Nuclear Safety
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# Nuclear Safety Research Support Facilities for Existing and Advanced Reactors: 2021 Update



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# Nuclear Safety Research Support Facilities for Existing and Advanced Reactors: 2021 Update


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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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- to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

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*Cover photos: PKL facility, Germany (Areva); The PANDA reactor pressure vessel (Paul Scherrer Institute, Switzerland).*

# Foreword

At its 61st meeting in June 2017, the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) decided to establish a Senior Expert Group on Safety Research (SESAR) to update previous assessments of capabilities and facilities required to support the safety of nuclear installations. The NEA issued a report on this activity in 2001 entitled *Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk*. Six years later, a follow-on activity resulted in the publication *SESAR/SFEAR: Nuclear Safety Research in OECD Countries – Support Facilities for Existing and Advanced Reactors* (2007).

The SESAR/SFEAR2's mandate is as follows: "The Senior Expert Group on Safety Research/ Support Facilities for Existing and Advanced Reactors 2 (SESAR/SFEAR2) is responsible for reviewing and updating the previous SESAR assessments of research facilities required to support the safety of nuclear installations. The group shall recommend actions to be taken by the CSNI and its member countries to facilitate broader use and sustained operation of essential research facilities required to support nuclear safety."

The activity described in this report builds upon and updates the previous work, expanding its scope to cover advanced reactors, including evaluations performed by the Task Group on Advanced Reactor Experimental Facilities (TAREF) and the emergence of several proposed molten salt and small modular reactor designs. In addition, although the need to maintain experimental databases was recognised as an important issue, it was not treated specifically in previous reports; the present report therefore makes some direct recommendations regarding database maintenance.

# Acknowledgements

The contributions of the following members of the Senior Group of Experts on Nuclear Safety Research/Support Facilities for Existing and Advanced Reactors 2 (SESAR/SFEAR2) to review and write this report are gratefully acknowledged. In particular, Joanne Ball provided capable leadership as chair of the group and ensured the timely and efficient completion of the report.

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# List of abbreviations and acronyms

| | |
|---|---|
| CEA | Alternative Energies and Atomic Energy Commission (France) |
| CFD | Computational fluid dynamics |
| CFP | Coated fuel particle |
| CHF | Critical heat flux |
| CNL | Canadian Nuclear Laboratories |
| CSNI | Committee on the Safety of Nuclear Installations (NEA) |
| CTF | Containment Test Facility (Canada) |
| D-LOFC | Depressurised loss-of-coolant |
| DOE | Department of Energy (United States) |
| ECCS | Emergency core cooling system |
| FBTR | Fast Breeder Test Reactor (India) |
| FP | Fission product |
| GFR | Gas fast reactor |
| HFR | High Flux Reactor |
| HTGR | High-temperature gas reactor |
| HTR | High-temperature reactor |
| I&C | Instrumentation and control |
| IAEA | International Atomic Energy Agency |
| IFE | Institute for Energy Technology (Norway) |
| IRSN | Institute for Radiological Protection and Nuclear Safety (France) |
| JAEA | Japan Atomic Energy Agency |
| JMTR | Japan Material Testing Reactor |
| LOCA | Loss-of-coolant accident |
| LOFC | Loss-of-forced circulation |
| LSCF | Large-Scale Containment Facility (Canada) |
| LSTF | Large Scale Test Facility (Japan) |
| LWR | Light water reactor |
| MCCI | Molten core concrete interaction |
| MOX | Mixed oxide |
| MSFR | Molten salt fast reactor |
| MSR | Molten salt reactor |
| MSRE | Molten Salt Reactor Experiment |

| | |
|---|---|
| NC | Natural circulation |
| NEA | Nuclear Energy Agency |
| NRU | National Research Universal (Canada) |
| NSC | Nuclear Science Committee (NEA) |
| NSRR | Nuclear Safety Research Reactor (Japan) |
| P-LOFC | Pressurised loss-of-coolant |
| PHWR | Pressurised heavy water reactor |
| PSI | Paul Scherrer Institute (Switzerland) |
| PWR | Pressurised water reactor |
| RIA | Reactivity insertion accident |
| RPV | Reactor pressure vessel |
| SA | Severe accident |
| SARNET | Severe Accident Research Network |
| SBO | Station blackout |
| SESAR | Senior Expert Group on Safety Research (NEA) |
| SETH | SESAR Thermal-hydraulics (NEA) |
| SFEAR | Support Facilities for Existing and Advanced Reactors (NEA) |
| SFP | Spent fuel pool |
| SFR | Sodium fast reactor |
| SG | Steam generator |
| SGTR | Steam generator tube rupture |
| SiC | Silicon carbide |
| SMR | Small modular reactor |
| SSC | Systems, structures and components |
| STEM | Source Term Evaluation and Mitigation (NEA) |
| TAREF | Task Group on Advanced Reactor Experimental Facilities (NEA) |
| TREAT | Transient Reactor Test Facility (United States) |
| VENUS | Vulcan Experimental Nuclear Study (Belgium) |
| VVER | Water-water energetic reactor |
| WGAMA | Working Group on Analysis and Management of Accidents (NEA) |
| WGFS | Working Group on Fuel Safety (NEA) |

# Executive summary

The OECD Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) decided in June 2017 to establish a Senior Expert Group on Safety Research (SESAR) to update previous assessments of capabilities and facilities required to support the safety of nuclear installations. The NEA issued a report on this topic in 2001 entitled *Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk* (NEA, 2001). In 2007, a follow-on activity resulted in the publication of a second report: *SESAR/SFEAR: Nuclear Safety Research in OECD Countries – Support Facilities for Existing and Advanced Reactors* (NEA, 2007).

This report builds on and updates the previous work, expanding its scope to cover additional advanced reactor types, including evaluations performed by the Task Group on Advanced Reactor Experimental Facilities (TAREF) for high-temperature reactors and sodium fast reactors. It also identifies general safety issues for several proposed molten salt and small modular reactor designs. The present report – *Nuclear Safety Research Support Facilities for Existing and Advanced Reactors: 2021 Update* – can be considered as a follow-on to the previous two reports. In addition, although the need to maintain experimental databases was recognised as an important issue, it was not treated specifically in previous reports; the present report makes some direct recommendations regarding experimental data preservation.

Several facilities have been shut down since the publication of the SESAR/SFEAR report in 2007. Accordingly, loss of critical research infrastructure (i.e. facilities, capabilities and expertise) remains a concern and was a major factor in conducting the current study. However, several key facilities were preserved between 2007 and 2019 thanks to the SESAR/SFEAR project. These are discussed in Chapters 1 and 4.

The present report discusses 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, water-water energetic reactors, and advanced or evolutionary designs of these types. The main purpose of this report is to:

- summarise current safety issues, whose resolution will depend on additional research work;
- provide the current status of research facilities that support resolution of the safety issues;
- recommend actions the CSNI could take in the short term to help maintain facilities which require a substantial investment of resources and are in danger of premature closure;
- provide recommendations on long-term nuclear safety research facility infrastructure needs and preservation.

The report also provides information on safety issues and research needs not unique to the nuclear industry and those associated with advanced reactor designs such as gas-cooled reactors, sodium fast reactors, molten salt reactors and small modular reactors. This information is presented for completeness and for use by designers, operators and researchers in planning and conducting future work.

This report addresses technical issues unique to the nuclear industry as well as those that are relevant to the nuclear industry as well as other industries.

- Issues unique to the nuclear industry:
  - thermal-hydraulics;
  - nuclear fuel;

  - reactor physics;
  - severe accident and containment phenomena;
  - integrity of equipment and structures.
- Issues not unique to the nuclear industry:
  - human and organisational factors;
  - plant control and monitoring;
  - cybersecurity;
  - external events;
  - fire assessment.

The second group of issues includes factors that are relevant to the nuclear industry but are also supported in a major way by other industries and thus may not require support from the CSNI. In general, in developing recommendations for consideration, the group focused on those facilities that have unique capabilities and high relevance to the resolution of safety issues for Generation II designs as well as the potential to be highly relevant in support of the resolution of safety issues for new and emerging Generation III and IV designs. The uniqueness, replacement cost (>EUR 5 million) and operating cost of facilities was an important consideration in both the evaluation and recommendations. Due to the costs of operating many larger facilities, further co-operative efforts will most likely be needed to maintain them in the longer term.

## Short-term conclusions and recommendations

To assess short-term concerns, the facilities were examined (see Chapter 3), and those that were unique, versatile and in danger of being shut down in the period 2021-2024. The CSNI is encouraged to support, to the extent possible, joint projects proposed for these facilities.

In the thermal-hydraulics area, three facilities are in short-term danger. These include PKL and Large Scale Test Facility (LSTF, Japan), which support PWR thermal-hydraulic work, and PANDA, which supports both light water reactors (LWR) and BWR thermal-hydraulic safety issues, as well as severe accident and containment phenomena. It should be noted that both PANDA and PKL were identified as being at short-term risk as early as 2001, but have been used to conduct several important CSNI research initiatives via NEA joint projects in the intervening years. Both PANDA (PSI, Switzerland) and PKL (Areva, Germany) proponents are anticipated to propose new projects before the current projects come to an end (2021 for PANDA and 2020 for PKL). The short-term prognosis of the LSTF (Japan) is similar: indications are that Japan will likely propose a new joint project in the near future.

In the severe accident and containment phenomena technical area, the aforementioned PANDA facility, the VERDON facility (France) and the THAI facility (Germany), identified as being at risk in the near term, had all been identified as being at risk, or to require long-term monitoring, in the 2007 report. There are future NEA joint projects planned for each of them in order to justify their operation for the coming years. The CHROMIA platform (fission product release and transport facilities at the Institut de Radioprotection et de Sûreté Nucléaire [IRSN] in France) is also at risk, and, like the THAI facility, has been extensively utilised by NEA projects (STEM 1, 2 and 3) in the 2010s. A new NEA joint project, ESTER, is now underway and will use both the CHROMIA and the VERDON facilities. Another new NEA joint project, THEMIS, is also underway and will use the THAI facility.

In the fuel and materials areas, several large, versatile reactors have been closed since 2007. The National Research Universal Reactor in Canada and the Halden Reactor at IFE Norway both closed in 2018, representing a significant loss worldwide, for both materials and fuel testing. Additionally, the Japan Material Testing Reactor (JMTR) in Japan, a test reactor used primarily for materials testing, has been shut down. Despite these facilities being previously identified as being at short-term risk (PHEBUS) or facilities to monitor in the long term, substantial national support is required to sustain or refurbish such ageing or specialised reactors.

Reactors worldwide, including CABRI, BR-2, LVR-15, MIR, TREAT, HFR, ATR and others have been identified as being suitable to replace some of the capabilities lost by the closure of these reactors. The NEA has recently taken some measures to protect existing infrastructure by proposing the establishment of the FIDES network, which, along with joint projects for a variety of irradiation activities, will allow members to access various reactors and test programmes for materials and fuels. This step towards collaborative irradiation projects will provide both short- and medium-term activities that can be supported by the CSNI to mitigate the current irradiation gap.

A review of sub-critical and zero-power reactors by the NEA Nuclear Science Committee found that several of these reactors have been closed since the last SESAR report, and recommended that they should closely monitor the remainder (VENUS in Belgium, ZED-2 in Canada, LR-0 in the Czech Republic and KUCA in Japan). The STACEY facility (Japan) was mentioned as a candidate to replace some capacity, but it is currently shut down. The CROCUS facility in Switzerland may also have some capacity. None of the aforementioned facilities have been identified as being in imminent danger of shutdown.

Finally, although the scope of the 2007 activity focused solely on facilities unique to the nuclear industry, the GALAXIE fire platform at the IRSN is identified in this evaluation as being at risk in the near term. Although other fire research facilities exist, the co-location of separate effects, intermediate and large-scale facilities in this platform, as well as the experience gained from three phases of NEA joint projects such as PRISME, make GALAXIE ideal for addressing fire safety issues unique to the nuclear industry (e.g. glovebox fires, propagation through ventilation systems, fire initiated in cable trays). An NEA joint project is planned to be proposed once the current PRISME project ends (in 2021).

## Longer-term conclusions and recommendations

Many of the factors used in the last two reports to arrive at conclusions and recommendations have resulted in effective measures for retaining key facilities at risk. These measures should continue to be used in the future, with consideration of the following factors:

- Cost of facility operation and replacement (i.e. limit CSNI involvement to large facilities needing multinational support).
- Consistency with SFEAR recommended list of facilities for long-term preservation (see 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).
- Long-term planning to ensure that 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 and ability to support NEA projects.

Building on previous SESAR evaluations 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. This list is discussed in Chapter 4. The facilities are those considered unique, hard to replace and identified as playing a significant role in resolving issues in their technical area. It is recommended that the CSNI continue to focus on these facilities in developing a strategy for long-term infrastructure preservation, continuing to take action, as appropriate, to ensure that critical facilities are available for each reactor type to meet the critical research infrastructure needs. The same should be done for facilities developed for new reactor types identified in Chapter 2. Similar to the short-term recommendations above, host country commitment and long-term facility plans will be important factors in determining which facilities to preserve.

## General conclusions and recommendations

The following conclusions and recommendations pertain to both the short and long term. They result from the group's observations and experience in carrying out the current activity and desire to develop a practical set of recommendations, with facility preservation being a co-ordinated effort among the NEA standing committees. Specific general conclusions and recommendations are listed below. It is worth noting that many of the conclusions reached from previous SESAR/SFEAR activities are still valid, and are included below.

- CSNI members are encouraged to continue their excellent support for facilities at risk, which has already resulted in several valuable projects (e.g. PKL, PANDA, GALAXIE).
- As recommended in the previous report, test reactor availability should be given special scrutiny, due to the high cost of operation and replacement. The new Framework for Irradiation Experiments (FIDES) framework and associated Joint Experimental Programmes (JEEPs) is an essential step in maintaining global capability.
- Regardless of FIDES and the success of JEEP, continued and ongoing attention must be focused on smaller unique facilities at risk.
- The Nuclear Science Committee (NSC) should maintain a close watch on facilities used to support criticality and reactor physics codes.

The shutdown of the Halden Reactor and subsequent project activities have highlighted the need for preserving key experiments in international databases. Consequently, the group developed a series of recommendations specifically targeting data preservation:

- NEA Joint Safety Research projects should clearly outline their plan for data preservation, and should stipulate that a copy of the primary data needs to be sent to the NEA for storage.
- CSNI working groups should be asked to identify key datasets in their areas. Some of this may have been done with code validation matrices and datasets to support the development and implementation of standards.
- There should be a cross-functional (CSNI, NSC, etc.) NEA task group established to consider what should be done to preserve the key experimental datasets. This could include possible options for data libraries, how to screen datasets, what information needs to accompany the primary data, etc.
- CSNI working groups should select an appropriate option for preserving each key dataset and develop an activity to put it in place (CSNI Activity Proposal Sheet [CAPS], joint project, etc.).

## References

NEA (2001), *Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk*, OECD Publishing, Paris, www.oecd-nea.org/SESAR.

NEA (2007), "Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors (SFEAR)", OECD Publishing, Paris, www.oecd-nea.org/SESAR-SFEAR.

# 1. Introduction

The Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) has used Senior Expert Groups on Safety Research (SESARs) to identify capabilities and facilities essential for the safety of nuclear installations, and to recommend how to preserve key capabilities and facilities. The CSNI has acted on SESAR recommendations and established co-operative research projects that focused research on areas where there were significant safety issues to be addressed, and that preserved key facilities which were in imminent danger of being shut down.

Tables 1.1 and 1.2 summarise the recommendations from the previous Senior Expert Groups, identify actions taken by the NEA as a result, and provide the status of facilities previously identified as being at risk or requiring monitoring.

This report describes the outcomes from the latest SESAR group established by the CSNI. The Senior Expert Group on Safety Research/Support Facilities for Existing and Advanced Reactors 2 (SESAR/SFEAR2) was asked to review and update the previous SESAR assessments. They were also asked to recommend actions to be taken by the CSNI and its member countries to facilitate broader use and sustained operation of research facilities essential to support nuclear safety.

The recommendations of the Senior Expert Group on Safety Research/Support Facilities for Existing and Advanced Reactors 2 (SESAR/SFEAR2) have prompted numerous 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 the CSNI as a result of two previous evaluations by the expert group, in 2001 and 2007, have kept a number of facilities active that would otherwise have been lost, and focused research on areas where there were significant safety issues that remained to be addressed.

Tables 1.1 and 1.2 summarise the recommendations from the previous Senior Expert Groups, identify actions taken by the NEA as a result, and provide the status of facilities previously identified as being at risk or requiring monitoring.

## 1.1. Purpose

The SESAR/SFEAR2 mandate defines the purpose as follows: "In conducting its review, SESAR/SFEAR2 should ensure that the facilities in all NEA member countries are considered, either through direct representation on the group, or through opportunities for non-group members to provide input. To the extent possible, consideration should also be given to infrastructure of other major nuclear countries like China and India. SESAR/SFEAR2 should interface with groups within the NEA that conduct similar reviews, with the International Atomic Energy Agency and with other international organisations as necessary."

For operating reactors, the purpose of the SESAR/SFEAR activity is to: summarise the currently identified safety issues whose resolution depends on 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 to help maintain the facilities; and provide recommendations on long-term nuclear safety research facility infrastructure needs and a strategy for its preservation. Note that many facilities, such as hot cells and test loops, that are owned by a commercial entity and primarily used to support nuclear vendors or operators are not typically included in this report, except where they provide unique infrastructure to resolve ongoing safety concerns. 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 summarise known safety issues, safety research and facility needs for use by designers, operators and researchers.

## 1.2. Scope

The mandate given to the SESAR/SFEAR2 defines the scope as follows: "The scope for the SESAR review should first focus on the support required for water cooled reactors (boiling water reactor [BWR], pressurised water reactor [PWR], pressurised heavy water reactor [PHWR]), particularly those currently deployed in NEA member countries. Nevertheless, recognising the increased importance of innovative reactor designs, requirements for non-water cooled reactors should be included with generic considerations in the various technical areas (e.g. materials, fuel, severe accidents, etc.)."

The scope of this activity is primarily limited to safety issues and facilities associated with nuclear reactor design, construction and operation. However, in contrast to the previous reports, the safety of spent fuel pools has also been added under the sub-chapter of nuclear fuel, given the events at Fukushima. In addition, the current report adds cybersecurity as an issue warranting a separate section because it has emerged as an area of concern for operating plants. Finally, the previous focus on seismic effects has been expanded to include other external events. The report focuses on the following technical areas:

- thermal-hydraulics;
- nuclear fuel;
- reactor physics;
- severe accidents and containment phenomena;
- integrity of equipment and structures;
- human and organisational factors;
- plant control and monitoring;
- cybersecurity;
- external events;
- fire assessment.

The first five address phenomena, safety issues and facilities unique to the nuclear industry. The last five address phenomena, safety issues and facilities that are relevant to the nuclear industry, but which are also relevant to other industries and where research and facilities may be supported by others. 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, PHWRs and water-water energetic reactors (VVERs) in member countries.
- Future designs including advanced light water reactors (LWRs), PHWRs and gas-cooled reactors, as well as sodium fast reactors, molten salt reactors and finally, small modular reactors that could belong to any of the aforementioned types.

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

## 1.3. Approach

A set of recommendations was developed for each of the technical areas above that are unique to the nuclear industry. No technical recommendations are provided for those technical areas previously identified as not unique to the nuclear industry, but the safety issue and facility information is provided for information only. The exception is for fire research, for which it was recognised that there is a need for facilities that can evaluate fire propagation and behaviour in confined spaces (ventilation, cable trays, gloveboxes) unique to nuclear facilities, and that there is a very versatile facility in France that should be maintained.

For the issues identified for operating and future plants, members re-evaluated the safety issues described in the previous SESAR/SFEAR report and identified research work that was still required. Further, the major facilities currently performing or available to perform research directly relevant to these issues have been identified. Those that were operable in the early 2010s but which have since been closed, or are due to be closed soon, are also specified, in order to provide an overview of recent developments.

## 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 of their organisations. However, in assembling the information contained in the report, the participants benefited from 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.

## 1.5. Organisation of report

Chapter 2 provides a short overview of the reactor designs and their safety issues addressed within the scope of this report. Chapter 3 addresses each of the technical areas listed above for operating reactors as well as advanced ones. It is split into two sections pertaining to those technical areas unique to the nuclear industry, and those that are not. It also describes in more detail the organisation and purpose of both of these sections. Chapter 4 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 previous report provided a detailed and systematic ranking of the importance of each safety issue as well as a ranking of facilities on the basis of their importance to resolving the issue, their uniqueness, their versatility to address more than one issue, and the cost of their operation or replacement. The same general approach was followed here; however, since the issues faced today are very similar to those faced in the early 2010s, it was unnecessary to rank the issues in terms of their safety significance. Furthermore, except where identified in the text, it is assumed that all of the issues identified in the tables have a medium to high safety relevance, and that additional research is required to resolve the issue.

The conclusions and recommendations are divided into near-term (CSNI action needed in the next one to three years) and long-term actions. However, any action by the 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. the CSNI does not intend to serve as a host country for facility preservation).

## Table 1.1. Impact of SESAR/FAP facility recommendations (2001) on SESAR/SFEAR findings (2007)

| SESAR/FAP recommendation | Resulting CSNI action | Impact on facilities until 2007 |
|---|---|---|
| 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, but identified as being in near-term danger in the 2007 SFEAR study.<br>PKL active and not in near-term danger. |
| 2) Monitor and maintain key thermal-hydraulic facilities in the long term. Thermal-hydraulic 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 remained active and was not deemed as being in near-term danger in 2007. Other thermal-hydraulic facilities continued 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 and MCCI remained active were and not in near-term danger in 2007. |
| 4) Develop centre of excellence on fuel coolant interaction in consideration of potential loss of the FARO and KROTOS facilities. | Initiated the SERENA programme (group of experts to discuss status of fuel coolant interaction and future experimental needs). FARO shutdown. KROTOS kept in standby. | KROTOS facility maintained through 2007. |
| 5) Develop centre of excellence on iodine chemistry and fission product behaviour. | Joint projects planned in Canada and France respectively. | New facilities developed in Canada and France to support NEA joint projects, Behaviour of iodine and STEM projects operating by 2007-08. |

## Table 1.2. Impact of SESAR/SFEAR findings (2007) on current facility status

| SESAR/SFEAR | Resulting CSNI action | Impact on facilities since 2007 |
|---|---|---|
| 1) Maintain the PANDA facility in the thermal-hydraulics and containment areas in the near term. | NEA joint project SETH-2 was planned. | Facility has remained operational, with a subsequent project, HYMERES-1, completed and HYMERES-2 underway. New joint project will be planned before HYMERES-2 is completed in 2021. |
| 2) Monitor the LSTF/ROSA, PACTEL, RD-14M, ATLAS, PKL and PSB-VVER thermal-hydraulics facilities in the long term. | Maintained facilities addressing issues of broad international interest through the establishment of international projects. Some of these projects have been completed (SETH, PSB-VVER, ROSA-1 and 2), while others are still ongoing (PKL-1 to 4, ATLAS-1 and 2). Some PKL tests were complemented by experiments performed in the PMK, ROCOM and (more recently) PACTEL facilities. | RD-14M has closed following a national decision. The other facilities remain open. PKL and LSTF/ROSA are currently identified to be at short-term risk, with proposals for NEA joint projects expected to be proposed in the near future. |
| 3) In the fuel area, PHEBUS was identified as being at short-term risk while Halden, NSRR, CABRI, NRU and MIR were identified as facilities to monitor in the long term. | Supported (and continued to support) the CABRI International Project. The Halden project will be supported until 2020. | National decisions resulted in PHEBUS, Halden and NRU being shut down. CABRI has been the subject of an NEA joint project since 2000. Both CABRI and MIR are planning joint projects under the new NEA FIDES network. The NSRR is available and not identified as being at risk. |
| 4) Numerous containment and severe accident facilities were identified as being at short-term risk, requiring long-term monitoring. These included CTF and LSCF in Canada, THAI, QUENCH and COMET facilities in Germany, CHROMIA platform and VERDON facilities in France, and the MCCI facility in the United States. | Initiated joint projects involving QUENCH, VERDON, CHROMIA and THAI facilities. | CTF, LSCF and COMET facilities closed based on national decisions. VERDON, CHROMIA, QUENCH and THAI facilities continue to be supported by national programmes and NEA joint projects. All are currently identified as being at short-term risk or requiring monitoring in the long term and are the subject of new NEA joint projects (THEMIS for the THAI facility, QUENCH-ATF for the QUENCH, ESTER for the CHROMIA facility). For the MCCI facility, a new joint project, ROSAU, makes use of some of the expertise and components employed by the MCCI, although a new facility will be used. |

## References

NEA (2001), *Nuclear Safety Research in OECD Countries: Major Facilities and Programmes at Risk*, OECD Publishing, Paris, www.oecd-nea.org/SESAR.

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# 2. Overview of reactor designs and safety issues

This section contains short overviews of each of the major reactor types within 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);
- water-water energetic reactors (VVERs);
- pressurised heavy water reactors (PHWRs) and advanced PHWRs (APHWRs);
- advanced light water reactors (ALWRs);
- gas-cooled reactors;
- sodium fast reactors (SFRs);
- molten salt reactors (MSRs);
- small modular reactors (SMRs).

## 2.1. Boiling water reactors

### *2.1.1. Introduction*

The direct-cycle 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 N-16, a very short-lived isotope (7 second 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 on the age of the design and product type. However, they all have some common characteristics:

- 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 refuelling 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.

The next section describes the main design features and safety issues associated with BWRs.

### *2.1.2. Boiling water reactor design features*

The nuclear core consists of fuel assemblies and control blades contained within the reactor pressure vessel and cooled by a recirculating water system. The fuel elements of an assembly are located within a box made of Zircaloy and the cross-shaped control blades are located within the gap between fuel assembly boxes. 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 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 (approximately 70 bars or 1 000 pounds per square inch) 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, which 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, the main steam line safety/relief valves and discharge piping, control rod drives and piping, piping and valves associated with the 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 loss-of-coolant accident (LOCA), and accommodates reactor blowdown through the safety/relief valve discharge piping to the suppression pool.

The containment is subdivided into drywell and wetwell or suppression pool. The suppression pool is a toroidal pool of demineralised water within the containment boundary and connected to the drywell by condensation pipes. The suppression pool provides: a means to condense any steam released in the drywell area during a LOCA; 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 heat exchangers; a heat sink for venting the nuclear system safety/relief valves; a source of water for emergency core cooling; and a source of water to the containment spray system.

Surrounding the containment is the shield building. The shield building is often 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. It contains the spent fuel pool.

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.

### *2.1.3. Boiling water reactor safety issues*

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

- anticipated transient without SCRAM;
- shroud, feedwater nozzle and recirculation pipe cracking;
- station blackout;
- core spray distribution;
- suppression pool dynamics;
- stability;
- severe accident concerns:
  - in-vessel melt progression and melt retention;
  - in- and ex-vessel fuel coolant interaction;
  - molten core concrete interaction (MCCI)/containment shell melt through (e.g. Mark I containment);
  - ex-vessel core debris coolability;
  - containment cooling/integrity/venting;
  - combustible gas control;
  - source term.

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

- plant ageing, including materials cracking/corrosion;
- power uprates;
- high-burnup/$MO_X$ fuel;

- installation of digital instrumentation and control (I&C);
- emergency core cooling system (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; others are shared with PWRs. Accident management programmes have been developed to prevent or mitigate many of the severe accident concerns mentioned.

Although many of 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, these differences need to be considered when assessing facility and programme needs.

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 research needs, such as:

- redefinition of large-break LOCA;
- break location and orientation.

## 2.2. Pressurised water reactors

### 2.2.1. Introduction

The pressurised water reactor (PWR) consists of a high-pressure reactor vessel (about 140 bar or 2 200 pounds per square inch) 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 pounds per square inch.

PWR core and plant designs can vary depending on the product type (2-, 3- or 4-loop plant). However, they all have some common characteristics:

- 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.

The next section describes the main design features and safety issues associated with PWRs.

### 2.2.2. Pressurised water reactor 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 of the assemblies. All fuel assemblies are without assembly boxes. 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 refuellings, 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 LOCA 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 and tend to slow the nuclear chain reaction.

Rod cluster control 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 rod cluster control 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;
- backup 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.

### *2.2.3. Pressurised water reactor safety issues*

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

- anticipated transient without SCRAM;
- loss-of-coolant accidents;
- loss-of-feedwater transients;
- steam line breaks;
- station blackout;

- severe accident concerns:
  - core debris criticality under reflood conditions;
  - in-vessel melt retention by late reflooding;
  - high-pressure vessel failure and direct containment heating due to RPV bottom head failure;
  - core concrete interaction (MCCI);
  - ex-vessel core debris coolability or external cooling of the RPV;
  - in- and ex-vessel fuel coolant interaction;
  - combustible gas control;
  - containment cooling/integrity/venting;
  - source term issue.

As is the case for BWRs, PWR safety issues are routinely re-evaluated in response to industry initiatives such as raising power or fuel burnup levels, increasing operating cycle lengths or extending plant lifetime. These issues have included:

- reactivity initiated events (e.g. boron dilution);
- plant ageing, including materials cracking/corrosion;
- power uprates;
- high-burnup fuel/$MO_X$ fuel;
- installation of digital I&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, like BWRs, emphasis on risk-informed regulation in some countries and more detailed assessments of accident scenarios and accident management actions has led to new issues needing research, such as:

- redefinition of large-break LOCA;
- break location and orientation.

Many of these issues have been largely resolved today due to the huge research efforts in the past decade and due to accident management programmes, which have been developed to prevent or mitigate many of the severe accident concerns mentioned. Finally, Fukushima evaluations have required consideration of design-basis extension and beyond design-basis accidents in PWR safety issues, some of which are discussed in Chapter 3.

## 2.3. Water-water energetic reactors

### *2.3.1. Introduction*

Nuclear power plants with VVER reactors of Soviet origin are operated in the Russian Federation and several other European countries. Some of these are NEA member countries such as the Czech Republic, Finland, Hungary and the Slovak Republic where nuclear power plants with VVER-440/213 and VVER-1000/320 reactors are in operation; older reactors of generation VVER-440/230 are already shutdown. VVER reactors are classified as a specific type of PWR reactor. However, there are some significant differences between the VVER and other types of PWR, both in terms of design and materials used. Distinguishing features of the VVER include: use of horizontal steam generators; use of hexagonal fuel assemblies; avoidance of bottom penetrations in the reactor vessel; use of high-capacity pressurisers. 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 other countries.

A next-generation of VVER-1000 reactors with larger reactor power and different passive safety features has already been designed and some plants are under construction in Russia, Belarus, the People's Republic of China (hereafter "China") and India. Drawing on the significant body of experience gained with the well-established VVER-1000/V-320, the AES-91 (or VVER-1000/V-428) was developed by Saint-Petersburg Atomenergoproekt and the AES-92 (or VVER-1000/V-412 and 466) by Moscow Atomenergoproekt. Along with upgraded technology and improved economics, these designs deployed the concept of beyond design-basis accident management based on a balanced combination of passive and active safety systems. Only small modifications were made in the basic power production systems – reactor, primary cooling circuit and turbine cycle. The main changes were in the safety systems and plant layout.

The VVER-1200 AES-2006 design is the latest evolution in the long line of VVER plants. It meets all the international safety requirements for Gen III+ nuclear power plants. The first AES-2006 units are under construction in Russia (Rosatom Overseas, 2018).

The latest project is the VVER-TOI (typical, optimised, with enhanced information) project to create a standardised VVER power plant, optimised in terms of technology and economics. It is being developed by Moscow Atomenergoproekt and is based on the AES-2006/V-392M design. It represents a further evolution of the VVER-1200 design and is designated V-510. This design has an upgraded pressure vessel, increased power to 3 300 MWt and 1 255-1 300 MWe gross (nominally 1 300 MWe), improved core design to increase cooling reliability, further the development of passive safety with a 72-hour grace period requiring no operator intervention after shutdown, lower construction and operating costs, and a 40-month construction time. It will use a low-speed turbine-generator (WWN, 2019).

### *2.3.2. Water-water energetic reactor design features*

The VVER is a pressurised water reactor, employing light water as a coolant and moderator. 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 a 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 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.

### *2.3.3. Water-water energetic reactor 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 nuclear power plants with VVER reactors was reviewed in the framework of the International Atomic Energy Agency's (IAEA) Extrabudgetary Programme (EBP) on the safety of VVER and high-power channel-type reactor (RBMK) nuclear power plants during the period 1990-98. 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.

Regarding the safety ranking of the more modern reactor type VVER-440/213, no safety issues of the highest category were identified. High safety concerns include 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 retrofitting 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, defence in depth has been taken into account from the beginning. The IAEA safety review for the standard series VVER-1000/320 did not identify any safety issues of the highest category. 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, especially of the first-generation VVER plants, have been identified as deviations from current standards and practices, which have evolved since these nuclear power plants were designed. In all countries, extensive back-fitting and upgrading programmes have been performed, which have resolved a great majority of the remaining safety issues; older plants of VVER-440/230 generation have been shut down.

The particular problem of these older VVERs was that their safety analysis was not validated via experimental facilities. It was for this reason that the experimental validation of computer codes and system behaviour was of great importance. As an example, the successful experimental validation of the bubble condenser system of VVER-440/213 was done, and the experimental validation of thermal-hydraulic computer codes for VVER-1000/320 at the PSB-VVER facility, as well.

In the future, it may be expected that the lifetime of the first generation of VVERs will be extended. This will lead to having to address many of the same issues as for 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 will not be specific to VVERs. The next generations of VVER reactors are now under construction.

## 2.4. Pressurised heavy water reactors and advanced PHWRs

### *2.4.1. Introduction*

A PHWR is a nuclear reactor commonly using natural uranium or slightly enriched uranium as fuel, that uses heavy water (deuterium oxide, $D_2O$) as its coolant and neutron moderator. The heavy water coolant is kept under pressure, allowing it to be heated to higher temperatures without boiling, much as in a PWR. While heavy water is significantly more expensive than ordinary light water, it creates greatly enhanced neutron economy, allowing the reactor to operate without fuel-enrichment facilities (offsetting the additional expense of the heavy water) and enhancing the ability of the reactor to make use of alternate fuel cycles.

Two generations of PHWRs of German design are under operation in Argentina. Atucha 1 entered commercial operation in 1974, and Atucha II in 2014. The Siemens design of the Atucha PHWR units is unique to Argentina. Both are PHWR with a different power employing a mixture of natural uranium and enriched uranium (0.85% of $^{235}U$), and use heavy water for cooling and neutron moderation. The nuclear area is based on the prototype reactor MZFR 56 MW(e), which was placed in operation at the Nuclear Research Center in Karlsruhe (Germany) in 1966. The secondary and auxiliary systems as well as the containment and other buildings were designed in a similar way as Konvoi PWR plants from Germany (around 1979). A main difference to the CANDU design is the vertical arrangement of the individual fuel channels in the reactor core like in a PWR. Day-by-day fuel loading is possible as in other PHWRs.

The CANDU (Canada Deuterium Uranium) is a Canadian PHWR design that has also been deployed worldwide. The acronym refers to its deuterium oxide (heavy water) moderator and its use of (originally, natural) uranium fuel. CANDU reactors were first developed in the late 1950s and 1960s by a partnership between Atomic Energy of Canada Limited (AECL), the Hydro-Electric Power Commission of Ontario, Canadian General Electric and other companies.

PHWRs of CANDU design are unique for containing the nuclear fuel and coolant in an array of horizontal fuel channels, rather than a pressure vessel like in the other type of PWR. 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°C, and serves as a backup 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.

### *2.4.2. Pressurised heavy water reactor CANDU 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 refuelled 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 ECCS to provide backup cooling in a LOCA. There are typically three modes of operation: 1) high-pressure injection; 2) intermediate/low-pressure injection; 3) 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: 1) single-unit containment; 2) multiple unit containment (incorporating a common vacuum building); and 3) 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 and recombiners prevent build-up of hydrogen to explosive levels.

In a multi-unit vacuum containment, four or eight 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. Multi-unit containments also feature hydrogen recombiners.

Current Indian PHWRs use a double concrete containment. The inner containment is a cylinder and dome of pre-stressed 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.

### *2.4.3. Pressurised heavy water reactor CANDU safety issues*

The main residual safety issues for operating PHWR CANDU reactors 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 the 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 sub-cooling requirements;
- flow distribution in headers during a LOCA.

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, including 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;
  - in-vessel retention.

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. However, the use of pressure tubes as a pressure boundary, the horizontal channels, and the existence of a water-filled calandria vessel and end-shield cooling tank make accident progression in-core very different than that of an LWR. Specialised facilities have been built to address these challenges.

## 2.5. Advanced light water reactors

### *2.5.1. Introduction*

ALWR designs constitute improvements of current generation pressurised water reactors (PWRs) and boiling water reactors (BWRs). For the purpose of this study, the ALWR designs considered are those being developed for deployment in the next five to ten years, which will likely exhibit some of the following design features:

- 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. pre-stressed 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 $MO_X$ fuel, and operate at conditions similar to present-day LWRs.

### *2.5.2. Advanced light water reactor 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.

## 2.6. Advanced pressurised heavy water reactors

### *2.6.1. Introduction*

Advanced PHWR designs based on the CANDU 6 design are also being developed to improve the safety and economics of the PHWR. Key features of the Enhanced CANDU (EC6) include:

- target life up to 60 years, >90% capacity factor;
- modern steam turbines with higher efficiency and output;
- increased safety and operating margins;
- additional accident resistance and core damage prevention features;
- addition of a reserve water system for passive accident mitigation;
- improved plant security and physical protection.

An additional advancement in the PHWR design is the Advanced Fuel CANDU Reactor (AFCR), which allows countries to use recycled uranium from the spent fuel of PWR plants to improve uranium resource utilisation. In addition, the AFCR can use thorium-based fuels, which will allow countries with indigenous thorium resources to use them as a near-term energy strategy to minimise dependence on uranium imports. The use of depleted uranium and enriched uranium to give a natural uranium equivalent (NUE) based fuel has been successfully demonstrated in China.

India is also developing an advanced heavy water reactor (AHWR) concept that aims to meet the objectives of using thorium fuel cycles for commercial power generation. The AHWR is a vertical pressure tube type reactor cooled by boiling light water under natural circulation. A unique feature of this design is a large tank of water on top of the primary containment vessel, called the gravity-driven water pool. This reservoir is designed to perform several passive safety functions.

The overall design of the AHWR is to utilise large amounts of thorium and the thorium cycle. The reactor design incorporates advanced technologies and several passive safety features, such as core heat removal through natural circulation; direct injection of ECCS water in fuel; and the availability of a large inventory of borated water in overhead gravity-driven water pool to facilitate sustenance of core decay heat removal. The ECCS injection and containment cooling can act (SCRAM) without invoking any active systems or operator action.

The reactor physics design is tuned to maximise the use of thorium-based fuel through the use of $PuO_2$-$ThO_2$ $MO_X$ and $ThO_2$-$^{233}UO_2$ $MO_X$ in different pins of the same fuel cluster, and the use of a heterogeneous moderator consisting of amorphous carbon (in the fuel bundles) and heavy water in 80-20% volume ratio. The core configuration lends itself to considerable flexibility and several feasible solutions, including those not requiring the use of amorphous carbon-based reflectors, are possible without any changes in reactor structure.

### *2.6.2. Advanced pressurised heavy water reactor safety issues*

As is the case for ALWRs, the safety issues for AHWRs are expected to be similar to those of their predecessor Generation II designs. However, the advanced reactor designs have incorporated passive safety systems and additional design changes to reduce core damage probability. These changes, and their effectiveness at addressing current safety issues, will require assessment.

## 2.7. Gas-cooled reactors

### *2.7.1. 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 high-temperature gas reactor (HTGR) concept, dates back in the United States to the late 1950s, when the design of the fully ceramic core and the use of the helium gas for cooling were pioneered by 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, Germany began designing the pebble bed type of HTGR. Two HTGRs were constructed in Germany, the experimental 15 MWe AVR and the 300 MWe THTR300. All of the HTGRs mentioned above have been decommissioned.

A Task Group on Advanced Reactor Experimental Facilities (TAREF) evaluated the facilities required for gas-cooled reactors in 2009 (NEA, 2009) and provided a high-level description of two different gas-cooled reactor concepts. This information is summarised below.

### *2.7.2. Design features of gas-cooled reactors*

*Design features of the high-temperature reactor*

The typical high-temperature reactor (HTR) design features include the following:

- High-performance coated fuel particles (CFPs) able to contain fission products for the full range of operating and postulated accident conditions, with a very low fuel failure fraction and subsequent release of fission products. The CFPs are embedded in either a rod compact inserted into a stacked prismatic block or a spherical compact, referred to as a pebble.
- An inert single-phase high-pressure coolant (helium).
- A graphite-moderated core with the characteristics of low power density, large heat capacity, high effective core thermal conductivity and large thermal margins to fuel failure.
- Negative fuel and moderator temperature coefficients of reactivity, which, along with the negative reactivity feedback of the fission product xenon-135, are sufficient to shut down the reactor during loss-of-forced circulation (LOFC) events. This aspect stabilises power-control feedback for most reactivity insertion events (for both startup and power operation) for the entire fuel life cycle and for all applicable temperature ranges.
- A design-basis accident decay heat removal system, typically a passive system utilising natural convection-driven processes (the reactor cavity cooling system).
- A confinement-style reactor building structure to accommodate depressurisations may be used instead of a leak-tight sealed containment.

- The balance of plant consists of an electrical power generation unit (most often a gas turbine) and, in some cases, a high-temperature process heat component potentially used for the production of hydrogen.

*Design features of the gas fast reactor*

The gas fast reactor (GFR) system features a high-temperature, helium-cooled, fast-spectrum reactor with a direct-cycle helium turbine or an indirect cycle using an intermediate heat exchanger for electricity production. It uses a closed fuel cycle. The GFR combines the advantages of fast-spectrum systems (long-term resource sustainability in terms of use of uranium and waste minimisation, through fuel reprocessing, recycling and burning of long-lived actinides) with those of the high-temperature reactors (high thermal cycle efficiency and possibly hydrogen production), and of the direct-cycle energy conversion option. Its development approach is to rely on technologies already used for the HTR but with significant advances, in order to reach the objectives stated above.

The main GFR design specifications as derived from the general objectives of Generation IV systems are:

- use of gas as a coolant as a means of reaching high temperatures;
- economic competitiveness by means of simplicity, compactness and efficiency;
- a robust safety demonstration, based on probabilistic safety assessment and defence in depth principles, and including severe accident management;
- fast neutron spectrum core with a zero (self-breeding) or positive breeding gain, with no or very limited use of fertile blankets in order to:
  - generate as much fissile material as it consumes, with an optimal use of uranium;
  - have a fuel cycle fed with only depleted or natural uranium;
  - achieve homogeneous recycling of all actinides in order to have no separation of plutonium from other actinides (proliferation resistance).
- core plutonium inventory not exceeding 10 tonnes/GWe, in order to have a realistic reactor fleet deployment (in a few decades) and high fuel burnup.

In contrast to the HTR, GFR cores have a low thermal inertia, with a special design feature to overcome this apparent unfavourable feature. These include:

- A fuel element based on refractory materials and high thermal conductivity, with the ability to ensure radioactive material confinement up to very high temperatures.
- A primary circuit design based on upward core cooling and a moderate pressure drop for all the primary components and circuit involved in accident scenarios. A gas-tight envelope encloses the primary circuit to limit the loss of pressure in the case of a primary loss of coolant. Maintaining high helium density allows the decay heat removal system to rely on moderate pumping power and even on passive natural convection in some situations.

The fuel element is able to withstand high operating temperatures and transients associated with the poor heat capacity of the gas coolant. The main temperature limits are the following:

- an operating temperature around 1 000°C that provides a sufficiently ample margin to failure;
- a boundary temperature of 1 600°C, below which fission product release is prevented;
- an upper temperature of 2 000°C, below which the core geometry can safely be cooled down.

The Generation IV objective of ultimate waste minimisation, proliferation resistance and natural resources optimisation (zero or positive breeding gain) are to be achieved by having no fertile blanket and multi-pass recycling of all actinides without separation.

Two primary fuel concepts are proposed to fulfil the above requirements: a ceramic plate-type fuel element and a ceramic pin-type fuel element. The reference material for the structure is reinforced ceramic, a silicon carbide (SiC) composite matrix ceramic. The fuel compound is made of pellets of mixed uranium-plutonium-minor actinide carbide. A leak-tight barrier made of a refractory metal or of Si-based multi-layer ceramics is added to prevent fission product diffusion through the clad.

The reactor pressure vessel is a large metallic structure with a thickness of 20 cm in the belt line region. The material selected, a martensitic 9Cr-1 Mo steel (industrial grade T91, containing 9% by mass chromium and 1% by mass molybdenum) undergoes negligible creep at operating temperature (400°C). The reference material for the internals is either 9Cr-1Mo or stainless steel, typically SS316LN. The global primary arrangement is based on three main loops (3×800 MWth), each fitted with one intermediate heat exchanger blower unit, enclosed in a single vessel.

The shutdown system has two redundant and passive shutdown systems (no power supply, gravity drop of absorber elements). Each main control rod and shutdown device and diversified shutdown device is individually driven, considering two independent groups each connected to a dedicated group of the instrumentation and control support system. Specific loops for emergency decay heat removal are directly connected to the pressure vessel and are equipped with heat exchangers and blowers.

The gas-tight envelope is designed to provide a sufficient pressure in the case of a large gas leak from the primary system. It consists of a metallic structure, initially filled with nitrogen slightly over atmospheric pressure to reduce the possibility of air ingress.

### *2.7.3. Safety Issues of gas-cooled reactors*

*High-temperature reactor*

The basic safety design approach for high-temperature reactors (HTRs) is different from most currently operating and advanced LWRs, in that HTRs rely on the retention of fission products in high-integrity ceramic CFPs in a relatively chemically inert environment to withstand accidents without fuel damage or fission product release. HTRs are designed with passive heat removal systems and inherent negative reactivity to limit fuel temperatures and maintain fuel particle integrity. However, with these novel design features and characteristics, HTR designers must provide proof of the safety performance of the equipment, including the CFPs; integrity of the reactor vessel; and supporting safety-related structures, systems and components. Fission product release and transport behaviour must be well understood and analysis tools must be validated against an adequate database if a vented confinement is to become an acceptable feature of an HTR.

The major HTR safety issues of concern that were identified and categorised as high importance combined with medium to low knowledge can be summarised as follows:

- core coolant bypass flows (normal operation);
- power/flux profiles (normal operation);
- outlet plenum flows (normal operation);
- reactivity feedback coefficients (normal operation and accidents);
- emissivity aspects for the vessel and the reactor cavity cooling system during depressurised loss-of-coolant (D-LOFC);
- reactor vessel cavity air circulation and heat transfer (D-LOFC);
- convection/radiation heating of upper vessel area during pressurised loss-of-coolant (P-LOFC) accidents.

*Gas fast reactor*

A specific characteristic of GFR systems is the lack of thermal inertia of the primary system, combined with the pressurised gas and the associated risk of leakage. From a thermal-hydraulic point of view, P-LOFC and D-LOFC are two main categories of transients that strongly influence the design of the system. Safety of the reactor mainly relies on active systems, and progression of a transient depends on the characteristics of several critical components. Hence, the behaviour of these safety critical components is a relevant GFR safety issue. In a fast neutron environment, the GFR core physics is sensitive to several reactivity effects that require specific attention as related to, for example, water ingress or reactivity effects resulting from accidental conditions. In this context, there will be a need for accurate nuclear data for GFR materials at representative neutron spectrum conditions.

A key conclusion from the gas-cooled reactor evaluation was that existing and planned facilities in member countries cover all technical areas of concern and most of the safety issues identified in these areas. Hence, there is no apparent need for a facility to be built (beyond what is currently planned in member countries).

Based on the responses received, the highest ranked facilities were identified and are shown in Table 2.1. Detailed descriptions of each of the facilities can be found in the TAREF report (NEA, 2009).

**Table 2.1. TAREF gas-cooled reactor summary ratings**

| | Accident and thermal fluids | Fission product transport | High-temperature materials | Graphite and ceramics | Fuel |
|---|---|---|---|---|---|
| Czech Republic | | HTHL | HTHL | HTHL | |
| France* | HEDYT<br>ENIGMA | MERARG | | HEDYT | PLINIUS |
| Germany | HELOKA<br>A2 | THAI | High Power Laser Lab | | |
| Italy | HE-FUS3 | | | | |
| Japan | HTTR | HTTR | HTTR | | NSRR |
| United States | | ATR | ORNL Materials Lab<br>INL High Temp Test Lab | MIT<br>HFIR | ACRR<br>ATR<br>MIT |

* For the longer term (2020 and beyond), the French gas fast reactor demonstration reactor ALLEGRO should also be considered.

## 2.8. Sodium fast reactors

The TAREF also conducted a study of sodium fast reactors (SFRs; NEA, 2011). The SFR system features a fast-spectrum, sodium-cooled reactor and a closed fuel cycle for efficient management of actinides and conversion of fertile uranium. This section provides a summary of the 2011 TAREF report.

### *2.8.1. Design features of sodium fast reactors*

The fuel cycle employs a full actinide recycle with two major options. One is an intermediate size (150-600 MWe) sodium-cooled reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor. The second is a medium to large (500-1 500 MWe) sodium-cooled reactor with mixed uranium-plutonium oxide fuel, supported by a fuel cycle based on advanced aqueous processing at a central location serving a number of reactors. The outlet temperature is approximately 550°C for both.

The SFR is designed for management of high-level wastes and, in particular, management of plutonium and other actinides. Important safety features of the system include a long thermal response time, a large margin to coolant boiling, a primary system that operates near-atmospheric pressure, and an intermediate sodium system between the radioactive sodium in the primary system and the water and steam in the power plant. With innovations to reduce capital cost, the SFR can serve markets for electricity.

The SFR's fast spectrum also makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal spectrum reactors with once-through fuel cycles.

The SFR system uses liquid sodium as a coolant, allowing high-power density with low coolant volume. The primary system operates at near-atmospheric pressure with outlet temperatures between about 500°C and 550°C. Austenitic and ferritic steel structural materials can be used, with a large margin (about 400°C) to coolant boiling. Despite these desirable features, one key characteristic of the various SFR designs that warrants special mention. This is the positive sodium void reactivity, except for very small cores, and the possibility of core re-criticality in the event of a core damage event.

Two main types of designs exist, the loop design, wherein the primary coolant and the heat exchangers are outside of the RPV, and the pool-type reactor, in which the coolant and heat exchangers reside inside the RPV. The loop design has advantages in ease of maintenance and the ability to have a significant height difference between the coolant and the heat exchangers, to promote natural circulation. Its disadvantages are that a loss-of-coolant could result in sodium fires. The pool-type reactor has advantages because it has a secondary vessel to contain any loss of coolant, nearly eliminating core uncover risks, and a large thermal inertia which mitigates accident consequences by absorbing excess core heat. Disadvantages are primarily related to ease of access to components residing within the pool.

Key results from the TAREF assessment are summarised below.

### *2.8.2. Safety issues of sodium fast reactors*

The TAREF group members agreed that for new SFR projects, the most important and top-tier R&D safety needs concern the technical areas with the following priority order:

- fuel safety and severe accident issues are of prime interest due to the lack of knowledge on new pin design and materials;
- thermo-fluids and reactor physics issues are of second priority, as one can live with the current knowledge when considering some margins to cover uncertainties;
- sodium risks and structural integrity issues may be considered a third priority, as they are more design-dependent.

Key to resolving the safety issues, the group stressed the need for fuel pin irradiation capabilities under representative conditions of fast neutron flux. They identified that a large body of work had been undertaken during the period 1970-95, followed by a slowdown, and that a large number of facilities operating in the past with sodium as coolant are no longer available.

A number of facilities appropriate to address SFR safety issues were evaluated. They are summarised below, almost verbatim from the TAREF report and identified in Table 2.2, along with facilities that may be available to address them in the short, medium and long term.

- The Indian FBTR fast reactor can be a valuable resource for irradiation of SFR fuel pins and new materials data; the American reactor ACRR (Sandia) would address issues related to fuel safety and severe accidents under specific conditions (provided there is confirmation of its availability for testing in the short term).
- The German KASOLA (Karlsruhe Institute of Technology) facility would provide data for the thermo-fluids issues in relation with computational fluid dynamics modelling approaches.
- The Japanese SWAT-1R-3R facility (Japan Atomic Energy Agency – JAEA) can be appropriate for studying sodium-water interaction in steam generator units; the Indian SFTF facility can be valuable for addressing several issues related to sodium fires; the SURTSEY facility (US Department of Energy – DOE) can be relevant to studies on sodium fires and sodium-water interaction in steam generators.

- The JOYO fast neutron reactor (JAEA, Japan) was identified as suitable to address fuel safety issues related to new fuel pin design (fuel pin performance and new materials database under irradiation, margin to fuel melting, impact of use of minor actinides) and some other issues; however, since it is under safety review by the nuclear regulation authority to ensure that new requirements prompted by the Fukushima accident are respected, uncertainty exists as to when operation will be permitted.
- Severe accident issues can only be addressed in a comprehensive way for the medium term and beyond due to the lack of available facilities for simulation of representative transient conditions in the short term with irradiated fuel pins. IGR (Kazakh facility used for JAEA programmes), which is addressing fresh fuel (controlled fuel relocation, debris bed formation), may be a suitable solution in the medium term as plans are under consideration for it to handle irradiated fuel. The VULCANO (Alternative Energies and Atomic Energy Commission, CEA, France) can also help for severe accident issues, provided it is refurbished for sodium use. The TREAT experimental reactor (US DOE) was also considered in the medium term for its relevance to severe accident issues (past experimental programmes simulating fast power transients).
- The MASURCA (CEA, France) was identified as being suitable for core physics issues for providing improved nuclear data of core materials (in relation with high burn-up level, use of minor actinides) and associated uncertainties; however, a decision was taken to shut down this facility subsequent to issuing the TAREF report.
- The CABRI experimental reactor (operated by the CEA) was recognised by members as the most appropriate facility to address irradiated fuel behaviour under incidental and accidental conditions (fuel safety issues such as margins to fuel melting and deterministic pin failure, severe accident issues such as consequences of various accidents leading to fuel melting, with associated consequences and risk of critical events and energy release). However, the sodium loop in the facility has now been decommissioned.
- In the case of innovative design for secondary circuits, the LIFUS5 Italian facility (National Agency for New Technologies, Energy and Sustainable Economic Development, ENEA) would address sodium interaction with alternative coolant species.

**Table 2.2. Highest ranked issues and facilities for sodium fast reactors**

| Thermo-fluids | Fuel safety | Physics | Severe accident |
|---|---|---|---|
| Flow regime transitions, transport properties, channel flow distribution, sodium boiling<br>KASOLA, NADYNE | Fuel pin performance under steady-state conditions (irradiation, cladding)<br>FBTR, JOYO, ASTRID, JHR | Doppler, fuel expansion reactivity feedback | Unprotected loss of flow with subsequent transient overpower accident and uncontrolled passage of gas bubble accident<br>MELT, IGR, ACRR, CAFÉ, TREAT, |
| Coolant structure interaction KASOLA, TTS, PLANDTL | Margin to fuel melting<br>JOYO, TREAT, JHR | Reactivity feedback due to sodium voiding | Local blockage accident<br>TREAT |
| | New fuel pin designs and materials (fuel, cladding)<br>FBTR, ACRR, JOYO | | Fuel coolant interaction<br>MELT, IGR, ACRR, MCCI, SURTSEY, VULCANO, TREAT |
| | Use of minor actinides<br>FBTR, ACRR, JOYO, TREAT | | Post-accident decay heat removal<br>MELT, IGR, ACRR, MCCI, VULCANO |
| | | | Release of fission products (source term uncertainty)<br>MERARG, ACRR, VERDON, TREAT |

## 2.9. Molten salt reactors

The molten salt reactor (MSR) is a concept that dates back to the 1950s, when the design was first proposed as the propulsion system of a nuclear-powered aircraft at the Oak Ridge National Laboratory. It is now one of the reactor concepts identified by the GEN IV International Forum as a candidate for co-operative development.

The GEN IV initiative, in conjunction with national reactor development drivers, revitalised safety research interest in the concept (Serp et al., 2013; Elsheikh, 2014; Watt-Logic, 2017), which can:

- minimise weapons useable material in storage;
- minimise need for high-level waste repository space;
- increase the proliferation resistance of nuclear energy;
- make beneficial use of spent fuel from LWRs;
- increase resource utilisation.

### *2.9.1. Design features of molten salt reactors*

MSRs are a class of nuclear fission reactors in which the primary coolant, or even the fuel itself, is a molten salt mixture. Two concepts are available. In the first, fissile material is dissolved in the molten salt. In the second, the molten salt serves as the low-pressure coolant to a coated particle fuelled core similar to that employed in HTRs. The solid fuel variant is typically referred to as a fluoride salt-cooled high-temperature reactor (FHR).

Since the 1940s, several MSR concepts have been proposed all over the world using different fuel (uranium [U], plutonium [Pu], thorium [Th]) and salt compositions (chlorides, fluorides). These research projects attempted to find the optimum solution for the fuel salt composition suitable simultaneously for optimising:

- neutronic properties (neutron moderation, breeding ratio, fissile inventory);
- operating temperature (melting temperature, radiation stability, transport properties);
- actinide and fission products solubility in the considered molten salt to guarantee a homogeneous core composition;
- materials compatibility and salt chemistry control;
- processing and low-waste options.

Design concepts for MSR abound, but some common themes can be identified. First, MSRs run at much higher temperatures (up to 700-750°C) than LWRs and operate at near-atmospheric pressure. Molten salts offer attractive characteristics as coolants, especially their high volumetric heat capacities and high boiling points. In liquid fuelled MSR designs, the nuclear fuel is dissolved in the molten fluoride salt coolant as fissile elements such as $UF_4$, $PuF_3$, minor actinides fluorides and/or fertile elements such as $ThF_4$ depending on the desired application (breeder reactor, actinide burner, etc.). Liquid fuelled MSRs are often associated with the $^{233}U$-$^{232}Th$ fuel cycle which, by using abundant thorium as a fertile element, enables breeding with a thermal spectrum, and is often considered as more convenient than the U-Pu fuel cycle in order to minimise the generation of highly radiotoxic transuranic elements.

Examples of different concepts demonstrated already include:

- In the first phase of the Molten Salt Reactor Experiment (MSRE) at Oakridge from 1965 to 1968, $^{235}U$ tetrafluoride ($UF_4$) enriched to 33% was dissolved in molten lithium, beryllium and zirconium fluorides at 600-700°C which flowed through a graphite moderator at ambient pressure, with the fuel comprising about 1% of the fluid. The second phase of the project (1968-69) used $^{233}U$, making MSRE the first reactor to use $^{233}U$. This programme paved the way for building an MSR breeder utilising thorium, which would operate in the thermal (slow) neutron spectrum.

- In the United Kingdom, plans for a 2.5 GW lead-cooled fast-spectrum MSR with the plutonium fuel dissolved in a molten chloride salt were only recently declassified (EPD, 2015). Theoretical work on the concept was conducted between 1964 and 1966, while experimental work was ongoing between 1968 and 1973.

There are a host of MSR designs currently under development, including several at the conceptual designs highlighted in Watt-Logic (2017). Examples, which vary widely with respect to technology readiness levels include:

- In Europe, concepts have been focused on fast-spectrum concepts with or without thorium support (e.g. molten salt fast reactor [MSFR] and molten salt actinide recycler and transmuter [MOSART] led by the Kurchatove Institute in Russia (Ignatiev et al., 2015), which have been recognised within the GEN IV International Forum as long-term alternatives to solid-fuelled fast neutron reactors with attractive features (strongly negative feedback coefficients, and smaller fissile inventory, simplified fuel cycle). Several considerations have demonstrated the potential of the MSFR and MOSART as systems with flexible configurations and fuel cycle scenarios which can operate within technical limits with different loadings and make up based on transuranic elements (TRUs) from spent LWR fuel with MA/TRU ratios up to 0.45 as special actinide transmuters, as self-sustainable system or even as a breeder.
- In the United States, Massachusetts-based Transatomic Power Corp is planning to develop a single-fluid MSR with a zirconium hydride moderator and a lithium fluoride – uranium fluoride salt, which can hold about 27 times as much uranium as an LWR.
- Newly incorporated Alpha-Tech Research Corp in Utah (United States) is developing a 30 MW thorium test reactor to produce medical isotopes, and has a consortium of seven Utah counties considering supporting the project.
- Flibe Energy, based in Alabama (United States), is studying a thorium-fuelled, graphite-moderated thermal reactor concept based on the 1970s MSRE. It uses lithium fluoride/ beryllium fluoride (FLiBe) salt as its primary coolant in both circuits. It starts with a pilot plant, followed by a 100 MW demonstration reactor entering operation in 2023, and ultimately a commercial plant in the 250-1 000 MW range ready by 2027.
- In the United States ThorCon International is developing small modular reactors based on the MSRE concept and believes it can build a full-scale 250 MW prototype in as little as four years, providing a suitable test site can be found. The company is collaborating with the Indonesian government launching a molten salt thorium reactor that is expected to be operating by 2025 (Dalton, 2017).
- In 2011, the Chinese Academy of Sciences announced its intention to commercialise a thorium-based MSR in 20 years (Mitchell, 2016) (non-thorium MSRs and solid fuel thorium reactors are also being developed).
- The Shanghai Institute of Applied Physics (SINAP) has two streams of thorium MSR development (Yu and Xu, 2016; Wang, 2016): solid fuel, in pebbles or prisms/blocks with a once-through fuel cycle, and liquid fuel (dissolved in FLiBe coolant) with reprocessing and recycle. A third stream of fast reactors to consume actinides from LWRs is planned.
- Canada's Terrestrial Energy is developing a modular liquid fuel MSR using low-enriched uranium. Each unit is envisaged to have a 190 MW capacity, and be deliverable in four years. A feasibility study is underway to explore siting the world's first commercial unit at Canadian Nuclear Laboratories.
- Thorium Power Canada claims to have a construction-ready, modular thorium reactor. A 10 MW plant is planned for Copiapa in Chile, while a 25 MW demonstration plant is slated for Indonesia.
- UK-based Moltex Energy's stable salt reactor is a conceptual reactor design with no pumps that relies on convection from static vertical fuel tubes in the core to convey heat to the steam generators. The reactors are designed to be modular in construction, with unit sizes from 150 MW.

- Several universities and companies propose reactor concepts of molten salt reactors in Japan, including 10 MW mini thorium molten salt power plants for supplying IT servers and electric vehicle charging stations, and integral molten salt fast reactor (IMSFR) proposed as a highly efficient transmutations system for tansuranium elements (Yamawaki and Koyama, 2015).
- Seaborg Technologies in Denmark is also developing modular thorium MSRs, and is planning a 50 MWt pilot unit with a view to 250 MWt commercial modular units fuelled by spent LWR fuel and thorium.
- In 2017, NRG (Netherlands) announced the use of thorium reactors in collaboration with the EU. The Salt Irradiation Experiment ("Salient"), based at the high flux reactor in Petten, will begin with experiments to melt thorium fuel and then bombard it with neutrons to transmute the thorium into $^{233}U$, which can sustain the chain reaction needed to generate energy. If this is successful, the next step will be to study tough, temperature-resistant metal alloys and other materials that can survive the high heat and corrosive conditions inside a molten salt reactor. Further investigations into handling of waste materials will follow.

### *2.9.2. Safety issues of molten salt reactors*

MSRs have some unique characteristics offering a potentially safer, more efficient and sustainable form of nuclear power associated with online fuel processing. For example:

- MSRs do not operate under high pressure, and are not cooled by water, making steam explosion impossible.
- For the case in which the fuel is dissolved in a molten salt, in case of reactor damage, the fuel salt can be drained into sub-critical storage tanks.

There are several unique safety and operational challenges associated with MSRs, however:

- The risks of corrosion by the impurities (oxygen and water primarily) dissolved into a molten salt coolant or by the fission products present in fuel salt. Advanced materials investigations for MSFR and FHR designs are currently underway in different countries, but qualification efforts for special materials involved in the MSR design are still needed. Experience gained at the ORNL in the 1960s and 1970s remains a source of knowledge that demonstrated many aspects of the technology; however, new designs with different salt mixtures present challenges.
- The sometimes conflicting requirements for corrosion resistance and limiting radiation damage. Nickel steels such as Hasteloy, while superior for molten salt corrosion resistance, undergo radiation-induced embrittlement.
- Beryllium is a proposed component of some fuel designs; it is highly toxic.
- Normal operation produces small amounts of fission product tellurium, which deposits on surfaces and leads to intergranular cracking.
- Use of lithium (Li) creates some issues. $^{7}Li$ must be used, since $^{6}Li$ is a neutron poison. This necessitates isotopic enrichment of the fuel salts to remove $^{6}Li$.
- Fission gas tritium is produced from irradiation of $^{7}Li$ directly, and indirectly from $^{6}He$. Hence, tritium management strategies must be adopted. The PHWR community has significant experience with tritium management; however, the complication of much higher operating temperatures allows rapid diffusion of tritium through the walls or the primary heat exchanger to the coolant salt and into the steam system. This could be discharged during a steam system blowdown. Investigations of the use of additional molten salt eutectic mixtures in a separate loop for tritium removal in MSR somewhat challenges current technologies.
- Waste disposal is complicated by fission products being in a water-soluble form as fluoride salts. These are not as amenable to long-term storage as insoluble waste forms.
- Even in insoluble waste forms, irradiation (self) of immobilised waste can produce $UF_6$ and fluorine gas.

## 2.10. Small modular reactors

Since 2010 there has been a flurry of activity in the development of small modular reactors. A recent post by the World Nuclear Association states:

> There is strong interest in small and simpler units for generating electricity from nuclear power, and for process heat. This interest in small and medium nuclear power reactors is driven both by a desire to reduce the impact of capital costs and to provide power away from large grid systems. The technologies involved are numerous and very diverse. (WNA, 2021)

As nuclear power generation has become established since the 1950s, the size of reactor units has grown from 60 MWe to more than 1 600 MWe, with corresponding economies of scale in operation. At the same time, there have been many hundreds of smaller power reactors built for naval use (up to 190 MW thermal) and as neutron sources yielding enormous expertise in the engineering of small power units. The IAEA defines "small" as under 300 MWe, and up to about 700 MWe as "medium" – including many operational units from the 20th century. Together, they are now referred to by the IAEA as small and medium reactors (SMRs). However, "SMR" is used more commonly as an acronym for "small modular reactor", designed for serial construction and collectively to comprise a large nuclear power plant. (In this report, the use of diverse pre-fabricated modules to expedite the construction of a single large reactor is not relevant.) A subcategory of very small reactors – vSMRs – is proposed for units under about 15 MWe, especially for remote communities.

### *2.10.1. Design features and safety issues of small modular reactors*

SMR designs cover a range of concepts, many of which were described in this chapter, and most of which have advanced safety features. The safety issues, therefore, are the same as those identified for the larger scale reactors. There are a few unique concerns that arise as the result of the use of SMRs in remote areas, however:

- The remote nature of some installation sites requires more extensive evaluation for implementation of emergency measures in the event of an accident.
- There are security concerns for remote operation, due to difficulties in ascertaining that fuel diversion is not occurring.
- Digital security/safety solutions may be difficult to access without appropriate IT infrastructure.
- Many proposed installations in the north are in areas that are environmentally vulnerable. Environmental concerns range from damage to permafrost coverage, lack of access to secondary heat sinks that conventional reactors rely on, enhanced radionuclide migration to the biosphere from routine or accidental releases, and ease of establishing rigorous environmental monitoring.

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# 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.

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

- thermal-hydraulics;
- nuclear fuel;
- reactor physics;
- severe accidents and containment phenomena;
- integrity of equipment and structures.

It lists the safety research issues currently associated with each technical area. In general, the safety research issues are relevant to all reactors, but those that are specific to one type of reactor are noted. The reactor types addressed in this section include boiling water reactors (BWRs), pressurised water reactors (PWRs), water-water energetic reactors (VVERs), pressurised heavy water reactors (PHWRs), advanced light water reactors (ALWRs) and advanced pressurised heavy water reactors (APHWRs). Other reactor types with issues corresponding to the above technical areas are addressed briefly in the second section.

A table for each technical area provides 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 table lists the facilities currently available that are conducting or have the potential to conduct research relevant to each issue.

The previous version of this document included a ranking of the importance of each safety issue, followed by a ranking of facilities on the basis of their importance to resolving the issue. Facilities at risk were considered on the basis of their versatility to address more than one issue, their uniqueness and the cost of their operation or replacement. This general approach was followed in the current assessment, although a detailed ranking was not done in the same systematic manner, largely because the previous thorough assessment established the importance of many of the facilities to resolving safety issues, and the cost of their replacement and operation. In most cases, the issues faced today are very similar to those faced at the end of the 2000s. Some have been largely resolved (e.g. boron dilution and wear) and have decreased in importance, but regulatory drivers, life extension of ageing nuclear power plants and advances in-reactor design have expanded the number of issues. The Fukushima accident has also identified issues previously not foreseen. Care has been taken to identify only those facilities that might require significant investment (>EUR 5 million) or medium investment for very unique facilities (>EUR 2 million). Furthermore, except where identified, it is assumed that all of the issues have a high to medium safety relevance, and that additional research is required to resolve them.

The technical areas addressed in Section 3.2 on issues and facilities not unique to the nuclear industry are:

- human and organisational factors;
- plant control and monitoring;
- cybersecurity;
- external events;

- fire assessment.

Each of these technical areas also includes a description of associated safety research issues and facilities that have been identified to resolve the issues. In general, data needs to resolve these issues can be met using facilities more widely available because of use by a broader industry base, because they are not unique to the nuclear industry. No recommendations were made in the previous report for the NEA Committee on the Safety of Nuclear Installations (CSNI) to consider related to preservation of these facilities, which were listed for information only. The current report does identify a unique fire assessment capability that is at risk, however.

## 3.1. Issues and facilities unique to the nuclear industry

### *3.1.2. Thermal-hydraulics*

*Introduction*

Thermal-hydraulics is one of the fundamental disciplines for the design and operation of water-cooled nuclear reactors. Achieving large core power densities requires a deep understanding of thermal-hydraulics. 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 light water reactors (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 thermal-hydraulic code validation as well as the experimental database needed for such validation with great attention. The previous Senior Expert Group on Safety Research (SESAR) reports give an overview of the large number of separate-effect and integral test programmes that have been carried out in the past (NEA, 1993; 1996; 2001; 2007). The results from these programmes provide a sound basis for model validation of traditional system codes, whereas they are insufficient for multidimensional and, especially, for computational fluid dynamics (CFD) codes.

Extensive research programmes in thermal-hydraulics were carried out from the 1970s to the 1990s and contributed to confirming the safety of existing reactors. Once these objectives were achieved, the necessity for some of these activities diminished, as did the available national funding streams. As a result, several large-scale facilities and even laboratories all over the world have been closed or have largely reduced their activities. On the one hand, the availability of data from those research programmes for future use became difficult or questionable; on the other hand, the expertise of experimentalists leading those programmes is diminishing.

In the early 2000s, the CSNI took the initiative 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. Some of these projects have been completed (SETH, PSB-VVER, ROSA-1 and 2), while others are still ongoing (PKL-1 to 4, ATLAS-1 and 2). From PKL-2 onwards, the PKL tests were complemented by experiments performed in the PMK (Hungary), ROCOM (Germany) and, more recently, PACTEL (Finland) facilities.

The SETH project focused on the capabilities of the PKL and PANDA facilities, which were recommended for international consideration (NEA, 2009a). The PKL experiments addressed the issue of potential boron dilution accidents in PWR reactors; this subject figured prominently on the programme of the follow-up PKL projects as well. The PANDA experiments provided data on containment three-dimensional (3D) gas flow and distribution that are important for code prediction capability improvements, accident management and design of mitigating measures. The PSB-VVER project had the objective of providing unique experimental data needed for the validation of thermal-hydraulic codes used for the safety assessment of VVER-1000 reactors. In the framework of the ROSA-1 and ROSA-2 projects, different issues were investigated, *viz.* temperature stratification and mixing, water hammer, natural circulation in the primary circuit and different LOCA cases (pressure vessel break, intermediate breaks and steam generator tube ruptures [SGTR]).

From the beginning, the different phases of the PKL project addressed the consequences of inherent boron dilution and proposed countermeasures to avoid critical situations, both during small-break LOCA and accidents in cold shutdown conditions. Cool down under asymmetric natural circulation conditions and fast cool down transients following a main steam line break were also evaluated (PKL-2). After Fukushima, beyond design-basis accidents became of primary interest. Accident scenarios such as station blackout (SBO), LOCA with additional system failures leading to significant core heat-up, became the central topics of the OECD/PKL-3 project. At the same time, the efficiency of using core exit temperature measurements to diagnose severe core heat-up was also investigated. The PKL-4 project, currently underway, is performing LOCA-related basic parameter studies (quench front propagation, swell level investigation) for code model development and validation, but is also expected to come to a final conclusion on inherent boron dilution and on boron precipitation following large-break LOCA. The PKL tests have been complemented by experiments performed in other facilities with the aim to investigate the system behaviour of different type of plants (PMK for VVERs) or the scale effect (PACTEL), but also more detailed 3D processes (ROCOM).

The first OECD ATLAS project started in 2014 and addressed thermal-hydraulic safety issues and accident management issues relevant for water reactors, by means of experiments in the Korean ATLAS integral effect test facility. Prolonged SBO with active or passive secondary cooling, small-break LOCA during SBO and total loss of feedwater assuming additional failures were investigated and, to address the scaling issue, two counterpart tests with ROSA were performed. ATLAS-2, initiated in 2017, is investigating safety key issues. Specifically, it will evaluate: long-term coolability with a failure of the residual heat removal system; passive core makeup with either a hybrid safety injection tank or passive emergency core cooling system during SBO; intermediate break of LOCA either at the pressuriser surge line or the direct vessel injection line; design extension condition scenarios such as steam line break followed by SGTR; and scaling issues by conducting a counterpart test for the previous reactor pressure vessel head break LOCA conducted at LSTF. Strong co-operation has developed among the ROSA, PKL and ATLAS projects in the past decade, materialising in counterpart tests and joint analytical workshops, the latter reflecting the strong commitment of the participants to perform code validation in parallel to the running projects. A computer code benchmark has always been a part of these projects assuring a fruitful co-operation between experimentalists and analysts.

*Scope*

As the scenarios of primary concern shifted from the large-break LOCA to small and intermediate breaks as well as to 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 a 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 bloc countries, provided additional needs for research and development. The emergence of advanced LWRs having passive safety systems opened another new area of less known phenomena and situations that had to be addressed.

The Fukushima accident raised a number of questions, but these concerned mainly issues or conditions which could lead to severe accidents. However, it also pointed out the potential vulnerability of spent fuel pools (SFP). In order to address this question, a joint activity by the Working Group on Fuel Safety (WGFS) and Working Group on Analysis and Management of Accidents (WGAMA) produced a status report on SFPs, followed by a phenomena identification and ranking table to identify the most important phenomena (NEA, 2015b; 2017). One of the conclusions was that SFP behaviour is significantly influenced by 3D effects, which need further experimental support for model development. The Fukushima accident also highlighted the need for understanding the reliability of cooling systems connected with the primary loop (e.g. the isolation condenser and the reactor core isolation cooling) and improved modelling via coupled thermal-hydraulic and mechanical modelling. Finally, Fukushima reinforced the already ongoing efforts to evaluate the consequences of beyond design-basis accidents: as a result, design extension conditions have to be taken into consideration in several countries in the design of nuclear power plants, and has improved accident management concepts. This has prompted the need for better modelling tools, particularly for new builds and reactor designs, as well as for less explored accident sequences.

System codes have reached maturity over the past two decades, but the demand to reduce safety margins, for example due to power upgrading of existing reactors, may also require refinement and further validation of existing analytical tools and additional experimental data. The better understanding of 3D processes within the primary system also calls for more precise experimental data for code validation purposes; this is also underscored by an ongoing WGAMA activity: Status on Simulation Capability of 3D System-scale T-H Analysis Codes (3DSYSTH).

Uncertainty and sensitivity assessments have become increasingly important in thermal-hydraulic analysis. In the last decade, a number of CSNI activities have addressed this issue (BEMUSE, PREMIUM, SAPIUM) with goals of reviewing available methods, assessing their capabilities and identifying further needs. It is important to note that “best estimate plus uncertainty” (BEPU) modelling requires, in many cases, more reliable test data, including reduced or well-understood measurement uncertainties.

CFD has made good progress in the last decade supported by computer development, and the use of CFD tools for nuclear safety applications has expanded considerably. The growing interest in CFD applications is also reflected by the high popularity of WGAMA actions in the field, like the series of CFD4NRS workshops and of benchmarks. The move towards uncertainty quantification of CFD results represents an important milestone in the area. All these developments have led to single-phase CFD being increasingly used in support of safety studies.

The *Scaling State of the Art Report* (S-SOAR), issued in 2017 (Bestion et al., 2017), constitutes a cornerstone for harmonising the knowledge and the understanding in relation to a vital issue in nuclear reactor safety. Scaling activities were considered essential in the 2001 SESAR (NEA, 2009a) report and scaling is part of most of the thermal-hydraulic issues identified in the 2007 SESAR report (NEA, 2007).

Based on the most important safety issues and a review of available experimental data to cover them, a list of prioritised thermal-hydraulic issues was proposed in 2007 (NEA, 2007). It is worthwhile to review them in order to measure the progress made in the last decade.

In a number of cases, CSNI projects or activities were initiated to address the issues:

- boron dilution has been investigated within several PKL projects, producing useful information both for safety evaluation and for code validation;
- mixing in cold legs and downcomer was extensively investigated in ROSA and ROCOM (as part of the PKL project);
- a WGAMA activity (3D-SYSTH) is presently reviewing the needs as required by 3D system codes;
- shutdown accidents have been one of the important items in the ROSA, PKL and ATLAS projects;
- the ECCS strainer clogging issue was investigated by a CSNI Task Group (NEA, 2013);

- a CFD benchmark was performed to study the maturity of CFD codes for the analysis of thermal cycling (NEA, 2011c);
- a recently started WGAMA activity will produce a status report in the field of passive system performance;
- a joint WGFS/WGAMA activity produced a status report on vulnerabilities of spent fuel pools.

It is expected that when these activities are completed, decisions can be taken as to whether to close these issues or to make recommendations for further actions.

No CSNI activities have been launched in the past decade on BWR stability or PHWR thermal-hydraulics. However, extensive studies have been performed for PHWRs by the CANDU Owner's Group and the Atomic Energy of Canada Limited for a series of postulated accident scenarios ranging from design basis to station blackout events using RD-14M. In addition, moderator cooling and flow experiments were performed to evaluate moderator analyses tools, and a series of experiments were performed to derive analysis parameters for critical heat flux and post dryout behaviour for various CANDU bundle types, with various degrees of diametral creep and axial power profiles. Freon experiments were also performed to support the high-temperature experiments in water, and a header facility was built to evaluate the 3D water flow through a CANDU core. Fukushima has also prompted evaluation of spent fuel cooling in CANDU designs.

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 and the need for design-basis extension and severe accident analyses. For example, as current plants continue to 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.

The remaining issues from 2007 are considered to still be pertinent, even if a slight shift in their content may be observed.

*Description*

Table 3.1 shows the list of current thermal-hydraulic safety issues that has been derived. It includes safety-related phenomena, accident scenarios and parameters. The list is based on the corresponding table of the previous SESAR report (NEA, 2011c), on a recent list of important thermal-hydraulic phenomena (Aksan, D'Auria and Glaeser, 2018; D'Auria et al., 2017) and on discussions of the members of the present SESAR group. The first column lists the high-priority issues; however, no priority is assigned to the row number. The second column provides a description of each issue and the third the motivation for inclusion in the present list. Identification of the facilities to resolve the issues is given in column 4 and additional notes are provided in the last column.

### *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. 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 nuclear power plant safety can vary depending on 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.

**Table 3.1. Current thermal-hydraulic safety issues**

| Issue | Description | Motivation | Facilities | Notes |
|---|---|---|---|---|
| 1) Anticipated transient without SCRAM (ATWS) | Each anticipated operational occurrence (AOO) may cause an ATWS. ATWS typically evolve at high pressure and high linear power. Situations of interest include void collapse following pressure increase in boiling water reactors and may include boiling transition during AOO. | Core heat transfer is concerned. The corresponding thermal-hydraulic phenomenon is "nuclear thermal-hydraulic feedback". Control rod ejection ATWS (see Issue 13) is not considered. See also Issue 14. | SPES, PMK, HIDRA. Facilities characterised by full power (and full linear power) in addition to full height and full pressure scaling are needed to simulate conditions expected after ATWS. | An attempt could be made to ask Russian scientists about the PSB facility in Electrogorsk and to use LOFT ATWS experiments. |
| 2) Natural circulation (NC) | NC is established when a heat sink and source co-exist at different elevations. Sub-topics include: stratification; flow stability; reliability of passive systems; application of computational fluid dynamics (CFD); boron dilution; CANDU NC between core and steam generator (SG); 3D effects including flow distribution and mixing in CL and in downcomer; over-spilling effect or siphon condensation; reflux condensation; non-condensable gas effect; horizontal heated channels and stratification of two-phase flow; NC in the presence of multiple interacting (passive) systems. | In the case of boron dilution, re-criticality might be of concern during a few days after the start of each fuel cycle (high boron concentration in the primary system). Need to model parallel U-tubes to catch flow reversal and stability phenomena. NC is a key phenomenon in case of steam generator tube ruptures in Section 3.1.4. | PKL, ATLAS, PACTEL, LSTF, FINCLS. Full height scaling is preferred for NC studies either in pressurised water reactor (including AP-1000) and in boiling water reactor conditions. | Several facilities are suitable for NC-related investigations. |
| 3) Interaction primary-system (PS)/containment (CO) | PS and CO interact at thermal-hydraulic level in case of most of the accident scenarios. The occurrence of two-phase critical flow (TPCF) at the break (e.g. in case of a loss-of-coolant accident) keeps PS and CO decoupled. However, when conditions for TPCF disappear, a tight interaction is established. The conditions of TPCF disappearance is of interest, as well as long-term cooling, including sump recirculation and potential for clogging and pump cavitation due to debris. TPCF through valves remains of concern since the writing of the CCVM on Separate Effects Test Facilities (non-pipe breaks). Additional interest derives from the possible occurrence of supersonic conditions in downstream valves discharging in pools. Additional importance for this issue is expected in various SMR and AP-1000 accident scenarios. | Few test data are available (somewhat ignored), e.g. from the Bubble Condenser facility in Russia (TACIS/PHARE project) and the multi-application small light water reactor (MASLWR) ITF at Oregon State University (United States). Interaction tests with PS and CO are planned in ATLAS, in which a new containment is under construction. | MASLWR (United States), ATLAS-CUBE (Korea), interactions between PS and CO are of interest in experiments investigating two-phase critical flow and transition to Bernoulli flow. Condensation, spray, stored energy distribution of internal structures, compartment arrangement are relevant phenomena and should be instrumented. Reverse flow from CO to PS in the long-term cooling phase, resulting in clogging and $N_2$ ingress also of interest. | The Bubble Condenser facility in Russia could be considered. |

**Table 3.1. Current thermal-hydraulic safety issues** *(cont'd)*

| Issue | Description | Motivation | Facilities | Notes |
|---|---|---|---|---|
| 4) Coefficients for pressure drop at geometric discontinuity (KPD) | Pressure drops at geometric discontinuity (direct flow and flow reversal) substantially affect any accident scenarios. Under many circumstances, errors in KPD result in inaccuracies in calculations which are incorrectly attributed to different origins and which may lead to modification of existing code models which are not responsible for those inaccuracies. | Limited data availability: CFD application might be used to derive suitable values for KPD, but errors are not negligible and derivations are cumbersome in two-phase conditions. Experimental data may be affected by large errors due to mixing of reversible and irreversible contributions. | PKL with high-quality instrumentation (pressure drop) is relevant. Reversible and irreversible pressure drop are of interest. Specific instrumentation needed (at least in one or a few geometric discontinuity locations). | All facilities have the potential to be relevant in this connection as long as suitable instrumentation is available. |
| 5) Main coolant pump (MCP) trip performance (mostly in two-phase conditions) | MCP affects LOCA and non-LOCA scenarios during the coast-down phase and long-term NC cooling because of pressure drops inside complex geometries (KPD) and TPCF, which are difficult to characterise experimentally and may not obey known scaling law. | Four-quadrant head curves used in computer codes feature large uncertainties, especially in two-phase conditions. | Reactor Coolant Pump Test (RCPT) Facility in Korea. Large-scale and two-phase conditions (including suitable range of parameters investigated); separate effects tests also relevant. | Other component performances may constitute an issue, e.g. separator, dryers, jet pumps, etc. |
| 6) Shutdown transient other than de-borated water injection and residual heat removal failures | During shutdown, the reactor is partially unprotected (e.g. removal of parts of PS) and more exposed to operator actions. Failure of the residual heat removal system (RHRS) can challenge the core coolability. In the mid-loop operation, air ingress into the RHRS pump due to the hot-leg water level decrease can result in failure. | A variety of situations may need investigation: experimental data are needed and modelling capabilities need to be demonstrated. Some test data are available from LSTF, ATLAS and PKL. | LSTF (Japan), ATLAS (Korea), PKL (Germany). Low pressure, openings in the primary loop and natural circulation are key words for the experiments. | |
| 7) Coolant flow distribution through horizontal channels in pressure tube reactors | In a PHWR, coolant is distributed to the fuel channels via individual feeder pipes from a distribution header. The distribution of the PHWR primary coolant through the channels as a function of ageing (diametral creep), axial power profile, bundle deformation and header distribution system needs to be assessed for a wide variety of possible scenarios and anticipated operational occurrences. | | Full-scale, instrumented water loops operated at PHWR temperature and pressure, full-scale PHWR freon loop full bundle CANDU header facility 1/8 scale (Canada). | |
| 8) Fluid structure interaction (FSI) | Also including thermal cycling leading to thermal-fatigue. Thermal-hydraulic oscillation may also trigger FSI, e.g. in case of fuel rods or SG tubes. Pressure wave propagation phenomena may also be affected by FSI, including load on internals following a LOCA. Turbulence and swirl formation at elbow, bend, orifice, etc. may affect erosion-corrosion of pipe wall. | Motivation is related to numerous problems with flow-induced vibration of nuclear power plants. SG tubes, fuel rod (or fuel assembly) and RPV internals are of main interest. In facilities, well-defined boundary conditions and simplified geometry facilitating easy meshing is needed for code (mainly CFD) validation. | Small-scale burst test facility. (Canada), MSUB facility (Canada). | The issue is more relevant for structural integrity, see Table 3.6, Item 8. Use of freon to induce vibrations in steam generator tube arrays. |

**Table 3.1. Current thermal-hydraulic safety issues** *(cont'd)*

| Issue | Description | Motivation | Facilities | Notes |
|---|---|---|---|---|
| 9) Passive system performance | The installations of passive systems poses new challenges to safety, which are strongly system dependent: leakages, minor installations errors for horizontal pipes, heat losses, small amounts of non-condensable gases. Activity connected with reliability of passive systems. | Various sensitivity studies are necessary in facilities to confirm the reliability of passive systems (flow instability, water hammer, condensation, boiling, common head effects, heat exchanger inclination of effects, and so on). Issue as NC and KPD are also applicable here. | PACTEL (Finland), LSTF (Japan), INKA (Germany), ATLAS-PAFS (Korea), PASCAL, FESTA (Korea). | Full height and full pressure facilities are relevant, but reduced height facilities are complementary. |
| 10) Power excursion during loss-of-coolant accident | Also covering header flow distribution and moderator sub-cooling and sub-cooling distribution phenomena. | Associated with positive void coefficient. | OMEGA loop (France). | OMEGA loop is a separate effects test facility where power supply (other than decay power) during a LOCA is investigated relevant to CANDU, PHWR of Atucha and RBMK. |
| 11) 3D global void, temperature and flow distribution | The trend towards higher power densities means that in open cores, departure from nucleate boiling in the maximum loaded fuel bundle and single rod may be prevented only due to the cross-flow connections inside the core. Especially for boiling water reactors, new core geometries such as partial length rod and swirl-type spacer significantly affect the core heat transfer. | For simulation, multidimensional analysis tools and improvement of subchannel codes are needed. Experiments focusing on multidimensional behaviour, and spacer effects are needed to improve primarily CFD 1-phase modelling of mixing in-core. | MCT (Korea, Moderator Facility). Stern Labs steam-water experiments, and CNL Freon experiments (Canada), study effects of spacers and appendages on flow and departure from nucleate boiling on PHWR horizontal bundle. | "Old" data available (partly) from UPTF (Germany), SCTF (United States) and CCTF (Japan) large-scale facility. |
| 12) Spent fuel pool (SFP) accidents | SFP behaviour, and deborated water injection are of interest. Because of large FP inventory in the SFP, its cooling should be maintained in all possible accident conditions, including those with an uncovered rod, where $H_2$ production, spray cooling and 3D pool fluid circulation are of interest. Water injection using the fire engine is also now considered as an important AM measure when plant-wide damage occurs. A variety of situations involve injection of deborated water. | Both identified key AS became of interest after Fukushima. Phenomena are still unclear during loss of cooling of SFP: convection, boiling in the pool, uncovering process of fuel assemblies, steam aspiration, criticality due to boiling, flow reversal, flow blockage, NC reduction by recovery of cooling system, spray cooling, CCFL, void distribution inside the fuel assembly, oxidation by air and steam. | MEDEA, ASPIC, MIDITH (France). | All part of the DENOPI programme being executed by the IRSN. |
| 13) Reactivity insertion accident (RIA) including control rod ejection (CRE). Also included under fuel. | The cladding temperature evolution plays an important role since it strongly affects the mechanical strength of the cladding. The knowledge of transient clad-to-coolant heat transfer coefficients are uncertain and need to be verified by experiments. Such heat transfer is also dependent on the transient power change. Sub-cooled boiling and boiling transition are of interest for CRE. Transient void formation is also important to assess the void feedback effect. | For high-burnup fuels, the enthalpy limit and PCMI failure threshold are reduced, resulting in safety margin reduction. Apart from fuel behaviour, from the viewpoint of thermal-hydraulics, applicability of the conventional post-CHF (critical heat flux) heat transfer models (CHF, transition boiling and film boiling) to the RIA conditions is of interest. Transient boiling data are not sufficient in case of CRE. | CABRI, BR-2, NSRR. Facilities to simulate RIA conditions in pool and/or convective boiling with various power pulse variations are necessary to validate thermal-hydraulics models and to reduce conservatism of the current regulatory practice. | There are facilities with simulated rods, but the others are research reactors using fuel rods. There are already some data from the now closed PATRICIA. |

**Table 3.1. Current thermal-hydraulic safety issues** *(cont'd)*

| Issue | Description | Motivation | Facilities | Notes |
|---|---|---|---|---|
| 14) Heat transfer at high temperature with radiation (HTR) | Combination of convection and radiation (HTR) in the presence of $H_2$ production possibly with partly damaged fuel (ballooned or embrittled clad) is safety relevant. | Experimental data not sufficient. Ranges of variations of this key parameter different from Issue 15. | High Temperature Fuel Channels Laboratories (Canada). | At high temperature, radiative heat transfer becomes more dominant than convection with a magnitude dependent on emissivity of the oxide layer of the fuel rod or pressure tube. Canada has a facility to measure emissivity from pressure tubes as a function of oxide thickness. |
| 15) Reflood/ quench front propagation (QFP) | Well-established phenomenon occurring with the transition from film to nucleate boiling or forced convection at low pressure under large-break LOCA conditions. Combination of QFP with situation of ballooning, $H_2$ production, crud formation, etc. pose safety concerns. Core heat transfer behaviour during AOO is of interest for boiling water reactors in terms of the acceptance criteria of the safety analysis, where the boiling transition (BT) is allowed as long as the fuel integrity is maintained. Models to predict the maximum temperature and duration of BT should be validated with the data, which are not sufficient, especially for rewetting propagation. | Revision of the acceptance criteria for AOO was proposed by the Atomic Energy Society of Japan in 2003. Investigation of ballooning (including fuel relocation and consequent thermal power effect) with simultaneous $H_2$ production. The rewetting mechanism at the quench front and its effects on the heat transfer are still unclear. Currently, empirical correlations are used for the heat transfer prediction, but new LOCA rulemaking and adoption of design extension conditions need integrated analysis of fuel performance and thermal-hydraulics. Fully coupled fuel behaviour and thermal-hydraulic experiments are required. | QUENCH (Germany), QUEST (Korea), ATHER (Korea), HIDRA (Japan), RBHT (United States), COAL (France). | QUENCH is used to study reflood/quenching at partly damaged fuel elements. COAL focuses on the coolability of a partially deformed fuel assembly, particularly in the ballooned area. HIDRA has test sections of a full-length 4×4 bundle and a short-length 3×3 length bundle. Studies focus on post-boiling transition heat transfer. The QUEST facility utilises IR thermometry and a total reflection visualisation method to obtain temperature distribution of the cladding. RBHT consists of 7×7 full-length rods with a highly detailed thermocouple distribution to measure quench profile and droplet size measurements up and down-stream of the spacer grid. |

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.

After the shutdown of major facilities – Osiris (France), PHEBUS (France), Halden (Norway), the National Research Universal (NRU, Canada) and the Japan Material Testing Reactor (JMTR, Japan) – there is a lack experimental facilities necessary for testing fuel performance and for safety assessment. This is occurring at a time when a new generation of fuels are under development, and at which there are open issues for current fuels which will still be in operation for decades.

*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 quite large, ranging from the "old generation" of fuels, still in operation ($UO_2$, $UO_2$-Gd, $MO_X$), with zirconium-based claddings and for which there are still pending issues, to accident-tolerant fuels, which include "evolutionary-fuels" (doped pellets with Cr coating cladding, for example) and "advanced-fuels (SiC/SiC cladding for example).

*Description*

Fuel designs vary by reactor type and technology. This section gives a summary of the fuel by reactor type.

**LWRs (including VVERs):** For LWRs, the uranium enrichment ranges from 2% to nearly 5%, although in some countries LWR recycled 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) 12-14 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 burnup level (which affects internal fuel pin pressure), the condition of the fuel cladding (e.g. corrosion, oxidation and embrittlement) and the location in the core. For economic reasons, increases in burnup levels are being considered in many countries. To support higher burnups, 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 on how fuel performance under accident conditions changes with changes in burnup, cladding material and service condition.

**ALWRs:** For ALWRs, it is expected that high-burnup levels will be desired for economic and possibly security reasons (e.g. proliferation resistance). To support such fuel designs, burnable poisons and additional advances in cladding materials will likely be needed. Ensuring that 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. Such testing will range from analyses and unirradiated clad testing to more extensive testing, depending on 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. The majority of the fuel is $UO_2$ in composition, but some PHWRs use recycled fuel where the plutonium oxide was never separated from the uranium oxide. There are also reactors utilising thorium oxide to flatten the power profile of the cores on a regular basis. PHWRs utilise a shorter fuel pin length (1.5 feet versus 12-14 feet), smaller pin bundles, low enrichment or natural uranium, and horizontal orientation in pressure tube core geometry. Generally, PHWR fuel is designed for lower burnups and higher linear powers than LWR fuel; however, the safety issues and performance concerns are essentially the same. PHWRs operate using an online refuelling system and thus the fuel handling (and potential refuelling accidents) of PHWR fuel is very different than that for LWRs.

With the exception of a new requirement for some countries to evaluate the impact of fuel failures on transient behaviour, and concerns about fuel integrity during interim storage, the fuel issues listed in Table 3.2 for existing fuel designs are the same as those compiled during the last writing of this report and all issues remain relevant today. A 2012 report on fuel safety criteria (NEA, 2012) states:

> "Complete or sufficient information is not available for a number of issues discussed in this report. These include CRUD deposition, cladding oxidation and hydriding, rod internal gas pressure, pellet-cladding and thermal-mechanical loads, fuel melting, fuel fragmentation, cladding embrittlement, gap activity, radioactive source term, high burn-up, mixed oxide fuel, slow or incomplete control rod insertion, axial offset anomaly, cladding elongation and cladding stability. Under the auspices of the OECD Nuclear Energy Agency, active research is being conducted in many of these areas through programmes including the Halden Reactor Project in Norway, the Studsvik Cladding Integrity Project in Sweden, and the CABRI International Project in France. These issues have been and will continue to be addressed by the Working Group on Fuel Safety, as directed by the NEA Committee on the Safety of Nuclear Installations."

Furthermore, a more recent state of the art report on LOCA written by the Working Group on Fuel Safety (NEA, forthcoming) has made several recommendations, summarised below:

- Ductility-based LOCA criteria derived from ring compression tests of double-sided oxidised specimens have been proposed by the US Nuclear Regulatory Commission to ensure long-term coolability of the core, whereas fracture-based LOCA criteria derived from integral rod testing have been chosen by the French regulators to ensure that the fuel rods are not excessively embrittled and maintain long-term coolability. Both approaches take into account the burnup effect through the level of hydrogen that has been picked up by the cladding prior to the transient. The significant differences between experimental techniques used in different countries to establish LOCA safety criteria call for further discussions between experimentalists and regulators. Development of standard LOCA criteria procedures could be a future objective.
- Fuel fragmentation, relocation and dispersal during a LOCA, observed in several in-pile and out-of-pile tests, needs further investigations. The effects of pellet burnup, the presence of additives in the pellet, cladding corrosion state, linear power during normal operation and axial gas transport phenomena should be addressed in new experiments. Fuel fragmentation, relocation and dispersal is more important today than breakaway oxidation of the cladding, and thus its impact on cladding temperature should be better assessed. As-fabricated cladding was used in most of the oxidation tests, but specific tests have been performed to demonstrate that pre-hydrided/pre-oxidised and sometimes thinned claddings are sufficiently prototypical. As such, a significant amount of new data has been accumulated on the effects of irradiation, alloying elements, pre-oxidising and pre-hydriding of the cladding, steam purity (addition of nitrogen or air), and high external steam pressure. Based on these findings, new licensing frameworks have been defined in most member countries. Any new cladding material has to be studied accordingly.
- It is important to continue to develop predictive computer codes associated with uncertainty analysis methods, to capture quantitatively the increased understanding of the phenomenology. A specific effort should focus on multi-rod configurations to predict core coolability.

Finally, since the last SESAR report was written, significant attention has been given worldwide to the development of accident-tolerant fuels, and advanced reactor designs, necessitating an additional need for irradiation testing of potential fuel designs from performance and safety perspectives. A thorough *State of the Art Report on Accident Tolerant Fuels* was published by the NEA in 2018 (NEA, 2018b) and versatile irradiation options for evaluation of such fuels and concepts were identified. Most of the facilities identified in the last SESAR report were listed, with BR-2, ATR, HFIR, CABRI, Halden, Hanaro, NSRR and HFR remaining on the list as available reactors at the time. Added to the list are the TREAT facility, recently refurbished, the LVR-15 research reactor in the Czech Republic, the Jules Horowitz Reactor in France (under construction) and the China Mianyang Research Reactor (CMRR, People's Republic of China). NRU and Halden were both shut down in 2018, and OSIRIS and JMTR were shut down in 2016.

It is encouraging to note that testing of entire assemblies of accident-tolerant fuels in an operating power reactor has recently been announced. Advanced fuel assemblies featuring chromia-doped fuel pellets and chromium-coated fuel cladding developed by Areva NP under the US Department of Energy's (DOE) Enhanced Accident Tolerant Fuel (EATF) programme, have been loaded into the Vogtle 2 plant in Georgia, which commenced operation on April 3 after a refuelling shutdown.

Table 3.2 lists fuel issues that are still considered to be of high or medium importance, along with the facilities available to address these issues. A comprehensive listing of research reactors worldwide capable of addressing these issues can be found in Table 3.3.

### Table 3.2. Current nuclear fuel issues and facilities

| Issues, scenarios, phenomena | Description | Facilities |
|---|---|---|
| Response to loss-of-coolant accidents (LOCAs) | As fuel designs change (minor evolution or advanced technologies for fuel pellets and claddings), 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 are needed, consistent with the design basis, and perhaps beyond the design-basis LOCA. Also, small- and intermediate-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. Fuel fragmentation, relocation and dispersal inside the primary circuit, cooling of area with ballooning fuels have to be included in research investigations. | ANL Hot cells (United States)<br>CABRI and CEA Hot Cells (France)<br>JAEA and Hot Cells (Japan)<br>MIR and Hot Cells (Russia)<br>RIAR and Hot Cells (Russia)<br>Studsvik Hot cells (Sweden)<br>TREAT and Hot Cells (United States) |
| Response to reactivity insertion accident | As fuel designs change (minor evolution or advanced technologies for fuel pellets and claddings), the response to reactivity insertion accidents needs to be investigated. Failure modes and criteria associated with new cladding materials and new fuel are currently being investigated through experimental programmes, consistent with design-basis and beyond design-basis events. Additional focus may be needed on fuel fragmentation and dispersal after boiling crisis. | BIGR (Russia)<br>CABRI and CEA Hot Cells (France)<br>NSRR and Hot Cells (Japan)<br>TREAT (United States) |
| Response to power oscillation events | As fuel designs change to achieve higher burnup (e.g. through the use of new cladding material), the response to power oscillation events (such as could occur due to an anticipated transient without sram or a loss of stability) needs to be determined. Experimental data are needed to establish failure modes and limits; however, conducting experiments simulating these conditions is difficult. | CABRI (France)<br>NSRR (Japan) |
| Fuel performance under steady-state conditions | 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). | ATR (United States)<br>BR-2 (Belgium)<br>Hanaro (Korea)<br>HFR (Netherlands) |
| New materials properties (fuel property database) | Improving plant performance in both existing and future plants will likely include improving fuel performance with respect to higher burnup 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. | ANL (United States)<br>BR-2 (Belgium)<br>Hanaro (Korea)<br>HFR (Netherlands)<br>TREAT (United States) |
| Response of defective rods to transient (reactivity insertion accident, steam generator tube ruptures) | Defective rods now have to be taken into account in safety assessments. Experimental data are needed, consistent with the design basis. | CABRI (France)<br>NSRR (Japan)<br>TREAT (United States) |
| Fuel integrity during interim storage | As final disposal of spent fuels delays, interim storages under dry condition and wet condition are expected to increase. Behaviour of fuel under the interim storage conditions needs to be investigated to support the regulatory requirements that ensure no release of radioactive materials from interim storage cask. The expected issues are generation of hydrogen gas by radiation decomposition under wet condition, cladding failure due to thermal creep and hydride reorientation under dry condition, etc. | ANL Hot cells (United States)<br>CEA Hot Cells (France)<br>CNL Hot Cells (Canada)<br>IFE Hot Cells (Norway)<br>JAEA Hot Cells (Japan)<br>Russia Hot Cells (Russia)<br>Studsvik Hot Cells (Sweden) |

### Table 3.3. Fuels and materials reactors worldwide

| Reactor | Type | Thermal power (MW) | Maximum thermal flux (n/cm²/s) E<0.625eV | Maximum fast flux (n/cm²/s) E>0.1MeV | Initial criticality/ design life | Irradiation capabilities/test conditions | Largest thermal flux test volume (n/cm²/s) | Largest fast flux test volume (n/cm²/s) | Fuel irradiation? | Transients? |
|---|---|---|---|---|---|---|---|---|---|---|
| Advanced Test Reactor (ATR)<br>United States | Light water tank | 250 | 1.0E+15 | 5.0E+14<br>2.0E+14<br>(E>1MeV) | 1967/<br>≥2040 | 6 loops<br>0 channels<br>47 in-core positions<br>24/36 reflector/pool positions<br>0 beam ports<br>PWR loops | 13.7 cm dia.<br>122 cm height<br>(1.0E+15) | 13.7 cm dia.<br>122 cm height<br>(5.0E+14) | Yes | Yes, limited |
| High Flux Isotope Reactor (HFIR)<br>United States | Light water tank | 85 | 2.5E+15 | 1.0E+15<br>6.0E+14<br>(E>1MeV) | 1965/<br>≥2050 | 0 loops<br>37 in-core positions<br>42 reflector positions<br>4 beam ports<br>Irradiation temperature up to 1 200°C | 7.2 cm dia.<br>61 cm height<br>(4.3E+14) | 7.2 cm dia.<br>61 cm height<br>(1.3E+13) | Yes | No |
| Massachusetts Institute of Technology Reactor – II (MITR-II)<br>United States | Light water tank with heavy water outer tank | 6 | 7.0E+13 | 1.7E+14 | 1975/<br>≥2050 | 1 loop<br>3 in-core positions<br>9 reflector positions<br>9 beam ports<br>In-core flow loops at PWR or BWR conditions, HTGR materials loop up to 1 600°C, gas-filled static capsule with instrumentation available | 4.57 cm dia.<br>55.9 cm height<br>(3.6E+13) | 4.57 cm dia.<br>55.9 cm height<br>(1.2E+14) | No | No |
| University of Missouri Research Reactor (MURR)<br>United States | Light water tank | 10 | 6.0E+14 | 1.0E+14 | 1966/<br>≥2056 | 0 loops<br>3 in-core positions<br>12/3 reflector/pool positions<br>6 beam ports<br>Static capsules only | 13.6 cm dia.<br>61 cm height<br>(6.0E+14) | 13.6 cm dia.<br>61 cm height<br>(6.0E+13) | No | No |
| US National Bureau of Standards Reactor (NBSR)<br>United States | Heavy water tank | 20 | 4.0E+14 | 2.0E+14 | 1967/<br>≥2065 | 0 loops<br>10 in-core positions<br>7 reflector positions<br>18 beam ports<br>Static capsules only | 8.89 cm dia.<br>73.7 cm height<br>(4.0E+14) | 8.89 cm dia.<br>73.7 cm height<br>(2.0E+14) | No | No |
| Transient Reactor Test Facility (TREAT)<br>United States | Graphite, pulse | 0.08 | 4.0E+11<br>Pulse: 1.0E+17 | | 1959<br>Refurbished 2018 | | | | Yes | Yes |

Note: PWR: pressurised water reactor; BWR: boiling water reactor; GCR: gas-cooled reactor; SFR: sodium fast reactor; VVER: water-water energised reactor.

Source: IAEA Research Reactor Database.

**Table 3.3. Fuels and materials reactors worldwide** *(continued)*

| Reactor | Type | Thermal power (MW) | Maximum thermal flux (n/cm²/s) E<0.625eV | Maximum fast flux (n/cm²/s) E>0.1MeV | Initial criticality/ design life | Irradiation capabilities/test conditions | Largest thermal flux test volume (n/cm²/s) | Largest fast flux test volume (n/cm²/s) | Fuel irradiation? | Transients? |
|---|---|---|---|---|---|---|---|---|---|---|
| RA-10<br>Argentina | Light water | 30 | 3.0E+14 | >3.0E14 | Expected in 2020 | | 8 x 8 cm section with 65 cm length<br>>1.0E+14 | 5 cm dia.<br>12 cm length<br>>3.0E+14 | Yes | ? |
| Jules Horowitz Reactor<br>France | Tank in pool | 100 | 5.5E+14 | 1.0E+15<br>5.5E+14<br>(E>1 MeV) | Expected in 2022<br>(50 years) | 1 corrosion loop<br>10 in-core positions<br>26 reflector positions<br>0 rabbits<br>0 beam ports<br>PWR, BWR, GCR, SFR | 10 cm in-core position<br>20 cm dia.<br>reflector position | 10 cm in-core position<br>20 cm dia.<br>reflector position | Yes | Yes |
| CABRI<br>France | Light water pool | 25<br>Pulse 20 GW | 2.65E+13<br>Pulse: 2.12E+16 | 7.34E+13<br>Pulse: 5.87E+16 | 1963<br>Totally refurbished in 2017 | 1 pressurised water loop<br>(PWR, BWR, VVER)<br>1 channel | | | Yes | Yes |
| Belgium Reactor-2 (BR-2)<br>Belgium | Light water tank | 100 | 1.0E+15 | 7.0E+14 | 1961/ projected life is 2026 or beyond | 1 loop under construction<br>80 in-core channels<br>0 rabbits<br>0 beam port<br>15.7 MPa, 340°C<br>PWR | 90 cm height<br>8.0 cm dia.<br>20 cm dia. | | Yes | Yes |
| High Flux Reactor (HFR)<br>Netherlands | Light water tank | 45 | 2.7E+14 | 5.1E+14<br>2.2E+14<br>(E>1 MeV) | 1961 | 0 loops<br>19 in-core positions<br>12 reflector positions<br>0 rabbits<br>12 beam ports<br>PWR, BWR, GCR | 60 cm height<br>(2.9E+14) | 60 cm height<br>(1.8E+14) | Yes | Yes |
| JOYO<br>Japan | Fast, sodium cooled | 140 | 5.7E+15 | 4.0E+15 | 1977<br>Currently shutdown, restart in 2021 | 0 loops<br>5 channels<br>21 in-core positions<br>1 reflector positions<br>SFR | | 60 cm height<br>Fuel bundle-sized capsules | | |
| BOR-60<br>Russia (RIAR) | Fast breeder, sodium cooled | 60 | 2.0E+14 | 3.7E+15<br>5.0E+14<br>(E>1MeV) | 1968 (operating license until 2020) | 0 loops<br>11 channels<br>15 in-core positions<br>10 reflector positions<br>0 beam ports<br>SFR | | 4.4 cm width, 45 cm height | | |

Note: PWR: pressurised water reactor; BWR: boiling water reactor; GCR: gas-cooled reactor; SFR: sodium fast reactor; VVER: water-water energised reactor.

Source: IAEA Research Reactor Database.

### Table 3.3. Fuels and materials reactors worldwide *(continued)*

| Reactor | Type | Thermal power (MW) | Maximum thermal flux (n/cm²/s) E<0.625eV | Maximum fast flux (n/cm²/s) E>0.1MeV | Initial criticality/ design life | Irradiation capabilities/test conditions | Largest thermal flux test volume (n/cm²/s) | Largest fast flux test volume (n/cm²/s) | Fuel irradiation? | Transients? |
|---|---|---|---|---|---|---|---|---|---|---|
| High Flux Research Reactor (SM-3)<br>Russia (RIAR) | Light water pressure vessel, trap-type | 100 | 5.0E+15 | 2.0E+15<br><br>6.0E+14<br>(E>1MeV) | 1961 | 2 loops<br>1 channel<br>6 in-core positions<br>30 reflector positions<br>0 beam ports | 6.8 cm dia. | 6.8 cm dia. | Yes | No |
| MIR-M1<br>Russia (RIAR) | Light water cooled, beryllium moderated | 100 | 5.0E+14 | 1.0E+14 | 1966/<br>>2020 | 7 loops<br>13 channels<br>11 in-core channels<br>20 MPa, 300°C, up to 27.8 kg/s | 12 cm dia. | 12 cm dia. | Yes | Yes |
| Nuclear Safety Research Reactor (NSRR)<br>Japan | Light water pool | 0.3 | 1.9E+12 | Fluence by pulse: 9.6E+14 (n/cm²) | 1975 | 0 loop<br>1 in-core position | 38 cm height | 38 cm height | Yes | Yes |
| PIK<br>Russia | Light water tank, heavy water reflector | 100 | 4.8E+15 | 8.0E+14 | 2011<br>Full power 2019? | 2 loops<br>17 channels<br>1 in-core channel<br>6 reflector channels | 10 cm dia.<br>4.8E+15 | 4.1 cm dia.<br>8.0E+14 | Yes | ? |
| BIGR<br>Russia | Air cooled | 500<br>(Pulse 70 GW) | | | 1977 | 1 loop<br>2 channels | | | No | Yes |
| Multipurpose Fast Research Reactor (MBIR)<br>Russia | Fast, power, sodium cooled | 150 | | 5.3E15 | BOR-60 replacement -target<br>Commissioning in 2020/<br>50 years | 3 loops (1 sodium loop)<br>14 channels<br>3 in-core channels<br>Inlet: 330-600°C<br>Outlet up to 850°C | 7.2 cm width<br>12.0 cm dia.<br>55 cm height | 7.2 cm width<br>12.0 cm dia.<br>5.0E+15 | Yes | ? |
| LVR-15 REZ<br>Czech Republic | Light water tank | 10 | 1.5E+14 | 3.0E+14 | 1957 | 4 loops<br>16 channels<br>2 in-core channels<br>2 reflector channels | | | Yes | No |
| TRIGA-II<br>Romania | Light water TRIGA dual cores | 14 | 2.6E+14 | 1.8E+14 | 1980 | 1 loop<br>8 channels<br>6 in-core channels<br>2 reflector channels<br>15.5 MPa, 280-300°C | | | Yes | No |

Note: PWR: pressurised water reactor; BWR: boiling water reactor; GCR: gas-cooled reactor; SFR: sodium fast reactor; VVER: water-water energised reactor.

Source : IAEA Research Reactor Database.

## Table 3.3. Fuels and materials reactors worldwide *(continued)*

| Reactor | Type | Thermal power (MW) | Maximum thermal flux (n/cm²/s) E<0.625eV | Maximum fast flux (n/cm²/s) E>0.1MeV | Initial criticality/ design life | Irradiation capabilities/test conditions | Largest thermal flux test volume (n/cm²/s) | Largest fast flux test volume (n/cm²/s) | Fuel irradiation? | Transients? |
|---|---|---|---|---|---|---|---|---|---|---|
| High Flux Advance Neutron Application Reactor (HANARO)<br>Korea | Light water pool | 30 | 4.5E+14 | 2.0E+14 | 1995 | 1 loop<br>39 channels<br>7 in-core channels<br>25 reflector channels<br>10 MPa, 290°C at outlet, 26.5 kg/s (CANDU mode) | 7.4 cm dia.<br>1.95E+14,<br>E>0.82Me | 7.4 cm dia.<br>4.30E+14 | Yes | No |
| China Advanced Research Reactor (CARR)<br>China | Light water tank | 60 | 8.0E+14 | 6.0E+14 | 2010 | >1 loop<br>31 channels<br>4 in-core channels<br>9 beam ports | | | | |
| High Flux Engineering Test Reactor (HFETR)<br>China | Light water tank | 125 | 6.2E+14 | 1.7E+15 | 1979 | 1 loop<br>11 channels<br>7 in-core channels<br>>11 MPa, >300°C, ~8 kg/s | | | | |
| CMMR<br>China | | | 2.4E-14 | 1E-9 | 2014 | | | | Yes | |
| DHRUVA<br>India | Heavy water | 100 | 1.8E+14 | 4.5E+13 | 1985 | 1 loop<br>22 channels<br>5 in-core channels<br>2 MW pressurised water test loop | | | | |
| MARIA POLAND<br>NCBJ, Świerk | Light water | 30 | 3.5E+14 | 1E+14 | 1974 | 6 horizontal channels<br>13 vertical channels | 3 E+14 | | Yes | No |

Note: PWR: pressurised water reactor; BWR: boiling water reactor; GCR: gas-cooled reactor; SFR: sodium fast reactor; VVER: water-water energised reactor.

Source: IAEA Research Reactor Database.

### *3.1.3. Reactor physics*

*Introduction*

Reactor physics issues are important to improve plants' performance through power uprates, higher burnup 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. 3D 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.

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 was 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 the CSNI.

*Scope*

The scope of work in reactor physics covers the current and future needs of nuclear power plants such as PWRs, VVER, BWRs, gas-cooled (thermal) reactors, PHWR reactors, gas-cooled (fast) reactors, liquid metal fast reactors, and molten salt reactors. The latter three are not the focus 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.

This section addresses reactor physics data (cross-sections, neutron spectra, reactivity coefficients, etc.) and facilities to measure data for code assessment. Table 3.4 lists the issues and facilities. Since the last SFEAR report, the PROTEUS, MINERVE and TCA facilities have stopped operation. PROTEUS (Switzerland) was a flexible facility that was capable of representing numerous types of reactors, such as LWRs, PHWRs and pebble bed modular reactors, among others. The shutdown of MINERVE (France) meant that the number of facilities with pile oscillators continued to diminish. Historically, pile oscillator experiments have been used to accurately determine the reactivity worth of small samples to support applications such as burnup credit and minor actinide burning; numerous comparisons had been done between the DIMPLE reactor and MINERVE in the past through the CERES programme. The TCA supported criticality assessments of potential fuel and materials for high-burnup configurations. VENUS was converted from an LWR to an ADS fast reactor facility with lead/bismuth as the coolant, but is returning to a configuration that can support LWR applications.

The International Atomic Energy Agency (IAEA) *Research Reactor Database* lists 238 operational and temporarily shut down reactors as of May 2019. Many of these reactors are not suitable to adequately test the current reactor physics issues identified. Facilities which may have some capabilities include Giacint (Belarus), IPEN/MB-01 (Brazil), Astra (Russia), CROCUS (Switzerland), ZED-2 (Canada), STACY (Japan, temporarily shut down) and RB (Serbia, temporarily shut down). Measurements from these facilities have been used as the basis for validation of criticality and reactor physics codes internationally. The status of the above facilities should be monitored closely.

*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 burnup and for enrichments higher than currently used in LWRs;
- 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 the 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 NSC together with the OECD/NEA Data Bank, in collaboration with member countries and other specialised institutions have developed databases 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);
- isotopic concentrations of spent nuclear fuel (SFCOMPO).

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.

More information on the activities and R&D needs identified by the NSC is provided in NEA (2003). An OECD expert group released a report in 2009 on the research and test facilities required in nuclear science and technology (NEA, 2009b).

**Table 3.4. Current reactor physics issues**

| Issues and relevant reactors | Description | Facility |
|---|---|---|
| 1) $MO_X$ fuel data:<br>PWR, BWR, VVER, ALWR, PHWR | Reactor physics data to support the use of $MO_X$ fuel in current and future reactors is essential to ensure safe operation. Such data include 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 plutonium (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. | VENUS (Belgium)<br>ZED-2 (Canada)<br>CABRI (France)*<br>TREAT (United States)* |
| 2) High-burnup fuel data:<br>PWR, BWR, VVER, ALWR, APHWR | Reactor physics data to support the use of high-burnup fuel in current and future reactors are essential to ensure safe operation. Such data include cross-sections and their uncertainties, delayed neutron generation, power distribution and power, temperature, and void coefficients. | VENUS (Belgium)<br>ZED-2 (Canada) |
| 3) Coolant void coefficient:<br>PHWR, APHWR | Loss-of-coolant accident 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) need to be understood and included in PHWR safety analysis. | VENUS (Belgium)<br>ZED-2 (Canada)<br>KUCA (Japan) |
| 4) Neutron flux and spectra:<br>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 are 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. | VENUS (Belgium)<br>ZED-2 (Canada)<br>LR-0 (Czech Republic) |
| 5) Shielding:<br>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 on predicting shielding performance. | LR-0 (Czech Republic) |
| 6) Moderator coefficients:<br>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. | VENUS (Belgium)<br>ZED-2 (Canada)<br>KUCA (Japan) |

* Use for temperature coefficient studies.

Note: PWR: pressurised water reactor; BWR: boiling water reactor; VVER: water-water energetic reactor; ALWR: advanced light water reactors; PHWR: pressurised heavy water reactors; APHWR: advanced pressurised heavy water reactors.

Source: NEA (2009).

## 3.2. Severe accident and containment phenomena

### *3.2.1. Introduction*

Severe accidents (SA) – often typically understood as (reactor) core melt accidents – are generally considered to be events beyond the traditional design basis of currently operating nuclear power plants. Today (after Fukushima), the terminology "severe accidents" is used in a more general sense and includes fuel melt accidents in the SFP as well. SAs are strongly linked with various phenomena inside the containment (thermal-hydraulics, combustible gas behaviour, fission product behaviour, molten core concrete interaction, etc.), as the containment is the last barrier against fission product release into the environment in a SA. Therefore, this section covers both severe accidents and containment-related phenomena and research issues.

Accident management which deals with the prevention or mitigation of SAs – both in the reactor core as well as in the SFP under all plant operating states – also focuses on maintaining containment integrity, as the largest contributor to reducing risk to the public from the operation of nuclear power plants. Many experimental series supported the development of accident management strategies or specific systems/components. Examples include experiments to understand the quenching of an overheated or partially molten core, experiments to assess the performance of passive autocatalytic recombiners for removal of combustible gases from the containment, experiments to qualify filtered containment venting systems, and assessments of the influence of engineered safety features (spray air coolers) on accident progression. Although generally not considered during initial licensing of most operating nuclear power plants, SAs have been assessed through specific plant reviews, generic analysis and the development of severe accident management programmes. Severe accident management programmes are, to some extent, plant design-specific, and have been significantly improved and extended after Fukushima, based on the latest stress test results and findings of SA research and analyses.

SAs involve an initiating transient, such as a loss-of-coolant accident or a loss of SFP heat removal 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 in the core or the SFP. These failures can result in damage and melting of the fuel, leading to the release of significant amounts of radioactivity from the reactor core into the containment or from the SFP into the surrounding building. In a core melt accident, the containment may also be postulated to fail or to be bypassed (e.g. through steam generator tube failure in a PWR) under certain circumstances, resulting in a major radioactive release to the environment. A fuel melt accident in the SFP may also result in a major radioactive release to the environment, as the SFPs in most nuclear power plants are typically in the "less protected" reactor building.

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." Many research programmes on various severe accident phenomena have been carried out since the severe accidents in Three Mile Island and Chernobyl and contributed to improve the safety of existing reactors. Once the objectives were achieved, interest and financing towards the related activities dropped down. Several important facilities all over the world have been closed down, the most prominent in the severe accident area being the PHEBUS FP facility at the IRSN. It is expected that at closure of the cited activities the question will be answered whether the issue/facility can be closed or what further actions need to be initiated. As a result, new research activities may be launched. The availability of data from those earlier research programmes for future use is an important topic, as is maintaining the expertise of experimentalists/organisation in leading those large experimental programmes.

Several severe accident-related activities are still continuing or are being extended, especially after the Fukushima accident. The CSNI is still playing a major role in organising and administering co-operative research programmes in the area of severe accidents.

Earlier experimental programmes included:[1]

- PHEBUS FP was conducted in France by the IRSN in the framework of EC programmes. These in-pile experiments to study key physical phenomena associated with a severe accident in PWRs specifically encompassed fuel rod degradation, release to and transport of radioactive materials in the primary system and the containment, and their physico-chemical behaviour. Results provide an experimental basis of vital importance to understanding fuel degradation and FP behaviour.
- RASPLAV/MASCA (conducted in Russia to assess the molten corium pool behaviour and thermal load on the RPV lower head).
- SNL-LHF/OLHF (conducted in the United States to assess the mechanical behaviour of the RPV lower head until failure under pressurised severe accident conditions).

1. NEA joint projects: www.oecd-nea.org/jointproj.

- MCCI (conducted in the United States to assess ex-vessel molten core concrete interaction and core debris coolability using the ANL test facility).
- RASPLAV/MASCA (conducted in Russia to measure the physical properties of molten core material within the lower head of the RPV on small-scale facilities).
- SERENA (a programme assessing the state of knowledge related to fuel coolant interactions in- and ex-vessel using the KROTOS [France] and TROI [Korea] facilities).

More recent and/or ongoing experimental programmes include:

- SESAR Thermal-hydraulics (SETH-1) made use of the PANDA (PSI Switzerland) and PKL (Areva, Germany). SETH-2 used the PANDA and MISTRA (CEA, France) facilities and consisted of thermal-hydraulic experiments in support of accident management.
- The HYMERES (Hydrogen Mitigation Experiments for Reactor Safety) Project made use of both PANDA and MISTRA and provided data for detailed analysis of the release and distribution of hydrogen in containment during severe accidents, the effect of activation of containment components (e.g. spray, cooler, heat sources – simulating the thermal effect of PAR), and hydrogen distribution issues related to the thermal stratification in BWR suppression pools.
- THAI (Thermal-hydraulics, Hydrogen, Aerosols and Iodine Project conducted at Becker Technologies, Eschborn [Germany] using the THAI/THAI+ facility, contributed significantly to hydrogen and fission product-related issues in a water cooled reactor containment in accident conditions).
- ISTP (International Source Term Program) by the IRSN in the framework of EC programmes: conducted at the CEA [France] using the VERDON facility to complete the existing database on FP release for high-burnup $UO_2$ and mixed oxide fuel; ruthenium release under oxidant conditions was also investigated, conducted at the IRSN using the CHIP facility to develop chemical models [thermodynamic and kinetic description] of the iodine, caesium, boron, hydrogen and oxygen chemical system to improve predictions of gaseous iodine release fractions at the RCS break.
- BIP (Behaviour of Iodine Project) conducted separate-effect tests at the CNL (Canada) and modelling studies of iodine behaviour in a nuclear reactor containment building following a severe accident.
- The STEM (Source Term Evaluation and Mitigation) project conducted at the IRSN (France) to improve the general evaluation of the fission product source term.
- The SFP (Sandia Fuel Project) used a 17x17 full-length PWR bundle to do experiments on ignition of Zirconium alloy in a prototypical PWR fuel assembly in an SFP during a complete drain-down.

Several of these projects provided data which are used for the development and validation of CFD codes, too.

The CSNI also sponsored efforts through senior expert groups to assess different research aspects and open topics. Examples are the Senior Expert group on Severe Accident Management, which reported SAMI published 2000) (NEA, 2011b).

Within Europe, a Network of Excellence SARNET (Severe Accident Research Network) was organised to co-ordinate European research on severe accidents in nuclear power plants. After 2013, it continued in Technical Area No. 2 of the NUGENIA association. Within SARNET, the SARP activity was related to define SA-related research priorities for future research (Klein-Hessling et al., 2014). An update of the earlier findings was recently made (Manara et al., 2019). The latest activities and projects under the NUGENIA framework are summarised in van Dorsselaere et al. (2017). One of them is the ALISA (Access to Large Infrastructures for Severe Accidents) project (Miassoedov et al., 2018), a four-year project that started in mid-2014 and which was led by the Karlsruhe Institute of Technology. It addressed the transnational access to large research infrastructures for optimal use of the R&D resources in Europe and China in the field of SA analysis for existing and future power plants. To optimise the use of the resources, the project provided access to experimental platforms in Europe to Chinese research institutes and access to Chinese experimental platforms for European research institutes. Activities focused on large-scale experiments under prototypical conditions for SA issues in LWRs, such as coolability of a degraded core, corium coolability in the reactor pressure vessel, possible melt dispersion to the reactor cavity, and hydrogen mixing and combustion in the containment.

After the Fukushima accident, the CSNI organised an SA-related analytical project, BSAF Phase I (NEA, 2015a) and II (Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station Project to improve severe accident codes and to analyse the accident progression and current status of units 1 to 3 of the Fukushima Daiichi). In 2015, the US DOE sponsored several efforts to gain consensus among US experts related to gaps in knowledge about severe accident progression (Farmer et al., 2015) and information needs from forensics examinations of the damaged reactors at Fukushima Daiichi (Rempe et al., 2015). Finally, the CSNI sponsored another so-called Senior Expert Group SAREF (NEA, 2016) (Senior Expert Group on Safety Research Opportunities Post-Fukushima), which addressed activities related to safety research knowledge gaps and data needs from Fukushima Daiichi decommissioning. The report was published in 2016 and contains useful information on past SA research and experimental facilities.

All of these programmes have contributed to increase the knowledge about severe accident phenomena, to the resolution of open questions related to severe accidents, and have identified potential for further improvement of SA management measures to successfully terminate or mitigate the accident progression. They have also served to maintain some of the key facilities from premature shutdown, but not all of them. However, important research issues remain, especially with regard to the late severe accident phase prior to and after RPV failure, to severe accidents in BWR plants in general and with regard to some aspects of fission product behaviour, which need to be studied to support the continued safe operation of nuclear power plants, to reduce the risk of severe accidents through severe accident management programmes, as well as to support the licensing of new nuclear power plant designs.

System and integral codes that simulate main relevant phenomena of SAs have reached maturity over the past two decades. Owing to continuous development of computing capability, the application of the CFD methodology can now be used for safety evaluation, especially in the area of containment thermal-hydraulics/phenomena, less in the SA domain in general.

Thermal-hydraulic behaviour inside the containment is generally 3D, for which analyses using system or integral codes is not always the best solution. However, further efforts are needed for efficient computation and enhancements of capability analysis for two-phase flow, gas distribution, hydrogen combustion and so on. Experimental data with sufficient instrumentation are required for the validation of the CFD codes.

### *3.2.2. Scope*

Based on the most important safety issues and a review of the available experimental data to cover them, the first Senior Expert Group on SFEAR published a list of prioritised severe accident issues (NEA, 2011a). It is worthwhile to review them in order to measure the progress made in the last decade. Information from the Senior Expert Group SAREF (NEA, 2016), SARNET/SARP and follow-up activities (Manara et al., 2019; Klein-Hessling et al., 2014), and the US DOE (Farmer et al., 2015; Rempe et al., 2015) on SA-related research priorities for future SA research have been used to update the severe accident research topics presented in the last SFEAR report to some extent. The issues in this section are listed in Table 3.5 and arranged still according to the main phases of progression of a severe accident in the reactor core and the phenomena present in each of those phases; SAs in SFP are added as a separate topic in addition to the same issue mentioned under the “thermal-hydraulics area” in Section 3.1.1:

- in-vessel phenomena (e.g. core heat-up, reflooding and quenching, clad/fuel melting and relocation, combustible gas generation);
- ex-vessel phenomena (e.g. reactor vessel failure, core concrete interaction, debris cooling, combustible gas phenomena);

- source term/fission product behaviour (e.g. quantity, chemical form, transport and timing of fission product release from the fuel [in- and ex-vessel], behaviour in RCS and containment);
- containment integrity (e.g. capability of containment to withstand severe accident conditions caused by combustible gas burning, decay heat, molten core concrete attack, as well as containment bypass scenarios);
- phenomena of SAs in an SFP (e.g. loss of fuel cooling, fuel heat-up and melting).

### *3.2.3. Description*

The severe accident issues and phenomena that could benefit from additional, ongoing or planned research are related to reducing the remaining uncertainties in SA progression, especially in the late phase prior to and after RPV failure and with regard to BWR, to the improvement of mitigation measures, and to the understanding of the safety implications caused by changes in plant design or operating characteristics (e.g. high-burnup or $MO_X$ fuel). The prevention and mitigation of severe accidents remain an important objective for the continued safe operation of nuclear power plants; the resolution of severe accident issues is tightly coupled to it.

To reduce uncertainties and close knowledge gaps, current research should be conducted at sufficient scale to investigate the important phenomena and use real materials, whenever possible. The Senior Expert Group SAREF (NEA, 2016), SARNET/SARP and follow-up activities (Manara et al., 2019; Klein-Hessling et al., 2014), and US DOE (Farmer et al., 2015; Rempe et al., 2015) on SA-related research priorities concluded that although the early phase of in-vessel core melt progression is fairly well understood, significant uncertainties remain in the later phase, including:

- melt relocation within the RPV, especially for BWR;
- late-phase core degradation, melt behaviour in the lower plenum and hydrogen generation, closely linked to accident management issues of in-vessel melt retention;
- RPV failure mode, especially for BWRs with many/large penetrations, and the question of whether or not molten core material will remain in-vessel after RPV failure;
- consequences of molten core material release from reactor vessel and retention/hold up by external structures below the RPV, especially for BWR;
- melt spreading even under water and melt coolability respectively termination of MCCI, and to some extent combustible gas generation;
- FP release, transport and behaviour in the containment, especially FP retention by pool scrubbing, effects of water chemistry on source term generation, linked to conditions which could lead to containment failure or bypass.

Improvement and determination of the best accident management strategies for preserving RPV and containment integrity and reducing the amount of radioactive material available for release to the atmosphere will continue and benefit from ongoing research.

Severe accident progression in BWRs is generally less well understood due to a lack of experimental data as most of the experiments have been conducted for PWRs. PHWRs have similar SA 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, the potential to overpressurise the calandria and cause expulsion of the moderator through calandria overpressure relief ducts and additional pressure tube rupture. Finally, severe accidents in SFP with air/steam access to the overheating/melting fuel is another open issue.

**Table 3.5. Severe accident and containment-related issues – past and current**

| Issues | Description | Facilities | Notes |
|---|---|---|---|
| **In-vessel phenomena** | | | |
| 1) Pre-core melt conditions | Understanding the conditions that can lead to core melt and the thermal-hydraulic conditions of the core prior to core melt (see Section 3.1.1) is essential to understand whether or not the implementation of accident management strategies will be successful in core melt prevention. Knowledge of pre-core melt thermal-hydraulic conditions will also help to refine accident management strategies so as to understand and be prepared for the outcome of actions taken by the operator. For CANDU-type reactors, the evaluation at calandria vessel subshell, penetrations and end-shield cooling system was an issue. | PHWR CANDU: Fuel Channel Safety Lab (Canada) | Several intermediate-scale studies evaluating critical heat flux in calandria vessel in PHWRs (CNL, Canada) have been done. Experiments can also be done to assess the integrity of fuel channels of PHWRs. |
| 2) Coolability of overheated core/ debris bed and quenching | The effects of adding water to an overheated or partly molten core as well as to a debris bed is investigated in order to understand the thermal-hydraulics phenomena, hydrogen generation and cooling (quenching), and to access whether accident management measures to prevent core melting or to stop melt progression are efficient. The issue of quenching of an overheated fuel bundle is closely coupled with Issue 15 of Section 3.1.1. | QUENCH, FZK (Germany) | Reflood/quenching of an overheated fuel bundle. Plans for operation until 2022. |
| | | PEARL (France) | Dedicated to debris bed reflooding. Its main features are: water injection flow rate up to 50 $m^3$/h, pressure up to 10 bar; initial debris bed temperature up to 900°C; debris bed of 500 kg, heated by induction. Currently used for the French PROGRES programme (up to 2021), and in parallel for the European IVMR programme. Use of PEARL facility in NEA projects may be envisaged in the near future. |
| | | CORDEB (Russia) | CORDEB can obtain material properties using real materials Used in the H2020 IVMR frame of the European project (2015-19). |
| 3) In-vessel melt progression | In-vessel melt progression includes melt relocation in the core and to the lower plenum of the reactor pressure vessel (RPV) and determines the heat load on the RPV wall during a core melt accident. The type of fuel ($UO_2$ or $MO_X$), cladding material, burnup and other factors which affect the composition of the melt are also important in this determination. 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. | CNL melt facilities (Canada) | Cold-crucible experiments for prototypical CANDU corium melts and simulant melt experiments to evaluate convection within the melt. |
| | | LIVE-FZK (Germany) | Uses simulant material; studies the late phase of core degradation, onset of melting, and the formation and stability of melt pools in the RPV lower plenum. Facility still exists, but currently not used. |
| | | PEARL (France) | Description provided under Issue 2. |
| 4) In-vessel fuel coolant interaction (FCI) | Molten fuel relocating within the reactor and 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. | KROTOS (in-vessel FCI) (France)<br>TROI (ex-vessel FCI) (Korea) | FCI tests using prototypic materials. Operation within OECD SERENA projects completed. Facilities still exist. |
| | | MFMI (Canada) | CANDU prototypical experiments with molten corium was constructed to study such extremely violent boiling phenomena and vapour generation during an energetic interaction between molten fuel and water. Facilities still exist. |
| 5) Effect of air on core melt progression | 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. | VERDON (LECA-STAR) (France) | Can conduct hot cell experiments with irradiated fuel. |
| | | CNL Hot Cells (Canada) | Can conduct hot cell experiments with irradiated fuel under various experimental conditions. |

**Table 3.5. Severe accident and containment-related issues – past and current** *(cont'd)*

| Issues | Description | Facilities | Notes |
|---|---|---|---|
| 6) Effect of high-burnup and $MO_X$ fuel | The use of high-burnup or $MO_X$ fuel could change the dynamics of melt progression and fission product release. Data on these effects are needed to properly assess consequences and risk from accident sequences involving high-burnup or $MO_X$ fuel. | VERDON (France) | Can conduct hot cell experiments with irradiated fuel. |
| | | CNL Hot Cells (Canada) | Can conduct hot cell experiments with irradiated fuel under various experimental conditions. |
| 7) RPV pressure | Depressurising the primary coolant system is important during the in-vessel melt progression phase to reduce stress or 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 the RPV and other RCS components' integrity and the subsequent effect on core melt progression. This is primarily an analysis issue. | None | |
| 8) Maintaining RPV integrity by in-vessel melt retention | 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 by external cooling 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. | LIVE-FZK (Germany) | Description provided under Issue 3. |
| | | Calandria Vessel Integrity Experiments (Canada) | Measurement of critical heat flux (CHF) on a scaled calandria vessel with external cooling and internal heating of the shell. Measurement of corium convection in a calandria vessel corium convection uses simulants with crust thickness measurements. |
| | | IVR2D (China) | The 2D IVR facility, REVECT-II at the CNPRI, is designed to investigate the external reactor vessel cooling of the lower plenum of the RPV in order to achieve in-vessel melt retention. Experiments focus on the influence of heat flux profiles either in a homogenous oxide pool or in a two-layer stratified pool in the RPV lower plenum on natural convection and CHF. |
| | | CORDEB (Russia) | Can obtain material properties using real materials. Currently used in the H2020 IVMR project. |
| | | IVR3D (China) | IVR3D test facility at the CNPRI studies 3D vessel external cooling under integrated reactor component mode. The aim is to compare CHF under different cooling channel geometries and two-phase coolant flow patterns. The research programme includes: investigation of two-phase coolant circulation process of natural convection, venting and condensation; and determination of the CHF in 3D geometry. |
| | | COPRA (China) | This facility at XJTU is designed to study the natural convection heat transfer in corium pools with high Rayleigh numbers up to 10^16. The test vessel is a two-dimensional ¼ circular slice with an inner radius of 2.2 m, it simulates full scale for the CNNC's ACP-1000. |

### Table 3.5. Severe accident and containment-related issues – past and current *(cont'd)*

| Issues | Description | Facilities | Notes |
|---|---|---|---|
| 9) Pressure tube integrity | Maintaining the integrity of the pressure tubes in a pressure tube reactor (PHWR) is important for maintaining cooling of the fuel in the tube and preventing overpressurisation and failure of the calandria due to high-pressure water injection and/or molten fuel injection and FCI. | High Temperature Fuel Channel Lab (Canada) | Fuel channel thermal-mechanical behaviour (pressure tube and calandria tube) at high-temperature conditions. |
| **B) Ex-vessel phenomena** | | | |
| 10) Ex-vessel melt progression and debris coolability | The amount, rate, timing and spreading of molten core material released following RPV failure are important to determine the ability of the concrete base-mat to maintain its integrity and the ability of an overlying pool of water or base-mat 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. | MCCI (United States)<br>Primarily MCCI | Large-scale test (1 $m^2$) with real materials, simulated decay heat and with or without overlaying water cooling. |
| | | VULCANO<br>Dry melt spreading | Can use prototypic materials. |
| | | PULiMS (KTH Sweden) | Melt spreading under water. |
| | Support of the development of new mitigation concepts (Gen 2 or Gen 3 reactors) with conditions as close as possible to the reactor case. First test is planned in 2021. | PLINIUS-2 (France) | The CEA built a new large mass prototypic corium experimental platform, PLINIUS-2, to support consolidation of modelling and validation of corium behaviour with prototypical corium. |
| 11) Core concrete interaction | When molten core material leaves the RPV, it can come into contact with concrete of the reactor cavity. Depending on the amount and depth of the material and the composition of the concrete, various amounts of combustible and non-combustible gas will be released into the containment, raising its pressure. These gases can also be a source of additional energy if they ignite, causing additional pressure and temperature rises. If not stopped, the core concrete interaction can also penetrate the reactor containment base-mat, thus failing containment. Understanding the rate and amount of gas generated from core concrete interactions is important to understanding the potential for containment failure and for success of mitigation strategies and, in the case of new plant designs, aid in selecting materials and configurations to minimise core concrete interactions. | MCCI (United States) | Large-scale test (1 $m^2$) with real materials, simulated decay heat and with or without overlaying water cooling. Facility used with OECD project. |
| | | MOCKA (Germany) | Core concrete interaction using simulant corium and decay heat simulation, modelling of concrete with reinforcing bars. |
| | | ARTEMIS (France) | Uses simulant material. |
| 12) Ex-vessel fuel coolant interaction | 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 on the amount, rate, fragmentation and mixing of the molten material, release a large amount of energy that should be taken into account when assessing the structural integrity of containment. | KROTOS (in-vessel FCI)<br>TROI (ex-vessel FCI) | FCI tests using prototypic materials. Operation within OECD SERENA projects completed. Facilities still exist. |

**Table 3.5. Severe accident and containment-related issues – past and current** *(cont'd)*

| Issues | Description | Facilities | Notes |
|---|---|---|---|
| **C) Source term/fission product behaviour** | | | |
| 13) Fission product chemistry and release | 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 $MO_X$) and burnup level. In addition, the effectiveness of source term attenuation measures (e.g. sprays, water chemistry, filters, pool scrubbing) needs to be understood. | VERDON (France) | Fission product behaviour in containment. |
| | | THAI (Germany) | Aerosol and iodine behaviour in containment. |
| | | *ARTIST (Switzerland)* | *The ARTIST Facility at PSI was a unique steam generator configuration: bundle, separator and dryer, used to study steam generator tube rupture accidents and FP retention. Facility is no longer in use.* |
| | | EPICUR (CHROMIA Platform, France) | Chemistry of iodine under irradiation |
| | | START (CHROMIA Platform, France) | Transport of ruthenium in thermo-hydraulic and chemical conditions representative of those of the primary system. |
| | | CHIP (CHROMIA Platform, France) | Transport of iodine in thermo-hydraulic and chemical conditions representative of those of the primary system. |
| | | CNL Hot cell Facility (Canada) | Fission product release from spent fuel, including leaching characteristics. |
| 14) Post-containment failure FP release to the environment | Containment failure can lead to additional FP release due to revolatilisation under depressurised conditions and/or due to air ingress.<br>Understanding these phenomena is necessary for predicting the consequences and risk from accident sequences that fail and depressurise the containment. | | |
| **D) Containment phenomena** | | | |
| 15) Containment integrity/venting | Understanding the conditions (e.g. pressure, temperature and equipment failure) which could lead to containment failure is important. Therefore, knowledge of the integrated effects of design basis and severe accident loads and the use of safety systems is necessary input to containment design. This issue provides input for structural analysis and containment failure modes testing (see Section 3.1.5).<br>Preventing containment overpressure failure by containment venting is one of the AM measures implemented during a severe accident. Since containment might have a significant amount of hydrogen, venting might cause large-scale hydrogen combustion with an unacceptable risk.<br>To reduce such a risk, the AM measure should be based on understanding of thermal-hydraulic behaviours during venting, and effects of geometry of the system and operational conditions. Several containment research facilities exist with different purposes.<br>Containment venting specific issues are not sufficiently established due to lack of related studies. For example, prediction of hydrogen concentration at the intake of the venting system is difficult when hydrogen is non-uniformly distributed in the containment. This is primarily an analysis issue. | PANDA (Switzerland) | Large-scale tests in multi-compartment configurations for decay heat removal from containment; originally designed to study BWR behaviour. |
| | | THAI (Germany) | Two-vessel facility built to study thermal-hydraulics, hydrogen, aerosol and iodine related containment issues. |
| | | MISTRA (France) | 100 $m^3$ PWR containment (0.1 linear scale) using helium as a simulant of $H_2$. Flexible free and compartmented volumes. Capability for steam and gas injection. Spray system, 3D instrumentation. |
| | | TOSQAN (France) | 7 $m^3$ facility to simulate $H_2$ (using helium) mixing under severe accident conditions, including steam, sprays and aerosols. |
| | | LSCF (Large-Scale Containment Facility, Canada) | 1 700 $m^3$ containment thermal-hydraulics facility, use of helium as a simulant. |
| | | CIGMA (Japan) | Facility for investigations of LWR containment thermal-hydraulics, built in 2015 for experiments addressing containment responses, separate effects and accident management. |

**Table 3.5. Severe accident and containment-related issues – past and current** *(cont'd)*

| Issues | Description | Facilities | Notes |
|---|---|---|---|
| 16) Containment bypass: Overheating and failing steam generator tubes | During core heat-up 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 (SGTR) to fail prior to failure of the hot leg, surge line or vessel lower head, or lead to a failure in a connected system outside the containment (ISLOCA). During these conditions, fission products could be released by the containment bypass scenario. Understanding the behaviour of steam generator tubes is important for preventing containment bypass scenarios. Many phenomena are to be studied such as aerosol retention inside/outside tube in steam generator under the wet/dry conditions, iodine retention in pool, etc. | AEOLUS (KAERI) | A scaled-down model of the SG in OPR1000 and APR1400 that can perform experiments of aerosol retentions in the break vicinity. |
| | | *ARTIST (Switzerland)* | *The ARTIST Facility at PSI was a unique SG configuration: bundle, separator and dryer, used to study SGTR accidents and FP retention. Facility is no longer in use.* |
| 17) Combustible gas control | Combustible gas ($H_2$ and CO) are generated from core oxidation during the in- or ex-vessel phase of a severe accident, from metal-water reactions or core-concrete interaction after RPV failure. Hydrogen combustion can further heat the containment atmosphere and/or pressurise the containment, thus challenging containment integrity. Hydrogen countermeasures, especially the use of passive autocatalytic recombiners, is a worldwide used SAM concept. Experiments are needed to show the efficiency of PARS under various conditions. | PANDA (Switzerland) | Large-scale-multi compartment capability for 3D mixing and studying the effect of safety systems. |
| | | THAI (Germany) | Two-vessel facility build to study thermal-hydraulics, hydrogen, aerosol and iodine related containment issues. Many PAR-related experiments have been performed.<br>Instrumentation can measure 3D thermal-hydraulics data. |
| | | MISTRA (France) | Large-scale tests in multi-compartment configuration for mixing and distribution of $H_2$ and studying the effect of safety systems. |
| | | TOSQAN (France) | Small scale, uses helium as a simulant of $H_2$. |
| | | HYMIT (China) | This facility at SJTU is a medium-scale hydrogen mitigation test facility designated for investigations of hydrogen recombination and combustion. It can operate with hydrogen concentrations between 0 and 30 vol.%, and steam concentrations between 0 and 50 vol %. The main part of the facility is a steel cylinder tank 4 metres high and 2 metres in diameter. |
| | | MCTHBT (China) | The MCTHBF is a hydrogen mitigation test facility at the NPIC with a steel cylinder tank 5 metres high and 2.8 metres in diameter. It is designated to test the hydrogen recombination and combustion behaviour. The topics of studies includes scaling from small to large volumes by comparing results of other previous experiments and validation of lumped-parameter and CFD codes. |
| | | CIGMA (Japan) | A facility with 60 $m^3$ containment for thermal-hydraulic experiments related to hydrogen risk and over-temperature containment damage under a broad range of thermal-hydraulic conditions (pressure up to 1.5 MPa, temperature up to 973 K). |

**Table 3.5. Severe accident and containment-related issues – past and current** *(cont'd)*

| Issues | Description | Facilities | Notes |
|---|---|---|---|
| 18) Containment cooling | Several cooling measures are considered in different containment designs as well as in severe accident management strategies to prevent over-temperature containment damage. Those may rely on water spray injections, fan coolers, stagnant water pool covering the containment top outer surface, free convection air flow along the outer surface and so on.<br>Effectiveness of a cooling measure is dependent on integral effects by the cooling on phenomena including gas specimen distributions, condensation, flows discharged from the pressure vessel, heat transfer deterioration due to the presence of non-condensable gas, etc.<br>Interactions with hydrogen risk issues should also be considered because condensation might increase hydrogen concentration. Therefore, knowledge of the integrated effects caused by cooling is relevant to validate the accident management measures to prevent over-temperature damage. | | In principle, the same facilities as listed under Issue 15. |
| **E) Spent fuel pool** | | | |
| 19) Core melting | Understanding the conditions which could lead to severe accidents in the SFP is important. Loss-of-coolant accidents are typically prevented by design of the SFP. Loss of cooling and evaporation of the water in the SFP can cause heat-up of the fuel assemblies dependent on their decay power level. Specific phenomena to be studied are oxidation, melting and fission product releases under air containing atmosphere. As the time available to prevent core melting under loss of cooling conditions are typically long, accident management concepts prefer the use of preventive measures to reinject water from outside. Mitigative measures are less developed. | SFP (Sandia Fuel Project) | Used a 17x17 full-length PWR bundle to do experiments on ignition of Zirconium alloy in a prototypical PWR fuel assembly in an SFP during a complete drain-down; experiments and programme completed. |

Note: PHWR: pressurised heavy water reactor; APHWR: advanced pressurised heavy water reactor; BWR: boiling water reactor; LWR: light water reactor; PWR: pressurised water reactor.

CFD was previously listed as an issue for the severe accident technical area (NEA, 2011a), although the real issue is that facilities in which severe accident research is conducted must have the appropriate instrumentation to evaluate 3D distribution of temperature, hydrogen and steam, and the possibility to measure condensation and gas flows. Thermal-hydraulic behaviour in containment is generally 3D, for which analyses are difficult using a conventional system code such as TRACE and MELCOR. With recent developments in computing capability, the application of the CFD methodology to safety evaluation is possible. However, further efforts are needed for efficient computation and enhancements of analysis capabilities for two-phase flow, hydrogen combustion and so on. For the validation of the CFD code, experimental data with sufficient instrumentation are required. PANDA, MISTRA, THAI and the LSCF can be used to obtain CFD grade data.

## 3.3. Integrity of equipment and structures

### *3.3.1. Introduction*

Many of the current problems with operating reactors are related to material degradation 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 still need to be investigated and solved. Accordingly, ensuring the condition of equipment and structures is monitored and known becomes increasingly important. Ageing mechanisms include stress corrosion cracking, corrosion, erosion, wear, fatigue, thermal and irradiation embrittlement, creep/stress relaxation, and irradiation-induced void swelling and deformation. These mechanisms can affect metallic and concrete equipment and structures as well as electrical and instrumentation and control (I&C) cables. Identifying, monitoring and controlling the ageing mechanisms are important for the continued safe plant operation.

### *3.3.2. Scope*

The issues addressed in this section are related to identifying the phenomena that can potentially cause problems, improving techniques for detecting and repairing problems, and using effective ageing management programmes to anticipate and prevent 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 plant designs, as listed in Table 3.6. The plant ageing mechanisms of wear, creep/stress relaxation, and void swelling have been added to Table 3.6 since the original SFEAR report to provide a more complete list of ageing mechanisms that can affect commercial nuclear power plants.

### *3.3.3. Description*

The timely detection and mitigation of ageing degradation in safety-significant plant systems, structures and components (SSC) are important to ensure SSC integrity and functional operability throughout plant service life. Maintaining SSCs is achieved through effective ageing management programmes. General guidance for such programmes related to nuclear power plant operation, inspection, testing, examination, maintenance and surveillance is provided by the International Atomic Energy Agency (IAEA) Nuclear Safety Standard (NUSS) Code. Specifically, the IAEA has developed the International Generic Aging Lessons Learned (iGALL) programme to support ageing management of safety-significant SSCs and support long-term nuclear power plant operation. The objective of iGALL is to provide a technical basis and practical guidance on managing the ageing of mechanical, electrical and I&C components, and civil structures of nuclear power plants important to safety. The programme is relevant for plants in operation, for plants considering long-term operation, as well as for new plants with both conventional and new designs.

Research programmes will continue to be needed to augment and improve this guidance by assessing how the reactor environment and operating conditions degrade the strength and integrity of equipment and structures over their operational lifetime. More specifically, in the area of integrity of equipment and structures, research programmes should address:

- The initial and potentially degraded state of materials that comprise equipment and structures. Degradation is affected by material composition, manufacturing processes and operational parameters.
- The loads imposed on the equipment and structures during operation (i.e. normal and transient operation, incidents, accidents and events), which are combined with the initial fabrication stresses.
- The historical and planned future environmental conditions associated with the equipment or structure.

- The overall susceptibility of SSCs to ageing degradation, which is determined by the combined material, loading and environment attributes.
- The presence of defects which result from manufacturing practice and environmental attack.
- The safety margins available in the design.
- The sensitivity and effectiveness of pre-service and in-service examination and testing methods.

The safety significance and state of knowledge related to either the plant ageing mechanisms or the application of new materials in existing reactors or advanced designs are ranked in Table 3.6. These rankings were provided in the original SFEAR report and the current rankings are largely consistent with the previous ones. Noted differences are the increases in safety significance on crack initiation and propagation (medium to high), in-service inspection (medium to high) and flow-induced vibration (low to medium). The significance of cracking has been elevated to reflect the greater propensity for cracking as the plant ages as evidenced by the continued discovery of cracking since the original SFEAR report. The importance of in-service inspection has been elevated to recognise its importance in finding flaws and other types of degradation before the degradation affects component functionality. The importance of flow-induced vibration was elevated from low to medium based on the fretting concerns with steam generator tubes and also the emergence of this issue in EPR surge lines. Further, the state of knowledge of erosion/corrosion, thermal and irradiation embrittlement, and materials for high-temperature advanced reactors has increased since the original SFEAR report to reflect work in these areas in the 2000s and 2010s.

Examples of facilities needed to address these issues are also provided in Table 3.6. The research needed to address irradiation-related issues typically requires test reactors to irradiate materials under conditions representative of those found in commercial nuclear power plants. Then the materials are transferred to hot cells with autoclaves and environmental loops or, in some cases, to other testing facilities. Other broad-based (e.g. nuclear and non-nuclear) issues may require large capacity systems to evaluate components at near-full scale, environmental chambers to accurately simulate plant conditions or specialised equipment to characterise the material properties and damage evolution. While all of these facilities are important, the focus of this report is on the test reactors and other unique large-scale experimental facilities needed to address issues specific to commercial nuclear power plants. Therefore, only those unique facilities are indicated in Table 3.6.

Additionally, due to the large number of hot cells and autoclaves and the fact that most member countries have those facilities, specific recommendations on their preservation is outside the scope of this evaluation. However, it is recommended that each member country continue to monitor the status of these facilities and identify any concerns regarding loss of critical infrastructure to the CSNI and its member countries. It should also be noted that the closure risk of major facilities needed to address the issues in Table 3.6 that are not highly safety significant is not of sufficient concern to be assessed in this report.

Since the last version of this report was published, several events have changed the research activities in this area and the facilities that are available to address them. First, the closure of two powerful research reactors, Halden and the NRU, have necessitated OECD actions and national task forces to fill the neutron gap, the expertise in designing in-reactor evaluations of fuel and materials, and facilities to examine irradiated fuels and materials. Second, the Fukushima accident, and numerous national initiatives to safely continue the operation of commercial nuclear reactors beyond their original design lifetimes, have led to increased activities in evaluating fitness for service, and research on the effects of ageing on structural materials and components. Third, numerous new reactor types and fuel types (including ATF) are under development, and which require new materials to realise performance requirements.

**Table 3.6. Current integrity of equipment and structures issues**

| A) Plant ageing issues | | Facilities | Notes |
|---|---|---|---|
| 1) Erosion/corrosion | As plants age, environmental conditions can cause some materials to corrode or erode more than was originally anticipated in the design. The corrosion can be either internal to the component or external due to leakage elsewhere within the plant. Corrosion can lead to cracks, leaks or even large failures. Mitigation of the conditions that lead to excessive degradation needs to be implemented to support continued operation and/or plant life extension. | N/A | Most erosion/corrosion issues are for balance of plant systems and do not require unique, large-scale nuclear-specific facilities.<br>Corrosion issues within the primary system (e.g. feeder tubes) require nuclear-specific facilities. |
| 2) Thermal and irradiation embrittlement | Embrittlement of steels, particularly in the reactor pressure vessel, reactor internals and other primary pressure boundary components, due to either prolonged exposure to relatively high operating temperatures and/or 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 continued confirmation by experimental data. Also, the effectiveness of corrective actions needs to be validated. | ATR<br>HFIR<br>BR-2<br>LVR-15<br>HFR<br>BOR-60 | Facilities listed are among the more prevalent test reactors with capabilities for irradiating materials. More information on these and other similar reactors can be found in Table 3.3.<br>No unique large-scale facilities needed to address thermal embrittlement issues. |
| 3) Crack initiation and propagation | Crack initiation and propagation (i.e. cracking) in steel, PHWR pressure tubes, and concrete equipment and structures has caused problems at operating reactors. Cracking is caused by a combination of stresses, environmental conditions (including irradiation) and material properties and may result from poor design, poor material selection, or an inaccurate knowledge of actual stresses and environmental conditions. Cracking may occur under constant or alternating loads (i.e. fatigue). The causes and corrective action for cracking need additional experimental confirmation. | ATR<br>HFIR<br>BR-2<br>LVR-15<br>HFR<br>BOR-60<br>PNNL | Material test reactors identified specifically here and more generally in Table 3.3 are needed to address irradiation-related issues.<br>The PNNL has developed capabilities for simultaneous corrosion cracking initiation and propagation testing. |
| 4) In-service inspection | Inspection techniques to look for cracks, erosion or the effects of other ageing degradation mechanisms is important to detect and correct ageing effects before they lead to a safety concern. Development and validation of inspection techniques is essential. | EPRI-N.C. (United States)<br>Pressure Tube Surveillance Facilities (Canada) | Performance demonstration initiative facility and extensive mock-ups for technical and personnel qualification.<br>Analysis of hydrogen ingress into pressure tubes by analysis of "scrapes" or removed pressure tubes. Hot cells with accompanying analytical laboratories required. |
| 5) Cable insulation degradation | Cable insulation (both power and I&C cables) can crack and become brittle over time. Environmental conditions such as irradiation, high temperature and moisture affect the rate at which this happens. This degradation can lead to shorts, fires or unexpected behaviour such that the cable cannot perform its intended function. Detection and mitigation techniques need to be verified. | N/A | Numerous CSNI countries have these types of facilities and some have been referenced in NEA (2018a). |
| 6) Long-term behaviour of concrete structures | As plants age, concrete properties change and/or cracks develop. Ageing can be exacerbated by moisture, irradiation, high temperatures and other environmental conditions. The safety implications of concrete ageing need to be understood to support the continued safe long-term operation of existing plants. | ODOBA (IRSN, France)<br>NIST (United States)<br>IETcc and Enresa (Spain)<br>University of Toronto (Canada) | 1 700 $m^2$ platform, 60 large-scale experimental blocks.<br>Large-scale facilities for testing full thickness components.<br>Examination of concrete from Jose Cabrera, a 160 MWe PWR decommissioned in Spain.<br>Facilities used in the OECD programme ASCET. |

**Table 3.6. Current integrity of equipment and structures issues** *(cont'd)*

| A) Plant ageing issues | | Facilities | Notes |
|---|---|---|---|
| 7) Containment integrity | The conditions under which containments fail, and the timing and modes of failure, are important to understand in order to assess safety margins, consequences (i.e. fission product release) and risk. Therefore, the structural analysis methods require experimental confirmation due to the complex nature of containment designs and penetrations. Much of the experimental validation data are dated and may not be sufficient to validate analytical codes attempting to address current safety issues. | VERCORS (France)<br>Various facilities studying related phenomena | A one-third scale mock-up of a reactor containment building in Paris. More than 500 sensors and 2 km of fiber optic cables are positioned in concrete slabs, both on rebar and pre-stressing cables. From the first concreting to the end of the programme, temperature, strain, water content of the concrete will be tested daily. Concrete samples will also be prepared and material behaviours (resistance and modulii, drying, shrinkage, creep and permeability) tested.<br>Facilities addressed in Section 3.1.4 and in conjunction with Item 6, above. |
| 8) Flow-induced vibrations* | As current plants continue to pursue power increases, flow distributions, particularly in-vessel, and their contribution to mechanical loads and vibration of equipment needs to be understood, monitored and evaluated to ensure continued functionality. Predicting such flow distributions and assessing vibratory loading from both mechanical and thermal-hydraulic sources needs experimental data to validate analytical tools. | MSUB facility (Canada) | For steam generator U-bend and tube bank evaluation. Replacement costs are high. |
| 9) Void swelling and deformation | Void swelling is the gradual increase in an equipment's or structure's volume caused by nucleation and growth of clusters of vacancies produced by irradiation. Void swelling may cause dimensional changes that exceed the tolerances of a component and may also induce stresses due to differential swelling. Void swelling susceptibility increases at higher temperatures and levels of irradiation and requires understanding and mitigation for some components that are expected to function beyond 40 years. | ATR<br>HFIR<br>BR-2<br>LVR-15<br>HFR<br>BOR-60 | Material test reactors identified specifically here and more generally in Table 3.3 are needed to address irradiation-related issues. |
| 10) Creep/stress relaxation | Creep (or alternatively stress relaxation) can occur in equipment or structures exposed to elevated temperatures and irradiation. Components such as bolts or pre-compression structures that need to maintain a preload for functionality are potentially susceptible to this ageing mechanism. Creep and stress relaxation may also lead to cracking due to other mechanisms discussed in this table. The complex nature and interrelationship to other degradation mechanism makes the understanding of this ageing mechanism important for certain components. PHWR designs with horizontal pressure tubes are subject to both axial and diametral creep, which has an impact on core flow. | ATR<br>HFIR<br>BR-2<br>LVR-15<br>HFR<br>BOR-60<br>Pressure tube creep research in Canada, see Section 3.1.1. | Material test reactors identified specifically here and more generally in Table 3-3 are needed to address irradiation effects.<br>Effects of creep on core flow and distribution in full-scale CANDU bundles.<br>No unique, large-scale facilities are needed to address thermal creep effects. |

**Table 3.6. Current integrity of equipment and structures issues** *(cont'd)*

| B) Performance improvement/new design issues | | Facilities | Notes |
|---|---|---|---|
| 11) New materials – existing plants and future conventional plants | Both current and future plants will consider using new materials that are intended to be less susceptible to relevant ageing degradation mechanisms to mitigate materials problems (e.g. cracking, corrosion) in both replacement and new components. There will be little service experience associated with the material in the intended application. Therefore, both short- and long-term application-specific performance of these materials needs to be understood and evaluated, and the materials will need to be appropriately qualified for the intended applications. | Facilities previously identified are also used to evaluate specific degradation issues. | The evaluation and qualification of new materials is also an important issue for nuclear fuels. See Section 3.1.2. |
| 12) New materials – ALWR and non-light water reactor (ANLWR) designs: e.g. APHWR, MSR, HTGR, LSR | A variety of advanced reactor designs that are being developed and considered throughout the world plan to operate at significantly higher temperatures than most existing commercial nuclear power plants. These designs will likely utilise both existing and new materials to operate under these conditions. However, the intended operating conditions are still uncertain and significant gaps are associated with the short- and long-term performance of candidate materials in these applications. Creep, creep-fracture and creep-fatigue behaviour are prominent gaps. Many ANLWRs operate in salt or highly corrosive environments such that corrosion control and stress corrosion cracking are important degradation considerations. See Chapter 2 for more information on the related design considerations for proposed ANLWRs. | New irradiation and testing facilities or demonstration reactors will be needed to appropriately simulate the high temperature, irradiation and environmental conditions in order to demonstrate and qualify material performance. | Addressing ageing issues other than those associated with irradiation effects may be partially achievable using existing facilities. Facilities to address ageing due to the aggressive environments being considered in some proposed reactor designs may also be costly. |

* During commissioning of Olkiluoto 3 (EPR), hot functional tests at full temperature and pressure were performed, during which strong vibrations were observed in pressuriser surge line. Analyses that have been done since have not yet revealed with certainty the root cause of the vibrations. The current assumption is that the excitations in the primary circuit match the same frequencies as the characteristic frequencies of the surge line. The excitations can be acoustic, flow-induced or mechanical and combinations thereof. According to preliminary analyses, the vibrations would not threaten the integrity of the surge line even during the design lifetime of 60 years. Furthermore, the break of surge line is included in the design basis of the unit. However, steps have been taken to damp these vibrations to values considered in the design to increase safety margins. This example highlights the importance of fluid structure interaction and needs for capabilities to model these interactions.

Note: PHWR: pressurised heavy water reactor; PWR: pressurised water reactor; ALWR: advanced light water reactor; APHWR: pressurised heavy water reactor; MSR: molten salt reactor; HTGR: high-temperature gas reactor.

## 3.4. Issues and facilities not unique to the nuclear industry

### *3.4.1. Human and organisational factors*

*Introduction*

The importance of human and organisational factors in nuclear reactor safety has been recognised for decades. Due to a series of accidents in the 1980s, it was increasingly accepted that human behaviour could not meaningfully be understood in isolation from the context in which it takes place, and that in complex systems, activities had to be organised in a way that matched human characteristics. Research in this area involves non-engineering disciplines such as social, psychological and organisational topics, and aims at generating insights about the interactions between humans, technology and work organisation, to help promote safety.

*Scope*

Human and organisational factors apply to both currently operating and future plants, and several areas may benefit from further research: staffing, human system interface, external influences on safety, operating experience, management of unanticipated events, management of decommissioning, as well as teamwork, conduct of operations and crew roles.

*Description*

- Currently operating plants

For currently operating plants, operating experience can identify root causes and contributing factors of events, while human reliability analysis identifies human error-related events to help enhance safety performance. From such analyses, solutions and remedial actions can be identified.

- Plant refurbishment

Several operating plants are currently being upgraded to include digital solutions. Early consideration of human and organisational factors in the design process is important in order to contribute to safe human performance in upgraded plants. Consideration of human and organisational factors is also crucial with regard to control room and human system interface verification and integrated system validation.

- New reactor design

For future plants, knowledge is needed regarding opportunities and potential new human-failure mechanisms when digital and modernised control rooms are introduced. Empirical research is central for acquiring knowledge that can be used to guide nuclear safety in future plants.

- Organisational factors

In addition, external organisational factors may influence human performance, and such factors should be considered regarding their impact on reactor safety. Organisations, social systems and ways of operating within organisations are 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 structures and behaviour patterns.

Table 3.7 shows the human and organisational factors safety issues that could benefit from additional research in current and future plants.

**Table 3.7. Current human and organisational factors issues and facilities**

| Issues | Description | Facility name | Capability | Notes |
|---|---|---|---|---|
| 1) Staffing | 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 are 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. 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. | Halden Man-Machine Laboratory (HAMMLAB) (Norway)<br>Human System Simulation Laboratory (HSSL) (United States) | Ability to simulate control room environment and conduct with plant operators. | Halden MTO programme, including HAMMLAB continues beyond 2020. |
| 2) Human-machine interface | 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:<br>– the role of the human vs. automation;<br>– navigation through software controlled displays;<br>– inputting commands.<br>Testing of new human-machine interfaces will be useful. This should include verification and integrated system tests. | Halden Man-Machine Laboratory (HAMMLAB) (Norway)<br>VTT Technical Research Centre of Finland<br>Human System Simulation Laboratory (HSSL) (United States)<br>CANDU Mock-Up Darlington Energy Centre (Canada) | Ability to simulate control room environment and conduct with plant operators. | Halden MTO programme, including HAMMLAB continues beyond 2020. |

**Table 3.7. Current human and organisational factors issues and facilities** *(cont'd)*

| Issues | Description | Facility name | Capability | Notes |
|---|---|---|---|---|
| 3) Organisational influences | 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 employees' 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 a key concern. Objective measures of organisational performance would also be useful. | - | - | - |
| 4) Human performances model | 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. | Halden Man-Machine Laboratory (HAMMLAB) (Norway)<br>Human System Simulation Laboratory (HSSL) (United States) | – | Halden MTO programme, including HAMMLAB continues beyond 2020. |
| 5) Review of operating experience | Many events at operating reactors have as their initiator or as an important element in the event human and/or organisational factors. A review of operating experience to identify and correct those human and organisational contributors is key to maintaining/improving safety. | – | – | – |
| 6) Management of unplanned/unanticipated events | Management of unplanned/unanticipated events are needed throughout the whole accident sequence (PSA levels 1, 2 and 3). Components required are:<br>– decision making under uncertainty;<br>training for the unforeseen, to help establish mental models/schema to support projections, diagnosis and detecting anomalies;<br>– guidance, support tools.<br>Events such as fire or flooding can lead to specific challenges such as main control room abandonment. | HAMMLAB (Norway)<br>Human System Simulation Laboratory (HSSL) (United States) | | |
| 7) Management of dismantling/ decommissioning operations | It would be of value to dedicate an old installation for testing practices, methods, tools, etc. applicable to dismantling, in an international project, in the same way as the experimental control room of the HAMMLAB is used in Halden for testing various HMI. | HBWR (IAEA training centre facility) | New technology such as VR, AR for training for decommissioning | |
| 8) Safety management and safety culture, including in inter-organisational project networks | Assuring and improving the safety culture in complex projects requires a systemic approach for leading safety culture change, based on contemporary views of safety science and management research. System dynamics modelling offered in an interactive simulation game could be used to support decision makers in gaining a holistic perspective on patterns of dynamic interrelations and potential safety effects in projects.<br>How to benchmark standards in safety culture – considering standards are not clearly defined. | VTT Technical Research Centre of Finland | | |
| 9) The role of automation and personnel: New concepts of teamwork in advanced systems | How do we decide when to have shared control between humans and automation and when to assign final authority to humans? | | | |

**Table 3.7. Current human and organisational factors issues and facilities** *(cont'd)*

| Issues | Description | Facility name | Capability | Notes |
|---|---|---|---|---|
| 10) Evolving concepts for the operation of nuclear power plants | What are appropriate models of teamwork for multi-agent systems? | | | |
| 11) Management of operational resilience | To identify the key characteristics of operational resilience and study how these characteristics are manifested in operator behaviour and how they can be operationalised and measured. | VTT Technical Research Centre of Finland | | |
| 12) Nuclear power plant human factor engineering (HFE) design and verification | New types of nuclear power plant control and operation system prototype development, i.e. operation concept research, MCR design, etc.<br>Advanced HSI design, verification and application.<br>Advanced operator supporting system research and development, i.e. computerised procedure system, online monitoring system, etc.<br>Human performance research and analysis.<br>Other HFE research, analysis and application.<br>HFE-related safety review work. | State Power Investment Company (SPIC) Nuclear Power Plant Human Factors Engineering Laboratory (China) | Ability to simulate control room and plant local environment, and conduct with nuclear power plant operators. | Long-time operation, as a company research laboratory. |
| 13) Human factors engineering methods and tools | What are the strengths and limitations of new HFE methods and tools, and what criteria should be applied to evaluating their acceptability for use in human factors R&D. | | | |
| 14) Complexity issues in advanced systems | How to assess and minimise the complexity of a new design. | | | |
| 15) Teamwork, crew roles and conduct of operations | How crews operate and interact in new and upgraded control rooms.<br>Input to conduct of operations:<br>– "out-of-the-loop" challenges with regard to plant automation;<br>– crew roles and teamwork;<br>– crew composition in new plant designs such as small modular reactors;<br>– independent peer check. | HAMMLAB<br>Human System Simulation Laboratory (HSSL)<br>CANDU Mock-Up Darlington Energy Centre | | |

A topic closely tied to human and organisational factors is maintaining expertise on reactor design and operation, including emergency operations. Human actions are needed to assess reactor safety and design and to control nuclear reactor operation including transient (or off-normal, including accident) situations. Urgent action may be required in case of a system crisis. This may require rapid access to detailed information through interactive media. Plant simulators and plant response software for emergency operations centres play a crucial role in this response.

An example of one of several facilities is NUTEMA, whose software has been designed to access comprehensive and systematic storage of knowledge at the basis of the design and safety (including licensing) of nuclear reactors, coupled with access to any piece of information (D'Auria et al., 2011). The reference information is organised into topics which include details such as blue-prints (now computer aided design drawings), 3D views, CFR (US Code of Federal Regulations) statements, experimental data, chemical compositions, thicknesses, notes from installation and maintenance of any system, results of numerical codes application, manuals of those codes, features of turbulence, chemical products of fire, etc.

The NUTEMA hardware consists of several dozen screens arranged in a hierarchic configuration suitable to be handled by one or a team of two to three specialists. The standard views for all the screens, at any time, provide the outlook of a given topic. Each screen is the tip of a "virtual pyramid" ending in more in-depth sections: accessible by the user, one may imagine several virtual screens as axial sections of the pyramid covering the available knowledge and connecting with other screens. Basically, any small "cube-of-information" part can be dug out following one or more coherent paths.

The presence of interactive screens in the NUTEMA facility make possible:

- the training of engineers and technicians, including nuclear power plant operators and analysts, as well as regulators;
- the working mode as a remote crisis centre should an event occur in an assigned reactor.

### *3.4.2. Plant control and monitoring*

*Introduction*

All nuclear power reactors require plant control, and monitoring and protection systems (commonly referred to as I&C systems), and there is 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 functionalities. 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 the 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.

Table 3.8 lists the plant control, monitoring and protection system safety issues that could benefit from additional research. There has been little change to the issues since the last time this document was updated; however, regulations regarding I&C safety have advanced significantly.

*Description*

Complex I&C systems are used to control complex nuclear power plants, and there is an issue with the need to model, or to simulate, I&C performance as part of safety and licensing analyses. Early nuclear power plants had relatively simple I&C and, as far as applicable, each related component was tested to show no impact or safe impact upon reactor transient (or off-normal and including accident) evolution (or performance).

- More and more I&C systems have been added: the complexity prevented comprehensive and systematic testing of interaction with transient reactor performance (here one could recall the recent case of the Boeing 737-800).
- Digital I&C transfers the safety issues related to analogue I&C and shifts the problem to a different region of phase-space (e.g. fast reaction of individual components) concerned by stability and bifurcation analysis, should any transient event occur.
- Proprietary information of I&C prevents the possibility of independent assessments from scientists who are not staff of vendor or designer organisations.

The I&C systems following any postulated initiating event, that is, events not requiring immediate SCRAM and the ATWS, have the potential to bring the operational status of the reactor to a condition that is unpredictable by safety analysts: this may impact the subsequent evolution of the accidents. I&C shall also be considered within the overall context of the defence in depth where the single failure principle is applied. Therefore, both the "postulated initiating event driving away" and the "single failure applications" impose the need to model I&C in safety analyses. As a side issue, avoiding the simulation of I&C prevents the possibility of performing independent assessments, or the fulfillment of another principle in nuclear reactor safety (D'Auria et al., 2012).

The challenges are to ensure that the new systems, primarily digital ones, 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. The challenges are the same for every reactor type, but it should be noted that remote and off-grid reactors (SMR) face additional challenges because of their need for remote monitoring and the implication of remote operating and monitoring with respect to safeguards, security and cybersecurity.

Common position papers have been developed by the Digital Instrumentation and Controls Working Group (DICWG) of the Multinational Design Evaluation Programme to obtain a consistent understanding of issues regarding digital I&C. For example, in 2016, it was agreed that a common position should be adopted regarding hazard identification and control for digital I&C systems given the increase of use of digital I&C in new reactor designs, its safety implications and the need to develop a common understanding from the perspectives of regulatory authorities. This action followed the DICWG's examination of the regulatory requirements of the participating members and of relevant industry standards and IAEA documents.

Several regulators worldwide have used the Multinational Design Evaluation Programme's common position papers as a framework against which to revise current, or inform new, regulation regarding digital I&C. Table 3.8 identifies issues for nuclear power plants and the facilities available to address them.

**Table 3.8. Current plant control and monitoring issues**

| Issues | Description | |
|---|---|---|
| 1) Software quality and reliability | With the increasing use of digital instrumentation and 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. | University of Virginia – Center for Safety Critical Systems (United States) |
| 2) Environmental qualification | The environment in which they operate (e.g. temperature, humidity, radiation, smoke, electro-magnetic/radiofrequency interference, etc.) can affect the performance (reliability, failure rate and failure mode) of digital I&C systems. Several standards developed by the IEEE and the 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. | Sandia National Laboratory – environmental qualification lab (United States) |
| 3) Digital system reliability | 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 the integration of digital system reliability models into current generation probabilistic safety assessments. | University of Virginia – Center for Safety Critical Systems (United States) |

**Table 3.8. Current plant control and monitoring issues** *(cont'd)*

| Issues | Description | |
|---|---|---|
| 4) Wireless communication | The use of wireless communication for monitoring and control in nuclear power plants is expanding rapidly. The technology that supports the current generation of wireless applications was not designed for the challenging environments in nuclear plants that have the potential to disrupt signals. Research is needed to understand the 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. | CNL National Innovation Centre for Cybersecurity (Canada) |
| 5) Online monitoring and advanced instruments | The use of advanced online monitoring and systems diagnostics in nuclear power plant instrumentation and control systems has added a higher level of complexity to the current generation I&C 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, these systems' new and complex failure modes need to be investigated. Facilities where these systems can be tested and reviewed in a prototypical fashion are needed. | Sandia National Laboratory – environmental qualification lab (United States)<br>Ohio State University – INL Academic Center of Excellence in Nuclear Instrumentation and Control and Safety Analysis (United States) |

### *3.4.3. Cybersecurity*

*Introduction*

New nuclear facilities and the modernisation of existing ones have led to a prominent role for digital systems to control and guard the safe operation of nuclear reactors. As a consequence, cyberattacks on these systems can have serious consequences on nuclear safety, on economics, due to down time, and on public perceptions of the safety and security of the nuclear sector. The evolutionary path of digital security mirrors that of digital safety, which, over a 30-year period, transitioned from new standards to regulation. Nuclear cybersecurity is just entering the regulation phase and is creating a demand for accredited and certified products and services. In the early 1980s, when technology moved from analogue to digital components, the transition created a multi-billion dollar industry in the "safety" of software and led to the development of software safety standards and "safety" devices that are now mandatory in any industrial environment where safety functions are performed. Similarly, digital security concerns and evolving standards and regulations are driving the demand for safe and secure solutions by both business and industrial consumers. By 2020, the cybersecurity industry is predicted to reach USD 170 billion.

Unlike IT systems, industrial control systems are often bound by strict regulatory requirements and rigorous change controls that introduce complexities, risks and costs that are substantially higher for any modernisation or modification and therefore requires specialists in I&C systems with cybersecurity expertise.

*Scope*

The risk of cyberattacks has increased significantly over the last decade. Several factors play a role in this increase:

- **Threat actors:** Digital attacks have evolved from hobbyist activities by hackers to profitable criminal activities and even state-sponsored research and development of digital weaponry for both defensive and offensive purposes. In order of competence, the main threat actors are nation states, cybercriminals and terrorist groups, and hacktivists. Unfortunately, all threat actors are relevant for the nuclear sector.
- **Connectivity:** The Internet of Things is growing rapidly, leading to all kinds of devices being accessible via networks and thus becoming potential gateways for network intrusion. Many of these devices have service lifetimes of ten or more years, but limited capabilities to protect themselves against digital threats that will emerge during their service lifetime. I&C systems as used in nuclear reactors are no exception.

- **Dark web:** A black market of hacking software, vulnerabilities unknown to vendors (zero days) and even hackers-for-hire is growing rapidly.

Alas, the cybersecurity risk is real and needs to be managed on several levels:

- **Governance and policy framework:** Nuclear operators must develop a governance model and a set of organisational measures, policies and procedures to guarantee that cybersecurity risks are managed and cybersecurity measures are implemented, maintained and regularly reviewed. As not all rules can be technically enforced, controlling the human factor by clear procedures (and awareness) is paramount.
  - Relevant standards: ISO 27K series, IAEA NSS-17.
- **Network security:** Networks of nuclear operators must be segmented into network zones with decoupling, access control and filtering mechanisms at zone borders making sure that sensitive equipment such as industrial control systems reside behind firewalls with respect to the normal business network to provide a trusted network area with a minimised threat landscape. Network activity crossing zone borders must be monitored.
  - Relevant standards: IEC 62443, IAEA NSS-17, NIST SP 800-82.
- **Asset risk management:** The impact of a successful cyberattack depends strongly on the role that a particular device plays in a nuclear installation. Therefore, the potential impact of losing the confidentiality, integrity and availability of a device should be evaluated in its context. Moreover, the impact of losing integrity and/or availability can increase with the time the device does not play its intended role. Hence, the impact of not having a device in its normal state in function of time together with the time required to bring a rogue device back to a normal state should be evaluated as well. For critical digital I&C systems for which loss of availability or integrity could lead to an unacceptable radiological consequence, a secondary analogue safeguard should be considered.
  - Relevant standards:
    - NIST: SP 800-39 (Corporate Risk Management), FIPS 199/SP 800-60 (Categorization), FIPS 200/SP 800-53 (Security Controls), 800-34/61/82/128/… (Implementation), SP 800-53A (Assessment/Monitoring Controls), SP 800-37 (Authorize/Monitor), SP 800-137 (Monitor Controls).
- **Asset security:** Security feeds provided by vendors must be followed in order to be aware of the potential impact of vulnerabilities that can be exploited by threat actors. Application of security patches should be integrated in the normal maintenance planning taking into account the risk induced by the vulnerability. Physical access is always a threat, but easier to control than logical access. For critical systems which are known to be vulnerable, logical access should be reduced to physical access by making sure that logical access can only be achieved from locations where physical access is strictly controlled and monitored.
  - Relevant standards: ISO/IEC 15408 Common Criteria.
- **Incident response:** Nuclear operators must have a cybersecurity incident response plan describing the actions to be taken in order to contain and mitigate the impact of a cybersecurity incident. In addition, a communication plan should also be developed to report the state of affairs to relevant stakeholders and, if required, communicate with the press.

As prescribed by all standards, a risk-based, graded approach should be applied: more risk implies more security measures to minimise the probability of a threat manifesting itself and/or the impact of the threat when it occurs. Protection of information systems boils down to protecting the three main pillars of information security: **c**onfidentiality, **i**ntegrity and **a**vailability – the so-called **CIA**. The priority of these pillars strongly depends on the environment: in a business network, confidentiality typically tops the list, while in an industrial network, availability is absolute key. Hence, security measures should always be tailored to the specific needs and risks of the equipment and processes present in the environment.

*Description*

There is one encompassing issue in this area, that of cyberthreats to the security of major digital assets. A main shortcoming is cybersecurity certification on the asset level. The ISO/IEC 15408 Common Criteria Standard is a major step in this direction, but has a number of shortcomings in providing the assurance required for nuclear operators. It does provide assurance that the specifications provided by vendors are rigorously implemented and evaluated in a standardised, repeatable manner, but leaves too many openings for vendors to restrict the scope of the evaluation. In particular, a vendor can select specific security attributes to be evaluated and specify (potentially unrealistic) assumptions with regard to the operating environment of their product and the threat landscape to which the product is exposed. Hence, such a certification only provides assurance for the selected attributes and only if the device effectively operates in an environment adhering to the **evaluated configuration**, i.e. the specific circumstances provided by the vendor for evaluation. In addition, the effort and time necessary to prepare all of the necessary documents before, during and after a Common Criteria evaluation makes it a very costly process, which takes so much time that the evaluation criteria and sometimes even the product itself are obsolete before the process is complete.

A number of test beds exist where the resilience of digital assets against cyberthreats can be tested in a controlled environment. The main challenge for such test labs is to configure the testing environment to emulate the real operating environment as closely as possible. Table 3.9 gives a non-exhaustive list of such labs.

**Table 3.9. Cybersecurity research facilities**

| Facility | Description | Reference |
|---|---|---|
| EPIC: Experimental Platform for Internet Contingencies (Joint Research Centre, European Commission, Ispra, Italy) | Test bed for networked critical infrastructure capable of emulating the real operating environment. It uses Emulab to emulate the topology and typical components of the network part of the operating environment such as servers, switches, routers, etc. The physical part of the operating environment is simulated by a generic PC with multi-tasking operating systems running real-time simulations of the physical components. The simulation software itself is built using Matlab Simulink and Matlab Real Time Workshop. At the time of writing, this test bed was only open for organisations collaborating directly with the Joint Research Centre, but they formally stated that this access policy will change in the future. | Christos Siaterlis (2013) |
| Cyber Security Platform (CEA-Leti Grenoble and List Paris-Saclay, France) | Platform staffed by 100+ experts in security of integrated circuits, embedded systems and mobile devices. Uses advanced tools to identify vulnerabilities and develop innovative ways to protect systems against cyberattacks, such as secure communication platforms for sensor networks and cryptographic implementations of hardware and software. Its Information Technology Security Evaluation Facility (ITSEF) [3] is a security evaluation laboratory certified by the ANSSI (French Cyber Security Agency) to conduct CSPN of hardware products (first level of ANSSI) and also accredited to do Common Criteria evaluations up to EAL7 for secure components with embedded software, hardware devices with security boxes, and manufacturing and production sites. In addition, it is also recognised as an evaluation laboratory for other certification bodies like EMVCo, VISA MASTERCARD, NXP MIFARE and BAROC. It is one of the laboratories that offers security tests of electronic components and equipment for industrial developers and also performs security audits of design and production sites for secure products. | www.leti-cea.com/cea-tech/leti/english/Pages/Applied-Research/Facilities/cyber-security-platform.aspx<br>www.leti-cea.fr/cea-tech/leti/english/Pages/Industrial-Innovation/Innovate%20with%20Leti/ITSEF.aspx |
| Cyber Security Training Centre (Thales, Tubize, Belgium) | Cyberlab that according to THALES is capable of reproducing information networks of an organisation to test its resistance against cyberattacks. It offers validation of the security level of information system architectures for its clients. Its test bed was delivered by DIATEAM, a French company that is a pioneer in virtualisation and simulation platforms for cyberdefense (both civil and military). | Thales (2017) |

**Table 3.9. Cybersecurity research facilities** *(cont'd)*

| Facility | Description | Reference |
|---|---|---|
| HNS Platform (DIATEAM, Brest, France) | Hybrid virtualisation and simulation environment allowing to combine real and simulated devices in a controlled environment for cybersecurity tests and benchmarking. It is capable of simulating IT and industrial control systems environments for several purposes related to cybersecurity:<br>– learning and training: simulated attacks in Red Team (attackers) and Blue Team (defenders) setup with realistic network traffic generators;<br>– analysis and prototyping of architectures in design or pre-production phase. | www.hns-platform.com |
| Cyber Security Lab (Sofia Tech Park, Sofia, Bulgaria) | Cyber security research group that, among other information security R&D activities, tests and audits the security of components, systems and even organisations. It is capable of simulating cyberattacks on complex infrastructures and systems. | https://sofiatech.bg/en/activities/laboratories/laboratoriya-po-kibersigurnost/ |
| SCADA Lab (Scada Laboratory, INTECO Data Centre, León, Spain and SCADA test bed, HQ Telvent Energy, Seville, Spain) | Test environment for assessing the security of an industrial control system environment split up into two sites: the laboratory area and the test bed area. The laboratory area hosts the hardware and software to launch a security assessment against the test bed that contains the real-time industrial control systems and physical systems. The testing methodology is based on a good practice guide for assessing the cybersecurity of industrial controls systems of the Centre for the Protection of National Infrastructure of the UK government [8] and financed by the European Commission. In [7], the authors state that operators can request a security assessment and that their framework is capable of providing a "first view" of the security level of critical infrastructures in a detailed, fast and simple way. | Sánchez Aragó, Redondo Martínez and Salán Clares (2014)<br>CPNI (2011),<br>Stouffer *Security Research*, 2014 |
| NIST Cybersecurity Testbed for Industrial Control Systems (Gaithersburg, Maryland, United States) | The main goal of this test bed is to measure the cyber resilience of industrial control systems setup and instrumented with cybersecurity protections according to best practices, in particular the recommendations in IEC 62443 and NIST SP 800-82. It is capable of emulating real-world industrial systems and a variety of industrial scenarios with slow and fast dynamics. Its main purpose is the validation of existing security guidelines and standards, but it also serves as a platform for academic, governmental and commercial research to experiment with security technologies and high assurance designs. | CPNI (2011)<br>R.C.T.Z.K. Stouffer |
| CNL National Innovation Centre for Cybersecurity (New Brunswick, Canada) | Focusing on industrial control systems, this facility contains a distributed control systems testing platform equipped with industrial controllers to simulate real-time process control functions which include a representative suite of field instrumentation such as smart devices and industrial communication network protocols. Technologies to be investigated include: boundary protection devices (e.g. firewalls), network intrusion protection and anomaly detection, endpoint protection, security information and event management tools, application sandboxing, vulnerability scanners, multi-engine virus scanners, network devices (e.g. smart switches, network taps), virtual servers, penetration testing tools.<br>Security operations centres equipped with situational awareness tools and dashboards that provide real-time monitoring, analysis, alerting and reporting capabilities. | CNL (2018) |

Many guidelines and standards exist with regard to managing cybersecurity threats of industrial control systems that are also applicable to the I&C systems of nuclear facilities. IAEA NSS-17 "Computer security at nuclear facilities" provides an interpretation of the two most prominent standards for cybersecurity relevant for nuclear operators specifically for the nuclear sector: ISO 27001 standard for information security management systems and IEC 62443 for network segmentation. Even when nuclear operators follow these recommendations to the letter, they are still for a large part dependent on the cyber resilience of the digital equipment they employ in the complete chain of a nuclear facility: firewalls, routers, switches, servers, human interface machines and software, industrial control systems, and so on. At the moment, ISO/IEC 15408 Common Criteria is the only standardised certification for cybersecurity that is generally accepted, but it leaves too many openings for vendors to restrict the scope of the evaluation to provide the assurance that nuclear operators require. In addition, the Common Criteria certification process is costly and so time-consuming that certification of a particular device often comes once the devices are already operating in the field. Cybersecurity test environments can fill this gap by simulating the setup and components of real-world industrial infrastructures as accurately as possible, allowing to test the behaviour and resilience of these infrastructures against known cyberattacks without the manifestation of the associated real-world risks.

### *3.4.4. External events*

*Introduction*

External events have the potential to impact plant safety by simultaneously affecting many plant systems, structures and components. To ensure that plants are designed for such events, data to confirm the design and safety evaluation methods are essential, depending on experimental facilities and simulations.

*Scope*

There is a wide range of external events, including natural phenomena like earthquakes, flooding and extreme weather phenomena, as well as man-made events. These phenomena are addressed by an equally wide spectrum of experimental facilities.

*Description*

- Seismic events

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. It is expected that future plants (which are likely to be standard designs to be marketed worldwide) 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. Table 3.10 lists the safety issues associated with the seismic behaviour of components and structures that could benefit from additional research.

**Table 3.10. Current seismic effects issues**

| Issues and relevant reactors | Description |
|---|---|
| 1) Confirmation of seismic design | New plant designs are incorporating new safety features (e.g. passive emergency core cooling system, 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 | 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 | As seismic events continue to occur and plants continue to age, data to confirm continued safe operation may be necessary. These data could be in the form of simulating the earthquake and the aged plant structure. |
| 4) Seismic isolation devices | 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. |

Shaking tables are used for investigating seismic effects on buildings. Most of them do not specifically deal with nuclear installations, with the exception of the SHAKESPEARE shaking table in Sint-Genesius-Rode near Brussels, Belgium, which was explicitly set up to address issues of future reactor designs (maximum load: 500 kg). This facility, however, is currently not at risk.

*Other external events*

Investigations of flooding and tsunami scenarios rely on high computer capacity used for modelling. A prominent example is the Tsunami Assessment Modeling System at the Joint Research Centre in Ispra (Italy). In addition, there are experimental facilities for storm surges and tsunamis, which allow addressing phenomena like runup, overland flow and wave-structure interaction, like Oregon State University's multidirectional wave basin.

The impact of projectiles against the walls of nuclear facilities may be caused by strong wind (tornadoes) or by man-made activities like an airplane crash. Damages resulting from tornado-borne debris were studied in the 1970s by Sandia Nuclear Laboratories (United States) using reinforced concrete panels. The Impact Testing Facility at VTT (Finland) is an example of an installation dedicated to investigate experimentally the possible outcomes of airplane crash scenarios, as in the ongoing international Impact project. Such issues are specifically related to containment integrity (see also Section 3.1.5).

In general, facilities investigating the above-mentioned events do not primarily depend on research of the nuclear sector. Initiatives launched by the nuclear sector do not constitute effective means to maintain such facilities. In view of its mandate, they were not the right targets for the SESAR group to focus on, and in line with an efficient approach, they were not further considered as candidates for actions to be taken by the CSNI.

*Further information about facilities related to external events*

- Earthquakes

A long list of shaking tables worldwide including their technical details can be found at: https://ipfs.io/ipfs/QmXoypizjW3WknFiJnKLwHCnL72vedxjQkDDP1mXWo6uco/wiki/Earthquake_shaking_table.html. It includes tables bigger than 2 m x 2 m or with a capacity of more than 4 tonnes.

Experimental facilities for earthquake engineering simulation worldwide are addressed NEA (2004), which lists a smaller number of shaking tables, but also more than 30 reaction walls worldwide.

Experimental facilities in Europe (shaking tables and reaction walls) are listed at: https://sera-ta.eucentre.it.

- Tsunamis and other flooding phenomena

The JRC Tsunami Assessment Modelling System at Ispra (Italy) is described in EC (2007) URI https://publications.jrc.ec.europa.eu/repository/handle/JRC41997.

Information on the multidirectional wave basin as part of the O.H. Hinsdale Wave Research Laboratory of the Oregon State University can be found at: wave.oregonstate.edu.

- Wind-driven projectiles

The test programme on tornado-borne debris by Sandia Nuclear Laboratories (United States) can be found in Stephenson and Sliter (1977).

- Airplane crashes

The VTT Impact Testing Facility (Finland) is described in Heckötter and Vepsä (2015) and Vepsä et al. (2017).

### 3.4.5. Fire assessment

*Introduction*

Fires present a very demanding generic problem to plant safety, which has been demonstrated by some serious incidents in 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 particular, zone models and multi-compartment zone models have shown promising results due to their ease of use and because they are not very time or memory consuming for simulations of compartment fires. In 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 source is known. Remaining problems include determination of fire sources such as the complex solid fuels (geometry, fire properties of solid combustibles, homogeneous/heterogeneous materials, porous/non-porous medium, effect of oxygen depletion for ventilated and confined fire scenarios, 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 have made the technology fairly reliable in general, and also helped to select the most useful numerical codes. The development of computational tools and accumulating experience is gradually enabling fire probabilistic safety assessment (PSA) on the same realistic level as in other branches of PSA.

However, additional experimental data are 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. Table 3.11 shows the fire assessment safety issues that could benefit from additional research.

*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 increased 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 are needed to fix crucial parameters of the modern fire models. The major problem is calculating the fire source, 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 to implement the promising models in CFD codes.

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

While the computing and simulation models tools are, to some degree, able to use already available knowledge and tools, in practice, several parallel development lines are needed: databases from the most safety-relevant fire scenarios; fire properties (ignition time, flame spread velocity, etc.) for the relevant materials; models of automatic fire protection; and quantitative assessment of manual fire protection.

### Table 3.11 Current fire assessment issues

| Issues and relevant reactors | Description | Facilities |
|---|---|---|
| 1) Fire growth and propagation | Accurate modelling of fire growth and propagation is 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, efforts should be made to simulate most safety-relevant typical scenarios as benchmarks, and make the results available in databases for plant-specific work. For example, it's important to study complex fuel such as electrical cabinets, electrical cables and gloveboxes.<br>Efforts should be made to utilise the emerging technology of flame spread modelling on solids. Actions to test various proposed models at all relevant scales, and implement the promising models in the best CFD codes are needed. Special efforts are needed to select the most suitable testing methods from existing or new concepts, which are necessary to determine flame spread parameters for practical commercial products. For example, there are no available methods for cables and glovebox. Benchmarking efforts are needed to transfer the technology from laboratories to industrial practice. For complex fuel, it is important to combine both analytical and full-scale R&D to study pyrolysis of each constituent and the behaviour of the equipment with all of its constituents in the full-scale conditions of a fire. | DIVA (France) GALAXIE platform: 5 rooms between 120 $m^3$ and 160 $m^3$, connected by a ventilation system network to investigate for different type of fires (oil, electrical cabinets, electrical cables, glovebox), heat and smoke propagation from room to room. Able to measure soot deposits or radioactive surrogates in outlet duct of ventilation network (to 2021).<br>PLUTON (France), GALAXIE platform: 1 room of 400 $m^3$ allowing confined and ventilated fire tests using different types of fuel. Ventilation up to 50 000 $m^3$/h for fire power up to 5 MW. Soot and radioactive surrogate measurement (to 2023).<br>US National Institute of Standards and Technology (NIST): Issues covered include fire growth and propagation (indefinite). |
| 2) Hot shorts | Fires in cable trays 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 are needed. Some data are already available for simple basic scenarios. Theoretical modelling is needed for assessing effects on systems performance. | Omega Point (United States)<br>DIVA, PLUTON |
| 3) Smoke propagation | The spread of smoke during a fire can have strong impacts. Smoke can affect the operability of certain equipment and inhibit firefighting efforts by limiting access and visibility. This is not unique to the nuclear industry; however, the specificity of the confined and ventilated environment encountered in nuclear power plants requires experimental data to increase knowledge, improve associated models and validation codes. | |
| 4) Equipment vulnerability | Understanding when and how equipment fails under fire conditions is essential to evaluate the fire consequences and the consecutive risk assessment on the nuclear facility. This includes failures from heat, smoke, suppression system activation, shorts, etc. Experimental data will likely be needed to address this issue.<br>Some data and basic calculation models are available on heat and smoke effects for some electrical and electronic equipment. Establishing a databank with benchmarking examples would be a good way to educate utilities to use that information. Establishing these databanks is mandatory for Monte Carlo analyses. | DANAIDES (France) GALAXIE platform test bench for testing electrical equipment operating under severe temperature and soot conditions similar to real compartment fire (gas temperature from ambient to 300°C and from 0 to 5g/$m^3$ for soot concentration) (to 2023).<br>CISCCO (France) GALAXIE: Experimental device for testing cable fire in various configurations (nature of cables, cable arrangement on the tray, distance between trays, etc.) (to 2025).<br>SIMBAG (France) GALAXIE test bench for testing all the combustion regimes, in particular air and thermal conditions dependent on various parameters evolving in time (geometrical evolutions of the glovebox and the ventilation), exchanges of flow of heat, relocation of the combustible materials and their pyrolysis). |
| 5) High-energy arcing faults | Fires caused by arcing from high-energy lines need to be modelled and included in risk assessments. | N/A |

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# 4. Conclusions and recommendations

## 4.1. Introduction

This chapter provides the conclusions and recommendations of the Expert Group on Support Facilities for Existing and Advanced Reactors (SFEAR2) regarding key facilities unique to nuclear safety research in danger of being lost in the short term (in the next one to three years) and those that should be monitored by the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) in the longer term (more than three years) to ensure a minimum facility infrastructure to support operating water cooled reactors, including advanced fuel concept developments in member countries. Unique facilities for new reactor designs are identified in a cursory manner in Chapter 2. This chapter also provides some general conclusions and recommendations (independent of short or long term) for the CSNI's consideration.

The overall strategy employed in developing recommendations was the same as followed in the past; namely, to identify minimum research infrastructure needs and the facilities that should be maintained to ensure that this infrastructure is available. As in previous efforts, the group took an approach that focused on those facilities with unique capabilities and that 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.

Specifically, the following factors were 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 an identified safety issue, within a given technical area;
- 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 costs and identified as unique were considered as candidates for CSNI action. The conclusions and recommendations summarised below are organised by the five technical areas discussed in Section 3.1.

## 4.2. Short-term conclusions and recommendations

To assess short-term concerns, the facilities in Chapter 3 were examined, and those in danger of being shut down in the next one to three years were identified. These facilities are shown in Table 4.1 by technical area, along with the issues and reactors they support. It is instructive to compare the facilities currently on the list with those identified in the last SESAR report as being either in short-term danger or warranting monitoring in the long term. These are also displayed in Table 4.1. Facilities that were identified in 2007 as being at risk in the short term appear in italics. Those that are now closed are highlighted in grey. A third category is facilities that were previously identified as requiring monitoring in the long term, and are now not identified to be at risk by member countries, either because more industrial funding has been obtained or because new facilities have obviated the need. These are discussed in earlier chapters. Actions being taken to preserve the capability of these facilities are also identified. Finally, the GALAXIE facility in France appears on the list of facilities at short-term risk, for the first time.

**Table 4.1. Facilities in danger in the short term (2019-21) and facilities identified in 2007 as being at risk**

| Facility | Applicable reactors | Issues addressed | Notes |
|---|---|---|---|
| **Thermal-hydraulics** | | | |
| *PANDA (Switzerland)* | *PWR, BWR, ALWR* | *Passive safety system performance.*<br>*Two-phase natural circulation.*<br>*Thermal stratification.*<br>*Thermal cycling.*<br>*Accidents initiated during shutdown.* | *Identified as being at short-term risk in 2001 but still operating as the result of NEA projects. A new proposal will be planned before the end of the current HYMERES project.* |
| *PHEBUS (France)* | *PWR, BWR, VVER, ALWR* | *Response to loss-of-coolant accidents.* | *Closed.* |
| *PKL (Germany)* | *PWR* | *Boron dilution.*<br>*Passive safety system performance.*<br>*Steam generator tube ruptures (SGTR).*<br>*Two-phase – natural circulation.*<br>*Thermal stratification.*<br>*Accidents initiated during shutdown.* | *Previously identified as a critical facility to monitor in the long term. Discussions for a next phase of the NEA PKL project have started with the aim of continuing soon after the end of current phase (September 2020).* |
| LSTF/ROSA (Japan) | PWR | Boron dilution.<br>Passive safety system performance.<br>SGTR.<br>Two-phase – natural circulation.<br>Thermal stratification.<br>Accidents initiated during shutdown. | Previously identified as a critical facility to monitor in the long term. Future joint projects will be discussed with the other organisations operating an integral test facility. |
| **Fuel** | | | |
| Halden (Norway) | All | | Previously identified as a critical facility to monitor in the long term.<br>Closed. |
| PHEBUS (France) | All | | Previously identified as a critical facility to monitor in the long term.<br>Closed. |
| NRU (Canada) | PHWR | | Previously identified as a critical facility to monitor in the long term.<br>Closed. |
| **Severe accidents and containment** | | | |
| CTF (Canada) | All | Combustible gas control. | Closed. |
| LSVCTF (Canada) | All | Combustible gas control. | Closed. |
| ARTIST (Switzerland) | PWR, PHWR, APHWR, ALWR | FP chemistry and release. | Closed. |

**Table 4.1. Facilities in danger in the short term (2019-21) and facilities identified in 2007 as being at risk** *(cont'd)*

| Facility | Applicable reactors | Issues addressed | Notes |
|---|---|---|---|
| PHEBUS (France) | BWR, PWR, VVER, ALWR, PHWR, APHWR | Pre-core melt conditions.<br>In-vessel melt progression.<br>Effect of air on core melt progression.<br>Effect of HB and $MO_X$ fuel.<br>Fission product chemistry and release.<br>Coolability of overheated cores. | Previously identified as at risk in the short term.<br>Closed. |
| THAI (Becker Technologies, Germany) | All | Combustible gas control.<br>Containment integrity. | Previously identified as a critical facility to monitor in the long term. A new NEA project, THEMIS, has been launched. |
| PANDA (Switzerland) | BWR, ALWR | Combustible gas control.<br>Containment integrity. | Previously identified as a short-term risk. Still operating as the result of NEA projects. A proposal for a new joint project will be planned before the end of the current HYMERES project (2021). |
| QUENCH (Germany) | PWR, BWR, VVER, ALWR | Pre-core melt conditions.<br>In-vessel melt progression.<br>Coolability of overheated cores. | Facility previously identified as a short-term risk. A new NEA project (QUENCH ATF) has been launched. |
| COMET-FZK (Germany) | PWR, BWR, VVER, ALWR | Core concrete interaction. | Closed. |
| VERDON (CEA, France) | All | Fission product release and transport at high temperature. | Previously identified as a facility requiring monitoring in the long term. A new NEA project (ESTER) has been launched. |
| CHROMIA (IRSN, France) | All | Fission product release in primary circuit and containment, EPICUR, CHIP and start are components. | Previously identified as requiring monitoring in the long term. A new NEA project (ESTER) has been launched. |
| **Integrity of equipment** | | | |
| Halden (Norway) | All | | Closed. |
| JMTR (Japan) | All | | Closed. |
| NRU (Canada) | PHWR | | Closed. |
| Halden (Norway) | All | | Closed. |
| **Others** | | | |
| GALAXIE Platform (IRSN, France) | All | Fire and soot propagation through ventilation, from cable trays, in gloveboxes. A number of large, medium and separate effects scale facilities co-located. Unique in ability to examine cable, glovebox fires and ventilation fires of particular concern to the nuclear industry. | Not previously identified as being at risk. A new NEA project (PRISME follow-up) is being contemplated. |

Notes: PWR: pressurised water reactor; BWR: boiling water reactor; VVER: water-water energetic reactor; ALWR: advanced light water reactor; PHWR: pressurised heavy water reactor; APHWR: advanced pressurised heavy water reactor. Gray shading includes those identified in 2007 as being at risk and that are now closed. Italics identify those previously identified as being at short-term risk, or requiring long-term monitoring.

In the thermal-hydraulics area, three facilities are in short-term danger. These include PKL and the Large Scale Test Facility (LSTF, Japan), which support pressurised water reactor (PWR) thermal-hydraulic work. PANDA, which supports both light water reactor (LWR) and boiling water reactor (BWR), as well as containment thermal-hydraulics. Both PANDA (PSI Switzerland) and PKL (Areva, Germany) proponents are anticipated to propose new NEA projects prior to the current projects coming to an end (2021 for PANDA and 2020 for PKL respectively). The situation for the LSTF (Japan) is more unclear, pending a decision on whether Japan will continue to support it.

RD-14M, a full-height pressurised heavy water reactor (PHWR) thermal-hydraulics facility was decommissioned in 2019, in consultation with both the Canadian industry and the regulator who identified no outstanding issues that warranted maintaining the infrastructure. A similar scaled facility (but lower power) has been constructed in India, but the status of its operation is unknown.

Numerous facilities have been closed in the containment and severe accident area since the last evaluation was published. These include PHEBUS, discussed in the fuels and materials section, combustion test facilities in Canada, the ARTIST aerosol retention facility (Switzerland) and the COMET facility in FZK Germany. The last report identified the QUENCH facility as being at risk and concluded that the long-term prognosis would depend on identifying a future experimental programme that could provide useful information beyond what had already been done in QUENCH. The facility is scheduled to operate until 2022, and a new NEA joint project on accident-tolerant fuel is being contemplated, but this facility will remain in short-term risk of closure unless the project is implemented.

The aforementioned PANDA facility, the VERDON facility and the THAI facility have been identified as being at risk in the near term. They had all been previously identified as being at risk, or critical for monitoring in the long term in the last report. There are NEA joint projects planned for each. The CHROMIA platform at the IRSN is also at risk. It has been supported extensively by NEA projects over the past 12 years (STEM) and a new joint project proposal, ESTER, is being prepared which also utilises the VERDON facility.

In the fuel and materials areas, several large, versatile reactors have been closed. However, regardless of having been previously identified as being short-term risks (PHEBUS) or facilities to monitor in the long term, substantial national support is required to sustain/refurbish ageing or specialised reactors. The National Research Universal Reactor in Canada and the Halden Reactor in IFE Norway both closed in 2018, and this represents a significant loss worldwide, both for materials and fuel testing. Additionally, the Japan Material Testing Reactor (JMTR) in Japan, a test reactor used primarily for materials testing, has been shut down as well. Reactors worldwide, including CABRI, BR-2, LVR-15, MIR, TREAT, HFR, ATR and others have been identified as being suitable to replace some of the capabilities lost by the closure of these reactors.

A review of sub-critical and zero-power reactors reported on in Section 3.1.3 has found that several of the reactors identified in the last report for monitoring in the long term have been closed. The Nuclear Science Committee has recommended that the remainder of the most versatile (VENUS, ZED-2, LR-0 and KUCA) be monitored closely. The STACEY facility (Japan) was mentioned as a candidate to replace some capacity, but it is currently shut down. The CROCUS facility in Switzerland also may have some capacity. Neither of these have been identified as being at risk, however.

Finally, although the scope of the previous 2007 activity excluded facilities that were not unique to the nuclear industry, the GALAXIE platform at the IRSN was identified in the current evaluation as being at risk in the near term. The collection of co-located separate effects, intermediate and large-scale facilities in this platform, as well as experience gained from three phases of PRISME projects make it ideal for addressing fire safety issues unique to the nuclear industry (e.g. glovebox fires, propagation through ventilation systems, fire initiated in cable trays). An NEA joint project is planned to be proposed once the current PRISME project ends in 2021.

Despite the large number of facilities that have closed since 2007, there are several which were previously on the list as being at short-term risk, but which are no longer on the list. At the same time, new facilities have been built that can be used to address the issues formerly requiring the closed facilities.

As an example of facilities which are no longer at risk, in the previous report, both the APEX (PWR, Oregon State University) and PUMA (BWR, Purdue University) thermal-hydraulic facilities were identified as being at risk in the short term, but are both currently operating, and were not identified as being at risk for the current assessment. This is presumably the result of strong industry support over the last several years. Also identified at risk were the VULCANO facility at KTH, which was not identified in the current evaluation as being at risk because of its continued use by national and EU projects.

An example of facilities that have been closed, but whose research purpose can be at least partially fulfilled by other facilities is the ARTIST integral facility in PSI, a scaled-down model of the FRAMATOME 33/19 SG. Although ARTIST has been closed, some of the more generic aerosol retention questions can be addressed by the AEOLUS integral facility in KAERI, a scaled-down model of the SG in OPR1000 & APR1400, which is currently performing aerosol retention studies in the break vicinity.

Some of the capability of the hydrogen combustion facilities in Canada, although unique in their scale and versatility, can be fulfilled by the THAI facility operated by Becker Technologies in Germany (PAR behaviour, hydrogen stratification and mixing). This highlights the importance of the THAI facility being maintained in the future. Nonetheless, issues such as deflagration to detonation if required by CSNI member countries, will require additional facilities to investigate. It is proposed that some of the large facilities identified by the EC Network of Excellence for Hydrogen Safety “HySafe” be assessed, should the need arise (HySafe, 2021).

Despite the large number of versatile reactors that have been shut down since 2007 when the last report was written, collaborative irradiation programmes, refurbished reactors and new builds are, or will be, available to close some of the gaps. The Jules Horowitz Reactor in France is planned to start by the mid-2020s and the MYRRHA reactor in Belgium in the mid-2030s. In addition, the pulse reactor TREAT in the United States has been successfully restarted and the US Department of Energy has launched its Versatile Fast Neutron Source Project to provide fast neutron testing capability to aid US development of advanced nuclear reactor technology. The Versatile Test Reactor, as it is also known, could be completed by 2026. Finally, the NEA has recently launched the FIDES network, with associated irradiation projects (JEEPs), which will allow member countries to access various reactors and test programmes for materials and fuels. This step towards collaborative irradiation projects is expected to somewhat mitigate the irradiation gap both in the short and long term.

## 4.3. Longer-term conclusions and recommendations

Many of the factors used in the last two reports to arrive at conclusions and recommendations have resulted in effective measures for retaining key facilities at risk. These measures should continue to be used in the future, with consideration of the factors below:

- Cost of facility operation and replacement (i.e. limit CSNI involvement to large facilities needing multinational 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).
- 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 the CSNI consider each factor when developing a strategy for facility preservation and in assessing and initiating future co-operative research programmes.

Building on previous SESAR reports 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 in Table 4.2 for various reactor types. For reference, facilities previously identified but no longer available are shown in grey shading and italics; those that have been identified for the first time are in bold. The facilities listed are those considered unique, hard to replace and identified as playing a significant role in resolving issues in their technical area, as discussed in Chapter 3. Accordingly, it is recommended that the CSNI continue to focus on these facilities in developing a strategy for long-term infrastructure preservation. The CSNI should monitor the status of these facilities 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 the new reactors and technologies identified in Chapter 2, 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.

In the thermal-hydraulics area, over the long term (beyond 2022), the Rod Bundle Heat transfer facility will require monitoring. This facility is unique, costly to maintain and has a high replacement cost. A unique situation exists in Korea, where R&D on pressurised heavy water reactor (PHWR) topics is on the decline, while advanced reactor design is continuing. This has prompted the inclusion of the Moderator Cooling Test facility as a thermal-hydraulic facility requiring monitoring in the long term.

In the containment and severe accident area, the TROI facility, which can be used to study molten corium interactions with water, has been previously utilised for NEA joint projects but had not been used for a few years in 2021. It is one of only a few facilities available that can use large amounts of real corium (20 kg), and should therefore be monitored in the long term as well, despite the fact that it is not in imminent danger of being decommissioned. The MCCI facility at Argonne National Laboratory is well advanced in planning new NEA joint projects, and if these are launched successfully, they will sustain the facilities for another few years. However, there are no plans past this project.

**Table 4.2. Critical facilities to be monitored in the long term**

| Technical area | BWR | PWR | VVER | PHWR/APHWR | ALWR |
|---|---|---|---|---|---|
| Thermal-hydraulics | PANDA[1] (Switzerland) | LSTF/ROSA (Japan)<br>PKL (Germany)<br>ATLAS (Korea)<br>RBHT (United States) | PSB-VVER (Russia)<br>PACTEL (Finland) | *RD-14-M (Canada)*<br>MCT (Korea) | LSTF/ROSA (Japan)<br>PKL (Germany)<br>PANDA[1] (Switzerland)<br>ATLAS |
| Fuels | *Halden (Norway)*<br>NSRR (Japan)<br>*PHEBUS (France)*<br>TREAT (United States)<br>BR-2 (Belgium) | *Halden (Norway)*<br>NSRR (Japan)<br>CABRI (France)<br>*PHEBUS (France)*<br>TREAT (United States)<br>BR-2 (Belgium) | *Halden (Norway)*<br>MIR (Russia)<br>CABRI (France)<br>*PHEBUS (France)*<br>TREAT (United States)<br>BR-2 (Belgium) | *Halden (Norway)*<br>*NRU (Canada)*<br>TREAT (United States)<br>BR-2 (Belgium) | *Halden (Norway)*<br>NSRR (Japan)<br>CABRI (France)<br>*PHEBUS (France)*<br>TREAT (United States)<br>BR-2 (Belgium) |
| Reactor physics | *PROTEUS*<br>Venus (Belgium) | *PROTEUS*<br>Venus (Belgium) | *PROTEUS*<br>LR-0 (Czech Republic) | *PROTEUS*<br>ZED-2 (Canada)<br>Venus (Belgium)<br>KUCA (Japan) | *PROTEUS*<br>Venus (Belgium) |

**Table 4.2. Critical facilities to be monitored in the long term** *(cont'd)*

| Technical area | BWR | PWR | VVER | PHWR/APHWR | ALWR |
|---|---|---|---|---|---|
| Severe accidents and containment | *Integral testing* | | | | |
| | *PHEBUS (France)* | *PHEBUS (France)* | *PHEBUS (France)* | *PHEBUS (France)* | *PHEBUS (France)* |
| | In-vessel phenomena | | | | |
| | QUENCH[1] (Germany)<br>VERDON (France)<br>KROTOS (France) | QUENCH[1] (Germany)<br>VERDON (France)<br>KROTOS (France) | QUENCH[1] (Germany)<br>VERDON (France)<br>KROTOS (France) | Fuel Channel Safety Facility (Canada)<br>MFMI (Canada)<br>VERDON (France)<br>KROTOS (France) | QUENCH[1] (Germany)<br>VERDON (France)<br>KROTOS (France) |
| | *Ex-vessel phenomena* | | | | |
| | MCCI (United States)<br>VULCANO (France)<br>THAI[1] (Germany)<br>KROTOS (France)<br>TROI (Korea) | MCCI (United States)<br>VULCANO (France)<br>THAI[1] (Germany)<br>KROTOS (France)<br>TROI (Korea) | MCCI (United States)<br>VULCANO (France)<br>THAI[1] (Germany)<br>KROTOS (France)<br>TROI (Korea) | MCCI (United States)<br>VULCANO (France)<br>THAI[1] (Germany)<br>KROTOS (France)<br>TROI (Korea) | MCCI (United States)<br>VULCANO (France)<br>THAI[1] (Germany)<br>KROTOS (France)<br>TROI (Korea) |
| | *Containment mixing/combustion* | | | | |
| | PANDA[1] (Switzerland)<br>LSCF (Canada)<br>THAI[1] (Germany)<br>MISTRA (France) | PANDA[1] (Switzerland)<br>LSCF (Canada)<br>THAI[1] (Germany)<br>MISTRA (France) | PANDA[1] (Switzerland)<br>LSCF (Canada)<br>THAI[1] (Germany)<br>MISTRA (France) | PANDA[1] (Switzerland)<br>LSCF (Canada)<br>THAI[1] (Germany)<br>MISTRA (France) | PANDA[1] (Switzerland)<br>LSCF (Canada)<br>THAI[1] (Germany)<br>MISTRA (France) |
| | *Accident management*<br>Uses data generated in resolution of other issues. No unique facility needs. | | | | |
| Integrity of equipment and structures | *Halden (Norway)*<br>*JMTR (Japan)*<br>LVR-15 (Czech Republic)<br>ATR (United States) | *Halden (Norway)*<br>*JMTR (Japan)*<br>LVR-15 (Czech Republic)<br>ATR (United States) | *Halden (Norway)*<br>*JMTR (Japan)*<br>LVR-15 (Czech Republic)<br>ATR (United States) | *Halden (Norway)*<br>*NRU (Canada)*<br>LVR-15 (Czech Republic)<br>ATR (United States) | *Halden (Norway)*<br>*JMTR (Japan)*<br>LVR-15 (Czech Republic)<br>ATR (United States) |

1. Assumes actions will be taken in the short term to preserve these facilities.

Note: BWR: boiling water reactor; PWR: pressurised water reactor; VVER: water-water energetic reactor; PHWR: pressurised heavy water reactor; APHWR: advanced pressurised heavy water reactor, ALWR: advanced light water reactor.

## 4.4. 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 current activity and desire to develop a practical set of recommendations with facility preservation being a co-ordinated effort among the NEA standing committees. Specific general conclusions and recommendations are listed below. It is worthwhile to note that many of the conclusions reached from the previous SESAR/SFEAR activities are still valid, and are included in the list below.

- **Recommendation**: CSNI members are encouraged to continue their excellent support for facilities at risk, which has already resulted in several valuable projects for current facilities at risk or to be monitored in the long term (e.g. PKL, PANDA, GALAXIE).
- **Recommendation**: As recommended in the previous report, test reactor availability should be given special scrutiny, due to the high cost of operation and replacement. The new FIDES network and associated JEEPs is an essential step in maintaining global capability.
- **Recommendation**: Regardless of FIDES and the success of JEEP, continued and ongoing attention must be focused on smaller unique facilities at risk.

- **Recommendation**: The Nuclear Science Committee should maintain a close watch on facilities used to support criticality and reactor physics codes.

The shutdown of Halden and subsequent project activities have highlighted the need for preserving key experiments in international databases. Consequently, the group developed a series of recommendations specifically targeting data preservation.

- **Recommendation**: NEA joint safety research projects should clearly outline their plan for data preservation, and should stipulate that a copy of the primary data needs to be sent to the NEA for storage.
- **Recommendation**: CSNI working groups should be asked to identify key datasets in their areas. Some of this may have been done with code validation matrices and datasets to support the development and implementation of standards.
- **Recommendation:** There should be a cross-functional (CSNI, NSC, etc.) NEA task group established to consider what should be done to preserve the key experimental datasets. This could include possible options for data libraries, how to screen datasets, what information needs to accompany the primary data, etc.
- **Recommendation**: CSNI working groups should select an appropriate option for preserving each key dataset and develop an activity to put it in place (CAPS, joint project, etc.).

## Reference

HySafe (2021), List of Experimental Facilities, www.hysafe.org/facilities (accessed 12 July 2021).

## NEA PUBLICATIONS AND INFORMATION

The full catalogue of publications is available online at www.oecd-nea.org/pub.

In addition to basic information on the Agency and its work programme, the NEA website offers free downloads of hundreds of technical and policy-oriented reports. The professional journal of the Agency, *NEA News* – featuring articles on the latest nuclear energy issues – is available online at www.oecd-nea.org/nea-news.

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# Nuclear Safety Research Support Facilities for Existing and Advanced Reactors: 2021 Update

Experimental facilities in nuclear energy are key to addressing safety issues. The recent loss of some critical infrastructure, from facilities to industry expertise, has therefore become a concern for many countries. In response, the Nuclear Energy Agency (NEA) has launched several efforts to address the matter as outlined in this report. Current safety issues, research needs and research facilities associated with currently operating water-cooled reactors in NEA countries are all addressed. Also included is an assessment of the present needs to maintain experimental databases. The Senior Group of Experts on Nuclear Safety Research, which produced this update of the 2007 report on the same issue, noted the success of previous reviews in helping maintain critical infrastructure and make a number of recommendations to preserve key research facilities and capabilities.

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