Chapter 3

Aspects of the design and safety demonstration of research reactors at international level

3.1. Convergence of practices on a few main safety objectives, principles and safety approaches

The construction of nuclear reactors (whether for research or power generation) began in the mid-20th century in a few countries (USA, former Soviet Union, France, UK, etc.) that were engaged in the research and development of technologies that could use the energy produced by nuclear fission to generate electricity.

In view of the safety and radiological protection issues raised by these facilities, which mobilized nuclear materials and radioactive fission products, and to avoid the exposure of workers and the public and the release of radioactive substances into the environment, the industry, in liaison with the safety organizations and authorities that were gradually set up, adopted various fundamental safety objectives, principles, procedures and criteria. These included, for example:

– compliance with "main safety functions", namely[37], for all reactors, control of core reactivity, removal of the heat released by the radioactive material, and confinement;

37. The wording used by the IAEA (e.g. in document SSR-3, which will be discussed in section 3.2.3) is as follows: *"The design for a research reactor facility shall ensure the fulfilment of the following main safety functions (…) for all states of the facility: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel storage; and (iii) confinement of the radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases."*

– the insertion of several physical confinement "barriers" between the radioactive materials or substances and the environment;

– the classification of equipment according to its importance for safety ("safety classification");

– or even the adoption of a redundancy principle[38] for the most important safety-related systems.

They also adopted methods and approaches for safety analysis and demonstration, e.g. the determination and analysis of normal, incident and accident events related to the facilities themselves, and of events (hazards) that could cause damage to the facilities, whether of internal or external origin (fire, floods, earthquakes, etc.).

In parallel the industry developed rules for the design (including the dimensioning[39]) and the construction of equipment, based on proven best practice and offering different requirement levels – to be chosen for each item of equipment on the basis particularly of its safety classification.

The practice of sharing experience feedback was gradually established, nationally then internationally, and in some countries, particularly France, the practice of carrying out periodic safety reviews (described below in sections 3.5 and 9.2) developed, including for French research reactors from the 1990s onwards.

The representative bodies – industrial, technical safety organizations, safety authorities, etc. – in these countries contributed their experience and expertise to the drafting of the IAEA safety standards for international use. These IAEA safety standards, which incorporated this "knowledge", were the subject of a consultation with all the Member States[40] to achieve a broad consensus.

The IAEA safety standards are documents which are not binding but are used as reference documents on the basis of which the IAEA conducts safety assessments when asked to do so by a Member State. Particularly in the case of research reactors, many Member States have written these safety standards into their own national regulations.

This chapter, which looks at some issues of the design and safety analysis of research reactors at international level, refers to the document reference base of the IAEA.

3.2. The IAEA safety standards

The IAEA's statute enables it to establish safety standards, to promote their application by its Member States and to provide assistance in this area to any Member States that request it.

38. Redundancy, or possibly more, for certain systems, equipment or components, in order to improve the reliability of their functions.
39. Determining the technical characteristics (geometry of equipment, flowrate of pumps…) of a facility during the design process to satisfy pre-established criteria and regulatory requirements.
40. On 23 October 1956, 81 States approved the Statute of the IAEA, which came into being on 29 July 1957. On April 30, 2018, the IAEA had 170 Member States.

A Code of (good) Conduct on the Safety of Research Reactors was adopted by the IAEA Board of Governors in March 2004. This Code, the text of which is similar to that of the Convention on Nuclear Safety, which applies solely to power reactors, is a high level document that is also not legally binding. It lays down guidelines for developing and harmonizing national practices as regards regulation and defines the ideal conditions for managing the safety of research reactors.

This Code of Conduct is a key element of the IAEA's programme of activities related to research reactors. This programme, approved by the Board of Governors, in particular involves the development of safety standards which inform or contribute to the application of the Code of Conduct, the organization and performance of safety assessments (INSARR), the organization of regional or international meetings on specific issues, and training carried out nationally or regionally to promote the guidance in the Code of Conduct. The IAEA intends that this programme will facilitate the sharing of operating experience feedback and lessons learned from events occurring at research reactors, especially through the Incident Reporting System for Research Reactors (IRSRR) and associated periodic meetings (see chapter 4 for more details). The IRSRR is managed by the IAEA, just like the IRS[41] for power reactors and the FINAS[42] system for fuel cycle facilities. However, it should be noted that only a few major or valuable incidents in terms of the lessons learned are entered into these databases.

3.2.1. Drafting process of the IAEA safety standards

The IAEA Secretariat organizes the drafting of the IAEA safety standards with the support of four specialist committees (competent respectively in the fields[43] of nuclear safety, radiation safety, transport safety, and waste safety), overseen by the Commission on Safety Standards (CSS). The work of the CSS has to be approved by the Member States within the Board of Governors. The process of drafting new standards or revising existing standards is shown in the diagram in figure 3.1. IRSN and the the French nuclear safety authority (ASN) are extensively involved in the development of these IAEA safety standards.

It is worth mentioning that other specialist international organizations sometimes take part in the development of these standards, either through direct involvement in their drafting or by commenting on the draft texts.

Through the process described above, a broad consensus is reached on the IAEA safety standards among the Member States. Consequently, implementation of the high level standards (safety fundamentals and requirements – see section 3.2.2) can be viewed as necessary to achieve an adequate level of safety for nuclear facilities, bearing in mind that responsibility for monitoring their safety remains a national responsibility. All safety standards (including guides) are generally reviewed five years after publication to decide whether revision is necessary.

41. International Reporting System for operating experience.
42. Fuel Incident Notification and Analysis System.
43. To be more specific, the Nuclear Safety Standards Committee (NUSSC), the Radiation Safety Standards Committee (RASSC), the Waste Safety Standards Committee (WASSC) and the Transport Safety Standards Committee (TRANSSC).

Figure 3.1. Process for the development or revision of the IAEA safety standards (note that for safety requirements and safety fundamentals, final endorsement is given by the Board of Governors). @ Georges Goué/IRSN.

3.2.2. Structure of the IAEA safety standards

The IAEA safety standards consist of three types of document: from the most general to the most specific, there are the safety fundamentals, the safety requirements and the safety guides.

The safety fundamentals present the objectives and general principles on which the IAEA's different standards for nuclear safety are based.

The safety requirements specify the requirements to be met to protect people and the environment.

The safety guides provide information and clarifications to help with applying the fundamentals and requirements; where appropriate, they are accompanied by examples of best practice.

The IAEA safety standards can be split into two main families: thematic standards and standards specific to a particular type of nuclear facility or activity. Separate safety requirements can therefore be drawn up for cross-cutting (thematic) fields and for specific facilities or activities (nuclear power plants, research reactors, fuel cycle facilities, radioactive material handling and transport, etc.). There are not very many thematic safety guides, but there are numerous guides for the different types of facility.

In 2006 the IAEA adopted a new structure for the safety standards (figure 3.2), which aims to ensure that there are clear, logical links between the safety fundamentals, the safety requirements and the safety guides.

Figure 3.2. Structure of the IAEA safety standards Series. @ Georges Goué/IRSN.

Within this structure, the general safety requirements have been grouped together in a single document, whereas the facility- and activity-specific safety requirements are separate documents. The new structure also takes the same approach to integration of the different fields (nuclear safety, radiation safety, waste safety and transport safety) as the approach used for the safety fundamentals.

3.2.3. Brief presentation of the safety standards *for research reactors*[44]

A set of safety standards has been established by the IAEA as part of its research reactor safety activities. Whereas the majority of these standards fall into the category

44. Published up to June 2018.

of standards specific to a type of facility, other important safety issues for research reactors, such as emergency preparedness and response[45], fall into the thematic category.

All organizations involved in research reactor safety, whether designers, operators or users, or indeed inspection bodies, can find these safety standards useful. They are written especially so that they can be used when drafting national regulations.

The standard SSR-3 (*Safety of Research Reactors – Specific Safety Requirements*) published in 2016 to replace the standard NS-R-4, contains safety requirements applicable to different types of research reactors cooled by (light or heavy) water with a thermal power of no more than a few tens of megawatts. For other types of research reactors and for research reactors with a higher thermal power rating, the safety requirements in the safety standards for power reactors can be used.

Compared to the standard NS-R-4, the standard SSR-3 introduces additional requirements on topics including:

– the "design extension conditions"[46]; this topic, which means that the account taken of postulated events in the design and for the safety demonstration of a nuclear reactor is extended, is discussed further later on;

– the use of a "graded approach"; this also is explained in more detail later on;

– the feedback of operating experience;

– the interface between nuclear safety and nuclear security[47] – safety measures and security measures should not compromise one another;

– the management of waste from research reactor operation.

The requirements in the standard SSR-3 deal with essential safety issues, including safety governance, regulatory control, safety demonstration and quality assurance, but also all the key stages in the life of these facilities, from choice of site, design (confinement barriers, main safety functions, defence in depth, etc.), construction, commissioning, operation, utilization and modification of research reactors, to final decommissioning.

The standard SSR-3 also requires that operators of research reactors have an independent safety committee (or advisory group[48]) to advise them on the safety aspects of

45. See the IAEA documents: General Safety Requirements No. GSR Part 7: "*Preparedness and Response for a Nuclear or Radiological Emergency*", and General Safety Guide No. GS-G.2.1: "*Arrangements for Preparedness for a Nuclear or Radiological Emergency*".

46. Accidents that are more severe than Design Basis Accidents, caused by internal or external hazards (based on the IAEA's definition of postulated initiating events).

47. This topic is not discussed in this report; for more information please refer to the document "*A comparative approach to nuclear safety and nuclear security*", Reference documents series, IRSN 2009/117, available on the IRSN website.

48. Independent of the operating organization or the reactor manager (the member of the reactor management team to whom the operator gives direct responsibility for, and authority over, operation of the research reactor and whose functions consist primarily of fulfilling this responsibility).

their reactor (design, commissioning, operation) and its utilization (experiments, training, etc.).

The committee members should be specialists in different fields on which the safety of the research reactor concerned depends; they may be external experts independent of the operating organization. The safety questions or issues to be considered by the committee include:

– the design, including the chemical composition, of nuclear fuel elements and reactivity control elements,

– modifications to operational limits and conditions,

– proposed new tests or experiments, and also systems, equipment or procedures that have significance for safety,

– proposed modifications to elements of the facility that have significance for safety,

– incidents that must be or have been reported to the regulatory body,

– periodic safety reviews of the facility,

– reports on radioactive discharges into the environment (in normal, incident or accident conditions) and on radiation doses to personnel at the facility and to the public.

There are a number of safety guides to help with the application of the requirements presented in standard NS-R-4 (and consequently those listed in standard SSR-3) for research reactors. A list of them, with comments, is provided in table 3.1 at the end of this chapter (guides in existence as at July 2018).

3.2.4. Application of the IAEA safety standards

As stated above, the IAEA safety standards are the expression of an international consensus aimed at protecting people and the environment. Despite this, the Member States are not legally bound to apply the safety standards. However, the IAEA does apply them to its own activities carried out under agreements to provide assistance or equipment signed with Member States. These agreements also stipulate that the country receiving assistance with acquiring or operating a research reactor must adhere to the IAEA safety standards.

More generally, the IAEA encourages its Member States to include in their national regulations and apply to their facilities the safety standards on research reactors as well as the standards on the regulatory and governmental infrastructure required for nuclear safety, radiation safety, the safety of radioactive waste and the safety of radioactive materials transport.

Finally, as stated above in section 3.1, all the standards are used as a reference by the IAEA when conducting safety assessments.

3.2.5. Supporting documents for application of the IAEA safety standards

Other documents known as *safety reports* and *technical documents* (TECDOC) are published by the IAEA in addition to the safety standards. They do not make any new recommendations and their only purpose is to facilitate the application of the safety guides by providing technical information, practical examples and detailed methods. There are many documents of this type related specifically to research reactors. They cover areas such as the technical and regulatory infrastructure to be set up by countries wishing to launch a nuclear power generation programme by building their first research reactor, the "conversion" of research reactors (to use low enriched with uranium-235 fuels), site evaluation, source term[49] evaluation and evaluation of the radiological consequences of accidents, implementation of an integrated management system, ageing, extended shutdown and decommissioning of facilities, and the corresponding safety analyses.

The process of preparing these documents is simpler than for the safety standards, because they do not have to undergo the full review and approval process used for the IAEA safety standards.

We saw earlier that a significant proportion of research reactors are in extended shutdown. In 2004 the IAEA wrote a technical document on this subject, TECDOC-1387 entitled *Safety Considerations for Research Reactors in Extended Shutdown*. This document makes some recommendations and suggests practices considered satisfactory in relation to the various safety issues raised by this situation, for example:

– maintenance of skills and a memory of the facility's technical history,

– qualification of the personnel used,

– human resources, retention of enough personnel to respond in an emergency,

– equipment (including instrumentation) that can be decommissioned,

– conditions for the preservation of equipment (which can mean removing it for storage in a less harsh environment [mothballing], e.g. unloading of the core for storage),

– surveillance, periodic testing and maintenance of structures, systems and components,

– preventing criticality risks, dealing with neutron moderators used to operate the reactor (e.g. in the case of heavy water reactors, removal of the water for safe storage),

– radiological protection,

– adaptation of the operating rules, the associated documentation, updating of documentation,

49. The expression "source term" refers to releases from a facility in an accident, expressed as Becquerels (Bq) of each radionuclide.

– procedures for restarting a reactor after an extended shutdown (especially pre-operational testing of equipment), etc.

3.3. IAEA exchange and assessment mechanisms

The IAEA uses the following resources in the context of its activities to improve safety at research reactors worldwide:

– international or regional meetings on the application of the Code of Conduct on the Safety of Research Reactors. Indeed, these meetings amount to discussions forums where the participants can share their experience and identify safety best practices. Self-assessments carried out during these meetings also enable the IAEA to identify more clearly the Member States' needs and any areas where safety management at research reactors could be improved. This information is then taken into account in the definition and performance of the IAEA's programmes of activities;

– national or regional training workshops on specific topics identified by the IAEA as important for the requesting country or the region;

– INSARR missions, which can be carried out at Member States' request to assess the safety of research reactors or to help resolve safety or radiological protection issues of a technical or organizational nature, including on inspection and regulation. These assessments cover around twenty topics. They are carried out by the IAEA with the participation of experts from operator organizations or safety bodies in different countries;

– more specific expert missions can also be organized to give the requesting bodies advice and assistance with resolving specific safety issues;

– periodic meetings, organized on average every 18 months as part of the IRSRR system, to exchange information about significant events that have occurred at research reactors and that could offer lessons for all research reactors.

Finally, the IAEA's technical cooperation programmes provide financial support to promote the participation of specialists from Member States that are (nuclear) developing countries in the meetings and workshops listed above. The IAEA's resources are also used to conduct INSARR missions and expert missions required as part of technical cooperation projects set up with the countries concerned.

3.4. Some general principles and approaches related to safety

3.4.1. Organization of safety control, safety culture

The fundamental safety principles and objectives are the subject of document SF-1 entitled "Fundamental Safety Principles", published by the IAEA in 2006. This document sets out the basis for the safety requirements. The ten safety principles discussed in this document concern nuclear safety and radiological safety. The document states that the fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation. The key principles presented concerning the organization of safety are:

– the prime responsibility for safety rests with the person or organization responsible for facilities and activities that give rise to radiation risks. The licensee retains the responsibility throughout the lifetime[50] of the facility or activity and cannot delegate it;

– an effective legal and governmental framework for safety must be established and sustained. The government is responsible for the establishment and adoption of the necessary legislation and regulations. It is also responsible for establishing a regulatory body independent of the operator organizations, with legal authority, technical and managerial competence, and the resources to fulfil its responsibilities;

– an effective integrated management system (for quality, safety, etc.) must be established, that ensures the promotion of a "safety culture" (a concept explained in more detail later on). Regarding accidents, the primary means of preventing them and mitigating the consequences of any that do occur is defence in depth (see section 3.4.2);

– the safety of facilities and activities that give rise to radiation risks must be assessed using a "graded approach" taking account proportionately of the potential risks associated with them (see section 3.4.4).

In practice, there are major disparities in the application of these safety principles and objectives at research reactors throughout the world. These disparities concern:

– the effectiveness and independence of the regulatory and inspection bodies, taking account of the skills and resources at their disposal;

– the updating of safety documentation to reflect the true state of facilities;

– the validity and the "envelope" nature of the safety analyses of these facilities.

However, it should be noted that, in countries where the construction of a new research reactor is considered to be an important step in preparations for a nuclear power generation programme, safety and regulatory texts generally refer these days to the IAEA safety standards and to international best practice.

The concept of safety culture emerged from the reflection process following the accident at the Chernobyl nuclear power plant on 26 April 1986. While the measures taken following the Three Mile Island accident in 1979 focused particularly on the ergonomic and cognitive aspects of workstations in reactors and other nuclear facilities, the Chernobyl accident raised questions of a different kind, concerning organizational factors. The development of a safety culture within organizations active in the nuclear

50. Including, at the end of its life, issues related to dismantling and waste management.

sector has generally been considered to be the appropriate response. As a result of the post-Chernobyl reflection process, it was felt that a more international vision of nuclear safety was necessary. This led to the issue of various reports by INSAG[51], an international group of experts in nuclear safety created within the IAEA. They included the *Summary Report on the Post-accident Review Meeting on the Chernobyl Accident* (Safety Series No.75-INSAG-1[52]) published in September 1986, which introduced the safety culture concept. The concept was discussed in more detail in 1991 in the report entitled *Safety Culture* (Safety Series No.75-INSAG-4). Safety culture is defined as "*that assembly of characteristics and attitudes in organizations and individuals which establishes that...) safety issues receive the attention warranted by their significance*". In particular, safety culture assumes that, within an organization, a questioning, careful and thorough approach and good communication between individuals are encouraged.

Two other INSAG reports are worth mentioning:

- the report entitled *Management of Operational Safety in Nuclear Power Plants* (Safety Series No.75-INSAG-13), published in 1999. This report discusses safety management issues of significance in promoting a safety culture, accompanied by recommendations and best practices. In particular, it contains recommendations on the maintenance of safety management during periods of organizational change, on monitoring safety performance and on early identification of declining safety performance before it has a significant impact on safety;
- the report entitled *Key Practical Issues in Strengthening Safety Culture* (Safety Series No.75-INSAG-15), published in 2002. This report, which includes some questions that can be asked as part of a safety culture self-assessment within an organization, discusses key issues such as: the importance of the manner of communication and of ensuring messages on safety are being received and understood, and especially of users understanding why procedures are used, the reporting culture and the attention that should be given to "near miss" incidents and to possible deviations ('*to tolerate is to validate*'[53]), an organization's ability to challenge itself at all levels (the "learning organization[54]").

The explanations and recommendations in these different INSAG reports are relevant to all types of facility, including research reactors – and also to operators, designers or other organizations that make a significant contribution to their operation. It is worth

51. International Nuclear Safety Group.
52. Updated in 1992 by report Safety Series No.75-INSAG-7.
53. Regarding the latter topic, it is worth mentioning here work done by an American sociologist, Diane Vaughan, on the Space Shuttle Challenger accident, published in 1996 in the book "*The Challenger Launch Decision, Risky Technology, Culture and Deviance at NASA*". It shows that what in hindsight appears to be a series of clearly identifiable errors is actually a series of decisions and interpretations that are perfectly understandable in the context in which they were made, but which are in fact slight deviations from normal limits and lead imperceptibly to the normalization of deviation.
54. INSAG-15 also points out that, although safety culture cannot be directly regulated, it is important that safety bodies understand how their actions affect the development of a safety culture and the improvement of the less formal human aspects of safety in organizations operating in the nuclear sector.

mentioning that, while there are production challenges for research reactors just as there are for power reactors (performing experiments, production of radioisotopes, etc. in the case of research reactors; production of electricity in the case of power reactors), the safety culture concerns two populations: the operating personnel, but also to a certain extent the experimenters. Regarding experience feedback from incidents, section 10.1.1 discusses the importance of the operators involved in experiments having sufficient awareness of nuclear safety and radiological protection.

3.4.2. Confinement barriers, fundamental safety functions, defence in depth

Historically, in terms of safety, reactor design has naturally followed the principle of placing multiple physical confinement barriers between radioactive materials and the environment and "fundamental safety functions" have been adopted. These were explained in section 3.1.

The deployment of multiple confinement barriers was already in itself a type of defence in depth. But over time this concept has taken on a much broader meaning and can now be described as follows.

The defence in depth principle can be summarised as the deployment of a succession of different "levels of defence" such that, if one level fails, the consequences of the failure are mitigated by higher levels. The independence of the different levels of defence is therefore a key element[55] in meeting this objective, and should therefore be sought as far as reasonably achievable[56].

The general objectives of defence in depth are:

– to compensate for human and component failures;

– to maintain the effectiveness of the barriers by averting damage to the facility and to the barriers themselves;

– to protect the public and the environment from harm in the event that these barriers are not fully effective.

A concept associated with the defence in depth principle has been developed over time and was formalised in the INSAG-10 report ("*Defence in Depth in Nuclear Safety*") published in 1996, as five levels. The five levels are shown in figure 3.3.

In defence in depth, the concept of level corresponds to a set of measures such as the intrinsic characteristics of the facility in question (reactor, fuel pool, etc.), physical measures (structures, systems and components) and procedures.

Even though the implementation of the levels may differ from country to country and may depend on facility design, the main principles are universal.

55. Expression used in INSAG-10.

56. Improving the independence of the levels of defence in depth *"as far as reasonably achievable"* is one of the safety objectives of WENRA for reactors of the future.

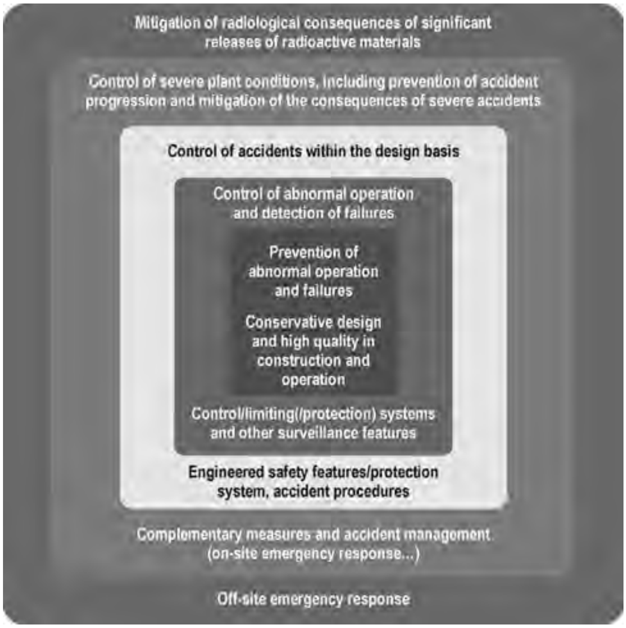

Figure 3.3. The defence in depth concept as developed in the INSAG-10 report: objectives and means. © Georges Goué/IRSN.

Because level 1 is the first level, it performs the main prevention function. Because levels 4 and 5 are the last levels, their main function is to limit the consequences of severe accidents.

There must also be a balance between the different levels of defence in depth. The INSAG-10 report stresses that accident management (level 4) may not be used to excuse design deficiencies at prior levels.

Conservative assumptions and safety margins (in relation to postulated phenomena) should be general features of the first three levels of defence in depth (choice of site, design and safety demonstration [e.g. when setting the limits for triggering protection and safeguard systems], construction, operation and modification, etc.). Measures should be taken to anticipate ageing (in the case of known mechanisms). At defence in depth levels 4 and 5, "best estimate", or reasonably conservative, considerations are used.

The INSAG-10 report also highlights the fact that, when implementing the defence in depth principle, internal and external hazards (fire, flooding, earthquakes, etc.) require particular attention because they could simultaneously impair several levels of defence in depth.

The INSAG-10 report also states that if it is not feasible to implement defence in depth against some events (such as sudden failure of a component under pressure), several levels of precautions should be introduced into the design and operation. These precautions could, for instance, be taken by choosing certain materials, by incorporating additional margins of safety during design, by reducing the length of welds, by using appropriate procedures for in-service monitoring, etc.

The various levels of defence-in-depth are explained below.

▶ Level 1: prevention of abnormal operation and failures

A nuclear facility such as a reactor (power or research reactor) should be intrinsically robust so as to reduce the risks of failure. This means that, once the initial definition of the facility (and the selection of the design options) is complete, the normal and abnormal conditions of operation should be clearly identified (i.e. as exhaustively as possible) in order to ensure the systems and components are sufficiently robust and resistant, including in accident conditions. According to the defence in depth concept, level 1 must offer an "initial basis of protection" against internal and external hazards (earthquake, plane crash, fire, explosion, etc.), though additional measures may be required at higher levels. The study of these hazards would lead, for example, to the selection of a seismic reference level, maximum weather conditions (expressed in terms of temperatures, wind speed, weight of snow), a maximum overpressure wave due to external industrial explosions, and the duration of exposure to these phenomena. The choice of site plays a key role in limiting these constraints.

The facility's various SSCs can then be designed, built, inspected, installed, tested and operated and be subject to appropriate preventive maintenance, following well established and qualified rules, with a sufficient margin beyond the limits defined for correct operation of the facility, and more specifically to ensure the SSCs perform the functions required for them in the various envisaged situations. These margins should prevent regular use of systems designed to cope with abnormal situations, particularly recourse to the measures taken for defence in depth levels 2 and 3.

Binding rules laid down in design and construction codes[57] define precisely the conditions for the design, procurement, manufacture, assembly, inspection, testing and preventive maintenance of safety-related equipment, to guarantee its quality in the broadest sense.

They are used to define the facility's authorised normal operating domain and its general operating rules.

A reactor technology[58] that changes state slowly and has automated controls can reduce the risk of stress for operating personnel. The design of the human-machine interface and the time available before manual intervention is required can make an important positive contribution.

57. Which reflect proven industrial best practice. Worth mentioning are the American ASME (American Society of Mechanical Engineers) code, the RCC-M (design and construction rules for mechanical components) for French PWRs, and the RCC-MRx (design and construction rules for mechanical components of nuclear installations: high-temperature, research and fusion reactors), applicable to research reactors in particular.
58. Designers often use the term "process" to refer to the technology of a reactor.

The choice of personnel involved at each phase in the life of a facility (design, manufacture, inspection and testing, operation, decommissioning), their training, the general organization of the different bodies involved – particularly in quality assurance and safety culture –, the sharing of responsibilities, and operating procedures, all help to prevent failures throughout the life of the facility.

Methodically taking account of experience feedback is also a key element that helps to improve prevention of failures.

▶ Level 2: control of abnormal situations and detection of failures

The facility should be prevented from leaving the authorised operating domain defined above and systems should be designed that are sufficiently reliable and that can stop abnormal changes before equipment is beyond design conditions chosen so that there is no failure risk.

A reactor design with a stable core and good thermal inertia makes it easy to keep the reactor within its authorised operating domain.

Surveillance of the facility's compliance with the design assumptions by means of in-service inspection and appropriate periodic testing of equipment is necessary to detect any degradation before it can affect the safety of the facility[59] and to carry out essential repairs (curative maintenance, replacement, etc.).

Systems that measure the radioactivity of different fluids and the atmosphere in different rooms can be used to verify the effectiveness of confinement barriers and purification systems.

Clear information in the control room about faults but also about the state or configuration of the facility's structures, systems and components makes it easier for operating personnel to deal with faults within an appropriate time scale.

Systems that limit adverse trends and can rapidly stop an undesirable phenomenon, not properly controlled by regulation, from occurring are deployed, including reactor shut down.

▶ Level 3: control of accidents within design basis

The first two levels of defence in depth – prevention and control of abnormal situations – are designed to prevent the occurrence of accidents.

However, in spite of the attention paid to these two levels, obviously for safety reasons a number of accidents are postulated, leading to failures such as a break in a reactor coolant pipe, regardless of the precautions taken to make them unlikely, or indeed

59. For French PWRs, in-service inspection of equipment is the subject of a document entitled RSE-M (equipment in-service surveillance rules), published by AFCEN (the French association that produces rules for the design, construction and in-service inspection of components for nuclear plants). There is no generic equivalent for research reactors because of the variety of their different designs.

very unlikely. This approach is usually described as “deterministic”, and it is an important part of the design of a facility and essential for the safety demonstration. These accidents have to be chosen at the start of the project design phase, so that systems can be designed that prevent severe damage to the core (e.g. core melt) and to ensure these systems are properly integrated into the other part of the facility. They have to be chosen with the utmost care because it is very difficult to integrate major systems at a later date into a facility that has already been built.

The systems defined are known as safeguard systems; they do not play any role in the normal operation of the facility. Where necessary, these systems start up automatically and they only require human intervention once sufficient time has elapsed for diagnostics to be carried out under calm conditions. The correct functioning of these systems ensures that, in the postulated situations, the integrity of the core structure is maintained and it can therefore continue to be cooled. Releases into the environment would be therefore very limited.

To ensure these safeguard systems are sufficiently reliable, particular attention should be given to the risk of common mode failures, hence the adoption of principles such as redundancy, geographical separation, diversification, etc. In-service monitoring and appropriate maintenance must also be carried out on the safeguard systems. Particular attention must be given to the procedures used to qualify these systems for accident conditions, which obviously cannot involve triggering an accident at the facility itself.

▶ Level 4: prevention of accident progression and mitigation of the consequences of severe accidents

The accident at the Three Mile Island power plant in 1979 prompted efforts to develop the means to cope with situations not covered by the first three levels of defence in depth, involving severe core damage. The challenge was to try to limit releases caused by situations where the core is badly damaged, for example in the event of core melt, and to buy time in which measures can be taken, if necessary, to protect the off-site population. Maintaining the containment function under the best possible conditions is essential in this case.

Special measures are taken by the operator of the damaged facility as part of the on-site emergency response plan: alerting the public authorities, monitoring the state of the damage facility, following appropriate operating procedures, implementing means of communication, response, etc. Periodic exercises are run with the different stakeholders that would be mobilised in an emergency, to ensure these measures will be effective should such situations occur.

▶ Level 5: mitigation of the radiological consequences of significant external releases of radioactive substances

The use of measures (off-site emergency response) to protect the population if significant releases occur ([enhanced] monitoring of activity levels and radiological exposure, sheltering, evacuation and control of foodstuffs, etc.), assumes that the previous measures have not been effective or have failed. The public authorities decide the conditions for evacuation or sheltering. They also take measures to control the consumption or marketing in the short, medium or long term of foods that could be contaminated. The decision to apply these measures is based on analyses of the situation by the operator and the safety bodies and on environmental radioactivity measurements.

Periodic exercises are also required, obviously involving the relevant public services, to ensure the intended logistical resources are effective.

Some specific aspects of the adaptation of defence in depth to French research reactors will be mentioned in chapter 7, giving examples.

3.4.3. The deterministic approach, design and safety demonstration basis – Situation regarding probabilistic studies for research reactors

In particular, to take account of safety objectives and requirements when designing research reactors and establishing their safety demonstrations, a deterministic approach (see above) must be taken, using conservative data that considers the most unfavourable configurations, for the reactor, of the core and the experimental devices or experiments. In accordance with this approach, to comply with the recommendations in the IAEA safety standards for research reactors, the safety analysis must consider a selection of postulated initiating events that can result from equipment failure, system malfunction, human error, or an internal or external hazard. A list of the initiating events generally postulated for research reactors is given in IAEA safety standard SSR-3; table 3.3 at the end of this chapter lists a selection of these events, grouping them into various families.

The deterministic approach involves studying a number of "facility states[60]", determined on the basis of the initiating events, some of which can be categorized according to the estimated frequency of the associated initiating events, has been refined over time. Most research reactors in operation were designed on a more cursory basis – though they are often still quite robust – than the more recent research reactors. The safety reviews discussed in sections 3.5, 4.3, 9.2.2 and 10.2 or other important administrative stages (e.g. see below for the HFR at Petten) enable the safety analyses to be extended by reference to more recent practices.

Probabilistic studies can be used in addition to the deterministic approach. However, it should be pointed out that the majority of research reactors are less complex than power reactors (e.g. pressurized water reactors), and that the value of probabilistic

60. According to IAEA terminology, especially in the standard SSR-3, the facility states are "*Normal operation*", "*Anticipated operational occurrences*", "*Design basis accidents*" and "*Design extension conditions*" (which includes "*Severe accidents*"). "*Normal operation*" and "*Anticipated operational occurrences*" are "*Operational states*". There is a limited number of facility states, with each one chosen to cover the corresponding family of events (events affecting core reactivity, core cooling, etc.).

safety analyses is therefore less obvious. But even with research reactors, probabilistic studies can be useful to identify relative weaknesses in their design or to assess quantitatively the contribution of implemented or planned improvements or modifications[61]. In addition, the use of probabilistic methods can give a better assessment of the relative importance of systems for the safety of a research reactor and identify more accurately their potential interactions.

It is worth mentioning in this regard that some safety authorities have asked operators of research reactors to conduct probabilistic safety analyses as part of the licensing process. For example, in 2003, probabilistic safety analyses (PSAs) at level 1 (assessment of the sequences leading to core damage and the overall probability of that damage) and level 2 (assessment of the different categories of radioactive releases into the environment and their probabilities) were carried out for the HFR reactor at Petten as part of the license renewal for operation of the reactor with low enriched uranium fuel. In particular, these probabilistic analyses were used to determine the dominant sequences of damage to the core (loss of off-site power, large-break loss of coolant outside the pool) or fuel elements (blockage of water circulation in the core, etc.)[62].

3.4.4. The graded approach[63]

IAEA Safety guide No. SSG-22, entitled *Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors*, published in 2012, presents recommendations for the "graded" application of the standard NS-R-4, and of the new standard SSR-3 that replaces it, which are applicable to research reactors and were mentioned above (in section 3.2.3).

The diversity of research reactors in terms of design, technical characteristics (power, quantity and nature of radioactive materials and substances, etc.), mode of operation and utilization, technological maturity and experience feedback is reflected in the diversity of associated risks. This diversity of risks has naturally led to the development of the graded approach concept.

The graded approach concerns many issues and applies at all stages in the life of a research reactor. For every research reactor, the design measures, application of the defence in depth principle, level of detail of the safety analyses, verifications of all kinds, documentation, activities and procedures used to implement the safety requirements, and more generally the resources dedicated to safety and safety monitoring, should be in proportion to the potential hazards posed by that reactor. The concept of potential hazards is very important for understanding and making correct use of the graded approach: grading should be based on the potential hazard of the facility in its environment due in particular to the inventory of radioactive material and substances, the

61. These uses are less affected by the lack of valid data on equipment reliability for the different research reactors, due mainly to the wide range of different research reactor designs, uses and modes of operation.
62. The probabilities of this damage are overall of the same order of magnitude as those for core melt in power reactors (a few 10^{-5} per year).
63. The expression "proportionality" is also used.

energy capable of disseminating that material, the site characteristics, the proximity of local populations, etc. For example, the resources dedicated to off-site emergency response plans should be proportionate to the robustness and containment capacity of the reactor building and to the radioactive releases envisaged in accident situations, their radiological impact on the populations likely to be affected, etc.

The purpose of the graded approach is to ensure the efforts made by operators and safety organizations are in proportion to the significance of the safety issues they address. According the IAEA safety guide, the graded approach can be applied to the following:

- the level of detail of procedures and operating instructions,
- the approval of documents or the authorization of modifications to the facility and experiments,
- training programmes,
- regulatory and other inspection programmes (e.g. frequency and duration of inspections),
- the integrated management system (safety, quality),
- emergency preparedness and response,
- the frequency of maintenance, equipment calibration, etc.

In some countries, like France, the application of the graded approach is written into national regulations.

3.5. *Periodic safety reviews*[64]

Internationally, periodic safety reviews are not a widespread practice at research reactors. Safety reviews are often carried out only for the purpose of renewing operating licences issued by safety authorities for a limited period of time. But for many research reactors worldwide, the licence does not set a maximum period of operation, and consequently periodic safety reviews cannot be carried out systematically, even though they are useful for:

- assessing whether continued operation is acceptable from a safety perspective, in view of any modifications made to facilities or to their operating procedures and any changes in their environment;
- identifying safety improvements to be made to these facilities on the basis of operating experience feedback (for the facility in question and similar facilities elsewhere in the world), better knowledge of certain risks and changing safety requirements or criteria.

IAEA safety guide No. SSG-25 entitled *Periodic Safety Review for Nuclear Power Plants*, published in 2013, makes recommendations for carrying out periodic safety

64. This is the expression used in the IAEA safety standards.

reviews on power reactors. The maximum recommended time between reviews is 10 years. With certain adaptations to reflect the specific nature of research reactors and the application of the graded approach, these recommendations can be used for research reactors.

In general, periodic safety reviews consist of the systematic review of safety at a nuclear facility at regular intervals, taking account in particular of the effects of ageing, modifications made to the facility, operating experience feedback, changes at the site, new knowledge acquired (e.g. concerning seismic risk), and best practices; changes to safety requirements are also considered. The aim of a review is to determine whether measures to ensure safety at a facility, which may have been modified in the light of other safety reviews, will still be adequate by the next safety review (or until the facility is permanently shut down). French practice – based on a ten-yearly frequency – is discussed in section 9.2 and illustrated by some of the most notable safety reviews conducted in France.

Periodic safety reviews carried out at research reactors generally cover:

– the safety management system, including quality assurance measures;

– the physical state, due to ageing, of structures, systems and components, which may have become brittle from the effect of radiation or could have eroded or corroded (e.g. components exposed to humidity if there is no air conditioning or if ventilation systems are not working);

– changes to safety requirements and the applicable criteria;

– changes to the site of the facility, such as an increase in population density, the advent of industries involving hazardous materials, the construction of highways to transport those materials, or changes in traffic (road and air traffic, etc.);

– experimental devices and experiments;

– maintenance programmes, test programmes and periodic inspection programmes;

– experience feedback, including international,

– organizational aspects concerning the operating personnel (recruitment, mobility, qualification, training, maintenance of skills and knowledge);

– the doses received by operating personnel;

– management of effluent and radioactive waste, the associated reports;

– the safety and operating documentation for the facility (safety report, general operating rules, on-site emergency plan, operating procedures).

Weaknesses and non-compliances identified during periodic safety reviews have in most cases led to safety improvement programmes at the facilities concerned, with precise timescales, subject to the safety authority's approval.

Although, during these improvement programmes, components of significance for the safety of research reactors may be replaced (in case of obsolescence or significant ageing), the configuration of civil engineering structures can, in some cases, make it difficult or even impossible to achieve adequate physical separation of the different "trains" of redundant safety systems during renovation work, and the safety analysis should take account of this.

The periodic safety reviews are an important step for maintaining a satisfactory level of safety. On the basis of these reviews, the safety authority may say whether the facility can continue operating or not.

The IAEA continues its efforts to promote and extend the practice of periodic safety reviews for research reactors with the forthcoming publication of a specific safety report and the organization of training activities on the subject.

3.6. Safety aspects of experimental devices

An experimental device[65] contains one or more samples to be irradiated in a neutron flux produced by a research reactor. The device contains the sample supports and the equipment for producing and controlling the desired irradiation conditions.

Experimental devices are generally installed in the core of a research reactor, in its reflector or around its periphery. A wide variety of experiments or irradiations are carried out with these devices. In particular, they can involve the irradiation of:

– fuel samples, subjected to pressure and temperature conditions and to coolant fluids that may be very different from those in the research reactor where they are irradiated; in this case the irradiation device is an experimental loop. The thermohydraulic conditions in the experimental loop may reflect incident or accident situations that the samples could be subject to in a power reactor. In these experiments, the fuel samples studied can be tested to the point of cladding failure and/or fuel melt;

– various materials for industrial applications;

– targets for the production of radioisotopes for medical or other uses.

An experimental device mainly consists of an "in-pile" part and an "out-of-pile" part.

The in-pile part contains the sample(s) to be irradiated and can be used to achieve and control the desired characteristics for the environment of these samples. In terms of safety, it has one or more barriers separating the sample from the reactor core coolant. The requirements for these barriers depend on the irradiation conditions and the risks posed by the experimental device as a whole.

The out-of-pile part consists in particular of the power supplies, the instrumentation and control units for the device, the fluid circuits and, with some irradiation loops, cells for analysing the fission products released by the fuel being tested. The out-of-pile part of an experimental loop helps to obtain the desired experimental conditions, particularly in terms of pressure and temperature, to which a sample should be subjected.

65. They are not experimental in themselves but are used for experiments. The name "experimentation device" would therefore be more appropriate; however, the usual term has been retained in this document.

It is important to emphasise that possible interactions between the experimental device(s) and the reactor where the irradiation is being performed, should be carefully examined from the point of view of safety.

Because there are common aspects to planning experiments in research reactors and planning modifications to those facilities, such as organization, safety analysis, management of authorizations and commissioning tests, in 2012 the IAEA published guide No. SSG-24 entitled *Safety in the Utilization and Modification of Research Reactors*. This guide recommends that the operator of a research reactor has responsibility for all aspects of reactor safety connected with preparing and carrying out experiments – even if other organizations (research organizations, universities, hospitals, industrial companies, etc.) are in charge of the design and programming of these experiments and even though the performance of certain tasks may be subcontracted to other organizations. The safety committees mentioned in section 3.2.3 (standard SSR-3) may be called upon to examine the suitability and safety of experiments and to make recommendations to the reactor manager. The guide recommends that the safety authority of the country where a research reactor is located should define and implement a licensing process (including the possibility for the operators of "internal" licensing under certain conditions) for experiments in research reactors and should check that operators are taking appropriate measures to manage and control the safety of these experiments.

The guide also recommends that:

– planned experiments be categorized on the basis of their significance for safety (as part of a graded approach);

– procedures be established for the safety analysis and approval of experiments;

– experiments with a major or significant effect on reactor safety be designed following the same principles as the reactor itself (defence in depth, single failure criterion, etc.) and be subject to formal licensing by the safety authority in the country concerned; experiments with only a minor effect on the safety of the reactor may be internally licensed by the operator.

The guide lists various safety aspects specific to experiments that should be examined for each:

– the reactivity worth[66] of the experimental device, which should remain within operational limits and conditions (negative reactivity of the core when the reactor is shut down, etc.);

– the protection system associated with the experiments, which can also be designed to protect the reactor;

– the heat produced in the experimental device and the ability of the device's cooling circuit to remove that heat, which should not affect the ability to cool the reactor;

66. The reactivity worth of any constituent of a reactor core is expressed in pcm (per cent mille). A fuel element has a positive worth because it contributes reactivity to the core, whereas a neutron absorbing rod has a negative worth. An experimental device can have positive reactivity (e.g. if it contains fissile material) or negative reactivity (e.g. if it is an irradiation capsule for steel samples).

– any risks associated with pressure in the experimental device, especially with regard to equipment important to reactor safety;

– the compatibility of the materials in the experimental device with one another and with the materials of the reactor (risk of corrosion, of eutectic formation, etc.);

– the possible interactions between the experimental device and the reactor (neutron flux disturbance, mechanical interactions, etc.);

– updating of the safety documentation for the facility (safety report, general operating rules, emergency procedures, etc.).

The guide also recommends application of the ALARA principle[67] to operator exposure when carrying out experiments and that the main risks associated with each experiment zone be displayed at the entrance to that zone.

Finally, the guide recommends that appropriate measures are taken to ensure that any equipment can be stored or disposed of safely on decommissioning or when the reactor is dismantled.

3.7. "Envelope" accidents taken into account in research reactor safety analyses

3.7.1. Definition and characteristics of "envelope" accidents

Certain aspects of research reactors and their uses, and experience feedback from their operation, very early on prompted designers and safety organizations to consider the possibility of accidents involving damage to the fuel in the reactor core or to the core as a whole, to the point of core melt. These aspects include:

– the fact that many handling operations take place in the reactor core or in proximity to it;

– some research reactors being sited close to populated areas;

– the occurrence of a number of reactivity accidents internationally, as explained in section 4.2 below.

Directly postulated "envelope" accidents determined from initiating events (single or multiple failures) of internal origin, are defined in order to verify the acceptability of the design[68] and the operating procedures. They are also used to define organizational and physical measures for emergency response. Studies of these accidents are

67. As Low As Reasonably Achievable. This principle, developed from the study of risk (cyndinics), was formulated for the first time in 1977 by the ICRP in its publication No. 26.

68. Architecture of systems, functional requirements of equipment, technical characteristics of this equipment (thickness of a concrete wall, rebar ratio, flow rate of a pump, thickness of a tank or vessel, materials used, type of welds, etc.).

conducted to assess radioactive releases and their radiological consequences for humans and the environment, based on the behaviour of the confinement barriers under the stresses they are subject to.

Various terms are used throughout the world to refer to the accidents in question at research reactors, which does not make understanding them any easier: envelope accident, reference accident (a term used particularly for French research reactors), maximum credible accident or maximum hypothetical accident, controlled severe accident, etc. The English terms Design Basis Accident (DBA) and Beyond Design Basis Accident (BDBA) are also used – for the newest research reactors or in the context of recent safety reassessments – to refer to concepts involved in the deterministic approach that has developed over time.

From a terminology point of view, it may be worth recalling the definitions given in the IAEA glossary (2007 version) for the different facility states:

– the *design basis accident* (DBA) is defined as *"accident conditions against which a facility is designed according to established design criteria"*;

– *beyond design basis accident* (BDBA) refers to *"accident conditions more severe than a design basis accident"*;

– *severe accident* refers to *"accident conditions more severe than a design basis accident and involving significant core degradation"* (making it a subset of the BDBA domain).

It therefore seems to be the case that the "envelope" accidents used for research reactors are mostly, by nature, beyond design basis accidents, or even severe accidents.

The term *Design Extension Conditions* (DEC)[69] has also been introduced by the international community – particularly the IAEA in document SSR-3 – for accidents that were previously described as beyond design basis (multiple failures, complex events, fuel melt accidents), and consequently the study of these accidents should aim to determine whether the design of the facility (including the ultimate confinement barrier) can adequately limit their consequences, or whether reinforcement (e.g. of the ultimate barrier) or the installation of extra equipment (additional power supplies, ultimate water make-up, etc.) should be envisaged.

A wide variety of "envelope" accidents are studied for the various research reactors throughout the world, a fact illustrated particularly by table 3.3 – though this presents only a selection. These accidents cover a wide range of states of core degradation, ranging from minimal damage to a fuel element to partial or total core melt. Although there are factors that partially explain this diversity (different designs and intrinsic characteristics [neutron feedback, etc.], varying robustness of safety systems [architecture, redundancy, diversification, etc.]), there is no denying that there are also disparities in the "envelope" accidents considered for research reactors that are technically similar.

69. The term in French is "*domaine complémentaire*", which has become "*domaine de conception étendu*" in more recent texts (see for example ASN Guide No. 22 "*Exigences de sûreté et recommandations pour la conception des réacteurs à eau sous pression"*).

The Nuclear Energy Agency (NEA) and IRSN have highlighted[70] the value of identifying and establishing best practices for defining "envelope" accidents for research reactors.

Specifically in the case of pool-type research reactors that use uranium-aluminium fuel, which are widespread throughout the world, "envelope" accidents initiated by a rapid and large-scale reactivity injection leading to core melt – so-called BORAX accidents[71] – are considered. However, the mechanical effects of the interaction between molten fuel and cooling water, in the form of a steam explosion, have not been uniformly taken into account for all these reactors, especially as regards the mechanical robustness of the reactor pool and containment; in addition, the potential consequences of projectiles hitting the containment wall due to a steam explosion have not always been examined.

Other differences concern the data used to calculate the radioactive releases from "envelope" accidents; this issue is discussed in the next section.

Chapter 8 of this report, on BORAX accidents, discusses the issues mentioned above, and how these accidents are taken into account in the case of French pool-type research reactors.

3.7.2. Source term evaluation for "envelope" accidents

Source term evaluation when studying the radiological consequences of an accident causing damage to the fuel in the reactor core (cladding failure, fuel melt) assumes that the nature and extent of the damage is known, as well as the pathways and quantities of fission products released by the fuel in the reactor building, and from that, the releases of fission products into the environment, and finally the doses and (long-term) contamination that could occur at various distances from the facility. For some reactors, the cancer risk from the radiation has been worked out[72].

These elements need to be evaluated on a case-by-case basis taking account of the specific characteristics of the reactor building (integrity, possible bypasses, etc.) and the ventilation (extraction rate, effectiveness of filtration systems), and specific characteristics of the site, considering that the accident could happen during a loss of off-site power, etc.

When fuel melt happens under water, the fission products are released in the pool water, from which a fraction is assumed to be released instantly into the atmosphere in the reactor building (particularly all the noble gases). The release of fission products then continues over time (partly through evaporation of the water in the pool – depending on the temperature difference between the water and the air in the reactor building and the evaporation surface area). When fuel melt happens in the air, the fission products are assumed to be released straight into the atmosphere in the reactor building.

70. In particular see the paper: "*Safety of research reactors: views of the NEA committee on the safety of nuclear installation*" – IAEA International conference on research reactors, Rabat, Morocco, 14–18 November, 2011.
71. BOiling water ReActor eXperiment.
72. For example, in 2003 in the case of the HFR reactor at Petten, when the operating license was being renewed to use the reactor with low enriched with uranium-235 fuel (see section 3.4.3).

There are differences in the assumptions used throughout the world to calculate the transfer of fission products from the fuel to the water, from the water to the air in the reactor building hall, and finally from the hall to the environment. As far as the release of radionuclides from fuel is concerned, the noble gases (xenon, krypton) are generally assumed to be released in their entirety. The differences between safety analyses concern the other species (iodine, caesium, ruthenium, strontium, actinides). They are often due to a lack of transposable experimental data (the transfer rates depend in particular on the burnup of the fuel, the maximum temperature reached by the fuel, and the medium it is in [water, air, steam-air, etc.]). Big differences have been observed, for example for iodine-131 and caesium-137, release rates from molten fuel into water vary from 0.1 in some reactors to 0.8 in others (OSIRIS reactor, etc.). The value 0.8 came from an analysis by the operator of the OSIRIS reactor after six fuel plates melted in the SILOE reactor in 1967 – attributed to a loss of cooling at the inlet of the affected fuel element (this event is described in section 10.1.2).

For transfer from the hall of the reactor building to the environment, the differences mainly relate to whether or not the deposition of fission products on surfaces and the effectiveness of filtration systems are taken into account.

3.8. Possible improvements to studies, research and development on research reactor safety

While research reactors can be used to acquire knowledge that is useful for assessing the safety of power reactors, their own safety does of course need to be justified by sufficient supporting data. Using very conservative values to study postulated initiating events can prove excessive and cause difficulties with design, construction or operation. Assumptions that are more informed (realistic) could be a way, provided that sufficient validated knowledge is available.

Acquiring further knowledge of the release rates of fission products from fuel elements in incident or accident conditions would be particularly useful, as previously turned out to be the case with evaluation of the source term linked to fuel melt accidents. Although the designers and operators of research reactors, and more specifically of the fuels to be used in these reactors, undertake experimental programmes to qualify these fuels, the programmes mainly explore temperature and pressure conditions, etc. in normal operation or during research reactor transients. The NEA (and IRSN) has drawn the attention of research reactor designers and operators to the value of using tests to improve knowledge of fuel behaviour in research reactors in incident and accident conditions[73].

In addition, various thermohydraulic simulation codes originally developed for power reactors have been adapted for studies related to research reactors in normal operating conditions, during transients, and in incident or accident conditions. However, disparities have emerged in the mathematical models and the correlations used in these codes, and in their degree of validation specifically for research reactors. This is because

73. See footnote 72.

data or knowledge were either not shared or not shared sufficiently, and because the connections between neutronics and thermohydraulics need to be improved. A Coordinated Research Project[74] (CRP) of the IAEA was run from 2003 to 2006 to compare simulations performed by different codes of operating transients in a reactor chosen as a reference (the Brazilian IEA-R1 reactor). The CRP mainly identified a need to benchmark the simulation codes using experimental data (qualification process), which led to a second CRP[75] run from 2008 to 2013, in which IRSN participated (see chapter 11). The aim of the second CRP was to assess the ability of simulation codes to reproduce a number of neutronics and thermohydraulics measurements taken straight from the cores of different research reactors[76]. In most cases, the neutronic data included core parameters such as the effective multiplication factor, the neutron flux distribution in the core, the fission rate in the fuel, some kinetic parameters, and the "worth" of the neutron-absorbing elements. The thermohydraulic data included, most notably, the temperature of the water measured at the inlet and outlet of the fuel elements. These data were for stable operating states and for reactivity and flow rate transients – including reductions in flow until natural convention was established, possibly with reversal of the direction of flow of the water in the core, for the ETRR-2, IEA-1 and RSG-GAS reactors. The final report of the second CRP is in preparation.

Other areas for improvement have been identified by the NEA and IRSN[77]; they are:

- acquisition and sharing of data on the mechanical characteristics of specific materials used in research reactors (e.g. the aluminium or zirconium alloys used for reactor tanks), and the changes in these characteristics over time and/or under irradiation;
- knowledge management, a particularly important subject given the long service life of many research reactors, extended shutdown periods, and related refresher training of operating personnel.

Among other IAEA initiatives, it is worth mentioning CRP T12029 *Benchmarks of Computational Tools against Experimental Data on Fuel Burnup and Material Activation for Utilization*, *Operation and Safety Analysis of Research Reactors*. This CRP was begun in 2015 and should end in 2019. It aims to contribute to the validation of the methods and codes used for computing fuel burnup and material activation by comparing experimental data collected from various operators. The results of this CRP will consist of a database of experimental results, measurements and associated facility specifications, and a publication comparing the experimental results with those from the different simulation codes and methods used.

74. IAEA CRP J7.10.10: "*Safety Significance of Postulated Initiating Events for Different Research Reactor Types and Assessment of Analytical Tools*".

75. IAEA CRP 1496: "*Innovative Methods in Research Reactor Analysis: Benchmark against Experimental Data on Neutronics and Thermal-hydraulic Computational Methods and Tools for Operation and Safety Analysis of Research Reactors*".

76. Measurements taken at the ETRR-2 reactor in Egypt, IEA-R1 in Brazil, McMaster Nuclear Reactor in Canada, MINERVE in France, MNSR in Syria, OPAL in Australia, RSG-GAS in Indonesia, and SPERT III and IV in the USA.

77. See footnote 72.

Table 3.1. IAEA guides for research reactors.

Reference	Description	Comments
SSG-10	Ageing Management for Research Reactors	This guide presents recommendations on establishing an ageing management programme depending on the actual state of facilities. This is a particularly important issue for research reactors, around two thirds of which are more than 40 years old.
SSG-20	Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report	This guide presents recommendations for preparing, examining and assessing the safety documents for a research reactor (safety report, general operating rules, on-site emergency plan, etc.). This safety guide focuses particularly on the design and construction stages of research reactors. It can be used not only as part of new reactor licensing procedures but also during periodic safety reviews of existing reactors.
SSG-22	Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors	Safety requirements should generally be applied in a way that is proportionate to the risks presented by a facility. This guide aims to clarify the graded approach and makes practical recommendations for the different phases in the life of a research reactor.
SSG-24	Safety in the Utilization and Modification of Research Reactors	This safety guide presents recommendations for the utilization and modification of research reactors. It is intended primarily for existing reactors but can also be useful to organizations considering carrying out new experiments at a research reactor.
SSG-37	Instrumentation and Control Systems and Software Important to Safety for Research Reactors	This guide presents recommendations for the design, production and qualification of instrumentation and control systems and the associated components and software, including the architecture of those systems, their safety classification, their human-machine interface, and security with regard to malicious acts. These recommendations are applicable both to instrumentation and control systems for new reactors and to the modernization of instrumentation and control systems at research reactors already in operation.
SSG-40	Predisposal Management of radioactive waste from nuclear power plants and research reactors	This guide provides recommendations on how to meet the requirements for the management of radioactive waste generated at nuclear power plants and research reactors (including subcritical or critical models). It covers all the stages of the management of such waste, from its generation until its elimination (but not its elimination), including its treatment (pretreatment, treatment and conditioning). Radioactive waste generated during normal operation and in the event of an accident is taken into account. This guide covers all phases of the life of waste management facilities, including the choice of site location, design, construction, implementation, commissioning, operation, closure, and decommissioning.

(*Continued on next page*)

Table 3.1. (*Continued*)

Reference	Description	Comments
NS-G-4.1	Commissioning of Research Reactors	Commissioning is one of the most important stages in the life of a reactor. Although this safety guide is more directly applicable to the commissioning of newly designed and built research reactors, it can also be used when re-commissioning a reactor after an extended shutdown or major modifications, and when commissioning new experimental devices in a research reactor.
NS-G-4.2	Maintenance, Periodic Testing and Inspection of Research Reactors	This safety guide presents various international practices considered satisfactory, particularly as regards the preventive and corrective maintenance of structures, systems and components of significance for safety, and periodic tests to ensure compliance with the operational limits and conditions defined for the facility.
NS-G-4.3	Core Management and Fuel Handling for Research Reactors	This guide presents recommendations on core management and fuel handling for research reactors in accordance with the applicable safety requirements and the service limits associated with the fuel. The guide covers core design and operation, core control parameters, the steps and processes involved in the receipt, handling and transport of fuel, core loading, and the handling and transport of new or irradiated fuel.
NS-G-4.4	Operational Limits and Conditions and Operating Procedures for Research Reactors	This safety guide presents recommendations for the establishment not only of operational limits and conditions (OLC), but also of operating procedures. Detailed recommendations are made for their development, formulation and implementation, both for the operation of the reactors and for the experiments conducted in the reactors.
NS-G-4.5	The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors	This guide is based on the premise that, for a reactor to be operated under satisfactory safety conditions, an appropriate and clearly defined organizational structure must be put in place with qualified personnel, and a safety culture must be developed. This safety guide makes recommendations concerning the operation of a research reactor, and the recruitment, training and qualification of the operating personnel (including those involved in maintenance operations), based on international best practice.
NS-G-4.6	Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors	This guide provides recommendations on radiation protection and the management of radioactive waste from research reactors. It identifies important elements that should be considered at the design stage with regard to facilitating radiological protection and radioactive waste management, and good practices in developing and implementing radiological protection programmes during facility operation.

(*Continued on next page*)

Table 3.1. (*Continued*)

Reference	Description	Comments
WS-G-2.1	Decommissioning of nuclear power plants and research reactors	This guide provides recommendations to ensure that the decommissioning process for nuclear power plants and research reactors is conducted in a safe and environmentally acceptable manner. It applies to these facilities and their sites. It mainly addresses the radiological risks resulting from the activities associated with the decommissioning of the reactors, and in particular the decommissioning after the planned final shutdown. A large number of provisions also apply to decommissioning as a result of an abnormal event that has resulted in contamination or severe damage to the reactor. In this case, the guide can serve as a basis for the development of special decommissioning provisions. The guide does not explicitly address non-radiological hazards, such as those due to potential sources of fire or those resulting from a release of abestos, which may be generated by decommissioning operations but which must also be managed.

Table 3.2. Illustration of the diversity of fuel melt accidents studied for research reactors[78].

Reactor year commissioned – year decommissioned	**Country**	**Power (MW)**	**Fuel**	**Fuel melt accidents considered** (*fuel melt under water, unless otherwise stated*)
HIFAR 1958–2007	Australia (Lucas Heights)	10	UAl enriched to around 60% ^{235}U	Total core meltdown (*loss of cooling*)
HFR 1961	Netherlands (Petten)	50	Originally UAl enriched to 91% ^{235}U, then U_3Si_2 enriched to around 20% ^{235}U	(*Reactivity accident excluded because water circulation direction would prevent ejection of neutron-absorbing rods* *Blockage of a cooling channel in a fuel element: does not lead to fuel melt*)
BR2 1963	Belgium (Mol)	100	UAl enriched to around 93% ^{235}U	Core melt (*power excursion* – with aluminium-water interaction)
SAFARI-1 1965	South Africa (Pelindaba)	20	Originally UAl enriched to 87%–93% ^{235}U, then U_3Si_2 enriched to around 20% ^{235}U	Core melt
OSIRIS 1966–2015	France (Saclay)	70	U_3Si_2 enriched to around 20% ^{235}U	Core melt (*power excursion* – with aluminium-water interaction)
RHF 1971	France (Grenoble)	57	UAl enriched to 93% ^{235}U	Melting of the fuel element in the core under water (*power excursion* – with aluminium-water interaction) Various accidents leading to the melting of one or more fuel elements, in the core, during handling or in a storage channel, under water or in air, in the short term (< 24 h) or longer term
ORPHEE 1980	France (Saclay)	60	UAl enriched to 93% ^{235}U	Core melt (*power excursion* – with aluminium-water interaction)
RSG-GAS 1987	Indonesia (Serpong)	30	U_3Si_2 enriched to around 20% ^{235}U	Melting of a fuel element (*blockage*) Melting of five fuel elements (*transient with postulated failure of the protection system* [*ATWS*])
FRM-II 2004	Germany (Garching)	20	U_3Si_2 enriched to around 90% ^{235}U	Total core meltdown (*loss of cooling or power excursion, no aluminium-water interaction*)
OPAL 2007	Australia (Lucas Heights)	20	U_3Si_2 enriched to around 20% ^{235}U	Melting of three fuel plates (*partial blockage of channels in a fuel element*) Melting of 36 UMo targets (*loss of cooling*)

78. This table is based on information that IRSN was able to gather.

Table 3.3. Selection of postulated initiating events for research reactors according to the IAEA SSR-3 standard.

Loss of electrical power supplies:

- Loss of normal electrical power.

Insertion of excess reactivity:

- criticality during fuel handling (due to an error in fuel insertion),
- startup accident,
- control rod failure or control rod follower failure,
- control drive failure or control drive system failure,
- failure of other reactivity control devices (such as a moderator or reflector),
- unbalanced rod positions,
- failure or collapse of structural components,
- insertion of cold water,
- changes in the moderator (e.g. voids, leakage of D_2O into H_2O systems, etc.),
- effects of experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material),
- etc.

Loss of flow:

- primary pump failure,
- reduction in flow of primarycoolant (e.g. due to valve failure or a blockage in piping or a heat exchanger),
- rupture of the primary coolant boundary leading to a loss of flow,
- fuel channel blockage or flow reduction (e.g. due to foreign material),
- improper power distribution due to, for example, unbalanced rod positions in core experiments or in fuel loading (power-flow mismatch),
- reduction in coolant flow due to bypassing of the core,
- deviation of system pressure from the specified limits,
- loss of heat sink (e.g. due to failure of a valve or pump or a system rupture).

Loss of coolant:

- rupture of the primary coolant boundary,
- damaged pool,
- pump-down of the pool,
- failure of beam tubes or other penetrations.

Erroneus handling or failure of equipement or components:

- failure of the cladding of a fuel element,
- mechanical damage to core or fuel (e.g. mishandling of fuel or dropping of a transfer flask onto fuel),
- failure of the emergency core cooling system,
- malfunction of the reactor power control,
- criticality in fuel in storage,
- failure of the means of confinement, including the ventilation system,
- loss of coolant to fuel in transfer or storage,
- loss or reduction of proper shielding,

(*Continued on next page*)

Table 3.3 (*Continued*)

- failure of experimental apparatus or material (e.g. loop rupture),
- etc.

Special internal events:

- internal fires or explosions, including internally generated missiles,
- internal flooding,
- loss of support systems,
- security related incidents,
- malfunctions in reactor experiments,
- improper access by persons to restricted areas,
- fluid jets or pipe whip,
- exothermic chemical reactions,
- drop of heavy loads.

External events (hazards):

- earthquakes (including seismically induced faulting and landslides),
- flooding (including failure of an upstream or downstream dam and blockage of a river and damage due to a tsunami or high waves),
- tornadoes and tornado missiles,
- sandstorms,
- hurricanes, storms and lightning,
- tropical cyclones,
- explosions,
- aircraft crashes,
- fires,
- toxic spills,
- accidents on transport routes (including collisions into the research reactor building),
- effects from adjacent facilities (e.g. nuclear facilities, chemical facilities and waste management facilities),
- biological hazards such as microbial corrosion, structural damage or damage to equipment by rodents or insects,
- extreme meteorological phenomena,
- electromagnetic interference (e.g. from solar events),
- lightning strikes,
- power or voltage surges on the external supply line.

Human errors