Western European WENRA WGRR Nuclear Regulators Association

Report

WENRA Safety Reference Levels for Existing Research Reactors

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Safety Reference Levels for Existing Research Reactors

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Foreword

A principal aim of the Western European Nuclear Regulators' Association (WENRA) is to develop a harmonized approach to nuclear safety within the member countries. One of the first major achievements to this end was the publication in 2006 of a set of Safety Reference Levels (RLs) for operating nuclear power plants (NPPs). Based on the lessons learned from the TEPCO Fukushima-Dai-ichi accident, results of the stresstests and its commitment to continuous improvement of safety, WENRA has updated the RLs for NPPs in 2014. Recently (2020), these RLs were updated again, addressing issues not revised in the 2014 revision.

This document provides for the first time a set of RLs for existing research reactors (RR). The development of these RLs was based on a decision by WENRA in 2016, upon a request from the European Nuclear Safety Regulators Group (ENSREG). This set for RR is based on the 2014 version of the RLs for existing NPPs, but includes the updates of Issue C (mainly the leadership aspect) and Issue I (to take into account the lessons from the ENSREG Topical Peer Review Ageing Management, e.g. obsolescence) from the 2020 version of RLs for NPPs. In addition, this set for RR includes Issue X (experimental devices and experiments), RLs specific for RR.

The RLs are agreed by the WENRA members. They reflect expected practices to be implemented in the WENRA countries. As the WENRA members have different responsibilities, the emphasis of the RLs has been on nuclear safety, primarily focusing on the main safety functions for ensuring the integrity of the research reactor. The RLs specifically exclude nuclear security and, with a few exceptions, radiation safety.

As RLs have been established for greater harmonization within WENRA countries, the areas and issues they address were selected to cover important aspects of nuclear safety where differences in substance between WENRA countries might be expected. They do not seek to cover everything that could have an impact upon nuclear safety or to form a basis for determining the overall level of nuclear safety in operating RRs.

The scope of application of these RLs covers all types of research reactors with the exeption of critical, sub-critical assemblies, homogeneous zero-power reactors and accelerator driven systems.

Given the various regulatory regimes and range of types of research reactors in operation in WENRA countries, the RLs do not go into legal and technical details.

For the application of the RLs for existing RRs, a graded approach shall be taken into account carefully as decribed in ANNEX A. In order to support the understanding and application of the RLs, a glossary has been added in ANNEX B.

There are significant interactions between some of the Issues and hence each Issue should not necessarily be considered self-standing and the RLs need to be considered as a whole set.

By issuing these RLs for RRs WENRA aims at further convergence of national requirements and safety improvements in WENRA member countries, as necessary.

Stakeholders were asked for comments on the Safety Reference Levels. All the comments were reviewed during the finalization process.

For further information, guidance on specific issues is also available on the WENRA website www.wenra.org. WENRA WGRR guidance on Issue O was specifically developed for RRs. Although the WENRA RHWG guidance for Issues F and T (both developed for NPPs) might already be useful for RRs, they will be customized, as necessary.

01
Issue A: Safety Policy

Safety area: Safety Management

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A1. Issuing and communication of a safety

A1.1 A written safety policy[1] shall be issued by the licensee.

A1.2 The safety policy shall be clear about giving safety an overriding priority in all *research reactor* activities.

A1.3 The safety policy shall include a commitment to continuously develop safety.

A1.4 The safety policy shall be communicated to all *site personnel* with tasks important to safety, in such a way that the policy is understood and applied.

A1.5 Key elements of the safety policy shall be communicated to contractors, in such a way that licensee's expectations and requirements are understood and applied in their activities.

A2. Implementation of the safety policy and monitoring safety performance

A2.1 The safety policy shall require directives for implementing the policy and monitoring safety performance.

A2.2 The safety policy shall require safety objectives and targets, clearly formulated in such a way that they can be easily monitored and followed up by the *research reactor* management.

A2.3 The safety policy shall require continuous improvement of nuclear safety by means of:

- Identifying and analysing any new information with a timeframe commensurate to its safety significance;
- Regular[2] review of the overall safety of the *research reactor* including the safety demonstration, taking into account operating experience, safety research, and advances in science and technology;
- Timely implementation of the reasonably practicable safety improvements identified.

Continuous improvement applies to all nuclear safety activities and hence it is relevant to all of the issues addressed in this document. Therefore, this requirement is not repeated in the other issues although it is applicable to all of them.

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[1] A safety policy is understood as a documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as a visible part of an integrated organizational policy or of an integrated management system.

[2] Regular is understood as an ongoing activity to review and analyse the research reactor design and operation and identify opportunities for improvement. Periodic safety review is a complementary tool to verify and follow up this activity in a longer perspective.

A3. Evaluation of the safety policy

A3.1 The adequacy and the implementation status of the safety policy shall be evaluated by the licensee on a regular basis, at least together with the periodic safety reviews.

02
Issue B: Operating Organisation
Safety area: Safety Management

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B1. Organisational structure

B1.1 The organisational structure for safe and reliable *operation* of the *research reactor*, and for ensuring an appropriate response in emergencies, shall be justified[3] and documented.

B1.2 The adequacy of the organisational structure, for its purposes according to B1.1, shall be assessed when organisational changes are made which might be significant for safety. Such changes shall be justified in advance, carefully planned, and evaluated[4] after implementation.

B1.3 Responsibilities, authorities, and lines of communication shall be clearly defined and documented for all *site personnel* with duties important to safety.

B2. Management of safety and quality

B2.1 The licensee shall ensure that the *research reactor* is operated in a safe manner and in accordance with all applicable legal and regulatory requirements.

B2.2 The licensee shall ensure that decisions on safety matters are timely and preceded by appropriate investigation and consultation so that all relevant safety aspects are considered. Safety issues shall be subjected to appropriate safety review, by a suitably qualified independent review function.

B2.3 The licensee shall ensure that all *site personnel* is provided with the necessary facilities and working conditions to carry out work in a safe manner.

B2.4 The licensee shall ensure that safety performance is continuously monitored through an appropriate review system in order to ensure that safety is maintained and improved as needed.

B2.5 The licensee shall ensure that relevant operating experience, international development of safety standards and new knowledge gained through R&D-projects are analysed in a systematic way and continuously used to improve the *research reactor* and the licensee's activities.

B2.6 The licensee shall ensure that *research reactor* activities and processes are controlled through a documented management system covering all activities, including relevant activities of vendors and contractors, which may affect the safe *operation* of the *research reactor*.

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3 The arguments shall be provided that the organisational structure supports safety and an appropriate response in emergencies.

4 A verification that the implementation of the organisational change has accomplished its safety objectives.

B3. Sufficiency and competency of *staff*

B3.1 The required number of *staff* for safe *operation*, and their competence, shall be analysed in a systematic and documented way.

B3.2 The sufficiency of *actual staff* for safe *operation*, their competence, and suitability for safety work shall be verified on a regular basis and documented.

B3.3 A long-term staffing plan[5] shall exist for activities that are important to safety.

B3.4 Changes to the number of *staff*, which might be significant for safety, shall be justified in advance, carefully planned and evaluated after implementation.

B3.5 The licensee shall always have in house, sufficient, and competent *staff* and resources to understand the licensing basis of the *research reactor* (e.g. *Safety Analysis Report* or Safety Case and other documents based thereon), as well as to understand the actual design and *operation* of the *research reactor*.

B3.6 The licensee shall maintain, in house, sufficient and competent *staff* and resources to specify, set standards, manage and evaluate safety work carried out by contractors.

[5] Long term is understood as 3-5 years for detailed planning and at least 10 years for prediction of retirements etc.

03
Issue C: Leadership and Management for Safety

Safety area: Management for Safety

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C1. Objectives

C1.1 Leadership[6] and management for safety shall be established, sustained and balanced in the licensee organisation to effectively foster a strong safety culture and enhance safety performance.

C1.2 The senior management shall ensure that the safety policy is implemented and that its objectives are fulfilled.

C2. Leadership for safety

C2.1 Leadership for safety shall be effective at all organisational levels within the licensee organisation.

C2.2 The senior management shall ensure that the developed goals, strategies, plans and objectives are consistent with the safety policy of the licensee organisation. Their collective impact on safety shall be understood and managed in such a way that safety is not compromised by other priorities.

C2.3 The senior managers shall ensure that decisions made at all levels take into account the priorities and accountabilities for safety.

C2.4 Managers at all levels shall develop competences for leadership for safety, demonstrate commitment to safety and foster a strong safety culture.

C2.5 Managers at all levels shall promote values and expectations for safety by means of their decisions, statements and actions.

C2.6 Managers at all levels shall ensure that relevant professional knowledge, skills and experience of individuals under their responsibility are used in making decisions.

C3. Management for safety

C3.1 An integrated management system[7] shall be established, implemented, assessed and continuously improved by the licensee. The main aim of the integrated management system shall be to achieve and enhance nuclear safety. Other demands[8] on the licensee and the licensee's management system shall be considered in unison with nuclear safety, in order to help preclude their possible negative impact on nuclear safety.

[6] Leadership is understood as a person's ability to give direction to and motivate individuals and groups and to influence their commitment to shared goals, values and behaviour.
[7] See IAEA Safety Standards Series No. SSR-3 (2016): Integrated management system
[8] Examples of such demands are health, environmental, security, quality and economic requirements.

C3.2 The licensee shall ensure that management at all levels demonstrate its commitment to the establishment, implementation, assessment and continuous improvement of the management system.

C3.3 The human and organisational factors[9] that influence safety shall be taken into account in the management system in an integrated approach.

C3.4 It shall be defined in the management system when, how and by whom decisions[10] are to be made within the organisation, ensuring that safety is taken into account in decision making and is not compromised by any decision taken.

C3.5 Provisions shall be made in the management system to collect, process and document operating experience. Internal and external experience shall be used to improve safety.

C3.6 The potential safety impact of changes to the management system shall be analysed prior to their implementation. Changes with potential impact on safety shall be justified, planned, executed and evaluated accordingly.

C3.7 All *site personnel* shall be trained in the relevant aspects of the management system with the aim to ensure its implementation and to foster their involvement in its continuous improvement.

C3.8 The licensee shall determine and provide the necessary resources to establish, implement, assess and continuously improve the management system.

C3.9 The application of management system requirements shall be graded so as to deploy appropriate resources, on the basis of the consideration of:

- The significance and complexity of each activity and its results;
- The hazards and the magnitude of the potential impact associated with each activity and its results;
- The possible consequences if an activity is carried out incorrectly or its objective is not achieved.

C3.10 The documentation of the management system shall include at least:

- The policy statements of the licensee;[11]
- A description of the management system;
- A description of the organisational structure of the licensee;
- A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;
- A description of the interactions with relevant external organisations and with interested parties;
- A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved.

C3.11 The documentation of the management system shall be understandable to those who use it. Documents shall be up to date, readable, readily identifiable and available at the point of use.

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9 Human and organisational factors (HOF) are understood as the factors which have influence, in a positive or adverse manner, on human performance in a given situation, keeping in mind that safety is the result of interaction of human, technology and organisation

10 With respect to decisions that impact on nuclear safety.

11 including values and behavioural expectations

C3.12 Documentation shall be controlled. Changes to documents shall be reviewed and recorded and shall be subject to the same level of approval as the documents themselves. It shall be ensured that document users are aware of and use appropriate and correct documents.

C3.13 Records shall be specified in the management system documentation and shall be controlled. All records shall, for the duration of the retention times specified for each record, be readable, complete, identifiable and easily retrievable.

C3.14 The processes[12] that are needed to achieve the goals, provide the means to meet all requirements and deliver the products of the licensee organisation shall be identified, their development shall be planned, and they shall be implemented, assessed and continuously improved. The sequence and interactions of the processes shall be determined.

C3.15 The methods necessary to ensure the effectiveness of both the implementation and the control of the processes shall be determined and implemented to achieve the organisation's goals without compromising safety.

C3.16 Arrangements for qualification, selection, evaluation, procurement, and oversight of the supply of products and services important to safety[13] shall be made on the basis of specified criteria.[14]

C3.17 Purchasing requirements shall be developed and specified in procurement documents. Evidence that products and services meet these requirements shall be available to the licensee before they are used.[15]

C3.18 The control of processes, or work performed within a process, contracted to external organisations shall be identified within the management system. The licensee shall retain overall safety responsibility when purchasing any products or contracting any services. It shall be ensured, that sufficient comprehension and knowledge about the product or service, that is being procured, are available within the licensee's organisation.

C4. Culture for safety

C4.1 Management, at all levels in the licensee organisation, shall consistently demonstrate, support, and promote attitudes and behaviours that result in an enduring and strong safety culture. This shall include ensuring that their actions discourage complacency, encourage an open reporting culture as well as a questioning and learning attitude with a readiness to challenge acts or conditions adverse to safety.

C4.2 The management system shall include provisions to systematically develop, support, and promote desired and expected attitudes and behaviours that result in a strong safety culture.

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12 This is not understood as a full process orientation of the management system. Also functional or organisational oriented routines and procedures could be used for certain activities together with cross cutting processes for other activities.

13 Products and services participating in the technical or organisational provisions on which the safety demonstration of the *research reactor* is based.

14 Procurement procedures shall include specific instructions for preventing, detecting, reporting and disposing of counterfeit, fraudulent and suspect items.

15 Through inspection, testing, verification and validation activities before the acceptance, implementation, or operational use of products.

C4.3 The licensee organisation shall ensure that suppliers, contractors, students, scientists and others whose *operations* may have a bearing on the safety at the *research reactor* comply with C4.1 and C4.2 in a way that ensures that the resulting interfaces with the *research reactor* support the standards and expectations.

C5. Measurement, assessment and improvement

C5.1 The senior management shall ensure that:

- The adequacy and effectiveness of the management system is monitored and measured;
- Self-assessments and independent[16] assessments are conducted regularly regarding:
  - the performance of work for which they are responsible,
  - leadership for safety, and
  - safety culture, including the underlying attitudes and behaviours.

C5.2 An organisational unit[17] shall be established with the responsibility for conducting independent internal assessments. This unit shall have sufficient authority to discharge its responsibilities. Individuals conducting independent assessments shall not assess their own work.

C5.3 The licensee shall evaluate the results of the assessments and take any necessary improvement actions, and shall record and communicate inside the licensee organisation the results, the decisions and the reasons for the necessary actions.

C5.4 Improvement plans shall include plans for the provision of adequate resources throughout all phases of implementation. Actions for improvement shall be monitored through to their completion and the effectiveness of the improvement shall be checked.

[16] By an external organisation or by an internal independent assessment unit.
[17] Depending on the risk profile of the *research reactor*, arrangements can be considered for an independent *staff* position, rather than an organisationalunit.

04
Issue D: Training and Authorization of Site Personnel (Jobs with Safety Importance)

Safety area: Safety Management

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D1. Policy

D1.1 The licensee shall establish an overall training policy and a comprehensive training plan on the basis of long-term competency needs and training goals that acknowledges the critical role of safety. The plan shall be kept up to date.

D1.2 A systematic approach to training shall be used to provide a logical progression, from identification of the competences required for performing a job, to the development and implementation of training programmes including respective training materials for achieving these competences, and to the subsequent evaluation of this training.

D2. Competence and qualification

D2.1 Only qualified persons[18] that have the necessary knowledge, skills, and safety attitudes shall be allowed to carry out tasks important to safety. The licensee shall ensure that all *site personnel* performing safety-related duties have been adequately trained and qualified.

D2.2 The licensee shall define and document the necessary competence requirements for their *site personnel*.

D2.3 Appropriate training records and records of assessments against competence requirements shall be established and maintained for each individual with tasks important to safety.

D2.4 *Site personnel* qualifying for positions important to safety shall undergo a medical examination to ensure their fitness depending upon the duties and responsibilities assigned to them. The medical examination shall be repeated at specified intervals.

D3. Training programmes and facilities

D3.1 Performance based training programmes shall be established for all *site personnel* with tasks important to safety. The programmes shall cover initial training in order to qualify for a certain position and regular refresher training.

D3.2 All *Site personnel* shall have a basic understanding of nuclear safety, radiation safety, fire safety, the on-site emergency arrangements and industrial safety.

[18] If any tasks important to safety are carried out during training and education, they shall be in accordance with the safety requirements and shall be supervised by the authorised *site personnel*.

D3.3 Refresher training for control room operators shall include especially the following items as appropriate:

- *Research reactor operation* in normal operational states, selected anticipated operational occurrences and accident conditions;
- Operational experiences and modifications of *research reactor* and procedures;
- Shift crew teamwork.

D3.4 *Site personnel* carrying out maintenance and technical support work shall have practical training on the required safety critical activities.

D4. Authorization

D4.1 *Operating personnel* controlling changes in the operational status[19] of the *research reactor* shall be required to hold an authorization valid for a specified time period. The licensee shall establish procedures for their *operating personnel* to achieve this authorization. In the assessment of an individual's competence and suitability as a basis for the authorization, documented criteria shall be used.

D4.2 If an authorised individual:

- Moves to another position for which an authorization is required;
- Has been absent from the authorised position during an extended time period;

Re-authorisation shall be conducted after necessary individual preparations.

D4.3 Work carried out by contractors or experimenters on structures, systems, or components that are important to safety shall be approved and monitored by a suitably competent member of *staff*.

[19] This RL concerns for instance the start-up of the reactor. Such a task shall only be carried out by *operating personnel* specifically authorized for that task.

05
Issue E: Design Basis Envelope for Existing Research Reactors

Safety area: Design

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E1. Objective

E1.1 The design basis[20] shall have as an objective the prevention or, if this fails, the mitigation of consequences resulting from anticipated operational occurrences and design basis accidents. Design provisions shall be made to ensure that potential radiation doses to the public and the *site personnel* do not exceed prescribed limits and are as low as reasonably achievable.

E2. Safety strategy

E2.1 Defence-in-depth[21] shall be applied in order to prevent, or if prevention fails, to mitigate harmful radioactive releases.

E2.2 The defence-in-depth concept shall be applied to provide several levels of defence including a design that provides a series of physical barriers to prevent uncontrolled releases of radioactive material to the environment, as well as a combination of safety features that contribute to the effectiveness of the barriers.

The design shall prevent as far as practicable:

- challenges to the integrity of the barriers;
- failure of a barrier when challenged;
- failure of a barrier as consequence of failure of another barrier.

E3. Safety functions

E3.1 During normal *operation*[22], anticipated operational occurrences and design basis accidents, the *research reactor* shall be able to fulfil the fundamental safety functions[23]:

- control of reactivity,
- removal of heat from the reactor core and from the spent fuel, and
- *confinement* of radioactive material.

[20] The design basis shall be reviewed and updated during the lifetime of the *research reactor* (see RL E11.1).
[21] For further information see IAEA Safety Standards Series No. SSR-3 (2016).
[22] Normal *operation* includes start-up, power *operation*, shutting down, shutdown, maintenance, testing and refuelling.
[23] Under the conditions specified in the following paragraphs.

E4. Establishment of the design basis

E4.1 The design basis shall specify the capabilities of the *research reactor* to cope with a specified range of *research reactor* states[24] within the defined radiation protection requirements. Therefore, the design basis shall include the specification for normal *operation*, anticipated operational occurrences and design basis accidents from Postulated Initiating Events (PIEs), the safety classification, important assumptions and, in some cases, the particular methods of analysis.

E4.2 A list of PIEs[25] shall be established to cover all events that could affect the safety of the *research reactor*. From this list, a set of anticipated operational occurrences and design basis accidents shall be selected using deterministic methods, complemented where appropriate and available by probabilistic methods, as well as engineering judgement.[26] The resulting design basis events shall be used to set the boundary conditions according to which the *structures, systems and components* important to safety shall be designed, in order to demonstrate that the necessary safety functions are accomplished and the safety objectives met.

E4.3 The design basis shall be systematically defined and documented to reflect the actual *research reactor*.

E5. Set of design basis events

E5.1 Internal events such as loss of coolant accidents, equipment failures, maloperation and internal hazards, and their consequential events, shall be taken into account in the design of the *research reactor*.[27] The list of events shall be *research reactor* specific and take account of relevant experience and analysis from other *research reactor*.

E5.2 External hazards shall be taken into account in the design of the *research reactor*. In addition to natural hazards[28], human made external hazards[29] – including airplane crash and other nearby transportation, industrial activities and site area conditions which reasonably can cause fires, explosions or other threats to the safety of the *research reactor* – shall as a minimum be taken into account in the design of the *research reactor* according to site specific conditions.

E6. Combination of events

E6.1 Credible combinations of individual events, including internal and external hazards, that could lead to anticipated operational occurrences or design basis accidents, shall be considered in the design. Deterministic assessment and probabilistic methods as well as engineering judgement can be used for the selection of the event combinations.

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24 Normal *operation*, anticipated operational occurrences and design basis accident conditions.

25 The IAEA Safety Standards Series No. SSR-3 (2016) may be used as guidance for establishing a list of PIEs.

26 Depending on the specific topic being under review, not all types of insights (deterministic, probabilistic or engineering judgement) may be relevant or needed.

27 Additional information on internal hazards is provided in IAEA Safety Reports Series No. 55. Safety Analysis for Research Reactors and IAEA Safety Standards Series No. SSR-3 (2016).

28 See Issue T.

29 The definition of a design basis event shall be justified and appropriate to the risk presented by the *research reactor*.

E7. Definition and application of technical acceptance criteria

E7.1 Initiating events shall be grouped into a limited number of categories that correspond to *research reactor* states, according to their probability of occurrence. Radiological and technical acceptance criteria shall be assigned to each *research reactor* state such that frequent initiating events shall have only minor or no radiological consequences and that events that may result in severe consequences shall be of very low frequency.

E7.2 Criteria for protection of the *fuel element* integrity, including e.g. fuel temperature, critical heat flux, and cladding temperature, shall be specified. In addition, criteria shall be specified for the maximum allowable fuel damage during any design basis accident.

E7.3 If applicable, criteria for the protection of the primary coolant boundary, *experimental devices* with high pressure and temperature loops as well as cold or hot neutron sources shall be specified, such as maximum pressure, maximum temperature, thermal- and pressure transients and stresses.

E7.4 Criteria shall be specified for protection of means of *confinement*, including temperatures, pressures and leak rates.

E8. Demonstration of reasonable conservatism and safety margins

E8.1 The initial and boundary conditions shall be specified with conservatism.

E8.2 The worst single failure[30] shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and its function remains unaffected by the PIE.

E8.3 Only systems that are suitably safety classified can be credited to carry out a safety function. Non safety classified systems shall be assumed to operate only if they aggravate the effect of the initiating event.[31]

E8.4 A stuck control rod shall be considered as an additional aggravating failure in the analysis of design basis accidents.[32]

E8.5 The safety systems shall be assumed to operate at their performance level that is most penalising for the initiator.

E8.6 Any failure, occurring as a consequence of a postulated initiating event, shall be regarded to be part of the original PIE.

E8.7 The safety analysis shall:

(a) rely on methods, assumptions or arguments which are justified and conservative;

(b) provide assurance that uncertainties and their impact have been given adequate consideration[33];

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30 A failure and any consequential failure(s) shall be postulated to occur in any component of a safety function in connection with the initiating event or thereafter at the most unfavourable time and configuration.

31 This means that non safety classified systems are either supposed not to function after the initiator, or supposed to continue to function as before the initiator, depending on which of both cases is most penalising.

32 This assumption is made to ensure the sufficiency of the shutdown margin. The stuck rod selected is the highest worth rod at Hot Zero Power and conservative values of reactor trip reactivity (conservative time delay and reactivity versus control rod position dependence) are used. A stuck rod can be handled as single failure in the analysis of design basis accidents (DBAs) if the stuck rod itself is the worst single failure.

33 Conservative assumptions, safety factors, uncertainty and sensitivity analysis are means to address uncertainties and their impact on safety assessment.

(c) give evidence that adequate margins have been included when defining the design basis to ensure that all the design basis events are covered;

(d) be auditable and reproducible.

E9. Design of safety functions

General

E9.1 The fail-safe principle shall be considered in the design of systems and components important to safety.

E9.2 A failure in a system intended for normal *operation* shall not affect a safety function.

E9.3 Activations and control of the safety functions shall be automated or accomplished by passive means such that operator action is not necessary within 30 minutes of the initiating event. Any operator actions required by the design within 30 minutes of the initiating event shall be justified.[34]

E9.4 The reliability of the systems shall be achieved by an appropriate choice of measures including the use of proven components[35], redundancy, diversity[36], physical and functional separation and isolation.

E9.5 For sites with multiple *research reactors*, appropriate independence between them shall be ensured.[37]

Reactor and fuel storage sub-criticality

E9.6 At least one shutdown system shall, on its own, be capable of quickly rendering the reactor sub critical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure.
The provision of a second independent, preferably diverse, shutdown system shall be given due consideration in the design.

E9.7 Sub-criticality shall be ensured and sustained:

- in the *research reactor* after planned reactor shutdown during normal *operation* and after anticipated operational occurrences, as long as needed;
- in the *research reactor*, after a transient period (if any) following a design basis accident[38];
- for fuel storage during normal *operation*, anticipated operational occurrences, and design basis accidents.

[34] The control room *staff* shall be given sufficient time to understand the situation and take the correct actions. Operator actions required by the design within 30 min after the initiating event shall be justified and supported by clear documented procedures that are regularly exercised.
[35] Proven by experience under similar conditions or adequately tested and qualified.
[36] The potential for common cause failure, including common mode failure, shall be appropriately considered to achieve the necessary reliability.
[37] The possibility of one unit supporting another could be considered as far as this is not detrimental for safety.
[38] Technical acceptance criteria have to be fulfilled during a transient period for which sub-criticality is not ensured.

Heat removal functions

E9.8 Means for removing residual heat from the core after shutdown and from spent fuel storage (if necessary), during and after anticipated operational occurrences and design basis accidents, shall be provided taking into account the assumptions of a single failure and the loss of off-site power.

Confinement functions

E9.9 Means of *confinement* shall be provided in order to ensure that any release of radioactive material to the environment in a design basis accident would be below prescribed limits. In the design consideration shall be given to implementing:

- leaktight structures;
- features for isolation of the reactor building including penetrations for the primary cooling system and ventilation system (including redundancy, automation, type and location);
- features for the management and removal of fission products, hydrogen, oxygen and other substances that could be released into the reactor building.

E10. Instrumentation and control systems

E10.1 Instrumentation shall be provided for measuring all the main variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, the barriers belonging to the means of *confinement*, and the state of the spent fuel storage. Instrumentation shall also be provided for obtaining any information on the *research reactor* necessary for its reliable and safe *operation*, and for determining the status of the *research reactor* in design basis accidents. Provision shall be made for automatic recording[39] of measurements of any derived parameters that are important to safety.

E10.2 Instrumentation shall be adequate for measuring *research reactor* parameters and shall be environmentally qualified for the *research reactor* states concerned.

Control room

E10.3 A control room shall be provided from which the *research reactor* can be safely operated in all its operational states, and from which measures can be taken to maintain the *research reactor* in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences and design basis accidents.

E10.4 Devices shall be provided to give in an efficient way visual and, if appropriate, also audible indications of operational states and processes that have deviated from normal and could affect safety. Ergonomic factors shall be taken into account in the design of the control room. Appropriate information shall be available to the operator to monitor the effects of the automatic actions.

E10.5 Special attention shall be given to identifying those events, both internal and external to the control room, which may pose a direct threat to its continued *operation*, and the design shall provide for reasonably practicable measures to minimize the effects of such events.

[39] By computer sampling and/or print outs.

E10.6 Depending on the design of the *research reactor*, there shall be sufficient monitoring and control equipment available, preferably at a single location that is physically, electrically and functionally separate from the control room, so that, if the control room is unavailable, the reactor can be placed and maintained in a shut down state, residual heat can be removed from the core and spent fuel storage, and the essential *research reactor* parameters, including the conditions in the spent fuel storages, can be monitored.

Reactor protection system

E10.7 Redundancy and independence designed into the protection system shall be sufficient at least to ensure that:

- no single failure results in loss of protection function; and
- the removal from service of any component or channel does not result in loss of the necessary minimum redundancy.

E10.8 The design shall permit all aspects of functionality of the protection system, from the sensor to the input signal to the final actuator, to be tested in *operation*. Exceptions shall be justified.

E10.9 The design of the *research reactor* protection system shall minimize the likelihood that operator action could defeat the effectiveness of the protection system in normal *operation* and anticipated operational occurrences. Furthermore, the *research reactor* protection system shall not prevent operators from taking correct actions if necessary in design basis accidents.

E10.10 Computer based systems used in a protection system, shall fulfil the following requirements:

- the highest quality of and best practices for hardware and software shall be used;
- the whole development process, including control, testing and commissioning of design changes, shall be systematically documented and reviewed;
- in order to confirm confidence in the reliability of the computer based systems, an assessment of the computer based system by competent persons independent of the designers and suppliers shall be undertaken; and
- where the necessary integrity of the system cannot be demonstrated with a high level of confidence, a diverse means of ensuring fulfilment of the protection functions shall be provided.

Emergency power

E10.11 It shall be ensured that the emergency power supply is able to supply the necessary power to systems and components important to safety, in any operational state or in a design basis accident, on the assumption of a single failure and the coincidental loss of off-site power.

E.11 Review of the design basis

E11.1 The actual design basis shall regularly[40], and when relevant as a result of operating experience and significant new safety information[41], be reviewed, using deterministic methods, complemented where appropriate and available by probabilistic methods as well as engineering judgement to determine whether the design basis is still appropriate. Based on the results of these reviews needs and opportunities for improvements shall be identified and relevant measures shall be implemented.

40 See RL A2.3.

41 Significant new safety information is understood as new insights gained from e.g. site evaluation, safety analyses and the development of safety standards and practices.

06
Issue F: Design Extension of Existing Research Reactors

Safety area: Design

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F1. Objective

F1.1 As part of defence in depth, analysis of Design Extension Conditions (DEC) shall be undertaken with the purpose of further improving the safety of the *research reactor* by:

- enhancing the *research reactor's* capability to withstand more challenging events or conditions than those considered in the design basis,
- minimising radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events or conditions.

F1.2 There are two categories of DEC:

- DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved;
- DEC B with postulated severe fuel damage.

The analysis shall identify reasonably practicable provisions that can be implemented for the prevention of *severe accidents*.[42] Additional efforts to this end shall be implemented for spent fuel storage with the goal that a *severe accident* in such storage becomes extremely unlikely to occur with a high degree of confidence. In addition to these provisions, *severe accidents* shall be postulated for fuel in the core and, if not extremely unlikely to occur with a high degree of confidence, for spent fuel in storage, and the analysis shall identify reasonably practicable provisions to mitigate their consequences.

F2. Selection of design extension conditions

F2.1 A set of DECs shall be derived and justified as representative, based on deterministic assessments and engineering judgement, complemented, where appropriate, by probabilistic methods.

F2.2 The selection process for DEC A shall start by considering those events and combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to severe fuel damage in the core or in the spent fuel storage. It shall cover:

- Events occurring during the defined operational states of the *research reactor*;
- Events resulting from internal or external hazards;
- Common cause failures.

[42] If it is demonstrated that a *severe accident* is physically impossible, reference levels related to the DEC-A can be considered as achieved and DEC B is not applicable.

Where applicable, all reactors and spent fuel storages on the site have to be taken into account. Events potentially affecting all nuclear facilities on the site, potential interactions between them as well as interactions with other sites in the vicinity shall be covered.

F2.3 The set of category DEC B events shall be postulated and justified to cover situations, where the capability of the *research reactor* to prevent severe fuel damage is exceeded or where measures provided are assumed not to function as intended, leading to severe fuel damage.

F3. Safety analysis of design extension conditions

F3.1 The DEC analysis shall:

(a) rely on methods, assumptions or arguments which are justified[43], and should not be unduly conservative;

(b) be auditable, paying particular attention where expert opinion is utilized, and take into account uncertainties and their impact;

(c) identify reasonably practicable provisions to prevent severe fuel damage (DEC A) and mitigate *severe accidents* (DEC B);

(d) evaluate potential on-site and off-site radiological consequences resulting from the DEC (given successful accident management measures);

(e) consider *research reactor* layout and location, equipment capabilities, conditions associated with the selected scenarios and feasibility of foreseen accident management actions;

(f) demonstrate sufficient margins to avoid “cliff-edge effects”[44] that would result in unacceptable consequences, i.e. for DEC-A severe fuel damage and for DEC-B a large or early radioactive release;

(g) reflect insights from PSA according to issue O1.1 or other probabilistic methods;

(h) take into account *severe accident* phenomena, where relevant;

(i) define an end state, which should where possible be a safe state, and, when applicable, associated mission times for *SSCs*.

F4. Ensuring safety functions in design extension conditions

General

F4.1 In DEC A, it is the objective that the *research reactor* shall be able to fulfil, the fundamental safety functions:

- control of reactivity;[45]
- removal of heat from the reactor core and from the spent fuel; and
- *confinement* of radioactive material.

[43] These methods can be more realistic up to best estimate. Modified acceptance criteria may be used in the analysis.

[44] A cliff edge effect occurs when a small change in a condition (a parameter, a state of a system…) leads to a disproportionate increase in consequences.

[45] Preferably, this safety function shall be fulfilled at all times; if it is lost, it shall be re-established after a transient period.

In DEC B, it is the objective that the *research reactor* shall be able to fulfil *confinement* of radioactive material. To this end removal of heat from the damaged fuel shall be established.[46]

F4.2 It shall be demonstrated that *SSCs* (including mobile equipment and their connecting points, if applicable) for the prevention of severe fuel damage or mitigation of consequences in DEC have the capacity and capability and are adequately qualified to perform their relevant functions for the appropriate period of time.

F4.3 If accident management relies on the use of mobile equipment, permanent connecting points, accessible (from a physical and radiological point of view) under DEC, shall be installed to enable the use of this equipment. The mobile equipment, and the connecting points and lines shall be maintained, inspected and tested.

F4.4 The *research reactor* may rely on common services and supplies (if any) that are shared with other facilities. If so, it shall be justified that common resources of personnel, equipment and materials expected to be used in DEC are sufficient and effective for the *research reactor* at all times.

F4.5 The nuclear site shall be autonomous regarding supplies supporting safety functions for a period of time until it can be demonstrated with confidence that adequate supplies can be established from off-site.

Long-term sub-criticality

F4.6 In design extension conditions, sub-criticality of the reactor core shall be ensured in the long term[47] and in the fuel storage at any time.

Heat removal functions

F4.7 There shall be sufficient means including necessary power supplies available to remove the residual heat from the core and the spent fuel. At least one of these means shall be effective after events involving external hazards more severe than design basis events.

***Confinement* functions**

F4.8 Isolation of the reactor building shall be possible in DEC. If this cannot be achieved in due time, severe core damage shall be prevented with a high degree of confidence.

If an event leads to bypass of the reactor building, severe core damage shall be prevented with a high degree of confidence.

F4.9 The threats due to pressure and temperature inside the reactor building shall be managed.

F4.10 The threats due to combustible gases shall be managed.

F4.11 The reactor building shall be protected from overpressure, unless it can be demonstrated that a *severe accident* leading to such an overpressure is physically impossible or extremely unlikely with a high degree of confidence.

If venting is to be used for managing the pressure, adequate filtration shall be provided.

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[46] For the fulfilment (or re-establishment) of the fundamental safety functions in DEC A and DEC B, the use of mobile equipment on-site can be taken into account, as well as support from off-site, with due consideration for the time required for it to be available.

[47] It is acknowledged that in case of DEC B, sub-criticality might not be guaranteed during core degradation and later on during some time in a fraction of the corium.

F4.12 In DEC A, radioactive releases shall be minimised as far as reasonably practicable.

In DEC B, any radioactive release into the environment shall be limited in time and magnitude as far as reasonably practicable to:

(a) allow sufficient time for protective actions (if any) in the vicinity of the *research reactor*; and

(b) avoid contamination of large areas in the long term.

Instrumentation and control for the management of DEC

F4.13 Adequately qualified instrumentation shall be available for DEC for determining the status of the *research reactor* (including spent fuel storage) and safety functions as far as required for making decisions.[48]

F4.14 There shall be an operational and habitable control room (or another suitably equipped location) available during DEC in order to manage such situations.

Emergency power

F4.15 Adequate power supplies during DEC shall be ensured considering the necessary actions and the timeframes defined in the DEC analysis, taking into account external hazards.

F4.16 Batteries shall have adequate capacity to provide the necessary DC power until recharging can be established or other means are in place.

F5. Review of the design extension conditions

F5.1 The design extension conditions shall regularly[49], and when relevant as a result of operating experience and significant new safety information, be reviewed, to determine whether the selection of design extension conditions is still appropriate. The revision shall be performed using a deterministic assessement and engineering judgement, complemented where appropriate by probabilistic methods. Based on the results of these reviews needs and opportunities for improvements shall be identified and relevant measures shall be implemented.

48 This refers to decisions concerning measures on-site as well as, in case of DEC B, off-site.

49 See RL A2.3.

07
Issue G: Safety Classification of Structures, Systems and Components

Safety area: Design

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G1. Objective

G1.1 All *SSCs* important to safety shall be identified and classified on the basis of their importance for safety.

G2. Classification process

G2.1 The classification of *SSCs* shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods (if available) and engineering judgment.

G2.2 The classification shall identify for each safety class:

- The appropriate codes and standards[50] in design, manufacturing, construction and inspection;
- Need for emergency power supply, qualification to environmental conditions;
- The availability or unavailability status of systems serving the safety functions to be considered in deterministic safety analysis;
- The applicable quality requirements.

G3. Ensuring reliability

G3.1 *SSCs* important to safety shall be designed, constructed and maintained such that their quality and reliability is commensurate with their classification.

G3.2 The failure of a *SSC* in one safety class shall not cause the failure of other *SSCs* in a higher safety class. Auxiliary systems supporting equipment important to safety shall be classified accordingly.

G4. Selection of materials and qualification of equipment

G4.1 The design of *SSCs* important to safety and the materials used shall take into account the effects of operational conditions over the lifetime of the *research reactor* and, when required, the effects of accident conditions on their characteristics and performance.

[50] In the case of *structures, systems and components* for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment having similar environmental and operational requirements may be applied, or, in the absence of such codes and standards, the results of experience, tests, analysis or a combination of these may be applied. The use of such a results based approach shall be justified.

G4.2 Qualification procedures shall be adopted to confirm that *SSCs* important to safety meet throughout their design operational lives the demands for performing their function, taking into account environmental conditions[51] over the lifetime of the *research reactor* and when required in anticipated operational occurrences and accident conditions.

[51] Environmental conditions include as appropriate vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, humidity, and combinations thereof.

08
Issue H: Operational Limits and Conditions (OLCs)

Safety area: Operation

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H1. Purpose

H1.1 OLCs shall be developed to ensure that *research reactors* are operated in accordance with design assumptions and intentions as documented in the *Safety Analysis Report* (*SAR*).

H1.2 The OLCs shall define the conditions that must be met to prevent situations that might lead to accidents or to mitigate the consequences of accidents should they occur.

H2. Establishment and review of OLCs

H2.1 Each established OLC shall be justified based on *research reactor* design, safety analysis and commissioning tests.

H2.2 OLCs shall be kept updated and reviewed in the light of experience, the current state of science and technology, and every time modifications in the *research reactor* or in the safety analysis warrant it, and changed if necessary.

H2.3 The process for making modifications or temporary modifications of OLCs shall be defined. Such modifications shall be adequately justified by safety analysis and independent safety review.

H3. Use of OLCs

H3.1 The OLCs shall be readily accessible to control room personnel.

H3.2 Control room operators shall be highly knowledgeable of the OLCs and their technical basis. Relevant operational decision makers shall be aware of their significance for the safety of the *research reactor*.

H4. Scope of OLCs

H4.1 OLCs shall cover all operational *research reactor* states including *operation*, shutdown, and refuelling, any intermediate conditions between these states and temporary situations arising due to maintenance and testing.

H5. Safety limits, safety systems settings and operational limits

H5.1 Adequate margins shall be ensured between operational limits and the established safety systems settings, to avoid undesirably frequent actuation of safety systems.

H5.2 Safety limits shall be established using a conservative approach to take uncertainties in the safety analyses into account.

H6. Unavailability limits

H6.1 Limits and conditions for normal *operation* shall include limits on operating parameters, stipulation for minimum amount of operable equipment, actions to be taken by the *operating personnel* in the event of deviations from the OLCs and time allowed to complete these actions.

H6.2 Where operability requirements cannot be met, the actions to bring the *research reactor* to a safer state shall be specified, and the time allowed to complete the action shall be stated.

H6.3 Operability requirements shall state for the various modes of normal *operation* the number of systems or components important to safety that should be in operating condition or standby condition.

H7. Unconditional requirements

H7.1 If *operating personnel* cannot ascertain that the *research reactor* is operating within operating limits, or the *research reactor* behaves in an unexpected way, measures shall be taken without delay to bring the *research reactor* to a safe and stable state.

H7.2 The *research reactor* shall not be returned to service following unplanned shutdown until it has been shown to be safe to do so.

H8. Staffing levels

H8.1 Minimum staffing levels for shift personnel shall be stated in the OLCs.

H9. Surveillance

H9.1 The licensee shall ensure that an appropriate surveillance[52] program is established and implemented to ensure compliance with OLCs and shall ensure that results are evaluated and retained.

H10. Non-compliance

H10.1 In cases of non-compliance with OLC, remedial actions shall be initiated immediately to re-establish compliance with OLC requirements.

H10.2 Reports of non-compliance shall be investigated and corrective action shall be implemented in order to help prevent such non-compliance[53] in future.

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[52] The objectives of the surveillance programme are: to maintain and improve equipment availability, to confirm compliance with operational limits and conditions, and to detect and correct any abnormal condition before it can give rise to significant consequences for safety. The abnormal conditions which are of relevance to the surveillance programme include not only deficiencies in *SSCs* and software performance, procedural errors and human errors, but also trends within the accepted limits, an analysis of which may indicate that the *research reactor* is deviating from the design intent. The programme prescribes the frequency for the performance of activities and establishes criteria for acceptable deviations. The specification may prescribe the frequency of activities in terms of average intervals with a maximum interval that is not to be exceeded. Deferrals that exceed the maximum interval shall be justified and made subject to approval, and safety measures shall be put in place where necessary.

[53] If the actions taken to correct a deviation from OLCs are not as prescribed, including those times when they have not been completed successfully in the allowable outage time, *research reactor* shall be deemed to have operated in non-compliance with OLCs.

09
Issue I: Ageing Management

Safety area: Operation

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I1. Objectives of ageing management

I1.1 The licensee shall establish suitable organizational and functional arrangements to manage physical ageing[54] and technological obsolescence[55] of in-scope *SSCs* with foresight and anticipation through the entire life time of the *research reactor*, including design, construction, commissioning, *operation* and decommissioning phases. The licensee shall mitigate ageing degradation effects and prevent them, where reasonably practicable.

I1.2 In order to accomplish the safety functions through the entire life time of the *research reactor*, the licensee shall, within the integrated management system:

- Implement an effective overall Ageing Management Programme[56] and
- Address technological obsolescence.

I1.3 The following *SSCs* shall be included in the scope of ageing management:

- *SSCs* important to safety;
- Other *SSCs* whose failure may prevent *SSCs* important to safety from fulfilling their intended functions.

I2. Technical requirements, methods and procedures

I2.1 The ageing management programmes shall identify in a systematic and knowledge-based manner all relevant potential degradation mechanisms and their ageing effects, determine their possible consequences and the necessary activities to ensure and monitor the availability and reliability of in-scope *SSCs*.

I2.2 The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects and to detect, in a timely manner, unexpected behaviour of the in-scope *SSCs* or degradation symptoms. Where necessary, corrective actions shall be taken in a timely manner, taking into account prioritization by safety significance. Acceptance criteria against which the need of corrective actions is evaluated shall be defined.

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[54] Physical ageing is considered as a process by which the physical characteristics of a *structure, system or component* (*SSC*) gradually change with time or use. It occurs due to physical, chemical and/or biological processes (degradation mechanisms).

[55] Technological obsolescence, relates to hardware availability, as a lack of spare parts and technical support (e.g. lack of suppliers) or to industrial policies (e.g. lack of industrial capabilities). Conceptual aspects of obsolescence relate to the evolution of knowledge, standards and regulations; conceptual aspects of obsolescence are not covered in Issue I.

[56] An overall Ageing Management Programme may include several ageing management programmes.

I2.3 The ageing management programmes for *SSCs* shall take into account design basis, manufacture, environmental and process conditions and operating history (duty cycles, maintenance schedules, service life, testing schedules and replacement strategy), as well as the outcomes of Periodic Safety Reviews. Due consideration shall be given to the outcomes from qualification processes for the service life of *SSCs*.

I2.4 In case of specific conditions resulting from e.g. *extended shutdown* the licensee shall implement measures to manage the potential impact on ageing of in-scope *SSCs*.

I2.5 The ageing management programmes shall be regularly reviewed and updated, in order to incorporate new information as it becomes available, to address new issues as they arise, to use adequate and proven tools and methods as they become available and to assess the effectiveness of these programmes.

10
Issue J: System for Investigation of Events and Operational Experience Feedback

Safety area: Operation

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J1. Programmes and Responsibilities

J1.1 The licensee shall establish and conduct a programme to collect, screen, analyse, and document operating experience and events at the *research reactor* in a systematic way. Relevant operational experience and events reported by other *research reactors* and other nuclear installations shall also be considered.

J1.2 Operating experience at the *research reactor* shall be evaluated to identify any latent safety relevant failures or potential precursors and possible tendencies towards degraded safety performance or reduction in safety margin.

J1.3 The licensee shall designate *staff* for carrying out these programmes, for the dissemination of findings important to safety and – where appropriate – for recommendations on actions to be taken. Significant findings and trends shall be reported to the licensee's top management.

J1.4 *Staff* responsible for evaluation of operational experience and investigation into events shall receive adequate training, resources, and support from the line management.

J1.5 The licensee shall ensure that results are obtained, that conclusions are drawn, measures are taken, good practices are considered and that timely and appropriate corrective actions are implemented to prevent recurrence and to counteract developments adverse to safety.

J2. Collection and storage of information

J2.1 The information relevant to experience from normal and abnormal *operation* and other important safety-related information shall be organized, documented, and stored in such a way that it can be easily retrieved and systematically searched, screened and assessed by the designated *staff*.

J3. Reporting and dissemination of safety significant information

J3.1 The licensee shall report events of significance to safety in accordance with established procedures and criteria.

J3.2 All *site personnel* shall be required to report abnormal events and shall be encouraged to report internally near misses relevant to the safety of the *research reactor*.

J3.3 Information resulting from the operational experience shall be disseminated to relevant *site personnel*, shared with relevant national and international bodies.[57]

J3.4 A process shall be put in place to ensure that operating experience of events at the *research reactor* concerned as well as of relevant events at other *research reactors* and other nuclear installations is appropriately considered in the training programme for *site personnel* with tasks related to safety.

J4. Assessment and investigation of events

J4.1 An initial assessment of events important to safety shall be performed without delay to determine whether urgent actions are necessary.

J4.2 The licensee shall have procedures specifying appropriate investigation methods, including methods of human performance analysis.

J4.3 Event investigation shall be conducted on a time schedule consistent with the event significance. The investigation shall:

- Establish the complete event sequence;
- Determine the deviation;
- Include direct and root cause analysis, including causes relating to equipment design, *operation* and maintenance, or to human and organizational factors;
- Assess the safety significance including potential consequences; and
- Identify corrective actions.

J4.4 The operating organisation shall maintain liaison as appropriate with the organizations (manufacturer, research organization, designer) involved in design and construction, with the aims of feeding back information on operating experience and obtaining advice, if necessary, in case of equipment failures or abnormal events.

J4.5 As a result of the analysis, timely corrective actions shall be taken such as technical modifications, administrative measures or personnel training to restore safety, to avoid event recurrence and where appropriate to improve safety.

J5. Review and continuous improvement of the OEF process

J5.1 Periodic reviews of the effectiveness of the OEF process based on performance criteria shall be undertaken and documented either within a self-assessment programme by the licensee or by a peer review team.

[57] Licensees are encouraged to disseminated operating experience with the relevant national and international facilities, especially with the facilities of the same type.

11
Issue K: Maintenance, In-Service Inspection and Functional Testing

Safety area: Operation

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K1. Scope and objectives

K1.1 The licensee shall prepare and implement documented programmes of maintenance, testing, surveillance, and inspection of *SSCs* important to safety to ensure that their availability, reliability, and functionality remain in accordance with the design over the lifetime of the *research reactor*. They shall take into account operational limits and conditions and be re-evaluated in the light of experience.

K1.2 The programmes shall include periodic inspections and tests of *SSCs* important to safety in order to determine whether they are acceptable for continued safe *operation* of the *research reactor* or whether any remedial measures are necessary.

K2. Programme establishment and review

K2.1 The extent and frequency of preventive maintenance, testing, surveillance and inspection of *SSCs* shall be determined through a systematic approach on the basis of:

- Their importance to safety;
- Their inherent reliability;
- Their potential for degradation (based on operating experience, research and vendor recommendation);
- Operational and other relevant experience and results of condition monitoring.

K2.2 In-service inspections of *research reactors* shall be carried out at intervals whose length shall be chosen in order to ensure that any deterioration of the most exposed component is detected before it can lead to failure.

K2.3 Data on maintenance, testing, surveillance, and inspection of *SSCs* shall be recorded, stored and analysed. Such records shall be reviewed to look for evidence of incipient and recurring failures, to initiate corrective maintenance and review the preventive maintenance programme accordingly.

K2.4 The maintenance programme shall be periodically reviewed[58] in light of operating experience, and any proposed changes to the programme shall be assessed to analyse their effects on system availability, their impact on *research reactor* safety, and their conformance with applicable requirements.

K2.5 The potential impact of maintenance upon *research reactor* safety shall be assessed.

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[58] It is anticipated that such reviews are carried out more frequently than the 10-yearly Periodic Safety Reviews.

K3. Implementation

K3.1 *SSCs* important to safety shall be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the *research reactor*, without undue risk to *workers* and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods shall be specified and adequate safety precautions taken to compensate for potential undiscovered failures.

K3.2 Procedures shall be established, reviewed, and validated for maintenance, testing, surveillance and inspection tasks.

K3.3 An adequate work planning and control system shall be implemented to ensure that maintenance, testing, surveillance and inspection work is properly authorized and carried out according to the procedures.

K3.4 Before equipment (including *experimental devices*) is removed from or returned to service, full consideration and approval of the proposed reconfiguration shall be ensured, followed by a documented confirmation of its correct configuration and, where appropriate, functional testing.

K3.5 The actions to be taken in response to deviations from the acceptance criteria in the maintenance, testing, surveillance and inspection tasks, shall be defined in the procedures.

K3.6 Repairs to *SSCs* shall be devised, authorized, and carried out as promptly as practicable. Priorities shall be established with account taken first of the relative importance to safety of the defective structure, system, or component.

K3.7 Following any event due to which the safety functions and functional integrity of any *SSCs* may have been challenged, the licensee shall identify and revalidate the safety functions and carry out any necessary remedial actions, including inspection, testing, maintenance, and repair, as appropriate.

K3.8 The primary coolant boundary shall be subject to a system leakage surveillance (leakage test if possible) at least after an outage in the course of which its leak-tightness may have been affected.

K3.9 All items of equipment used for examinations and tests together with their accessories shall be qualified and calibrated before they are used. All equipment shall be properly identified in the calibration records, and the validity of the calibration shall be regularly verified by the licensee in accordance with requirements of the management system.

K3.10 Any in-service inspection (ISI) process shall be qualified[59], in terms of required inspection area(s), method(s) of non-destructive testing, defects being sought and required effectiveness of inspections.

K3.11 When a detected flaw that exceeds the acceptance criteria is found in a sample, additional examinations shall be performed to investigate the specific problem area in the analysis of additional analogous components (or areas). The extent of further examinations shall be decided with due regard for the nature of the flaw and degree to which it affects the nuclear safety assessments for the *research reactor* or component and the potential consequences.

[59] See e.g. IAEA TECDOC No. 1263 “Application of Non-Destructive Testing and In-service Inspection to Research Reactors”

K3.12 Surveillance measures shall be carried out to verify the integrity and efficiency of the *confinement* function of the reactor building. The surveillance activities shall include (where appropriate):

(a) leak tests;

(b) tests of penetration seals and closure devices;

(c) inspection and testing of ventilation system and high efficiency filters;

(d) inspections for structural integrity.

12
Issue LM: Emergency Operating Procedures and Severe Accident Management Guidelines

Safety area: Operation

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LM1. Objectives

LM1.1 A comprehensive set of procedures and guidelines, including emergency operating procedures (EOPs) and, if needed[60], *severe accident* management guidelines (SAMGs) shall be provided, covering accident conditions initiated during all operational states.

LM2. Scope

LM2.1 EOPs shall be provided to cover Design Basis Accidents. These EOPs shall provide instructions for recovering the *research reactor* to a safe condition.

LM2.2 EOPs, with other specific procedures or guidelines when applicable, shall be provided to cover DEC A. The aim shall be to re-establish or compensate for lost safety functions and to set out actions to prevent severe fuel damage in the core or in the spent fuel storage.

LM2.3 SAMGs with other specific procedures or guidelines when applicable, shall be provided to mitigate the consequences of *severe accidents* for the cases where the responses to events including the measures provided by EOPs have not been successful in the prevention of such accidents.

LM2.4 EOPs for design basis accidents shall be symptom based or a combination of symptom based and event based procedures. EOPs for DEC A shall be symptom based unless an event based approach can be justified.

LM2.5 The set of procedures and guidelines shall be suitable to manage accident conditions that simultaneously affect the reactor, *experimental devices* and spent fuel storages, and shall take potential interactions between reactor, *experimental devices* or spent fuel storages into account.

LM2.6 Possibilities for one *research reactor*, without compromising its safety, supporting another *research reactor* in *operation* on the site shall be covered by the set of procedures and guidelines.

LM2.7 The set of procedures and guidelines shall be such that they are able to be implemented even if all nuclear installations on a site are under accident conditions, taking into account the dependencies between the systems and common resources.

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[60] If it is demonstrated that a *severe accident* is physically impossible, reference levels related to the SAMGs are not applicable (see also F1.2).

LM3. Format and Content of Procedures and Guidelines

LM3.1 EOPs shall be developed in a systematic way and shall be supported by realistic and *research reactor* specific analysis performed for this purpose. EOPs shall be consistent with other operational procedures, such as alarm response procedures and, if any, *severe accident* management guidelines.

LM3.2 EOPs shall enable the operator to recognise quickly the accident condition to which it applies. Entry and exit conditions shall be defined in the EOPs to enable operators to select the appropriate EOP, to navigate among EOPs and, if needed, to proceed from EOPs to SAMGs.

LM3.3 SAMGs, if needed, shall be developed in a systematic way using a *research reactor* specific approach. SAMGs shall address strategies to cope with scenarios identified by the *severe accident* analyses.

LM3.4 EOPs for design basis accidents shall rely on adequately qualified equipment and instrumentation. EOPs for DEC and SAMGs shall primarily rely on adequately qualified equipment.

LM3.5 The set of procedures and guidelines shall consider the anticipated on-site conditions, including radiological conditions, associated with the accident conditions they are addressing and the initiating event or hazard that might have caused it.

LM4. Verification and validation

LM4.1 The set of procedures and guidelines shall be verified and validated in the form in which they will be used in the field, as far as practicable, to ensure that they are administratively and technically correct for the *research reactor*, are compatible with the environment in which they will be used[61] and with the human resources available.

LM4.2 The approach used for *research reactor* specific validation and verification shall be documented. The effectiveness of incorporating human factors engineering principles in procedures and guidelines shall be judged when validating them. The validation of EOPs shall be based on representative simulations.

LM5. Review and updating

LM5.1 The set of procedures and guidelines shall be kept updated to ensure that they remain fit for their purpose.

LM6. Training and exercises

LM6.1 Control room personnel shall be regularly trained and exercised in using EOPs and, if any, SAMGs.

LM6.2 *Emergency workers* shall be regularly trained and exercised, commensurate with their expected role in managing an emergency, for situations and conditions covered by the set of procedures and guidelines.

LM6.3 If SAMGs exist, the transition from EOPs to SAMGs for management of *severe accidents* shall be regularly exercised.

[61] In particular, expected manual operation of equipment shall be possible.

LM6.4 Interventions called for in the set of procedures and guidelines and needed to restore necessary safety functions, including those which may rely on mobile or off-site equipment, shall be planned for and regularly exercised. The potential unavailability of instruments, lighting and power and the use of protective equipment shall be considered.

13
Issue N: Contents and Updating of Safety Analysis Report (SAR)

Safety area: Safety Verification

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N1. Objective

N1.1 The licensee shall provide a *SAR* to demonstrate that the *research reactor* fulfils relevant safety requirements and use it as a basis for continuous support of safe *operation*.

N1.2 The licensee shall use the *SAR* as a basis for assessing the safety implications of changes to the *research reactor*, or to operating practices.

N2. Content of the *SAR*

N2.1 The *SAR* shall describe the site, the *research reactor* layout, normal *operation* and utilization and demonstrate how safety is achieved.[62]

N2.2 The *SAR* shall contain detailed descriptions of the safety functions; all safety systems and safety-related structures, systems, components; their design basis and functioning in all operational states, including shut down and accident conditions.

N2.3 The *SAR* shall identify applicable regulations codes and standards.

N2.4 The *SAR* shall describe the relevant aspects of the *research reactor* organization and the management of safety.

N2.5 The *SAR* shall contain the evaluation of the safety aspects related to the site.

N2.6 The *SAR* shall outline the general design concept and the approach adopted to meet the fundamental safety objectives.

N2.7 The *SAR* shall include justification that it adequately demonstrates that the *research reactor* fulfils relevant safety requirements. The *SAR* shall describe the safety analyses performed to assess the safety of the *research reactor* in response to anticipated operational occurrences, design basis accidents and design extension conditions against safety criteria and radiological release limits. Safety margins shall be described.

N2.8 The *SAR* shall describe the emergency *operation* procedures and *severe accident* management guidelines, if any, the inspection and testing provisions, the qualification, and training of personnel, the operational experience feedback programme, and the management of ageing.

N2.9 The *SAR* shall contain the technical bases for the operational limits and conditions.

[62] Guidance on the specific aspects that need to be addressed in the *SAR* is given in IAEA Safety Standards Series No. SSG-20.

N2.10 The *SAR* shall describe the policy, strategy, methods, and provisions for radiation protection.

N2.11 The *SAR* shall describe the on-site emergency preparedness arrangements and the liaison and co-ordination with off-site organizations involved in the response to an emergency.

N2.12 The *SAR* shall describe the on-site radioactive waste management provisions.

N2.13 The *SAR* shall describe how the relevant decommissioning and end-of-life aspects are taken into account during *operation*.[63]

N2.14 The descriptions, assessments and arrangements mentioned in the *SAR* shall consider the site as a whole, to take into account hazards:

- which may challenge all installations within a short period of time;
- which arise from harmful interactions between installations.

N3. Review and update of the *SAR*

N3.1 The licensee shall update the *SAR* to reflect modifications, changes in utilization, new regulatory requirements, new information relevant for the safety assessment (including those related to characteristics of the site and the site environment), and relevant standards, in a timely manner after the new information is available and applicable.

[63] Guidance on the specific aspects that need to be addressed in the *SAR* is given in IAEA Safety Standards Series No. SSG-20

14 Issue O: Probabilistic Safety Analysis (PSA)

Safety area: Safety Verification

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O1. Scope

O1.1 A PSA shall be considered as a complementary tool to deterministic method to determine significant contributing factors to the radiological risks arising from the *research reactor* and to evaluate the extent[64] to which the overall design is well balanced.

[64] The development of a PSA is encouraged, as it may contribute to enhancing the safety of the *research reactor*. The PSA is indeed complementary to the deterministic approach, e.g. identifying ways of improvements. See also the dedicated WENRA guidance "Issue O: Probabilistic Safety Assessment of Existing Research Reactors".

15
Issue P: Periodic Safety Review (PSR)

Safety area: Safety Verification

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P1. Objective of the periodic safety review

P1.1 The licensee shall have the prime responsibility for performing the Periodic Safety Review.

P1.2 The review shall confirm the compliance of the *research reactor* with its licensing basis and any deviations shall be resolved.

P1.3 The review shall identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices taking into account operating experience, relevant research findings, and the current state of technology.

P1.4 All reasonably practicable improvement measures shall be implemented by the licensee as a result of the review, in a timely manner.

P1.5 An overall assessment of the safety of the *research reactor* covering the period until the next PSR shall be provided, and adequate confidence in *research reactor* safety for continued *operation* demonstrated, based on the results of the review in each area. This assessment shall highlight any issues that might limit the future safe *operation* of the *research reactor* and explain how they will be managed.

P2. Scope of the periodic safety review

P2.1 The review shall be made periodically, at least every ten years.

P2.2 The scope of the review shall be clearly defined and justified. The scope shall be as comprehensive as reasonably practical with regard to significant safety aspects of an operating *research reactor* and, as a minimum the following safety factors shall be covered by the review[65,]:

(a) *Research reactor* design;

(b) Actual condition of *structures, systems and components* (*SSCs*) important to safety;

(c) Equipment qualification;

(d) Ageing;

(e) Utilization;

(f) Deterministic safety analysis;

(g) Probabilistic safety assessment, if available (see O1.1);

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[65] Radiation protection is not regarded as a separate safety factor since it is related to most of the other safety factors. Alternatively, it may be decided to review radiation protection as a separate safety factor.
As far as there are other nuclear facilities at the site, interactions between them should also be covered by the review.

(h) Hazard analysis;

(i) Safety performance;

(j) Use of experience from other *research reactors*, NPPs, other facilities and research findings;

(k) Organization, the management system and safety culture;

(l) Procedures;

(m) Human factors;

(n) Emergency planning;

(o) Radiological impact on the environment.

P3. Methodology of the periodic safety review

P3.1 The review shall use an up to date, systematic, and documented methodology, taking into account deterministic assessment as well as probabilistic methods, if available.

P3.2 Each area shall be reviewed and the findings compared to the licensing requirements as well as to current safety standards and practices. The safety significance of all findings shall be evaluated using an appropriate approach. A global assessment shall consider all findings (positive and negative) and their cumulative effect on safety, and shall identify what safety improvements are reasonably practicable.

16
Issue Q: Research Reactor Modifications

Safety area: Operation

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Q1. Purpose and scope

Q1.1 The licensee shall ensure that no modification to a *research reactor*, whatever the reason for it, degrades the *research reactor's* ability to be operated safely.[66]

Q1.2 The licensee shall control *research reactor* modifications using a graded approach with appropriate criteria for categorization according to their safety significance.[67]

Q2. Procedure for dealing with plant modifications

Q2.1 The licensee shall establish a process to ensure that all permanent and temporary modifications are properly designed, reviewed, controlled, and implemented, and that all relevant safety requirements are met.

Q2.2 For modifications to SSC, this process shall include the following:

- Reason and justification for modification;
- Design;
- Safety assessment;
- Updating *research reactor* documentation and training;
- Fabrication, installation and testing; and
- Commissioning the modification.

Q3. Requirements on safety assessment and review of modifications

Q3.1 An initial safety assessment shall be carried out to determine any consequences for safety.[68]

Q3.2 A detailed, comprehensive safety assessment shall be undertaken, unless the results of the initial safety assessment show that the scope of this assessment can be reduced.

Q3.3 Comprehensive safety assessments shall demonstrate all applicable safety aspects are considered and that the system specifications and the relevant safety requirements are met.

Q3.4 The scope, safety implications, and consequences of proposed modifications shall be reviewed by competent persons not immediately involved in their design or implementation.

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[66] RL Q2.2 specifically addresses modifications to *SSCs*, all other RLs relate to all type of modifications in the sense of IAEA Safety Standards Series No. SSR-3 (2016)

[67] IAEA Safety Standards Series No. SSG-24 contains information about possible categories.

[68] This assessment is performed for the purpose of categorizing the intended modification according to its safety significance.

Q4. Implementation of modifications

Q4.1 Implementation and testing of *research reactor* modifications shall be performed in accordance with the applicable work control and *research reactor* testing procedures.

Q4.2 The impact upon procedures and training, shall be assessed and any appropriate revisions incorporated.

Q4.3 Before commissioning a modified *research reactor* or putting a *research reactor* back into *operation* after modification, *operating personnel* and other *site personnel*, as appropriate, shall have been trained and all relevant documents necessary for *research reactor operation* shall have been updated.

Q5. Temporary modifications[69]

Q5.1 All temporary modifications shall be clearly identified at the point of application and at any relevant control position.[70] *Operating personnel* shall be clearly informed of these modifications and of their consequences for the *operation* of the *research reactor*.

Q5.2 Temporary modifications shall be managed according to specific *research reactor* procedures.

Q5.3 The number of simultaneous temporary modifications shall be kept to a minimum. The duration of a temporary modification shall be limited.

Q5.4 The licensee shall periodically review outstanding temporary modifications to determine whether they are still needed.

[69] Examples of temporary modifications are temporary bypass lines, electrical jumpers, lifted electrical leads, temporary trip point settings, temporary blank flanges, temporary defeats of interlocks and *experimental devices*. This category of modifications also includes temporary constructions and installations used for maintenance of the design basis configuration of the plant in emergencies or other unanticipated situations. Temporary modifications in some cases may be made as an intermediate stage in making permanent modifications.

[70] By relevant control position it is meant any control point important for the modified system and also any administrative aspect related to the system in which the temporary modification has been implemented.

17 Issue R: On-site Emergency Preparedness

Safety area: Emergency Preparedness

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R1. Objective

R1.1 The licensee shall provide arrangements for responding effectively to events requiring protective measures at the scene for:

(a) Controlling an emergency situation arising at their site, following any reasonably foreseeable event, including events related to combinations of hazards as well as events involving all nuclear installations and facilities on the site;

(b) Preventing or mitigating the consequences at the scene of any such emergency; and

(c) Co-operating with external emergency response organizations in preventing adverse health effects to *workers*, all other persons on the site and the public.

R2. Emergency Preparedness and Response Plan

R2.1 The licensee shall prepare an on-site emergency plan and establish the necessary organizational structure for clear allocation of responsibilities, authorities, and arrangements for co-ordinating internal activities and co-operating with external response agencies in a timely manner and throughout all phases of an emergency. The emergency plan shall be commensurate with the hazard assessed and the potential consequences of an emergency should it occur.

R2.2 The licensee shall provide for:

(a) Prompt recognition and classification of emergencies, consistent with the criteria set for alerting the appropriate authorities;

(b) Timely notification and alerting of *emergency workers*;

(c) Ensuring the safety of all persons present on the site, including the protection of the *emergency workers*;

(d) Informing the authorities and the public, including timely notification and subsequent provision of information as required;

(e) Performing assessments of the current and foreseeable situation on the technical and radiological points of view (on- and off-site);

(f) If relevant, monitoring radioactive releases;

(g) Treatment and first aid of a limited number of contaminated and/or overexposed *workers*/persons on site; and

(h) *Research reactor* management and damage control.[71]

R2.3 The emergency plan shall be based upon an assessment of reasonably foreseeable events and situations that may require protective measures on- or off-site. The plan shall:

- clarify how the resources (human and material) are used (including these at the site and being common to several installations);
- be co-ordinated with all other involved bodies;
- address long-lasting situations if appropriate.

The plan shall be capable of extension, should more severe events occur.

R3. Organization

R3.1 During *operation*, the licensee shall have *staff* on-site at all times with the authority and responsibilities to classify and declare an emergency and, upon classification, to initiate promptly the appropriate on-site response.[72] For shutdown state, if it is justified by safety analysis, it may be sufficient to have an on-call duty to reach the authorized and responsible *staff*.

R3.2 Sufficient numbers of qualified personnel shall be available at all times for staffing appropriate positions promptly following the declaration and notification of an emergency. If needed, arrangements shall be established to ensure that sufficiently qualified *site personnel* can *staff* appropriate emergency positions in long-lasting situations.[73]

R3.3 Arrangements shall be made to provide technical assistance to *operating personnel*. Teams for mitigating the consequences of an emergency (e.g. radiation protection, damage control, fire fighting, etc.) shall be available.

R3.4 Arrangements shall be made to alert off-site responsible authorities promptly.

R3.5 The licensee shall identify those who are authorized to carry out the response functions assigned in the emergency plan.

R3.6 The licensee emergency response shall be functional in cases where infrastructures at the site and around the site are severely disrupted.[74]

R3.7 Any necsessary arrangements to support on-site actions shall be in place with considerations for large-scale destruction of infrastructure in the vicinity of the site due to external hazards.

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71 Understood as urgent mitigatory repairs, controls, and other actions that are carried out, primarily at the site, while the emergency is still in progress.

72 The on duty shift supervisor could be among those authorised to declare an emergency and to initiate the appropriate on-site response. For *research reactors* that do not operate continuously 24 hours per day, it may be sufficient to have an on-call duty to reach the responsible personnel.

73 For *research reactors* that do not require a permanent presence of *operating personnel* any necessary arrangements with off-site organisations shall be established. See also T5.6.

74 For *research reactors* with higher risk potential, these are basically the same measures as for NPPs, e.g. (own) *emergency workers*, separate (at sufficient distance from the reactor) *emergency control centre*, ensuring emergency water and power supply (sufficient number of redundancies; mobile devices), redundant monitoring of reactor parameters.
For *research reactors* with lower risk potential, these actions may be correspondingly less extensive, e.g. they may have contract with external organisations for disaster control, and may have less redundancies and less capacities for water and power supply.
See also LM 3.1, LM 3.2, LM3.5, LM4.1.

R4. Facilities and equipment

R4.1 Appropriate emergency facilities shall be designated for responding to events on site and that will provide co-ordination of off-site monitoring and assessment throughout different phases of an emergency response.

R4.2 A room separate from the control room shall be available as an emergency control centre. Important information about the *research reactor* and radiological conditions on and around the site shall be available there. The emergency control centre shall have means of communicating with the control room, any supplementary control room, other important points on site, and with the on-site and off-site emergency response organizations.

R4.3 Emergency facilities shall be suitably located, designed and protected to

- remain operational for accident conditions to be managed (including design extension conditions) from these facilities;
- allow the protection from radiation as well as control of radiation exposure of *emergency workers*;
- appropriate measures shall be taken to protect those occupying emergency facilities for a protracted time from hazards resulting from accident conditions.[75]

R4.4 Instruments, tools, equipment, documentation, and communication systems for use in emergencies (including necessary mobile equipment and consumables such as fuel, lubrication oil etc.), whether located on-site or off-site, shall be stored, maintained, tested and inspected sufficiently frequently so that they will be available and operational during DBA and DEC. Access to these storage locations shall be possible even in case of extensive infrastructure damage.

R5. Training, drills and exercises

R5.1 Arrangements shall be made to identify the knowledge, skills, and abilities needed for *site personnel* and other external organizations, if these are involved in the emergency preparedness to perform their assigned response functions.

R5.2 Arrangements shall be made to inform all *site personnel* of the actions to be taken in the event of an emergency.

R5.3 Training arrangements shall include basic emergency training and ongoing refresher training on an appropriate schedule and shall ensure that *emergency workers* and other external organizations, if these are included in the emergency preparedness, meet the training obligations.

R5.4 The site emergency plan shall be regularly exercised, with a frequency depending on the risk profile of the *research reactor*. Some exercises shall be integrated to include as many as possible of the off-site organizations concerned. For sites with multiple nuclear installations, some exercises shall address situations affecting multiple facilities on the site. Exercises shall also include the use and connection of mobile equipment, if any.

[75] This refers, primarily, to ensuring that the *Emergency Control Centre* and other locations where *staff* are expected to spend a significant time are located somewhere that the *staff* can reach and work throughout an extended emergency with minimum risk to health. This will require location away from areas that are likely to be damaged or affected by radiation fields and, where appropriate, this will include provision of re-circulatory air conditioning and continuous radiation monitoring systems.

R5.5 Emergency exercises shall be evaluated systematically, and the emergency preparedness arrangements and the plan shall be subject to review and updating in the light of experience gained.

18 Issue S: Protection against Internal Fires

Safety area: Emergency Preparedness

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S1. Fire safety objectives

S1.1 The licensee shall implement the defence in depth principle to fire protection, providing measures to prevent fires from starting, to detect and extinguish quickly any fires that do start and to prevent the spread of fires and their effects in or to any area that may affect safety.[76]

S2. Basic design principles

S2.1 *SSCs* important to safety shall be designed and located so as to minimize the frequency and the effects of fire and to maintain capability for shutdown, residual heat removal, *confinement* of radioactive material and monitoring of state of the *research reactor* during and after a fire event.

S2.2 Buildings that contain *SSCs* important to safety shall be suitably[77] fire resistant.

S2.3 Buildings that contain equipment that is important to safety shall be subdivided into compartments that segregate such items from fire loads and segregate redundant or diverse trains of a safety system from each other. When a *fire compartment* approach is not practicable, *fire cells* shall be used[78], providing a balance between passive and active means, as justified by fire hazard analysis.

S2.4 Buildings that contain radioactive materials that could cause radioactive releases in case of fire shall be designed to minimize such releases.

S2.5 Access and escape routes for fire fighting and *operating personnel* shall be available.

S3. Fire hazard analysis

S3.1 A fire hazard analysis shall be carried out and kept updated to demonstrate that the fire safety objectives are met, that the fire design principles are satisfied, that the fire protection measures are appropriately designed and that any necessary administrative provisions are properly identified.

S3.2 The fire hazard analysis shall be developed on a deterministic basis, covering at least:

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76 In this context, safety refers to all sources of nuclear safety risk, including radioactive waste facilities.

77 In accordance with the results of the fire hazard analysis.

78 In the *fire cell* approach the spread of fire is avoided by substituting the fire resistant barriers primarily with other passive provisions (e.g. distance, thermal insulation, etc.), that take into account all physical and chemical phenomena that can lead to propagation. Provision of active measures (e.g. fire extinguishing systems) may also be needed in order to achieve a satisfactory level of protection. The achievement of a satisfactory level of protection is demonstrated by the results of the fire hazard analysis.

- For all normal operating and shutdown states, a single fire and consequential spread, anywhere that there is fixed or transient combustible material;
- Consideration of credible combination of fire and other PIEs likely to occur independently of a fire;
- Fire hazards due to *experiments.*

S3.3 The fire hazard analysis shall demonstrate how the possible consequential effects of fire and extinguishing systems operation have been taken into account.

S4. Fire protection systems

S4.1 Each *fire compartment* or *fire cell* shall be equipped with fire detection and alarm features, with detailed annunciation for the control room *staff* of the location of a fire. These features shall be provided with non-interruptible emergency power supplies and appropriate fire resistant supply cables.

S4.2 Fixed or mobile, automated or manual extinguishing systems shall be installed. They shall be designed and located so that their rupture, spurious or inadvertent *operation* does not significantly impair the capability of *SSCs* important to safety to carry out their safety functions.

S4.3 The distribution loop for fire hydrants outside building and the internal standpipes shall provide adequate coverage of areas of the *research reactor* relevant to safety. The coverage shall be justified by the fire hazard analysis.

S4.4 Ventilation systems shall be arranged such that each *fire compartment* fully fulfils its segregation purpose in case of fire.

S4.5 Parts of ventilation systems (such as connecting ducts, fan rooms and filters) that are located outside *fire compartments* shall have the same fire resistance as the compartment or be capable of isolation from it by appropriately rated fire dampers.

S5. Administrative controls and maintenance

S5.1 In order to prevent fires, procedures shall be established to control and minimize the amount of combustible materials and minimize the potential ignition sources that may affect items important to safety. In order to ensure the operability of the fire protection measures, procedures shall be established and implemented. They shall include inspection, maintenance and testing of fire barriers, fire detection and extinguishing systems.

S6. Fire fighting organization

S6.1 The licensee shall implement adequate arrangements for controlling and ensuring fire safety, as identified by the fire hazard analysis.[79]

[79] Such arrangements must include nominating persons to be responsible for or have duties with respect to fire protection. The arrangements must set out the requirements for control of all activities that can have impact on fire safety, e.g. maintenance; control of materials; training; tests and drills; modifications to layouts and systems – such as fire detection, fire extinguishing, ventilation, electrical and control systems.

S6.2 Written emergency procedures that clearly define the responsibility and actions of *site personnel* in responding to any fire in the *research reactor* shall be established and kept up to date. A fire fighting strategy shall be developed, kept up-to date, and trained for, to cover each area in which a fire might affect items important to safety and protection of radioactive materials.

S6.3 When reliance for manual fire fighting capability is placed on an offsite resource, there shall be proper coordination between the licensee's response group and the off-site response group, in order to ensure that the latter is familiar with the hazards of the *research reactor*.

S6.4 If *site personnel* is required to be involved in fire fighting, their organization, minimum staffing level, equipment, fitness requirements, and training shall be documented and their adequacy shall be confirmed by a competent person.

19
Issue T: Natural Hazards

Safety area: Design

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T1. Objective

T1.1 Natural hazards shall be considered an integral part of the safety demonstration of the *research reactor* (including spent fuel storage). Threats from natural hazards shall be removed or minimized as far as reasonably practicable for all operational *research reactor* states. The safety demonstration in relation to natural hazards shall include assessments of the design basis and design extension conditions[80] with the aim to identify needs and opportunities for improvement.

T2. Identification of natural hazards

T2.1 All natural hazards that might affect the site shall be identified, including any related hazards (e.g. earthquake and tsunami). Justification shall be provided that the compiled list of natural hazards is complete and relevant to the site.

T2.2 Natural hazards shall include:

- Geological hazards;
- Seism tectonic hazards;
- Meteorological hazards;
- Hydrological hazards;
- Biological phenomena;
- Forest fire.

T3. Site specific natural hazard screening and assessment

T3.1 Natural hazards identified as potentially affecting the site can be screened out on the basis of being incapable of posing a physical threat, incapable to produce an event that could lead to radiological releases or being extremely unlikely with a high degree of confidence. Care shall be taken not to exclude hazards which in combination with other hazards[81] have the potential to pose a threat to the *research reactor*. The screening process shall be based on conservative assumptions. The arguments in support of the screening process shall be justified.

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[80] Design extension conditions could result from natural events exceeding the design basis events or from events leading to conditions not included in the design basis accidents.

[81] This could include other natural hazards, internal hazards or human induced hazards. Consequential hazards and causally linked hazards shall be considered, as well as random combinations of relatively frequent hazards.

T3.2 For all natural hazards that have not been screened out, hazard assessments[82] shall be performed using deterministic and, as far as practicable, probabilistic methods taking into account the current state of science and technology. This shall take into account all relevant available data, and produce a relationship between the hazards severity (e.g. magnitude and duration) and exceedance frequency, where practicable. The maximum credible hazard severity shall be determined where this is practicable.

T3.3 The following shall apply to hazard assessments:

- The hazard assessment shall be based on all relevant site and regional data. Particular attention shall be given to extending the data available to include events beyond recorded and historical data.
- Special consideration shall be given to hazards whose severity changes during the expected lifetime of the *research reactor*.
- The methods and assumptions used shall be justified. Uncertainties affecting the results of the hazard assessments shall be evaluated.

T4. Definition of the design basis events

T4.1 Design basis events[83] shall be defined based on the site specific hazard assessment.

T4.2 The exceedance frequencies of design basis events shall be low enough to ensure a high degree of protection with respect to natural hazards.

A common target value of frequency, not higher than 10^{-4} per annum, shall be used for each design basis event. If higher values of frequency are used for the definition of a design basis event, these shall be justified, taking into account the hazard potential of the *research reactor*. For the specific case of seismic loading, as a minimum, a horizontal peak ground acceleration value of 0.1g (where 'g' is the acceleration due to gravity) shall be applied, even if its exceedance frequency would be below the common target value.

Where it is not possible to calculate the exceedance frequencies of the design basis events with a sufficient degree of certainty, or the effort is disproportionate to the hazard potential of the *research reactor*, an event shall be chosen to reach an equivalent level of safety. This event may be a covering event which ensures a high degree of protection. The definition of a design basis event shall be justified and appropriate to the risk presented by the reactor. The definition of a design basis event shall provide a sufficient degree of certainty to reach an acceptable level of safety.

T4.3 The design basis events shall be compared to relevant historical data to verify that historical extreme events are enveloped by the design basis with a sufficient margin.

T4.4 Design basis parameters shall be defined for each design basis event taking due consideration of the results of the hazard assessments. The design basis parameter values shall be developed on a conservative basis.

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[82] For *research reactors* with low potential hazard, the amount of detail to be provided can be substantially reduced below that required for a medium power research reactor or high power *research reactors*.

[83] These design basis events are individual natural hazards or combinations of hazards (causally or non-causally linked). The design basis may either be the original design basis of the *reseach reactor* (when it was commissioned) or a reviewed design basis for example following a PSR.

T5. Protection against design basis events

T5.1 Protection shall be provided for design basis events.[84] A protection concept[85] shall be established to provide a basis for the design of suitable protection measures.

T5.2 The protection concept shall be of sufficient reliability that the fundamental safety functions are conservatively ensured for any direct and credible indirect effects of the design basis event.

T5.3 The protection concept shall:

(a) apply reasonable conservatism providing safety margins in the design;

(b) rely primarily on passive measures as far as reasonable practicable;

(c) ensure that sufficient measures to cope with a design basis event remain effective during and following this event;

(d) take into account the predictability and development of the event over time;

(e) ensure that procedures and means are available to verify the *research reactor* condition during and following design basis events;

(f) consider that events could simultaneously challenge several safety system, multiple *SSCs* or several nuclear facilities at multi-nuclear facilities sites, site and regional infrastructure, external supplies and other countermeasures;

(g) ensure that sufficient resources remain available at multi-unit sites considering the use of common equipment or services;

(h) not adversely affect the protection against other design basis events (not originating from natural hazards).

T5.4 For design basis events, *SSCs* identified as part of the protection concept with respect to natural hazards shall be considered as important to safety.

T5.5 Monitoring and alert processes shall be available to support the protection concept. Where appropriate, thresholds (intervention values) shall be defined to facilitate the timely initiation of protection measures. In addition, thresholds shall be identified to allow the execution of pre-planned post-event actions (e.g. inspections).

T5.6 During long-lasting natural events, arrangements for the replacement of *operating personnel* and supplies shall be available, if needed for keeping the *research reactor* in a safe state. For *research reactors* that do not require a permanent presence of *operating personnel*, measures shall be taken to ensure a correct surveillance of the reactor as long as needed.

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[84] If the hazard levels of RL T4.2 for seismic hazards were not used for the initial design basis of *the reseach reactor* and if it is not reasonably practicable to ensure a level of protection equivalent to a reviewed design basis, methods such as those mentioned in IAEA Safety Standards Series No. NS-G-2.13 (2009) may be used. This shall quantify the seismic capacity of the *research reactor*, according to its actual condition, and demonstrate the *research reactor* is protected against the seismic hazard established in RL T4.2.

[85] A protection concept, as meant here, describes the overall strategy followed to cope with natural hazards. It shall encompass the protection against design basis events, events exceeding the design basis and the links into EOPs and SAMGs.

T6. Considerations for events more severe than the design basis events

T6.1 Events that are more severe than the design basis events shall be identified as part of DEC analysis. Their selection shall be justified.[86] Further detailed analysis of an event will not be necessary, if it is shown that its occurrence can be considered with a high degree of confidence to be extremely unlikely.

T6.2 To support identification of events and assessment of their effects, the hazards severity as a function of exceedance frequency or other parameters related to the event shall be developed, when practicable.

T6.3 When assessing the effects of natural hazards included in the DEC analysis, and identifying reasonably practicable improvements related to such events, analysis shall, as far as practicable, include:

(a) demonstration of sufficient margins to avoid “cliff-edge effects” that would result in loss of a fundamental safety function;

(b) identification and assessment of the most resilient means for ensuring the fundamental safety functions;

(c) consideration that events could simultaneously challenge several safety systems, multiple *SSCs* or several nuclear facilities at multi-nuclear facilities sites, site and regional infrastructure, external supplies and other countermeasures;

(d) demonstration that sufficient resources remain available at multi-nuclear facilities considering the use of common equipment or services;

(e) on-site verification (typically by walk-down methods).

[86] See issue F section 2.

20 Issue X: *Experimental Devices* and *Experiments*

Safety area: Design

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X1. Objective

X1.1 *Experimental devices* for a *research reactor* shall be designed so that they will not adversely affect the safety of the *research reactor* in any operational states or accident conditions.

X1.2 Neither the assembly, the insertion into or removal from the core, the *operation*, the maintenance, nor the failure of an *experimental device* shall compromise:

- the control of Reactivity;
- reactor protection system;
- cooling capacity;
- means of *confinement*

or lead to unacceptable radiological consequences.

X1.3 A design basis shall be established and a safety analysis performed for each *experimental device*. This shall at least take into account:

- the radioactive inventory of the *experimental device;*
- the potential for the generation or release of energy;
- the damage caused to the *experimental devices* by the postulated initiating events of the *research reactor*;
- the interaction between the *experimental devices* and the *research reactor*.

X2. Safety of *experiments* and role of the operating organization

X2.1 *Experiments* with safety significance shall be a subject to safety analysis, and procedures for design, fabrication, installation, commissioning, and *operation* equivalent to those applied to the reactor itself.

X2.2 Where necessary for the safety of the reactor and the safety of the *experiment*, the design shall provide appropriate monitoring of parameters for *experiments* in the reactor control room.

X2.3 The operating organisation shall have the overall responsibility for all safety aspects of the preparation and performance of *experiment*.

X2.4 The operating organisation shall develop procedures in accordance with the management system for review and approval of:

- proposals of *experiments*;
- the control of their performance;[87]
- the utilization programme;

[87] Effects on the *research reactor*, particularly changes in reactivity and radiation levels

- *experimental devices* disassembly and maintenance.

Annex A: Implementation of a Graded Approach for the Application of the Reference Levels for Research Reactors

Need of a graded approach

Given the various types of existing RR, and considering that they can represent very different risks for the workers, the population and the environment, there is a need for applying a graded approach in the application of the SRLs.

Indeed, the SRLs may not be required to every research reactor in the same way. For example, a RR with a high thermal power, and for which a severe core degradation is physically possible, may apply some SRLs in a similar manner as NPPs comply with their SRLs, whereas the same dispositions may appear disproportionate for a RR with a low thermal power, and for which a severe core degradation may practically be excluded, allowing some adaptations in its application.

Criteria for determining the risk represented by a research reactor

In order to implement a graded approach, it is necessary to first identify the risk represented by the research reactor. Several criteria can have an impact on the risk profile of the RR. Although the thermal power of the reactor might give a first indication of this risk, there can be some particular risks related to other characteristics of the RR that need to be considered.

A list of criteria to consider for determining the risk profile of a RR is provided in 2.7 of the IAEA guide SSG 22 [1]. These criteria are:

(a) the thermal power of the reactor;
(b) the radiological source term;
(c) the amount and enrichment of fissile material and fissionable material;
(d) spent fuel storage areas, high pressure systems, heating systems and the storage of flammables, which may affect the safety of the reactor;
(e) the type of fuel and its chemical composition;
(f) the type and mass of moderator, reflector and coolant;
(g) the amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and engineered safety features;
(h) the quality of the containment structure or other means of confinement;

(i) the utilization of the reactor (experimental devices, tests, radioisotope production, reactor physics experiments);
(j) the location of the site, including the potential for external hazards (including those due to the proximity of other nuclear facilities) and the characteristics of airborne and liquid releases of radioactive material;
(k) proximity to population groups and the feasibility of implementing emergency plans.

Implementation for a graded approach

The RRs within the scope of application of these SRLs have to address and apply all of them (waiving a SRL is not allowed). However, after identifying the risk profile of a RR, the way reference levels are applied may be graded (i.e. proportional), as long as sufficient levels of safety are reached.

A graded approach may be used in many of SRLs, the adaptations may be material, or organizational.

The licensee has to propose to and justify for approval by the national regulator for which SRLs they want to apply grading.

Annex B: Glossary

Introduction

This glossary defines key terms used in the WGRR RLs in order to harmonize terminology within the given document and supports the understanding and application of the RLs. It is intended only for the use in combination with the WGRR RLs.

As the WGRR RLs are based on the RHWG RLs, analogies are obvious but especially when it comes to utilization and the persons working at or using a research reactor, differences between research reactors and NPPs were found and are accounted for in the definitions given here. For consistency, the original definitions of RHWG RLs were moved to the glossary.

The glossary lists terms applied by the WGRR in the reference levels. Many definitions were taken from the IAEA Safety Glossary or IAEA Safety Standards Series No. SSR-3; the sources used are given at the end of each entry.

In order facilitate the utilization of the glossary, terms defined in here are given in *italics* in the main body of this document.

Term	Definition
Confinement	*Confinement* is the prevention or control of releases of radioactive material to the environment in *operation* or in accidents (IAEA Safety Glossary: 2018 Edition). *Confinement* is a basic safety function that is required to be fulfilled in normal operational modes, for anticipated operational occurrences, in design basis accidents and, to the extent practicable, in selected design extension conditions. The function of *confinement* is usually fulfilled by means of several barriers surrounding the main parts of a nuclear reactor that contain radioactive material. For a *research reactor*, the reactor building may be the ultimate barrier for ensuring *confinement*. Consideration may be given to the use of other structures (e.g. the reactor block in a fully enclosed *research reactor*) for providing *confinement* where this is technically feasible.
Containment	For most designs of large nuclear reactor, a strong structure housing the reactor is the ultimate barrier providing *confinement*. Such a structure is called the *containment* structure or simply the *containment*. (IAEA Safety Standards Series No. SSR-3 (2016))

Emergency Control Centre	The *Emergency Control Centre* is the office accommodation and associated office services set aside on or near to the site for staff who are brought together to provide technical support the *operations staff* during an emergency or where the licensee emergency response is directed. It may have information on *research reactor* systems available, but is not expected to have any reactor controls.
Emergency workers	This definition includes *workers* from the operating organisation and, if necessary, contractors, as well as off-site emergency responders that may be needed on-site.
Experimental devices	Devices used for research, development, isotope production or any other purpose, that could have an impact on the safety of the reactor or other safety relevant *SSCs*. This includes e.g. devices installed in or around the reactor to utilize the neutron flux or ionizing radiation from the reactor. (Adopted from IAEA Safety Standards Series No. SSR-3 (2016))
Experiments	Activities performed in the *research reactor* which use the neutron flux or ionizing radiation produced by the reactor. *Experiments* include for example radioisotope production, irradiation for materials testing, doping of silicon, or examination and neutron activation realized thanks to neutron beams. Education, training and validation of computational models performed with the reactor are also considered as *experiments*.
Extended shutdown	State of a *research reactor* that is no longer operating, with no decision on its decommissioning, and where there is no clear decision about the future of the reactor as to whether it will be brought back into *operation* or decommissioned. Long shutdown periods for maintenance or for implementation of refurbishment and modification projects are not considered as an *extended shutdown* state. (IAEA Safety Standards Series No. SSR-3 (2016))
Fire cell	*Fire cells* are separate areas in which redundant items important to safety are located. Since *fire cells* may not be completely surrounded by fire barriers, spreading of fire between cells shall be prevented by other protection measures. (IAEA Safety Standards Series No. NS-G-1.7 (2004))
Fire compartment	A building or part of building that is completely surrounded by fire resistant barriers of sufficient rating so that a total

	combustion of the fire load can occur without breaching the barriers (barriers comprise doors, walls, floors and ceilings).
Fuel element	A *fuel element* includes the nuclear fuel, its cladding and any associated components necessary to form a structural entity. (IAEA Safety Glossary: 2018 Edition)
Operating personnel	The *operating personnel* comprise the reactor manager, the shift supervisors, the operators, the maintenance *staff* and the radiation protection *staff*. (IAEA Safety Standards Series No. SSR-3 (2016))
Operation	All activities performed to achieve the purpose for which a *research reactor* was designed and constructed or modified. Besides operating the *research reactor*, this includes: maintenance, testing and inspection; fuel handling and handling of radioactive material, including the production of radioisotopes; installation, testing and *operation* of *experimental devices*; the use of neutron beams; the use of the *research reactor* systems for the purposes of research and development as well as education and training and other associated activities. (adapted from IAEA Safety Standards Series No. SSR-3 (2016))
Research reactor	Nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiation for research and other purposes, including experimental facilities associated with the reactor and storage, handling and treatment facilities for radioactive materials on the same site that are directly related to safe *operation* of the *research reactor*. (IAEA Safety Standards Series No. SSR-3 (2016)) This definition excludes nuclear reactors used for the production of electricity, naval propulsion, desalination or district heating. Facilities commonly known as critical assemblies and subcritical assemblies, homogeneous zero-power reactors and accelerator driven systems are out of the scope of this publication.
Safety analysis report (SAR)	A consistent safety document or integrated set of documents constituting the licensing basis of the *research reactor* and updated under supervision of the regulatory body.

Severe Accident	Accident more severe than a design basis accident and involving significant *fuel element* degradation.[88]
Site personnel	All persons working in the site area of an authorized facility, either permanently or temporarily (IAEA Safety Glossary: 2018 Edition). This comprises e.g. *workers* from contractors, scientists, students.
Staff	Any person employed by the licensee.
Structures, systems and components (SSCs)	A general term encompassing all of the elements (items) of the *research reactor*, except human factors. Within the scope of this document, *experimental devices* are also part of the *SSCs*. **Components**: Discrete elements of a system. Examples of **components** are wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves. **Structures**: Passive elements (e.g. buildings, vessels and shielding). **Systems**: Several components assembled in such a way as to perform a specific (active) function. *SSCs* include their support functions and related instrumentation as well as software for I&C.
Worker	Any person who works, whether full time, part time or temporarily, for an employer and who has recognized rights and duties in relation to occupational radiation protection. A self-employed person is regarded as having the duties of both an employer and a *worker*. (IAEA Safety Glossary: 2018 Edition)

88 For research reactors, there might be other *severe accident*s not involving severe fuel/core degradation. The need for a precise definition for a *severe accident* will be taken into account in a future update.