems with insertion of control rods. At still higher doses, beyond turn-around, where the swelling makes the volume considerably greater than the original volume, graphite structures and fuel balls will start to disintegrate and experience total loss of integrity.

Accordingly, there is a need to conduct confirmatory research to establish an information base related to the long-term performance and behaviour of nuclear-grade graphite under the temperatures, radiation, and environments expected during normal operating and accident conditions. Potential loss of strength and of resistance to fatigue and creep, shrinkage, swelling, cracking, and corrosion during operation could impact the performance and function of the graphite core components. Various variables, including coke source, size, impurity, and structure; manufacturing processes; density; grain size; and crystallite size and uniformity determine the as-received and irradiated properties of the graphite component and need to be correlated to the irradiated graphite properties.

To evaluate the suitability of a particular graphite for HTGR application, property change data due to irradiation is needed in addition to the as-received properties. Development of irradiation data on graphite is difficult, expensive, and time consuming. Therefore, reactor designers/vendors have proposed to use radiation data from studies conducted on older graphites and attempt to use graphites produced in a similar manner. However, the as-received and irradiated graphite properties depend strongly on the raw materials and manufacturing processes. Small variations in these may have strong effects on the graphite properties. Since the exact raw materials and processes have changed and may continue to change in the future, there is a need to independently confirm whether a particular graphite will behave the same as the old graphites under operating irradiation conditions. To accomplish this without irradiation testing every time a change occurs in the graphite raw materials or processing, correlations are needed for predicting irradiated graphite properties and changes from the as-received graphite raw materials characteristics, composition, processing, and properties.

Graphite corrosion and oxidation can occur in HTGRs from oxidising impurities in (or added to) the helium coolant, from in-leakage during normal operation, or from air or water ingress during accidents. The oxidation of graphite is an exothermic reaction, and it is important to know the rate of heat generation, particularly during accidents. Oxidation also will remove the surface layers of graphite components resulting in loss of structural integrity. Further, oxidation will change the thermal conductivity and reduce the fracture toughness and strength of graphite components. The loss in strength may be due to attack of the binder. The oxidation rates vary for different graphites, and can be greatly affected by the impurities in the original graphite. Therefore, oxidation rate data is needed for the graphites proposed for new reactors.

The graphite structures will consist of thick and relatively thin pieces. The relatively thin structures may be manufactured differently from large structural blocks of graphite, and the mechanical and other properties may be different. Furthermore, the properties of the large block graphite will vary through the thickness of the block. The difference in properties between the sleeves and large blocks and through-thickness variations need to be established. The potential for different irradiated properties of sleeve graphite and large block graphite also needs to be evaluated.

There is a lack of standards for nuclear grade graphite. To ensure consistency, nuclear graphites should meet certain minimum requirements with respect to important properties, such as strength, density, and thermal conductivity, as is the case for materials used in other reactor systems. Specific impurities in the graphite might be detrimental to irradiation properties of the component, and they should be limited in nuclear graphites. Other elements, such as halides, which can be released during operation and cause degradation of other components in the reactor, should also be limited in nuclear grade graphite. Thus, standards need to be developed to establish the acceptable physical, thermal, and mechanical properties, composition, and manufacturing variables for nuclear grade graphite.

Metallic structures

National codes and standards for the design and fabrication of metallic structures and components for HTGR service conditions are needed. Although methodologies could be assembled from existing knowledge for calculating fatigue, creep, and creep-fatigue lives of components in high-temperature applications, appropriate data bases are needed for these calculations. Based on past experience and research, environmental effects play an important role in reducing fatigue lives and in enhancing degradation of materials. For example, small levels of impurities, such as less than 1 part per million of oxygen in the high-purity water coolant of LWRs, can greatly decrease fatigue life and resistance to stress corrosion cracking of metallic components. Because helium is inert, there has been a tendency top obtain design data in pure helium; in impure helium (but not all impurities were included) or in air. The effects of all important impurities, such as oxygen, in helium need to be taken into account with respect to reductions in fatigue and creep life and such data and understanding need to be developed.

To address degradation and ageing of metals in HTGRs, the effects of high-temperature helium with impurities, including oxygen, at levels present in HTGRs need to be evaluated with respect to stress corrosion crack initiation and growth rate, crevice corrosion crack initiation and growth rate, and cyclic crack growth rate. Low levels of impurities in high-temperature, high-purity aqueous environments are know to cause these types of degradation and to accelerate the crack growth rates. The potential exists for these phenomena to occur in a high-temperature helium environment with low levels of impurities.

Many alloys undergo solid state transformation and precipitation during elevated temperature exposures. These transformation reactions are known as ageing and can lead to embrittlement of the alloy. Ageing and embrittlement occurs, for example, in cast stainless steel components under temperatures and time conditions experienced in operating LWRs. At the operating temperatures of HTGRs, the reaction rates can be much higher (i.e., the ageing and embrittlement would occur sooner). The different alloys and higher temperatures of HTGRs would indicate potentially different ageing reactions and mechanisms, some of which could occur relatively rapidly and render the material embrittled and susceptible to cracking. The ageing reactions, as a function of time and temperature, in the different alloys used in important components of HTGRs need to be studied to establish the potential for material property degradation and embrittlement during the operating lifetime of HTGRs.

Another solid state reaction that occurs in stainless steels (and austenitic alloys) is called sensitisation. Sensitisation is caused by the precipitation of chromium carbides at the grain boundaries of the stainless steel. This precipitation normally occurs during slow cooling of the metal through high temperatures, such as when cooling from the high temperatures associated with welding. Formation of the carbides depletes the chromium from the grain boundary areas, rendering the stainless steel susceptible to intergranular stress corrosion cracking (cracking along the grain boundaries) in oxidising and impurity environments. The sensitisation rate is exponential with temperature, and at the higher operating temperatures of HTGRs, there is a potential for sensitisation during the lifetime of these plants, thus rendering the stainless steel components susceptible to stress corrosion cracking.

In some HTGR designs, the connecting pipe which carries hot helium from the core to the power conversion system is treated as a vessel because this pipe is designed, fabricated, and inspected to the same rules as a reactor pressure vessel. The consequence of this assumption is that a design-basis double-ended break is not considered for the connecting pipe, and therefore no mitigating systems are incorporated in the design. Considering this pipe as a vessel will require further investigation, because the pipe is of a much smaller diameter and therefore possesses a much thinner wall than a reactor pressure vessel designed to the same working pressure. If an unexpected degradation mechanism should initiate in the pipe, because of the thin wall, it can propagate through the wall in a relatively short time and possibly not be detected by ISI. Conversely, if an unexpected degradation mechanism were to initiate in a pressure vessel, it would require a long time to propagate through the greater wall thickness, allowing enough time to be detected by ISI.

Carburisation, decarburisation, and oxidation of metals in HTGRs are other phenomena that can lead to degradation caused by the operating gaseous and particulate environment. Carburisation is a phenomenon where carbon, either as a particulate or from carbon containing gases, diffuses into steel to form a surface layer with high carbon content. This surface layer may be hard and brittle, and have higher strength than the substrate. Differences in strength and other physical properties between the surface layer and substrate may lead to high stresses in the surface layer when the component is under load. In addition, carbides may form in the high carbon surface layer of stainless steel leaving the matrix depleted of chromium and susceptible to stress corrosion cracking and oxidation. Cracking, stress corrosion cracking, and oxidation can more easily develop in the surface layer which could then propagate into the component.

Decarburisation is a process whereby carbon is depleted from the steel depending on the composition of the gaseous environment. Depletion of carbon results in a softer steel and in reduced fatigue and creep lives. The presence of oxygen results in the formation of scale and general corrosion of metallic components. More importantly it can oxidise the graphite and render metallic components susceptible to stress corrosion cracking. To control the phenomena of carburisation, decarburisation, and oxidation, a very careful control of the level of different impurities in the coolant is required. Further, conditions that lead to avoidance of one of the above phenomena can lead to development of another. For example, to avoid carburisation, some HTGRs might use slightly oxidising conditions created by the addition of oxygen to the gas stream. However, this can lead to oxidation of graphite, general corrosion of metals, and an increased susceptibility to stress corrosion cracking. Some research has been conducted to study the phenomena described above; however, additional confirmatory research is needed to better define the conditions under which the phenomena occur for important metallic components of HTGRs. In addition, much of the previous research did not include oxygen in the gaseous environment. Since oxygen may be present in HTGRs at high enough levels to affect the progression of the above phenomena and to rescue fatigue life, creep life, and resistance to stress corrosion cracking, oxygen needs to be included n new experimental studies.

A number of potential issues related to the inspection of HTGR reactor components exist. Because some of these reactors are designed to operate for long periods of time between scheduled shutdowns for maintenance or refuelling, intervals between ISI may be long and the amount of inspection conducted limited. Therefore, the effectiveness of various ISI programmes as a function of the frequency of inspections and the number and types of components inspected needs to be evaluated. Additionally, many internal components are not easily accessible for ISI, and the impact of not inspecting these components needs to be assessed. An alternative to conducting periodic ISIs during reactor shutdowns is to conduct continuous online, nondestructive monitoring for structural integrity and leakage detection of the entire reactor or reactor components during operation. Techniques for continuous monitoring have been developed, validated, and codified for use in LWRs. If ISI for HTGRs cannot be conducted on a frequent enough basis and certain components cannot be inspected, then continuous monitoring may become necessary. The continuous monitoring techniques need to be evaluated and validated for the materials, environments, and degradation mechanisms of the HTGR.

In summary, the technical issues that need to be addressed for metallic structures are:

(1) Availability and applicability of national codes and standards for design and fabrication of metallic components for service in HTGR high-temperature helium environments.
(2) Lack of appropriate databases for calculating fatigue, creep, and creep-fatigue interaction lifetimes of components in high-temperature helium on degradation of components.
(3) Ageing behaviour of alloys during elevated temperature exposures.
(4) Sensitisation of austenitic alloys.
(5) Treatment of pipe as a vessel.
(6) Degradation by carburisation, decarburisation, and oxidation of metals in HTGRs.
(7) Issues related to inspection of HTGR reactor components.

For information related to recent and on-going work in the field of high temperature materials, the OECD Proceedings from a September 1999 conference on *Survey on Basic Studies in the Field of High Temperature Engineering* are a good reference.

Concrete structures

In HTGRs, concrete structures may be subjected to sustained high temperature. Research is needed to accumulate and expand existing data on the effects of high temperatures on the properties of concrete. The objective of additional research would be to investigate the change in concrete properties when it is subjected to sustained high temperatures. In the current American Concrete Institute (ACI) Code, the temperature limits specified for concrete are 150EF for long term, 200EF for normal use, and 300EF for abnormal conditions.

The operating temperatures of HTGR reactor vessels may be greater than those for currently licensed nuclear power reactors. Therefore, depending on the effectiveness of the reactor vessel insulation and cooling system, the concrete reactor building could experience a high-temperature environment. Elevated temperatures can reduce the strength of concrete due to additional shrinkage effects, as well as cause degradations such as cracking and spalling.

Additional research would include data accumulation and expansion of existing data bases. Significant information regarding high-temperature effects is available in the literature, including journals, conference transactions, and proceedings. Earlier research on LWR severe accidents conducted by Sandia National Laboratories also accumulated significant data on the effects of high temperatures on the properties of concrete. Oak Ridge National Laboratory has also assembled information on concrete subjected to high temperature. Lessons learned from facilities at which concrete was found to be subjected to high temperatures for long periods of time should also be investigated and used.

Chapter 4

CONCLUSIONS AND RECOMMENDATIONS

4.1 Introduction

This chapter provides the SFEAR group's conclusions and recommendations regarding key facilities unique to nuclear safety research in danger of being lost in the short term (next 1-2 years) and those that should be monitored by CSNI in the longer term (> 2 years) to ensure a minimum facility infrastructure to support operating LWRs and PHWRs in member countries and the licensing of ALWRs and APHWRs. This chapter also provides some general conclusions and recommendations (independent of short or long term) for CSNI consideration.

The overall strategy employed in developing recommendations was to identify minimum research infrastructure needs and the facilities that should be maintained to ensure that infrastructure is available. The group took an approach that focused on those facilities with unique capabilities and represented a substantial investment of resources such that, in the current climate of reduced funding for safety research, they would likely not be replaced, if lost.

This approach was considered the most practical given the large number of facilities discussed in Section 3.1 and the limited resources available for research.

Specifically, the following factors have been considered in determining whether or not to recommend CSNI action to preserve a facility or to recommend longer term monitoring of status.

- The importance of the facility to resolving the identified safety issues, based upon the relative ranking of a facility within a given technical area using the ranking approach described in the introduction to Chapter 3.[5]
- The versatility of the facility.
- The importance of the facility to maintain a minimum infrastructure of safety research capability (i.e. uniqueness and replacement cost).

Only facilities with medium or high replacement cost and high relative rankings were considered candidates for CSNI action. The conclusions and recommendations are summarised below, organised by the 5 technical areas discussed in Section 3.1. No conclusions or recommendations are provided for Sections 3.2 or 3.3, since the information therein is for information only.

4.2 General conclusions and recommendations

The following conclusions and recommendations pertain to both the short term and long term. They result from the group's observations and experience in carrying out the SFEAR activity and

5. It is important to note that the relative ranking is only valid within a given technical area. Comparison of relative ranking values between technical areas should not be done.

desire to develop a practical set of recommendations with facility preservation being a coordinated effort among the NEA standing committees. Specific general conclusions and recommendations are listed below:

- CSNI efforts at facility preservation should focus on large facilities, whose loss would mean the loss of unique capability as well as the loss of substantial investment that in the current climate of tight resources, would not likely be replaced. Such preservation also includes maintaining the expertise, knowledge, capabilities and personnel essential to infrastructure preservation. In this regard, it should be noted that due to previous CSNI efforts, several large facilities (i.e. PANDA, PKL, MACE, ROSA) have been kept active over the past 5 years, thus helping the current SFEAR effort. However, many large, expensive and unique facilities are projected to close over the next 1-5 years. Examples include thermal-hydraulic and severe accident facilities. In addition, many of the test reactors are old and will reach their end of life without substantial refurbishment. The loss of such facilities would severely detract from the nuclear safety research infrastructure. Additional discussion on a strategy for long-term facility preservation is discussed in item c) below.

- The NEA-Nuclear Science Committee (NSC) should take the lead to monitor the status of and make recommendations for actions to preserve key facilities in the reactor physics area. The facilities and information in this report in the reactor physics area represent the SESAR group's views on the safety issues and facilities important to nuclear safety research and are for NSC use in carrying out this responsibility.

- To help stimulate industry interest in facility and infrastructure preservation, it is recommended that both CSNI and CNRA take steps to encourage industry co-operation by emphasizing 1) the responsibility of industry to develop sufficient data to support their applications, 2) the benefits of co-operative research and 3) the value of preserving critical research infrastructure.

- Hot cells and autoclaves are essential to nuclear safety research. However, due to the large number of hot cells and autoclaves, it is impractical for CSNI to monitor their status. Accordingly, each country should monitor the status of these facilities and bring to CSNI's attention any concerns regarding loss of critical infrastructure.

- Certain safety issues have no large-scale facilities identified for the conduct of relevant research. The appropriate CSNI Working Groups should evaluate whether or not large-scale facilities are needed to support resolution of these issues. The issues that fall in this category are:

  – ECCS strainer clogging (Thermal-Hydraulic issue #6).
  – 3-D core flow distribution (Thermal-Hydraulic issue #12).
  – Long-term behaviour of concrete structures (Structural Integrity issue #7).
  – Flow induced vibrations (Structural Integrity issue #9).

4.3 Short-term conclusions and recommendations

The following recommendations are directed toward those actions that CSNI could take in the short term (2006-2007) to prevent the loss of key facilities in imminent danger of closure. To assess short-term concerns, the facilities in Section 3.1 were examined as to which ones are in danger of being shut down in the next 1-2 years. For the purposes of this report, that time period includes through year 2007. These facilities are shown in Table 4.1, by technical area, along with the issues and reactors they support. Also shown are their replacement cost, and relative ranking scores for that technical area. As can be seen from Table 4.1 there are a large number of facilities in danger of being shut down in the next 1-2 years. Discussed below, by technical area, are the short-term conclusions and recommendations for CSNI consideration. These are based upon trying to preserve those facilities having high relative ranking and versatility that would be hard to replace, if lost.

In the thermal-hydraulics area, four facilities are in short-term danger. Two of these facilities support PWR thermal-hydraulic work (PKL and APEX). However, there are other facilities for PWR-T/H work not in short-term danger (e.g., ROSA). Thus no recommendation for short-term action is needed for PWR T/H facilities. For BWR T/H facilities, both existing large integral BWR thermal-hydraulic test facilities (PANDA and PUMA) are in danger of being closed in the next 1-2 years. These facilities are unique and expensive and at least one should be maintained to be available for supporting research related to current or future BWR safety issues. Accordingly, preservation of one integral BWR thermal-hydraulic test facility (either PANDA or PUMA) is considered essential for preserving a BWR thermal-hydraulic research infrastructure. SESAR is of the view that PANDA is the preferred facility for preservation due to its scale, replacement cost and versatility (i.e. it is useful in the severe accident as well as thermal-hydraulic area). Accordingly, CSNI action is recommended in the short term to support a co-operative research programme in PANDA. It should be noted that CSNI actions resulting from the SESAR/FAP report played a major role in the preservation of PANDA over the past 5 years.

In the severe accident area, most facilities supporting the resolution of the following safety issues for BWRs, PWRs, VVERs and ALWR are in danger in the short term:

- Pre-core melt conditions.
- Combustible gas control.
- Coolability of over-heated cores.

Based upon a review of the facilities in short-term danger, listed in Table 4-1, the group concluded that the following facilities should be preserved due to their importance to resolution of the above issues (as illustrated by their high relative ranking), replacement cost, versatility, and value in long-term infrastructure preservation.

- PHEBUS.
- QUENCH.
- MISTRA.

Each of these is discussed further below.

PHEBUS is a unique facility representing a substantial financial investment, capabilities and expertise. Due to the high cost of its operation and the long timeframe necessary to plan and conduct experiments, it is not considered practical to propose that the CSNI organise a co-operative research programme in PHEBUS. Accordingly, it is recommended that PHEBUS be treated as a special case, with the French authorities taking the lead to propose and organise a future research programme using PHEBUS. In this regard it should be noted that a PHEBUS Expert Group has been organised to assess future experimental programmes in PHEBUS in the areas of LOCA (fuel response to LOCAs) and of severe accidents (fuel degradation and fission product release and transport). Both separate affects and integral tests are included in the assessment. The recommendations from this group should provide valuable input for justifying and planning future programmes in PHEBUS.

The QUENCH facility has been used extensively in the past to investigate pre-core melt conditions in LWRs. Although it is a unique facility in near-term danger, the group has concluded that any effort to preserve it for the long term should be dependent upon identifying a future experimental programme that can provide useful information beyond what has already been done in QUENCH. Accordingly, it is recommended that QUENCH be treated as a special case, with the German authorities taking the lead to propose a future research programme in QUENCH capable of generating useful new information. In this regard, it is also recommended that the appropriate CSNI Working Group (WGAMA) be requested to consider future uses for QUENCH and provide a recommendation to CSNI that can be factored into deliberations on the future of QUENCH.

The MISTRA facility has the capability for conducting experiments on combustible gas mixing and transport in multi-compartmental configurations with detailed instrumentation and helium as a simulant for H_2. As such, it can measure 3-D effects useful for assessing 3-D analytical tools. MISTRA complements the THAI facility which uses H_2 and can conduct experiments on H_2 combustion and aerosol distribution. THAI is not in near-term danger and is recommended for long-term preservation (see below). Accordingly, it is recommended that CSNI take action to reserve the MISTRA facility (so as to maintain the complementary infrastructure and expertise) by organising and conducting a co-operative research programme in MISTRA.

In the other technical areas (fuels, and integrity of equipment and structures) no short-term CSNI actions are recommended (i.e., note, PHEBUS was being addressed under severe accidents).

It should be recognised that implementation of the above recommendations are dependent upon interest and commitment of the “host countries” to provide sufficient resources to attract participation of other interested parties and the ability to propose experimental programmes relevant to resolution of the issues and of interest to member countries.

4.4 Longer-term conclusions and recommendations

In the longer term (beyond 2007), it is recommended that CSNI adopt a strategy for the preservation of a research facility infrastructure, based upon preserving unique, versatile and hard to replace facilities. The number and nature of these facilities should be based upon supporting currently operating LWRs and PHWRs and the licensing of future ALWRs and APHWRs. The strategy should include consideration of short and long-term priorities, cost of preservation (e.g., would the cost of preservation detract substantially from other programmes/facilities) and contingency plans in case of facility loss.

In this regard, many of the factors used in the report to arrive at conclusions and recommendations could be useful in developing a long-term strategy for assessing and initiating future co-operative research projects. These factors are:

- Cost of facility operation and replacement (i.e., limit CSNI involvement to large facilities needing multi-national support).
- Consistency with SFEAR recommended list of facilities for long-term preservation (discussed below).
- Ability to define a useful experimental programme (i.e., one that will provide information useful to the resolution of one or more safety issues).
- Relative priority, if there are multiple co-operative programmes proposed.
- Long-term planning to ensure the most important facilities receive the highest priority for long-term preservation (i.e. not first come first served). This would include assessing the long-term resource implications (i.e. consider impact of cost of a co-operative programme on resources available for other projects) and the host country’s long-term plans for the facility.
- Industry participation.
- Host country commitment.

It is recommended that CSNI consider factors such as the above in developing a strategy for facility preservation and in assessing and initiating future co-operative research programmes.

In addition, critical research capabilities and expertise are defined qualitatively in the OECD/NEA 2004 publication *Collective Statement Concerning Nuclear Safety Research*. Using this publication and the safety issues contained in Section 3.1 of this report, a table of critical research facility infrastructure needs was developed along with a list, by reactor type, of existing facilities that could fulfil those needs. These are shown as Tables 4-2 and 4-3, respectively. The facilities listed in Table 4-3 are those considered unique, hard to replace and identified as having high relative importance in their technical area, as discussed in Chapter 3 of this report. Accordingly, it is recommended that CSNI focus on these facilities in developing a strategy for long-term infrastructure preservation. CSNI should monitor the status of these facilities in the longer term with a goal of taking action, as appropriate, to ensure that critical facilities are available for each reactor type to meet the critical research infrastructure needs. In addition, for investigating safety issues associated with new reactors and technologies, CSNI should take an active role in encouraging and organising co-operative research efforts. This will also contribute to infrastructure preservation. Similar to the short-term recommendations above, host country interest will be an important factor in determining which facilities to preserve.

It should be noted that the SESAR/FAP report recommended that in the long-term thermal hydraulic facilities for each major reactor type should be maintained in North America, Europe and Asia. However, given the current situation with respect to safety research programme funding, the SFEAR group is of the opinion that this recommendation is no longer practical and recommends that the long-term strategy for facility preservation focus on ensuring at least one thermal-hydraulic facility for each reactor type be maintained worldwide.

Finally, it should be noted that test reactors (TRs) have for several decades supported the development and safety of nuclear power plants. However, most existing TRs have been in operation for a considerable period of time. This is the case for all relevant TRs in Europe, for which the operational lifetime is currently in the range 40-50 years. With the exception of the HANARO test reactor in the Republic of Korea, which has been in operation for about 10 years, the situation in Japan, in the U.S. and Canada, is practically the same as in Europe, i.e., the current TRs have, considerable age. Although one foresees that some of these reactors might continue operation for several years ahead, it is apparent that there will be a need to gradually update or supplement the existing ageing test reactors in the years to come. Due to the critical nature of test reactors to nuclear safety research and the fact that they are the most costly to replace, particular attention should be paid to test reactor preservation.

Table 4.1. **Facilities in danger in the short term** (2006-2007)

Facility	**Applicable reactors**	**Issues addressed**	**Replacement cost**	**Relative ranking**
Thermal-hydraulics				
PKL (Germany)	PWR	1. Boron dilution. 2. Passive safety system performance. 3. Non-pipe breaks. 4. SGTR 8. Two phase natural circulation. 9. Thermal stratification. 13. Flow distribution in cold legs and downcomers. 14. Accidents initiated during shutdown.	High	2.9

Table 4.1. **Facilities in danger in the short term** (2006-2007) (Cont'd)

Facility	Applicable reactors	Issues addressed	Replacement cost	Relative ranking
Thermal-hydraulics				
APEX (USA)	PWR, ALWR	1. Boron dilution. 2. Passive safety system performance. 4. SGTR. 8. Two phase – natural circulation. 9. Thermal stratification. 14. Accidents initiated during shutdown.	High	1.6
PUMA (USA)	BWR, ALWR	2. Passive safety system performance. 5. Stability and power oscillations. 8. Two-phase natural circulation. 9. Thermal stratification. 14. Accidents initiated during shutdown.	Medium	1.6
PANDA (Switzerland)	PWR, BWR, ALWR	2 Passive safety system performance. 8. Two-phase natural circulation. 9. Thermal stratification. 10. thermal cycling. 14. Accidents initiated during shutdown.	High	1.9
Fuels				
PHEBUS	PWR, BWR, VVER, ALWR	1. Response to LOCAs.	High	1.0
Reactor physics: NSC lead no near-term concerns				
Severe accidents				
CTF (Canada)	All	12. Combustible gas control.	Medium	0.6
LSVCTF (Canada)	All	12. Combustible gas control.	Medium	0.6
ARTIST	PWR, PHWR, APHWR, ALWR	13. FP chemistry and release.	Medium	0.6
VULCANO	PWR, BWR, VVER ALWR	9. Ex-vessel melt progression. 10. Core-concrete interaction.	High	1.20
PHEBUS (France)	BWR,PWR,VVER,ALWR,PHWR,APHWR	1. Pre-core melt conditions. 2. In-vessel melt progression. 4. Effect of air on core melt progression. 5. Effect of HB and MOX Fuel. 13. FP chemistry and release. 17. Coolability of overheated cores.	High	2.3
MISTRA (France)	PWR, BWR, VVER, ALWR	12. Combustible gas control. 15. Containment integrity.	High	0.7
PANDA (Switzerland)	BWR, ALWR	12. Combustible gas control. 15. Containment integrity.	High	1.0
QUENCH (Germany)	PWR, BWR, VVER, ALWR	1. Pre-core melt conditions. 2. In-vessel melt progression. 17. Coolability of overheated cores.	Medium	1.0
COMET-FZK (Germany)	PWR, BWR, VVER, ALWR	10. Core concrete interaction.	Medium	0.4
Integrity of Equipment and Structures – No near-term concerns				

Notes:

Replacement cost: High $\geq$ 10 Million USD.
Medium = 2-10 Million USD.

Table 4-2. **Critical research facility infrastructure needs**

Technical expertise needed	Facility capability needs	Important factors for facilities
Thermal-hydraulics: modelling and analysis	Large-scale integral test facility for each reactor type.	Scale, temperature and pressure capability are key factors. Also, the completeness of the facility with respect to factors such as: auxiliary systems, number of loops and instrumentation capability are important.
Fuels: performance and phenomena	Test reactor for steady state and reactivity insertion testing. Hot cell for PIE and simulated LOCA testing.	Ability to achieve representative values of energy deposition in transient testing. Adequate linear heat rating, burn-up and in-core instrumentation for steady state testing. Ability to do experiments with MOX and high burn-up fuel. Hot cells for full length and pin segment PIE.
Reactor physics: modelling, cross sections, parameters and analysis	Critical facility for measuring physics parameters and performing benchmark experiments.	Ability to do experiments with MOX and high burn-up fuel.
Severe accidents: phenomena, progression, modelling and analysis	In-reactor or ex-reactor testing of FP release and transport, core debris cooling, combustible gas control and AM strategies.	Use of prototypic materials and large scale are important.
Integrity of equipment and structures: materials behaviour, structural design	Test reactors for irradiating material samples under controlled conditions. Hot cells for examining large and small irradiated material samples. Autoclaves for material testing.	Ability to achieve fluence and other prototypic conditions (e.g. temperature, simulate impurities, stress, etc.). Hot cells and autoclaves for ex-reactor testing of irradiated materials.

Table 4.3. **Critical facilities to be monitored in the long term**

Technical area	BWR	PWR	VVER	PHWR/ APHWR	ALWR
Thermal-hydraulics	PANDA[1]	LSTF/ROSA PKL ATLAS	PSB-VVER PACTEL	RD-14-M	LSTF/ROSA PKL PANDA[1] ATLAS
Fuels[2]	Halden NSRR PHEBUS	Halden NSRR CABRI PHEBUS	Halden MIR CABRI PHEBUS	Halden NRU	Halden NSRR CABRI PHEBUS
Reactor physics*	Proteus Venus	Proteus Venus	Proteus LR-O	Proteus ZED-2 Venus	Proteus Venus
Severe accidents	*Integral testing* PHEBUS[1]	PHEBUS[1]	PHEBUS[1]	PHEBUS[1]	PHEBUS[1]
	In-vessel phenomena – QUENCH[1] – VERDON – KROTOS	– QUENCH[1] – VERDON – KROTOS	– QUENCH[1] – VERDON – KROTOS	– Fuel Channel Safety Facility – VERDON – KROTOS	– QUENCH[1] – VERDON – KROTOS
	Ex-vessel phenomena – MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]	– MCCI – VULCANO – THAI[2] – KROTOS[4]
	Containment mixing/combustion – PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]	– PANDA[1] – LSCF – THAI[2] – MISTRA[1]
	Accident management Uses data generated in the resolution of other severe accident and thermal-hydraulic issues. No unique facility needs				
Integrity of equipment and structures[3]	Halden JMTR LVR-15 ATR	Halden JMTR LVR-15 ATR	Halden JMTR LVR-15	Halden NRU LVR-15	Halden JMTR LVR-15 ATR

* (Included for completeness; NSC to monitor status).

Notes:
1. Assumes actions will be taken in the short term to preserve these facilities.
2. Assumes on-going effort to initiate a co-operative research programme will be successful.
3. Due to the large number of hot cells and autoclaves, each country should monitor the status and identify concerns.
4. Experimental programme under discussion in the CSNI-SERENA programme.

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