Safety Reports Series

No. 94

Approaches to Safety Evaluation of New and Existing Research Reactor Facilities in Relation to External Events

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APPROACHES TO SAFETY EVALUATION OF NEW AND EXISTING RESEARCH REACTOR FACILITIES IN RELATION TO EXTERNAL EVENTS

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SAFETY REPORTS SERIES No. 94

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FOREWORD

The safety of a research reactor requires that it be suitably sited, designed, constructed and operated to protect workers, the public and the environment against the uncontrolled release of radioactive material. External events play a major role in challenging facility defences and therefore appropriate provisions need to be taken to ensure adequate safety in case of such events.

IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors, establishes the safety requirements for research reactors, including those for siting and design. The IAEA Specific Safety Guides on site evaluation address all nuclear installations. Safety Reports Series No. 41, Safety of New and Existing Research Reactor Facilities in Relation to External Events, published in 2005, provides specific information to treat different types of research reactor subjected to external events.

This publication is a revision and update of Safety Reports Series No. 41. It provides approaches for evaluating the safety of new and existing research reactors in relation to the hazards posed by external events. It also provides updated information and incorporates feedback from the Fukushima Daiichi nuclear power plant accident relating to site evaluation and design aspects, especially site investigations, evaluation of external event hazards, re-evaluation of existing facilities and emergency preparedness. It elaborates safety categorization and the use of a graded approach for different types and sizes of research reactors.

The IAEA wishes to thank all contributors to this publication. The IAEA officers responsible for this publication were H. Mahmood, O. Coman, A. Shokr and D.V. Rao of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Siting and design of research reactors in relation to external events, from the use of national building codes to nuclear power plant codes, vary greatly among Member States. One of the main reasons for such broadly different approaches is the difficulty associated with the safety categorization of research reactors. The research, experiments, production, testing, education and training activities which are carried out in research reactors lead to facility specific safety cases and layouts.

IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [1], establishes the requirements on siting, design, construction and operation for research reactors. IAEA Safety Standards Series No. NS-R-3 (Rev. 1), Site Evaluation for Nuclear Installations [2], establishes the safety requirements on site evaluation for nuclear installations. The Specific Safety Guides [3–5] on site evaluation address all nuclear installations, and the methodologies are also applicable to research reactors through the use of a graded approach, commensurate with the potential hazards of the facility [6]. However, specific guidance is not provided in these publications on how to treat the different types of research reactor subjected to external events.

Safety Reports Series No. 41, Safety of New and Existing Research Reactor Facilities in Relation to External Events (2005), covers topics related to siting and design of research reactors, including safety concepts in siting, design, hazard categorization, site investigations, evaluation of external event hazards, qualification of equipment, ageing aspects, simplified approaches for seismic design, anchorage and interaction aspects. However, the technical content of the publication needed to be updated to provide more clarity to safety concepts and incorporate lessons learned from different external events, including those related to the accident at the Fukushima Daiichi nuclear power plant, to consider beyond design basis scenarios and the simultaneous occurrence of two or more extreme external events at a multi-unit nuclear installation site. Moreover, there was a need to elaborate on the use of a graded approach in application of the safety requirements.

IAEA safety publications [2–5, 7] consider hazards for nuclear installations with respect to siting and external events. IAEA technical publications that have been widely used in recent years have addressed only isolated aspects relating to siting and design of research reactors. References [8, 9] cover different aspects of external events that are also applicable to research reactors, providing examples of practices in various Member States, and contain the preliminary information on a graded approach, which is examined in detail in this publication.

1.2. OBJECTIVE

The main objective of this publication is to present approaches for conducting a safety evaluation of new and existing research reactors in relation to the hazards posed by external events, consistent with the safety requirements established in Refs [1, 2]. The publication provides updated information on different aspects related to site investigations, evaluation of external event hazards, re-evaluation of existing facilities and emergency preparedness for research reactors.

This publication is intended for use by regulatory bodies, operating organizations and designers. It provides a technical basis for the safety aspects of self-assessment, in line with IAEA safety standards. It can be used to develop guidelines for conducting design and safety assessments in relation to external events.

This publication can also be used as a background in preparing training material for research reactor staff. Such training tends to encourage self-assessment by facility staff of the vulnerability of existing structures to external events.

1.3. SCOPE

From the standpoint of the safety of the public and workers, research reactors are difficult to categorize due to their broad range of application, their design characteristics and their power levels. This publication is intended to apply to all types of research reactor, including critical assemblies.

The external events considered in this publication include both natural and human induced hazards from accidental sources external to the site or external to the buildings related to safety. In this report, explicit reference is made to external event scenarios considered in the design of research reactors (earthquake, volcanoes, wind, precipitation (snow, rain, hail), flood, explosions, aircraft crash and external fire), for which specific methodologies are provided. Other safety hazards, such as the chemical hazard posed by the research reactor on the environment, are not included in the scope of this publication. This publication addresses both the design of new facilities and the re-evaluation of existing ones. A re-evaluation can be necessary due to a modification (e.g. for accommodation of new experiments), a periodic safety review and design re-evaluation, life extension of a facility or modification of the licensing requirements by the regulatory body.

This publication is based on the experience of Member States in evaluating the safety of research reactors: it spells out a coherent framework for the use of a graded approach to apply the safety requirements for siting and design.

Security issues are not within the scope of this publication. References [10–12] provide details regarding design provisions for security related scenarios and guidelines for identifying vital areas.

1.4. STRUCTURE

Sections 2 and 3 provide a detailed categorization of the radiological hazards posed to workers, the public and the environment, prior to the selection of suitable methods and procedures to evaluate the site for a research reactor facility, its design and safe operation. Sections 4 and 5 discuss site investigations and the development of a site specific hazard evaluation. Sections 6, 7, 8 and 9 discuss the general design approach, qualifications, evaluation of margins beyond design and safety re-evaluation of existing facilities. Section 10 looks at the environmental impact of radioactive dispersion from a research reactor facility following an accident. Sections 11 and 12 explore suitable measures for monitoring, alerts, event management, post-event inspection and implementation of an emergency plan in the event of external initiating events. Section 13 describes how to develop a management system for siting and design.

The appendices provide examples of feedback from operating experience in support of the approach proposed in this publication, along with some examples of applications that are considered useful for implementing the proposed approach. The annex provides an example of siting studies that were carried out at the Open Pool Australian Lightwater research reactor at Lucas Heights, Australia.

2. SAFETY CONCEPTS IN SITING AND DESIGN

2.1. GENERAL

This section covers safety concepts with the aim of developing the relationship between hazard categorization, external event categorization and performance goals for application to siting and design of research reactors in relation to external events according to the hazard they pose to workers, the public and the environment. IAEA Safety Standards Series No. SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report [13], provides the basic steps of the safety assessment process.

The following reactor states have to be considered in any safety assessment of a facility:

(a) Normal operation;
(b) Anticipated operational occurrences;
(c) Design basis accidents;
(d) Design extension conditions[1].

In addition, the following needs to be considered:

(i) Long term shutdown behaviour with or without need for an active cooling of radioactive fuel;
(ii) Refuelling;
(iii) Maintenance.

Postulated initiating events (PIEs) have to be identified and selected as required in SSR-3 [1]. Special care has to be taken in the PIEs relevant to external events that have a credible probability of being combined with internal events.

2.2. SAFETY OBJECTIVES FOR RESEARCH REACTORS

Paragraph 2.1 of IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [14], states:

> "To ensure that facilities are operated and activities conducted so as to achieve the highest standards of safety that can reasonably be achieved, measures have to be taken:
>
> (a) To control the radiation exposure of people and the release of radioactive material to the environment;

[1] Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.

(b) To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;
(c) To mitigate the consequences of such events if they were to occur."

IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [15], specifies the acceptable radiological consequence to workers and the public (*footnotes omitted*):

"III.1. For occupational exposure of workers over the age of 18 years, the dose limits are:

(a) An effective dose of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years) and of 50 mSv in any single year"

and

"III-3. For public exposure, the dose limits are:

(a) An effective dose of 1 mSv in a year;
(b) In special circumstances, a higher value of effective dose in a single year could apply, provided that the average effective dose over five consecutive years does not exceed 1 mSv per year…."

2.3. TECHNICAL SAFETY REQUIREMENTS FOR RESEARCH REACTORS

Technical safety requirements have to be met for any research reactor with reference to the three main safety functions (see Requirement 7 of SSR-3 [1]).

"Requirement 7: Main safety functions

"The design for a research reactor facility shall ensure the fulfilment of the following main safety functions for the research reactor for all states of the facility: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel storage; and (iii) confinement of the radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases."

For each selected PIE, a list of structures, systems and components (SSCs) important to safety have to be developed and their safety functions identified. Spatial and other possible interactions with the items important to safety have to be examined, since an external event can alter the behaviour of a number of items simultaneously. In particular, external events could induce chemical or biological hazards or could result in consequences for safety such as reductions in available personnel, limitations on transportation and restriction of access.

External events can be either human induced or caused by natural phenomena. In some external event scenarios, the development of the consequences on a facility is not proportional to the increase in the cause. In these cases, a 'cliff-edge effect'[2] may be significant. The technical safety requirements aim at preventing accidents in research reactors and mitigating their consequences if they occur. In the safety analysis report it is necessary to demonstrate the:

(i) Capability of shutting down the reactor and maintaining it in a safe shutdown state during and after an external event[3]: either automatically or through operator action. Redundancy, diversity and robustness (against an external event) in the reactivity control system of the reactor have to be demonstrated.

(ii) Cooling of core and spent fuel after the external event is possible with dedicated and reliable systems when necessary (though for research reactors natural convection or heat accumulation in the coolant is often sufficient).

(iii) Radiological consequences would be within acceptable limits, with a high level of confidence and for all design basis accidents and design extension conditions. Among others, the following measures need special attention:
— Avoiding any failure of SSCs which could directly or indirectly cause accident conditions as a consequence of an external event, particularly with respect to reactivity control, decay heat removal and confinement;
— Monitoring the important reactor parameters during and after an external event, in particular, the reactivity and cooling;
— Monitoring the radiological dispersion parameters;
— Ensuring access to the reactor site;

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[2] A cliff-edge effect is a sudden increase of the consequences as a result of a small increase of the causes. A typical example is the flooding scenario in a site protected by a dam: as soon as the water overtops the protection, the whole site is flooded up to the maximum level.

[3] The safety analysis has to consider the duration of the event and the time needed to return to a normal condition. In case the facility is not put into operation again, the total duration of the event corresponds to the time needed to come to a new stable and sustainable condition. The safety analysis needs to consider this scenario also.

- Facilitating the emergency functions of the operating personnel (e.g. habitability of the control room/emergency control room, communication within the site and with the off-site emergency team, and emergency alarm system);
- Monitoring external events through warning system(s) and their consequences in the vicinity of the facility, and alerting the emergency response team(s).

2.4. MAIN STEPS FOR THE SAFETY ASSESSMENT IN RELATION TO EXTERNAL EVENTS

The following main steps are suggested for a systematic safety assessment process for research reactors in relation to external events; these steps are explained in detail in Sections 2.5–2.10:

(i) *Hazard categorization (HC) of a facility*: The facility is categorized according to the risk that it may pose to public and workers.

(ii) *Safety classification of items*: Items are classified according to their safety functions. Items classified as “not important to safety” are traditionally designed according to non-nuclear standards and are not credited in the reactor safety analysis.

(iii) *External event categorization (EEC) of items*: In addition to their safety classification, items may be categorized according to their role in external event scenarios, either because they are directly affected by the events or because they are important to mitigate the consequences of the events. This categorization is complementary to the safety classification and is particularly important for non-seismic scenarios (when all components are somehow affected) as it allows applying appropriate safety margins only to SSCs important to safety, reducing the design effort. The case of items relevant for facility protection against external flooding is a typical example: only components either directly exposed to flooding or required for performing a safety function are considered.

(iv) *Calculation of the performance goal (PG) for the overall facility*: This is the safety target (expressed in probability of failure) associated with the potential consequences of a release from the facility, as a consequence of component failure, and of its role in prevention and mitigation. The PG is an overall measure of the reliability that the facility has to achieve and it is not related to the probability of occurrence of the external event scenarios.

(v) *Choice of the design class (DC)*: Given the PG, a choice has to be made on the safety margin to be used in the design process (i.e. design standards to be applied) of the safety components. According to the design basis chosen for the external event, a design standard has to be chosen in order to meet the required PG.

(vi) *Defence in depth*: This has to be applied, with account taken of the use of a graded approach that is commensurate with the potential hazard of the facility.

The process helps to select the appropriate standard for design and qualification, in line with the overall safety targets, and an adequate application of defence in depth.

2.5. USE OF A GRADED APPROACH

Considering the wide variation in the design, types and sizes of research reactor, associated experimental facilities and utilization programmes, the consequent potential hazard of the research reactor in relation to external events also varies. A graded approach is needed in the safety assessment commensurate with the potential hazard of the research reactor [6].

Reference [16] provides the following information on the use of a graded approach for safety reassessment of research reactors:

> "Aspects of the reassessment that may be subjected to grading include the scope, extent and details of the analysis, and the required human and financial resources, which may be significantly less for low power research reactors than for high power research reactors.
>
> "Factors affecting the application of a graded approach are those related to the risk and the potential hazard, including, for example:
>
> — The reactor power;
> — The fission product inventory and the radiological source term;
> — The amount and enrichment of fissile material;
> — Fuel design;
> — Inherent safety features of the design;
> — The presence of high pressure or high energy piping (experimental loops);
> — The quality of the means of confinement (containment and ventilation systems);
> — The presence of experimental facilities and experimental devices, and the reactor utilization programme;

— The stage of the lifetime of the reactor facility, ageing of the reactor, and upgrades and modifications;
— Any other special hazard (e.g. hydrogen, chemical and fire hazards);
— Siting (regional characteristics);
— The structural concept (above or below ground);
— The proximity of the reactor facility to populated areas."

Grading may be applied to the methodology for calculating design basis events and assessing design extension conditions of the research reactor facility. A simplified approach and less rigorous design codes could be used for low hazard category facilities, while reactors in the high hazard category (HC-1) may need to use design codes similar to those for nuclear power plants. Low hazard category research reactors may need only limited analysis under certain accident scenarios compared with high hazard category research reactors.

The graded approach may also be applied to the selection of site related design basis events (and design extension conditions) such that the examination of events may show that a minimal hazard to the research reactor facility is posed on a particular site. Reference [16] also provides the following information on the use of a graded approach in the application of safety requirements, emergency arrangements [17] and organizational factors [18, 19]:

> "A graded approach may also be used in the application of the safety requirements related to the levels of the defence in depth, in the sense that level 5, and sometimes level 4, may be met by the inherent safety characteristics of the reactor instead of through engineered safety features of the design. If the research reactor is designed without confinement or containment, for example, this needs to be justified on the basis that, under accident conditions, there is no potential for release of radioactive material from the facility that may result in unacceptable off-site consequences.
>
> "Grading may be applicable to the emergency arrangements to be established based on the potential hazard associated with the research reactor facility in line with the requirements established in [IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [17]]. Grading may also be applied to the number and types of escape routes, based on the layout and size of the reactor facility. It may also be applied to the necessary emergency equipment, and to the scope and frequency of the emergency drills and exercises.
>
> "A graded approach can be also applied to the organizational aspects, including human and financial resources, of performing the safety reassessment and to the management of implementation of the findings of the reassessment. Application of the graded approach should be based on

the potential hazard of the research reactor facility, and should take into account the existence of other nuclear installations on the site, including those facilities associated with the research reactor [IAEA Safety Standards Series No. SSG-22, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors [6]]".

Appendix II provides an example of the use of a graded approach for the seismic safety evaluation of a facility.

2.6. CATEGORIZATION OF FACILITIES

References [3–5] address nuclear installations and provide generic guidance on the graded approach for hazard evaluation in relation to external events based on the following categorization of facilities:

(i) For the least hazardous research reactors, the hazard input for the design may be taken from national building codes and maps in relation to hazardous industrial installations.
(ii) For research reactors in the highest hazard category, methodologies for hazard assessment as described for nuclear power plants may need to be adopted, with a few simplifications.
(iii) For research reactors categorized in the intermediate hazard category, further guidance is provided in IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [4], para. 10.10(c).

In general, the probability that external events will cause a radiological consequence depends on the characteristics of both the facility (utilization, layout, design, operation) and of the initiating events. Quantitative generic rules are difficult to formulate to address this concern, but a set of qualitative criteria are summarized to drive the facility categorization. Paragraph 10.5 of SSG-18 [4] lists the factors that need to be taken into account in deriving the characteristics of a nuclear installation, information that is also applicable to research reactors.

A general evaluation of the risk associated with a research reactor facility is rather difficult due to the high number of variables in its design. A reasonable and reliable risk classification (graded) can be made only on a case by case basis.

Therefore, in the framework of this publication, the factors listed in para. 10.5 of SSG-18 [4] could be interpreted as criteria for risk classification, driving the final evaluation of the risk associated with the facility, ranging from a minimum risk (conventional building) to the highest risks (similar to those of a nuclear power plant).

A reasonable and much simplified approach could entail a reduction in the number of criteria described above, as for most research reactors the application of the criteria for facility classification shows a strong correlation between risk associated with the facility and either its power or its overall radioactive inventory. This correlation might simplify the classification process at the beginning of the design. Clearly, such assumptions have to be evaluated in the safety assessment phase and justified in the safety analysis report. Table 1, which is reproduced from Safety Reports Series No. 41, with inventory values normalized for the reactor powers, provides an example of a simplified approach based on the radioactive inventory of a research reactor. While categorizing the facilities, consideration has to be given to all the criteria enumerated in this section. In case of conflict, the most stringent criteria have to be applied.

TABLE 1. EXAMPLE OF HAZARD CATEGORIZATION FOR RESEARCH REACTORS

Hazard category of the facility	Inventory[a], TBq (10^{12} Bq) (I)	
	γ[b]	α[c]
HC-1 (High)	$I > 4 \times 10^6$	$I > 20$
HC-2 (Medium)	$4 \times 10^5 < I < 4 \times 10^6$	$2 < I < 20$
HC-3 (Low)	$4 \times 10^4 < I < 4 \times 10^5$	$0.2 < I < 2$
HC-4 (Very low)	$I < 4 \times 10^4$	$I < 0.2$

Notes:

HC-1 includes research reactors with a power rating in the range of $P \geq 20$ MW.

HC-2 includes research reactors with a power rating in the range of $2 \leq P < 20$ MW.

HC-3 includes research reactors with a power rating in the range of $0.1 \leq P < 2$ MW.

HC-4 includes research reactors with a power rating in the range of $P \leq 0.1$ MW.

[a] These values are indicative.

[b] These values correspond to normalized values from a 20 MW(U^3Si^2) equilibrium core.

[c] These values correspond to 207 irradiation days, approximately.With this assumption, a 10MW reactor would have about 10 TBq.

For the purposes of this publication, facilities with a power or radioactive inventory in hazard category 1 (HC-1) can be designed on the basis of relevant codes and standards applicable to nuclear power plants using a graded approach. Where initial categorization of research reactors is based on radioactive inventory and power alone, it has to consider other factors that may change the hazard category, such as:

(i) If the reactor has inherent safety features, such as a strong negative temperature coefficient and passive safety systems providing a high degree of reliability against release of radioactive material, the hazard category defined by the power can be decreased by one. The same can be done in pool type reactors if the cladding material of the fuel is stainless steel or zirconium alloy.

(ii) If the reactor is categorized as HC-2, HC-3 or HC-4, the hazard category could be increased by one when there are one or more of the following conditions: the reactor has high temperature and high pressure experimental facilities, or experiments with fuel are carried out in the reactor. The same applies for prototype power reactors.

(iii) HC-4 is intended to include facilities where core damage does not occur and therefore the source term is particularly small. HC-4 facilities can be regarded as industrial installations at special risk and therefore they are not discussed in this publication. HC-3 research reactors can be downgraded to HC-4 if they show such an intrinsic feature in relation to the core damage.

(iv) Site characteristics also affect the hazard categorization of research reactor facilities. A facility which would normally be classified as HC-2, if located away from residential or industrial facilities, may be upgraded to HC-1 if located in a populated area. An HC-3 facility may be recategorized as HC-2 for similar reasons. This approach is applied in some countries, where the hazard categorization is based on the need for emergency systems in case of an extreme event. However, this approach is not applied in other countries where the hazard is associated only with the facility and the design is independent of external factors such as population density in the surrounding area. The definition of limits for emergency preparedness is crucial in this regard and may influence the final categorization of the facility.

(v) A facility located at a site where other nuclear installations are in operation may increase the likelihood of risk and result in the revision of its hazard category, adapted to the mutual risk of interaction (‘collocation of facilities’) and therefore to the hazard posed by the whole site to the surroundings.

Any hazard categorization approach has to be carefully evaluated and its outcome judged in the framework of a risk informed approach, as described in Ref. [20].

2.7. SAFETY CLASSIFICATION FOR SSCs

After a facility is categorized on the basis of the hazard posed to the surroundings in case of releases, a safety classification for SSCs has to be developed.

Paragraph 6.29 of SSR-3 [1] states that (*footnote omitted*):

> "The method for classifying the safety significance of items important to safety shall be based primarily on deterministic methods complemented, where appropriate, by probabilistic methods (if available), with due account taken of factors such as:
>
> (a) The safety function(s) to be performed by the item;
> (b) The consequences of failure to perform a safety function;
> (c) The frequency with which the item will be called upon to perform a safety function;
> (d) The time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function."

The failure of an SSC important to safety by itself may not lead to a release of radioactive material. However, the safety classification has to consider all the scenarios initiated by the PIEs. A typical example is represented by the emergency ventilation system: its failure may lead to a release if there is a coincident failure of other SSCs which requires the emergency ventilation system to perform properly.

2.8. EXTERNAL EVENT CATEGORIZATION OF SSCs

In addition to facility categorization and SSC safety classification, an external event categorization is useful to drive a rational design process [9, 21] with the appropriate use of a graded approach. The EEC may be identified with reference to the selected PIEs for external events.

The EEC is a quantitative index of how much external events can trigger hazardous consequences at a facility. SSCs important to safety may not be exposed to or affected by external events and therefore their external event categorization lists may be quite low.

An example of external event categorization may encompass the following categories:

(i) An EEC-1 item is an SSC pertaining to external event safety groups or systems which, during and after an external event, interact with items in the safety group of the external events.
(ii) An EEC-2 is an SSC important to safety, which is not in external event safety groups and does not interact with EEC-1 items during and after an external event (i.e. not needed for external events).
(iii) EEC-3 items are not themselves important to safety but could impair functions of EEC-1 and EEC-2 items or operator action.
(iv) EEC-4 is conventional risk.

In this publication, the EEC is used in conjunction with the safety classification, meaning that design provisions are provided for both external event categorized (safety classified) and external event non-categorized (non-safety classified) items.

Beyond the general list provided in annex 1 to SSR-3 [1], a description of the typical items important to safety of a research reactor and the effects of their failure on the relevant safety functions in relation to external events is provided in Table 2. These items are typically EEC–SSCs.

2.9. PERFORMANCE GOALS

For design purposes, the hazard category of the facility has to be converted into a performance goal target value. The performance goal for a facility in relation to a specific external event is defined as the probability of failure (release) of the facility to meet the safety requirements (shut down, containment, cooling) in case of that external event.

The performance goal affects the design of the safety classified items, while it is intended that non-safety classified items be designed according to conventional design standards.

The probability of failure of all SSCs in relation to external events is computed using the full range hazard curve of the external events and the fragility of the SSC under consideration, as shown in Appendix II.

The fragility of SSCs is defined as the cumulative conditional probability of failure (unacceptable performance) versus the selected hazard parameter. The hazard parameter is typically represented by factors such as: the peak ground

TABLE 2. TYPICAL SSCs OF A RESEARCH REACTOR CONSIDERED IN THE DESIGN OR RE-EVALUATION AGAINST EXTERNAL EVENTS

Item	Affected characteristics of the item	Effect of the failure of the item
Reactor building	Structural integrity[a] and stability[b], leaktightness[c]	Damage to the reactor and reactor safety systems
Reactor pool (with or without pool lining) or tank	Structural integrity and stability, leaktightness	Inability to maintain reactor pool water level
Control room	Structural integrity and stability	Inability to monitor and control safety activities
Ventilation stack	Structural integrity and stability	Damage to items important to safety
Shielding structures, protection dams	Structural integrity and stability	Loss of shielding or protection
Reactor vessel and reactor internals or reactor block	Structural integrity and stability	Possible core damage
Reactor cooling system	Structural integrity, functionality[d] (when required)	Possible core damage
Effluent filtration system	Structural integrity, functionality	Radioactive material releases
Emergency power supply	Functionality	Inability to perform safety functions
Reactor protection system	Functionality	Inability to perform safety functions, possible core damage
Instrumentation and control systems important to safety	Functionality	Inability to perform safety functions, increased potential for human error
Beam tubes penetrating into reactor pool	Structural integrity and stability, leaktightness	Inability to maintain reactor pool water level — radiation exposure, possible core damage
Experimental facilities (fuel test loops)	Structural integrity and leaktightness	Release of radioactivity

Notes:

[a] Structural integrity means that the SSCs will continue to maintain their geometry and transfer load.

[b] Stability means that the SSCs will not collapse.

[c] Leaktightness means that the SSCs will maintain fluid inventory under acceptable limits.

[d] Functionality means that the SSCs will continue to perform their required safety functions during and following an external event.

TABLE 3. TARGET PROBABILITY OF FAILURE AND SAFETY CLASS OR EXTERNAL EVENT CATEGORY

Hazard category of the facility	Target probability of failure for safety class items (or EEC-1, 2, 3)	Target probability of failure for non-classified components
HC-1	10^{-5}/a	10^{-3}/a
HC-2 or HC-3	10^{-4}/a	10^{-3}/a
HC- 4	10^{-3}/a	10^{-3}/a

acceleration for earthquakes, the water depth for floods and the maximum wind speed for winds.

Typical values for performance goals for research reactors in relation to generic external event scenarios are presented in Table 3. For comparison, the target probability of failure for nuclear power plant components in the highest safety class is normally established at 10^{-6}/a.[4]

2.10. DESIGN CLASS FOR SSCs

A design class (DC) for SSCs can be defined based on the level of safety margin[5] and reliability[6] that can be used in the design/qualification of an SSC with reference to a specific external event scenario in order to reach the assigned failure probability. Therefore, it is associated with the SSC as the result of the product of both the external event probability of exceeding[7] the external event scenario and the target probability associated with the same SSC.

[4] This value (probability of failure per year) is provided in many countries for an uncontrolled large release. In research reactors, an unmitigated release should be used to define the performance goal if the containment system is not available.

[5] *Safety margin* can mean one of the following:

- The availability of one or more lines of defence, before radionuclides are released to the environment, which is demonstrated by capacity divided by a demand value greater than 1.0;
- The capability of detection — such as the *'leak before break'* concept — which would allow prevention measures to be taken in time;
- The availability of mitigation measures.

[6] The probability that a system will meet its minimum performance requirement when called upon to do so.

[7] Reciprocal of the return period, in case of a stationary process.

The design class is the consequence of the design strategy chosen for the component: for the same value of the target probability, if a high probability of exceeding scenario is chosen, a bigger safety margin (higher ‘nuclear grade’ code) has to be used in the design and siting process in order to meet the target failure probability (linked to the facility performance goal).

The selection of design codes and the probability of exceeding scenario seem broad, but the combinations are rather limited in the engineering practice of research reactors.

The two extreme cases are the following:

(i) ‘Low’ probability of exceeding for design basis external events coupled with a ‘conventional’ design code;
(ii) ‘High’ probability of exceeding for design basis external events coupled with a ‘nuclear grade’ design and siting code.

This limited choice is governed by physical considerations which may affect the choice of probability of exceeding the design basis value for the external event, P: for some events the evaluation of a very low probability hazard is feasible because physical evidence is available (typically earthquakes), but for some scenarios this may not be the case (e.g. precipitation). Furthermore, some scenarios are associated not with a statistical period on a site, such as tornadoes, but rather with the probability of occurrence in a wide region (e.g. rotational winds). The hazard curve may also deviate strongly from linearity in the low probability range, which may suggest considering a site specific, low probability event instead of a more conventional high probability scenario.

In conclusion, the professional practice, the availability of complete historical records in the available database and the way the probability is calculated are important considerations to be developed in this process, which may drive the selection of the most convenient design class for the facility components.

Examples for probability of exceeding the external event design basis values are therefore suggested in the following practice from the Member States; the values have to be interpreted as minima in order to have a reliable estimation of the associated physical description of the external event scenario and association with a region (and not with the site) for localized events:

— Earthquake: $10^{-3} - 10^{-4}$/a.
— Straight wind: $10^{-2} - 10^{-3}$/a.
— Rotational wind: 10^{-5}/a.

— Flood: $10^{-3} – 10^{-4}$/a.
— Human induced events (e.g. explosion, aircraft crash): 10^{-5}/a.[8]

Even if, in principle, any combination of design class and probability of exceeding may be chosen, in practice, it is convenient to define four design classes (see also Section 6.3 for further details), as follows:

— DC-1 (nuclear grade): component probability of failure in the range 10^{-2}/a – 10^{-3}/a.
— DC-2 and DC-3 (hazardous industrial installations): component probability of failure in the range 10^{-1}/a – 10^{-2}/a; the classes may be defined as in the following:
  - DC-2 is a design class which is developed from DC-1 with some relaxations of the code; for example, through augmented ductility;
  - DC-3 is the enhancement of DC-4; for example, through limitation to non-linear behaviour.
— DC-4 (conventional risk facilities, building codes): component probability of failure by the application of design codes and standards for conventional risk facilities, traditionally around 10^{-1}/a.[9]

Table 4 provides an example of how a design class could be evaluated simply as a ratio between the performance goal and the probability of exceeding the external event design basis (P(EE)). This table also provides an acceptable range for the values of the DC and P(EE).

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[8] For human induced external events, such as aircraft crashes, this value represents a screening threshold. A deterministic approach is recommended.

[9] In the case of civil structures, conventional standards [ASCE43-05] require a generic safety factor of 1.5–2 between the design value and the structural failure. The analytical rationale behind the methods proposed by conventional design standards can show that, in the case of design basis earthquakes corresponding to a probability of occurrence of 10% in 50 years (with reference to a conservative hazard shape), those safety factors correspond to a 10% probability of unacceptable performance for a ground motion equal to 150%–200% of the design basis earthquake (DBE) ground motion, in relation to low ductility failure modes. Similarly, it can be shown that nuclear design codes correspond to a less than 1% probability of unacceptable performance for the design basis scenario.

TABLE 4. EXAMPLE OF DESIGN CLASS DETERMINATION

Hazard category	Performance goal	P(EE) = 10^{-4}/a	P(EE) = 10^{-3}/a	P(EE) = 10^{-2}/a
HC-1	10^{-5}/a	DC-2, DC-3	DC-1	Not applicable
HC-2, HC-3	10^{-4}/a	DC-4	DC-2, DC-3	DC-1
HC-4	10^{-3}/a	Not applicable	Not applicable	DC-4 codes and standards for conventional risk facilities

Note: /a — per annum, DC — design class, HC — hazard category, P(EE) — probability of exceeding the external event severity.

2.11. APPLICATION OF DEFENCE IN DEPTH

Defence in depth needs to be incorporated into the analysis and design of a new research reactor and into the re-evaluation of an existing research reactor. It aims at providing a suitable level of reliability of the required safety functions, according to Refs [1, 2, 13, 22, 23].

In relation to external events, defence in depth aims at a balance between two major aspects of safety concepts, namely:

— Prevention of deviation from normal operation, as the external event might induce unavailability of safety, remote control and surveillance systems;
— Mitigation of consequences of significant events for any postulated accident. Passive SSCs of external event categories 1 and 2 are preferred to address this concern.[10]

Defence in depth is a means of ensuring that the basic safety functions have been incorporated into the design, and that design extension conditions have been adequately addressed. There are five levels of defence in depth, as illustrated in Table 5.

Defence in depth may be subject to grading in the sense that level 5 and sometimes level 4 may be met by the inherent safety characteristics of the research reactor instead of through engineered safety features.

[10] Passive SSCs are those whose functioning does not depend on an external input (structural items, shielding, etc.).

TABLE 5. DEFENCE IN DEPTH

Level	Objective	Essential means
1	Prevention of deviation from normal operation and prevention of system failures	Conservative design High quality construction and operation
2	Control (by detection and intervention) of deviation from operational states to prevent anticipated operational occurrences from escalating to accident conditions	Control systems Protection systems Surveillance systems
3	Control of accidents within the design basis	Engineered safety features Emergency procedures
4	Control of severe facility conditions, including prevention of accident progression and mitigation of the consequences of design extension conditions	Complementary measures and accident management
5	Mitigation of radiological consequences of potential releases of radioactive material that may result from accident conditions	Off-site emergency response

Source: IAEA Safety Standards Series No. SSR-3 [1].

The defence in depth approach implies a deterministic definition of defence levels and barriers. This approach allows consideration of administrative measures and operating procedures as part of the defence in depth levels. Therefore, in the proposed framework, the number of levels and their reliability is a function of the facility hazard categorization, and this has to be addressed by means of additional robustness applied to the design.

It is important to note that the need for systems at any defence in depth level has to be defined in connection with the safety analysis of the facility and therefore with the safety classification of its SSCs. Many safety issues have to be considered at a research reactor that are not always explicitly correlated with component failures, such as most of the items listed in Section 2.6, which are part of the hazard categorization of the facility. Therefore, a comprehensive proposal needs to be developed that synthesizes relevant factors.

While the number of defence in depth levels is not a consequence of the choice of external event design basis, such levels are expected to be designed against external events if the external events are shown to induce internal accidents or if internal accidents have a significant probability of being contemporaneous with a design basis external event.

In all cases, the three basic safety functions listed in Section 2.3 have to be ensured and the defence in depth has to be demonstrated.

Depending on the reactor type, defence in depth levels can be provided by reactor inherent safety features or engineered safety features or both. For example, some reactors have inherent reactivity control by design, where the power and temperature excursions are limited due to a negative coefficient of reactivity. Other designs, however, may require an engineering regulating system.

Applying defence in depth to a research reactor in case of external events calls for certain clarifications:

- Protection of the facility against external events is established through robust and reliable design.
- Robust design has to be understood as high quality design with low sensitivity to variation in design parameters. It is usually achieved by high and demonstrable conservatism, including consideration of design extension conditions.
- Safety systems and/or physical barriers needed for prevention of deviations from normal operation are designed for external events only if there is a causal relationship between the accident they are designed for and an external event. A generic integrity is ensured, however, particularly to prevent any interaction with external event items that have been categorized (as is true for the containment/confinement building structure, which needs to be designed to withstand external events even though its function has to do with an internal accident).
- Barriers and defence in depth levels have to provide adequate reliability. The single failure criterion has to be applied to safety systems. Passive barriers may represent exceptions. Special attention needs to be given to external events with respect to both common mode effects on SSCs in the same facility and on different facilities at the same site.[11]
- The margins for external events in design extension conditions are usually specified in the design of the facility. In particular, cliff-edge effects in the structural response of passive systems can be investigated in order to determine whether a small increase in the design basis parameters could have dramatic effects on safety. When such effects are detected, additional engineering provisions are implemented on safety systems, such as warning, monitoring and operating procedures to maintain the safety of the reactor.
- As external events may have dramatic effects on workers, the public and the environment through, for instance, prevention of access to the site, loss of power supply, impairment of accident management at the site or hindrance of access by rescue teams, special attention has to be paid to any

[11] Site shared networks or emergency equipment could be required to deal with external event effects. Specific site assessment has to be carried out.

analysis of the implementation of emergency procedures during and after an external event.
— HC-4 facilities require only a robust design and the implementation of an emergency plan.

In general, the safety analysis of the facility supports the need for dedicated systems in any defence in depth level. For research reactors with low hazard potential, for example where the postulated design basis accidents do not lead to unacceptable releases, the third level of defence in depth may not be needed.

The scope of safety analysis related to PIEs is different for various reactors and has to be more comprehensive, to simulate the effects of interfacing systems and experimental facilities.

Few PIEs would be applicable for TRIGA reactors, for example, and the consequences would become apparent in the passive nature of the reactivity feedback during a temperature excursion. Conversely, larger reactors may have a greater number of applicable PIEs that would require specific safety analyses.

In conclusion, even if the levels of defence in depth comply with the generic safety requirements as stated in SSR-3 [1], the number of physical barriers and emergency systems to be designed at a research reactor has to be derived from a rigorous safety analysis, with reference to both internal PIEs and external events. Consistent grading in the number and content of the safety levels has to be applied in relation to the hazard categorization of the facilities.

3. GENERAL APPROACH TO SITING AND DESIGN

Siting and design of a nuclear installation are complementary activities that are used to protect the installation from the effects of external event hazards and to optimize protection in terms of radiological impact on workers, the public and the environment [24].

The graded approach is used to identify the needs for protecting the facility from the effects of natural and human induced external event hazards.

For research reactors that may be graded for application of nuclear power plant codes and standards, the design/site balance for external event hazard protection may differ from those for low potential hazard research reactor facilities. A research reactor of this grade is designed for large internal loads such as pressure, internal impacts and temperature. This gives a facility adequate robustness for its protection from at least some external event hazards. This means that for higher hazard category research reactor facilities, site selection may be performed in a similar manner to that for nuclear power plants and external event hazards may be screened out using similar criteria (both for distance and probability). However, these criteria need to be adapted for lower hazard category research reactor facilities. For example, designing a low potential hazard research reactor facility to withstand an airplane crash (even for small military planes) could be prohibitively expensive. Therefore, these need to be sited at locations where the screening distance and screening probability values are strictly observed.

The situation is the reverse for siting nuclear installations near bodies of water. While higher hazard category research reactor facilities may need to be near rivers, lakes or seas, smaller research reactor facilities do not. This makes it easier for these facilities to avoid river and coastal flooding issues.

For new nuclear installations, Member States may use a deterministic or risk informed approach in their regulations and licensing process. This will govern the way in which the operating organization conducts an external event hazard analysis. The IAEA safety standards generally provide for the needs of the Member States in this regard through the inclusion of alternative methodologies. However, there are established international practices for existing facilities that take advantage of a mixed approach such as the seismic margin assessment (SMA) based probabilistic safety assessment (PSA) approach.

This means that even if the regulatory regime of a Member State is deterministic, it is still necessary to derive probabilistic hazard curves for external events. The IAEA recommends external event probabilistic safety assessments (not only for seismic events) as one of the lessons learned from the Fukushima Daiichi accident.

The proposed sequence relies on some degree of conservatism in classification and design in order to avoid any further iteration of the design as a consequence of radioactive dispersion analysis. The approach is more straightforward, even if it relies on engineering judgement in the selection of levels of conservatism.

This methodological approach is also accompanied by some measures to control the safety margin embedded in the deterministic procedures for site investigation and design and the level of conservatism needed to reduce investigation effort, simplify design methodologies and reduce long term monitoring needs.

Two aspects, safety margin and conservatism (or ‘robustness’), are validated by the engineering experience and are driven by the method of the design class and by a series of deterministic assumptions at all phases of the siting and design process.

Details of the multistep approach are provided in the following (see also the flow chart in Fig. 1):

— Initial categorization of the hazard the facility poses to the environment, the public and workers in case of accident (not necessarily triggered by an external event) is shown in Box 1 of Fig. 1. This step categorizes the facility on the basis of guidance provided in Section 2.6 of this publication. Final categorization is a function of excessive radioactive release to the public and the environment (HC-1), workers (HC-2) or limited hazard (HC-3). The hazard category defines the need for:

- The level of detail in the facility's safety analysis report (SAR) (SSG-20 [13]).
- The level of quality assurance to be applied to materials, siting/design/construction/surveillance activities and documentation [19, 25].
- The extension of facility specific site investigation requirements is not needed for the lowest hazard category of facilities, while conservative, simplified arbitrary approaches are not acceptable for HC-1 facilities.
- The applicability of site screening criteria: for the lowest hazard category some sites may be excluded a priori in relation to external event scenarios.
- The applicability of conventional standards and codes for hazard evaluation and design of SSCs is allowed only for the lowest hazard category.
- A performance goal is finally associated with the overall facility according to its hazard category.

— The safety classification of SSCs reflects the internal postulated events and external events, as set forth in the safety analysis of the research reactor (Box 3 of Fig. 1). The applicable defence in depth levels and barriers [13], application of the single failure criterion and the assessment of the potential for common cause failures are developed in Box 2 of Fig. 1. Next is the evaluation of the need for emergency procedures, both on and off the site. This is followed by the identification of the internal events to be considered as a consequence of an external event or concurrent to an external event and of the safety functions to be maintained in case of an external event (e.g. cooling radioactive material, reactivity control, confinement).

— An external event categorization of SSCs (Box 4 of Fig. 1) may be developed to identify the safety related SSCs (required for the safety functions identified in the previous step) that are relevant for protection against external events. This categorization is affected by the potential of external events to induce radiological dispersion from a facility through failure of the affected components. This step completes the facility classification and SSC categorization and leads to defining performance goals for SSCs in relation to external events (Section 2.9). Simplified deterministic safety criteria for the SSCs, which defend the facility against external events, may be specified at this point (redundancy, diversity, quality, robustness).

— Defining the site specific hazard level and the design class for any SSC to be used in the external event design basis is subject to the performance goals assigned to any SSC (Box 6 of Fig. 1). Such an evaluation aims at minimizing the combined efforts required in the siting and design tasks, providing confidence in the required safety margin. As these definitions are preliminary to the evaluation of the external event hazard, they rely on simplified assumptions for hazard curves and fragility curves for SSCs that have been categorized for external events. Simplified tables are also suggested for ease and speed of evaluation.

— The evaluation of the design basis reflects the hazard level defined at Step 4 (Section 5; Box 8 of Fig. 1). The process may be site specific or based on national standards according to the facility categorization developed in Step 1. The site investigation campaign has to be carried out according to the requirements defined in Step 1 (Section 4; Box 7 of Fig. 1).
— Design and/or qualification of SSCs that have been categorized for external events reflect the design class identified at Step 4 and the design basis developed at Step 5 (Box 9 of Fig. 1). The methodologies to be used for design and qualification can be selected according to the facility categorization developed in Step 1 (Sections 6, 7).
— In the analysis of the dispersion of radioactive material (Box 10 of Fig. 1), the source term is selected consistent with the assumptions made in Step 2 as to the functions to be maintained during an external event. The radiological doses to the environment, the public and workers in an external event are evaluated with suitable conservatism (land use, population distribution and topography are modelled only if needed) and compared with the acceptable limits for normal operation and accident conditions. In this step, the final requirements for containment or confinement and emergency procedures are developed for accident mitigation (Section 12).
— The final safety assessment of the research reactor and the evaluation of the failure probability for the SSCs that have been categorized for external events are based on the actual hazard and design methodologies used in the design/qualification (Box 11 of Fig. 1). This step aims at the final tuning of the engineering safety features to assure that SSCs can provide the required safety function with the required reliability (Section 4). This step replaces a full scope PSA with simplified probabilistic methodologies.

The approach for re-evaluating an existing facility is expected to be the same. However, modifications in the technical details can be applied in some steps, as discussed in the following sections.

An appropriate level of conservatism may be applied in most of the steps defined above, particularly in the extension of the site investigation procedures, in the design/qualification methodologies and in the simulation of radioactive dispersion into the environment. The use of simplified methodologies, suitably graded on the basis of the research reactor facility hazard, has to be adequately documented and agreed with the regulatory body.

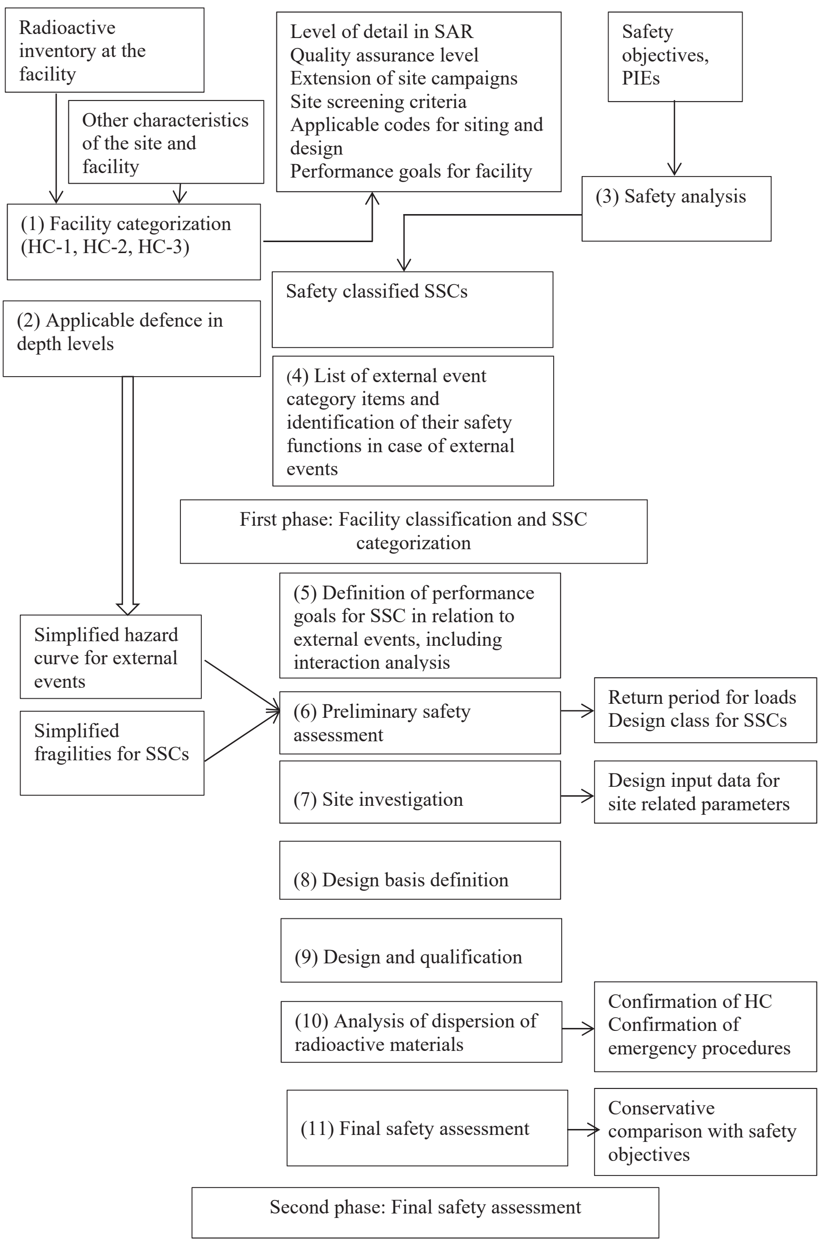

FIG. 1. Flow chart for the siting and design process.

The following sections detail the implementation of the steps defined in this section, while the appendices provide sample values and examples of application.

4. SITE INVESTIGATION

4.1. GENERAL

The investigation of the site has to cover all disciplines affecting site safety, including geology, seismology, geotechnics, volcanism, hydrology, and meteorology as well as take into account population distribution, marine environment and human induced external events [2].

The hazard evaluation for external events needs to follow the IAEA recommendations for siting and design of nuclear installations using a graded approach [1–9, 13, 16, 21, 26–30]. However, the site selection process for smaller research reactors has to consider more restrictive site exclusion criteria than those described in IAEA Safety Standards Series No. SSG-35, Site Survey and Site Selection for Nuclear Installations [24], as a compromise with the limited investment required for facility design, construction and operation. Some events from which it is difficult or expensive to protect the facility, for example, may be used as site screening criteria, such as aircraft crashes (low probability, thick shielding and special equipment qualification would be required for facilities without a containment structure), major accidental explosions (blast resisting structures would be required), and flooding (site protection engineering structures have to be built and maintained). Generally, in low power research reactors, internal accident scenarios do not imply high demand on the structures as they would for nuclear power plants, where, for example, a robust containment is normally part of the design features. Therefore, the added protection for some external events would imply significant cost increases to the construction of the facility, which might be incompatible with a rational approach to design.

A graded approach in siting and design for research reactors may be used in accordance with their hazard classification [1]. In particular:

(i) The safety margin in the design has to be easily proven, even in cases when codes applied are different from the codes for nuclear power plant design.

(ii) An adequate level of conservatism has to be guaranteed to compensate for a reduced database (with respect to a nuclear power plant), including site investigation campaigns and for simplified analysis methods.

Section 2.5 provides information on the use of a graded approach for hazard evaluation. The subsequent sections provide information on how such grading can be applied to reduce the investigated area during the siting phase.

4.2. EVALUATION OF SITE CHARACTERISTICS

The extent of the site survey can be defined in relation to the hazard category of the facility. The following criteria can be followed.

4.2.1. Seismic hazard and fault displacement hazards

4.2.1.1. Geological, geophysical and geotechnical database

Recommendations related to the geological, geophysical and geotechnical database are provided in paras 3.6–3.23 of IAEA Safety Standards Series No. SSG-9, Seismic Hazards in Site Evaluation for Nuclear Installations [3] for research reactor facilities in HC-1 and HC-2. However, depending on the adopted frequency of exceeding, the database to be collected may be altered in the following manner.

In the four scale approach recommended in SSG-9 [3], the size of the region, near region, site vicinity and site area need to be preserved. However, the investigation scales (corresponding to the actual scales of the mapping to be produced) may be reduced from the recommended values without losing important information.

Geophysical studies can be made on an as needed basis without resorting to generic grids covering unnecessarily large areas.

A full scope fault capability analysis has to be developed for HC-1 and HC-2 facilities, according to SSG-9 [3]. However, for an HC-2 facility it may be carried out on the basis of a solely deterministic approach. For HC-3 and HC-4 facilities a geological judgement may replace a formal evaluation of capability based on detailed investigations.

The detailed geotechnical investigations to establish geotechnical design parameters have to be initiated after the possible hazards that could result in permanent soil deformation (including liquefaction, collapse and slope instability) have been ruled out. Similarly, boreholes can be drilled on an ad hoc basis, taking into account the type of foundation material and the depth needed for the various safety related structures of the research reactor facility.

4.2.1.2. Seismological database

Recommendations related to the geological, geophysical and geotechnical database that are applicable to HC-1 and HC-2 research reactor facilities are provided in paras 3.24–3.33 of SSG-9 [3]. However, depending on the adopted frequency of exceeding, the database to be collected may be adapted as depicted in Sections 4.2.2–4.2.6.

If available, the catalogue of earthquakes developed by official national institutions in charge of this type of work may be used in lieu of developing a project specific catalogue.

The decision to deploy a site specific local earthquake monitoring network can be made on a case by case basis according to need.

4.2.2. Volcano hazard

Volcano hazards are described in IAEA Safety Standards Series No. SSG-21, Volcanic Hazards in Site Evaluation for Nuclear Installations [5]. Table 1 of SSG-21 [5] provides a comprehensive list of volcanic phenomena and the corresponding potential effects to be considered for nuclear installation safety. Many of these effects are considered exclusionary within certain screening distances. These distances are generally determined taking into account the inherent robustness of the engineered structures of the nuclear installation and therefore have to be greater for HC-3 and HC-4 facilities.

Recommendations related to the volcano hazard evaluation database that are applicable to HC-1 and HC-2 research reactor facilities are provided in paras 4.1–4.36 of SSG-21 [5]. However, depending on the adopted frequency of exceeding, the database to be collected may be altered in the following manner:

- Geophysical studies can be made on an ad hoc basis without resorting to generic grids covering unnecessarily large areas.
- For topics such as “monitoring for unrest and eruption” (para. 4.36 of SSG-21 [5]) and “emerging techniques” (para. 4.37 of SSG-21 [5]), the results of studies that are conducted on a national or international scale may be used instead of launching site specific studies for the research reactor facility.

4.2.3. Meteorological hazards

The following meteorological hazards are considered in SSG-18 [4]:

Meteorological variables:

- Air temperature;
- Wind speed;
- Precipitation (liquid equivalent);
- Snowpack.

Rarely occurring meteorological phenomena:

— Lightning;
— Tropical cyclones, typhoons and hurricanes;
— Tornadoes;
— Waterspouts.

Other possible phenomena:

— Dust storms and sandstorms;
— Hail;
— Freezing precipitation and frost related phenomena.

Recommendations related to the meteorological database that are applicable to HC-1 and HC-2 research reactor facilities are provided in paras 3.11–3.26 of SSG-18 [4]. However, depending on the adopted frequency of exceeding, the database to be collected may be altered in the following manner: some of the above mentioned hazards may be screened out depending on the site climatological conditions. The on-site meteorological programme needs to respond to the requirements of the nuclear installation design (e.g. points of release).

4.2.4. Hydrological hazards

The following hydrological hazards are considered in SSG-18 [4]:

— Storm surges;
— Waves;
— Tsunamis;
— Seiches;
— Extreme precipitation;
— Sudden release of water from natural or artificial storage;
— Water level rising upstream or falling downstream;
— Landslides or avalanches into water bodies;
— Waterspouts;
— Deterioration or failure of facilities on the site or near site facilities (e.g. canals, water retaining structures or pipes);
— Swelling of water in a channel due to a sudden change in the flow rate;
— Variation of groundwater levels;
— Subsurface freezing of super cooled water (frazil ice).

In general, research reactor facilities do not need the very large quantities of water that nuclear power plants do. Therefore, these need not be sited at coastal locations, and, through a prudent siting process, major hydrological hazards such as storm surges, seiches and tsunamis may be screened out. Similarly, these facilities do not need to be near major rivers and therefore some other scenarios mentioned above may also be screened out. As flooding is a major safety issue for nuclear installations in general, especially after the experience of the Fukushima Daiichi accident, it is recommended that flooding issues be avoided through the site selection process.

Recommendations related to the hydrological database are provided in paras 3.27–3.40 of SSG-18 [4] for research reactor facilities in HC-1 and HC-2. However, depending on the adopted frequency of exceeding, the database to be collected may be altered in the following manner: monitoring programmes related to hydrological or hydrogeological data collection may be tailored according to the needs of the research reactor facility.

4.2.5. Human induced external hazards

The following human induced hazard sources are considered in IAEA Safety Standards Series No. NS-G-3.1, External Human Induced Events in Site Evaluation for Nuclear Power Plants [28]:

Stationary sources:

— Oil refineries;
— Chemical plants;
— Storage depot;
— Broadcasting network;
— Mining or quarrying operations;
— Forests;
— Other nuclear facilities;
— High energy rotating equipment;
— Military facilities (permanent and temporary).

Mobile sources:

— Railway trains and wagons;
— Road vehicles;
— Ships;
— Barges;
— Pipelines;

— Airport zones;
— Air traffic corridors and flight zones (military and civil).

These sources may cause one or more of the following initiating scenarios or load cases that need to be considered:

— Explosion;
— Fire;
— Release of flammable, explosive, asphyxiant, corrosive, toxic or radioactive substances;
— Ground collapse, subsidence;
— Projectiles;
— Electromagnetic interference;
— Eddy currents into the ground;
— Blockage;
— Contamination (such as from an oil spill);
— Impact;
— Aircraft crash (involving impact, vibration and fire).

Many of these effects are considered exclusionary within certain screening distances. These distances are generally determined taking into account the inherent robustness of the engineered structures of the nuclear installation and therefore have to be greater for HC-3 and HC-4 facilities.

Recommendations related to the human induced event database are provided in paras 3.12–3.31 of NS-G-3.1 [28] for HC-1 and HC-2 facilities.

4.2.6. Dispersion of radioactive material in air and water and consideration of population distribution

4.2.6.1. Dispersion in air

The data to be collected generally include the following:

— Wind vectors (i.e. wind directions and speeds);
— Specific indicators of atmospheric turbulence;
— Precipitation;
— Air temperatures;
— Humidity;
— Air pressure.

These data, in general, involve on-site instrumentation and monitoring.

Recommendations related to the atmospheric dispersion database that are applicable to HC-1 and HC-2 research reactor facilities are provided in paras 2.10–2.37 of IAEA Safety Standards Series No. NS-G-3.2, Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants [31].

4.2.6.2. Dispersion in surface water

Dispersion in surface water is relevant when the research reactor facility is located near:

— Rivers;
— Estuaries;
— Open shores of large lakes and oceans;
— Human made impoundments.

In general, research reactor facilities do not need large quantities of water; hence this issue can easily be avoided by siting the facility away from large bodies of surface water. The data to be collected vary with the body of water in question and are provided in paras 3.12–3.19 of NS-G-3.2 [31].

4.2.6.3. Dispersion in groundwater

Considerations for dispersion of radioactive effluents in groundwater are relevant for all research reactor facilities because small leaks may occur which, depending on the designed and natural barriers between the leakage and the aquifer, may pose a potential risk to health and safety. This may be caused through direct or indirect (such as fallout) paths.

The data to be collected generally involve (both on a regional and a local scale):

— Climatological data;
— Initial concentrations of radionuclides;
— Major hydrogeological units, their hydrodynamic parameters and the ages or mean turnover times of groundwater;
— Recharge and discharge relationships;
— Data on surface hydrology.

Recommendations related to the groundwater dispersion database that are applicable to HC-1 and HC-2 research reactor facilities are provided in paras 3.26–3.34 of NS-G-3.2 [31].

The instrumentation and monitoring related to atmospheric dispersion, surface water dispersion, and groundwater dispersion have to be tailored to the needs and characteristics of the research reactor facility. The type and number of instruments, their location and data sets required as well as the duration of observation all depend on the needs of the installation and its potential impact on the population, workers and the environment.

4.2.6.4. Population distribution

Recommendations given for population distribution are provided in paras 5.1–5.15 of NS-G-3.2 [31]. In general, the type of data to be collected regarding population would not differ from one type of facility to another. However, the size of the region (the radius) has to be adjusted according to the source term and the engineered safety features of the research reactor facility.

5. DERIVATION OF EXTERNAL EVENT HAZARD

5.1. EVENTS CONSTITUTING DESIGN BASIS AND DESIGN EXTENSION CONDITIONS

Design basis events are those for which a research reactor facility is designed using conservative rules and procedures. The events are first defined by their parameters. These parameters are then used to define the design basis load cases for the facility’s safety related SSCs.

By definition, a design process that uses pessimistic assumptions, sophisticated methods of analysis and adequate safety margins is conservative. The conservatism is expressed both in the consideration of the systems (such as the principles of redundancy, diversity and single failure criterion) and in the behaviour limits used in the analysis of structural and mechanical components.

The purpose of defining design extension conditions is different. In particular, after the accident at the Fukushima Daiichi nuclear power plant, checking for cliff-edge effects has gained further importance. One established way of checking for these effects is by defining design extension conditions.

As stated above, the design process is conservative by definition and therefore a design basis external event is not expected to cause any damage to the safety related SSCs of a research reactor facility. The question is how much the loads can be increased due to the external event before they start causing real damage, such as contributing to the likelihood of core damage.

The design extension condition is used as a sufficiently greater event (compared with design basis events) to check whether or not significant damage occurs. The criteria used in this analysis are different from the conservative assumption used in the design process. The ratio of the loads that the facility can bear for design extension conditions to the design basis events can be thought of as the facility's margin in relation to the particular external event under consideration. When this ratio is small there is a greater chance of cliff-edge behaviour than when the design basis levels of external events have been exceeded.

If the same database and methodology recommended in safety guides for nuclear power plants are used, then the acceptance criteria can be relaxed (1 to 1.5 orders of magnitude). If this is not the case (i.e. a reduced database and simplified methods have been used), then the reduction can only be less than one order of magnitude.

A preliminary screening of the external event to be considered in the design of a research reactor can be conducted on the basis of a detailed hazard categorization. As mentioned in the previous section, for facilities in the lowest hazard category (e.g. HC-3), some extreme scenarios (aircraft crash, blast loads, tornado) have to be screened out by the appropriate site selection.

The screening process has to then consider the potential for off-site and on-site consequences induced by the external event. Table 6 presents possible consequences of external events to be analysed for screening. The analysis needs to also consider the following issues:

(a) Effects from and to the collocated facilities;
(b) Concurrence of uncorrelated external events (e.g. earthquake and frequent sandstorm);
(c) Complex external event scenarios made up of consequent external events of different natures (e.g. earthquake and tsunami).

Once the external event scenarios potentially affecting the site have been identified, a screening phase has to be carried out on the basis of their probability of occurrence. Thresholds have to be set up depending on the hazard category of the facility and the external event category assigned to that scenario (see Sections 2.6, 2.8).

The engineering practice described in Refs [2, 24] has to be referred to in the development of the screening phase, with special emphasis on the potential consequences associated with any scenario and therefore on the performance goal associated with the facility. Scenarios with low probability but high risk have to be retained for further analysis.

TABLE 6. EXAMPLES OF POSSIBLE CONSEQUENCES OF EXTERNAL EVENTS ON RESEARCH REACTORS

External event	Off-site damage	On-site damage	Safety function/ items affected	Severity of potential damage and warning
Geological and geotechnical hazard	Loss of off-site power and other utilities; Loss of access roads; Damage to emergency planning infrastructure	Loss of building stability; Operator action impaired	Structure stability and integrity	High; Sudden action without warning
Earthquake	Loss of off-site power; Communication and other utilities; Damage to emergency planning infrastructure	Collapse of non-classified buildings; Seismic induced fire and flood; Operator action impaired	Shutting down; Heat removal; Confinement; SSC integrity and operability	High; No warning
Extreme wind	Loss of off-site power	Collapse of non-classified buildings; Operator action impaired	Partial structure integrity	Moderate; Monitoring system warning
Extreme rain, snow and ice	Loss of off-site power; Damage to emergency planning infrastructure	Operator action impaired; Water egress to buildings and underground structures	Structure integrity	Low; Monitoring system warning
Flooding	Loss of off-site power and other utilities	Operator action impaired; Water egress to buildings and underground structures	Reactivity control; Confinement system	High-Moderate; Monitoring system warning
Abrasive dust and sandstorm	Loss of off-site power and other utilities	Operator action impaired	Ventilation system	Low; Monitoring system warning

TABLE 6. EXAMPLES OF POSSIBLE CONSEQUENCES OF EXTERNAL EVENTS ON RESEARCH REACTORS (cont.)

External event	Off-site damage	On-site damage	Safety function/ items affected	Severity of potential damage and warning
Lightning	Loss of off-site power	Operator action impaired	Structure integrity	Low; No warning
External fire	Loss of off-site power; Loss of access	Operator action impaired; Smoke and flame egress to classified areas	Ventilation system	Low; Monitoring system warning
External off-site explosion	Loss of access roads; Loss of off-site power	Damage to non-classified structures; Operator action impaired	Structure integrity	Moderate; No warning
Aircraft crash		Large fire (and smoke) at the site; Structural damage	SSC integrity and operability	High-Moderate; No warning
Release of hazardous liquids/gas	Affects the personnel of the research reactor	Operator action impaired	Ventilation system	Low; Monitoring system warning
Electromagnetic interference from off and on the site	Difficulties in communication	Electronic control systems; Difficulties in communication	Reactivity control	Low; Monitoring system warning

In the following sections references are provided on developing an external event design basis for a nuclear installation through the following steps:

(i) Retrieval of a proper data base of site information from both existing records and site investigation campaigns organized on purpose (see Section 4);
(ii) Selection of suitable methodologies for hazard development;
(iii) Design basis identification corresponding to the applicable return period.

In some cases, a full probabilistic hazard is not developed due to identifiable source characteristics (e.g. human induced events from fixed industrial facilities); therefore, deterministic parameters for the scenario are suggested from engineering practice and discussed in the following sections.

5.2. DESIGN BASIS FOR EXTERNAL EVENTS

5.2.1. Seismic input

For HC-1 and HC-2 facilities, site specific design response spectra (including site effects) have to be developed according to the relevant safety guide (SSG-9 [3]).

For HC-2 facilities, either a deterministic or probabilistic hazard may be developed. The design spectrum may be evaluated from the envelope of response spectra (at 5% damping) calculated from recorded data, with a suitable margin, and extrapolated to the required return period. The relevant epistemic tree may be limited to consider 'sustainable' values for variables and weights, without application of expert elicitation methodology.

For HC-3 facilities, if the regional tectonics are well understood and there is no geological evidence of tectonic structures in the near region able to affect the site, the design spectrum may be evaluated by extrapolating the hazard maps to the required return period value. In any case, a minimum value of design free-field acceleration for firm bearing strata has to be assigned.[12]

5.2.2. Extreme meteorological events

For HC-1 and HC-2 research reactors, the design value for precipitation has to be based on the probability of exceeding, compatible with the performance goals assigned to SSCs.

Alternatively, scaling factors can be applied to the prescribed value of the building code to account for the difference in the return period of the equivalent load.

Maximum and minimum values for air temperature have to be evaluated on the basis of statistical analysis of a representative time record. The analysis has to identify absolute maxima and minima as well as the same quantities for reference time intervals, traditionally taken at 6 hours and 1 day.

Water temperature has to be evaluated, also for its maximum and minimum values, with reference to operational limits and ice formation. For most research

[12] This value is usually set at 0.1g.

reactors that scenario does not challenge any related operational issues but emergency plans, at least, have to be assessed with that concern in mind. Further guidance is provided in SSG-18 [4].

5.2.3. Extreme straight wind

The extreme, normal (rather frequent) and frequent values for wind speed can be determined either from site monitoring data or the wind speed standards of the national building code.

For HC-1 and HC-2 research reactors, the design basis wind has to be evaluated on the basis of the selected probability of exceeding the external event hazard, according to the performance goal assigned to their SSCs important to safety. For more sophisticated investigations and analysis, further guidance is provided in SSG-18 [4].

For HC-3 facilities, the design basis wind can be determined according to the national building codes for hazardous facilities, scaled up to the required occurrence probability, if needed.

National building codes typically give design basis wind velocities and pressure distributions, including variation in the height above ground and relative values with respect to a building's geometry. These assumptions may be applied to the research reactor design, provided site specific topography is evaluated.

If site effects are expected to be significant, a monitoring system is usually installed and operated for comparison with regional data.

5.2.4. Flooding

Research reactors generally do not need large amounts of cooling water. Therefore, it is not important for them to be located close to large bodies of water such as the sea, a lake or a river. It is often possible to select 'dry sites', that is, sites which are well above flood level at all times. If it is not possible to select dry sites, it is necessary to construct all safety related items at an altitude above the reference level of the flood, which can be evaluated using the methods given below.

Floods may be generated by different sources: meteorological, hydraulic (collapse of water barriers), or seismic (tsunamis).

The design basis has to be evaluated by analysing the source and its propagation to the site, as the analysis of historical records is usually not sufficient to support a detailed hazard evaluation. Hazard maps developed on a national basis (especially for civil protection purposes) may be used as background information on occurrence probability and scenario characteristics. However, local effects have to be well understood, especially for HC-1 and HC-2 sites, and added to the overall information.

In all scenarios (i.e. for river and coastal sites), special emphasis has to be paid to the definition of a realistic but conservative combination of events; for example, on coastal sites the combination of wind wave, tide, storm surge, effects from rivers and sea is rather frequent. Sustainable values for the occurrence probability of the individual events have to be chosen, with reference to the probability of their combination and to the correlation effects. Further guidance is provided in SSG-18 [4].

5.2.4.1. River sites

The boundaries of the region to be investigated for river flooding hazard depend primarily on whether the rivers could cause floods large enough under extreme conditions to contribute to flooding at the site. Regional investigations have to be conducted for rivers relatively close to the site (in general, rivers with flood plain boundaries closer than a few kilometres from the site).

For sites located near rivers, the reference flood can be evaluated in two ways:

(1) Empirical formulas that have been developed for various parts of the world give a relationship between drainage basin parameters and potential flood characteristics.
(2) Extrapolated hazard curves, based on a series of maximum annual flows, can be used for evaluating the reference flood. These hazard curves can be derived from the available data and take into account random components, trends and jumps. If properly used, this method allows a reasonable evaluation of a reference flood.

Results evaluated for HC-1 and HC-2 research reactors have to be higher than any recorded historical occurrence, with some added margin. Based on the reference flood flow, a reference level can be obtained with appropriate hydraulic formulas which take into consideration the average river channel slope, channel cross section and friction factors. Due consideration has to be given to the presence of river channel obstructions close downstream from the site, since they can provoke backward elevation at the site.

The effect of a dam failure upstream from the site has to be evaluated by assuming simultaneous failure of all dams on the same stream.

5.2.4.2. Coastal sites

For coastal sites, the best protection is to use a dry site. To establish the reference level for such a site, the potential for coastal flooding has to be evaluated first. If the region of the site is subject to tropical storm effects (typhoon, hurricane, cyclone) or if there is a history of tsunamis, then historical data on the phenomena have to be collected. An analysis of the available data can give a good indication of the maximum flood level at the site. An adequate margin provides the minimum level for dry sites.

If flooding of the site is not precluded, either an embankment may be constructed or the design water load suggested in applicable national building codes may be used for design purposes. Analytical models which include both hydrostatic and hydrodynamic loads on safety related SSCs have to be used to correct the reference water elevation provided in the regional maps.

5.2.5. Rotational wind

Tornadoes, hurricanes, typhoons and cyclones are violently rotating winds which can reach speeds in excess of some hundreds of km/h. However, for the purposes of this publication, cyclonic winds may be regarded as straight winds when the scale of the phenomena is very large compared to the site. Tornadoes are traditionally addressed as rotational winds and the design basis is usually developed with reference to the equivalent wind velocity, which averages maximum velocity in the eye and peripheral speed.

High probability rotational wind sites are those where rotational wind velocities exceed extreme straight winds at a 10^{-4}/a probability of exceeding. Moderate probability rotational wind sites are those where rotational wind velocities exceed extreme straight winds at a 10^{-5}/a probability of exceeding. Low probability rotational wind sites are those where rotational winds exceed straight velocities at a 10^{-6}/a probability of exceeding.

Rotational winds can be excluded from the design basis if the rotational wind probability of exceeding is less than the probability of exceeding for the selected external event. For sites with HC-3 research reactors, tornadoes do not have to be considered unless they are included in national building code requirements for that site.

For HC-1 and HC-2 facilities, a proper hazard evaluation has to be carried out according to SSG-18 [4] and design basis values selected according to the relevant PG value (Table 3).

5.2.6. Wind-borne missiles

Usually a thorough hazard analysis is not carried out for wind-borne missiles and some experience-driven deterministic evaluations are applied to define envelope values for their action associated with the wind scenario selected for the site.

Missiles to be considered in design are typically of two types: penetrating and impacting. Penetrating missiles typically have relatively high velocity, are rigid and have small impact areas. A typical penetrating missile would be a 10 cm diameter pipe weighing 30 kg travelling at 0.6 times the maximum wind velocity. An impact missile typically has a relatively large mass, slow velocity and large impact area. A typical impact missile would be an 1800 kg automobile travelling at 0.2 times the maximum wind velocity [32].

5.2.7. Accidental chemical explosions

The site has to be located in an area where the effects from explosions are not significant; conversely, the facility has to be protected against these events.

The design can be developed on the basis of the data on fixed and mobile sources, their potential and nature, following the approach of an equivalent explosion, of TNT, particularly if the source is relatively far from the facility. For this purpose, two coefficients are applied to the identified mass of explosive material:

- An equivalent TNT mass ratio is applied to the mass of explosive product and gives the equivalent mass of TNT for its explosive effects.
- The detonation factor defines the ratio of the total mass that participates in the explosion, depending on storage or transport conditions. If a more rigorous estimation is not done, for hydrocarbons this ratio is taken to be equal to 20%.

According to the specialized literature, for the estimated equivalent mass of TNT and distance from the facility, an overpressure triangular wave can be postulated which includes the value and duration of the instant overpressure. When applying the derived pressure wave to the building, it is important to take into account reflection effects on walls, depending on the relative direction of walls and pressure wave propagation (this coefficient, depending on the proximity of the explosive source to the wall, can reach amplification factors which typically vary from 2 to 8) and dynamic effects due to the rise time of the blast wave relative to the period response and ductility of the structure (this coefficient can also reach a value of 2).

If the explosion risk is evaluated to be significant in terms of the pressure wave, then further studies need to be conducted or the site has to be rejected.

5.2.8. Aircraft crash

The site for the research reactor may be located in an area where the risk associated with an aircraft crash (enveloped over all potential aircraft categories) is not significant. In general, if this probability is higher than 10^{-5}/a, the facility design has to consider the impact characteristics corresponding to their category.

The aircraft crash design scenario has to be derived from an extensive analysis of the air traffic in the vicinity of the facility and over its lifetime.

From the geometry of the identified aircraft with an associated significant probability of crash at the site, a virtual area of the facility has to be defined as the mean normal section of cylindrical projection of the facility under the different crash angles. Finally, the probability of an aircraft crash on the facility has to be evaluated as the product of the probability of impact per unit surface and per year multiplied by the virtual surface of the facility.

5.2.9. External fire

The hazard for fire originating outside the facility has to be considered. The scenario may refer to a brushfire, forest fire or yard fire, according to the terrain conditions at the site boundary.

The design has to include appropriate fire protection measures (for prevention, alert, mitigation and suppression) both in the site vicinity and at the site itself, in coordination with local authorities.

5.3. HAZARD MODIFICATION OVER TIME

Sufficient margin in the design basis parameters can accommodate hazards severity change with time. The extra cost linked to the overestimation of such parameters in the design phase must be balanced with the hypothetical cost of future retrofitting of the research reactor to accommodate evolution.

Evolution of air traffic has to be anticipated, as well as of infrastructure, such as the introduction of dam equipment on an unequipped river or highway construction in valleys which then leads to modification of the flooding parameters. Climate change may also induce significant changes in meteorological and hydrological hazards.

An anticipation of the evolution of human activity has also to be taken into account in the design parameters. Further information is provided in Ref. [16].

6. GENERAL DESIGN APPROACH

6.1. GENERAL CONSIDERATIONS

Both the design of new research reactors and the re-evaluation of existing ones in relation to external events follow a common process, which can be summarized as follows:

(a) Evaluation of the design basis of the facility in relation to external events.
(b) Evaluation of loads and other induced effects of external events on each SSC.
(c) Evaluation of other loads and effects related to normal operation, normal environmental conditions (concurrent with the given external event), anticipated operational occurrences and accident conditions (if any, concurrent to the given external event).
(d) Selection of acceptable design or re-evaluation approaches (for each SSC and each external event), among the following:
    — Qualification by analysis: use of code-based stress and strength analysis (A in Table 7);
    — Qualification by testing (T in Table 7);
    — Qualification by experience (E in Table 7);
    — Qualification by special investigation when A, T or E is not applicable (S in Table 7); special analysis (beyond the conventionally used standard based stress/strength analyses); and/or special testing (beyond the conventionally used test procedures).
(e) Selection of acceptable codes (standards) for design and re-evaluation purposes (for each SSC and each external event).
(f) Development of the design and re-evaluation (for each item and each external event) which means:
    — Selection of an appropriate design and re-evaluation methodology.
    — Identification of load combinations to be considered.
    — Qualification by analysis:
        • Demand determination for a qualified item and for specified load combinations;
        • Capacity determination for a qualified item;
        • Comparison of demand to capacity.
    — Qualification by testing.
    — Qualification by experience.
(g) Calculation of the available safety margin for design extension conditions, with reference to suitable acceptance criteria.

TABLE 7. SUMMARY OF ACCEPTABLE DESIGN APPROACHES FOR SSCs OF RESEARCH REACTORS IN RELATION TO EXTERNAL EVENTS

External event		Acceptable design and re-evaluation methods
Natural	Earthquake, including other seismic induced effects	A, T, E
	Extreme wind	A
	Extreme snow	A
	Flooding	A (limited), design rules/provisions
	Extreme temperature	A
	Extreme frost, subsurface freezing, drought, hail	Design rules/provisions only
	Cyclones (hurricanes, tornadoes, tropical typhoons)	S
	Abrasive dust and sand storms	S
	Landslides and avalanches	S
	Lightning	Design rules/provisions only
	Volcanism	S
Human induced	Explosions (deflagrations and detonations)	A and/or S
	Aircraft crash	A and/or S
	Release of hazardous gas	S
	Release of corrosive gas and liquid	S
	External fires	S and/or design rules/provisions
	Collision of ships and floating debris	S
	Electromagnetic interference	S and/or design rules/provisions
	Combinations of the events above as the result of a common initiating event	A, T, S and/or design rules/provisions

Note: A — analysis, E — experience, S — special investigations, T — test.

TABLE 8. SELECTION OF ACCEPTABLE CODES (STANDARDS) BASED ON THE DESIGN CLASSIFICATION OF SSCs

Design class of SSC [a]	Codes (standards) for design purposes	Codes (standards) for re-evaluation purposes
DC-1	Nuclear codes (e.g. ACI349, ASME section III, RCC, KTA, PNAE)	Nuclear codes + best up to date engineering knowledge and experience
DC-2 DC-3	Nuclear codes (with some relaxations) or Conventional non-nuclear codes (with some enhancement) (e.g. Eurocode, ACI318, ASME code, KTA) (see Section 2.9)	Nuclear codes + best up to date engineering knowledge and experience or Conventional non-nuclear codes + best up to date engineering knowledge and experience
DC-4	Conventional non-nuclear codes	Usually not needed

Note: [a] For the definition of design class, see Section 2.10.

ACI — American Concrete Institute, ASME — American Society of Mechanical Engineers, DC — design class, KTA — Kern-Technischer Ausschuss (Nuclear Safety Standards Commission), PNAE — rules and norms in nuclear industry, RCC — reinforced cement concrete.

6.2. SELECTION OF DESIGN APPROACH

A single or combination of analytical, experience based and/or special analysis/testing is used in the design/qualification of SSCs. Table 7 summarizes the general methods, as defined in Section 6.1, for selecting acceptable design approaches for SSCs of research reactors in relation to external events.

6.3. SELECTION OF CODES AND STANDARDS

According to the procedures followed for the hazard evaluation, and in agreement with the design classification, the codes (standards) to be applied for design and re-evaluation purposes can be selected in accordance with Tables 4 and 8. The recommendations in Table 8 have to be assessed against the values in Table 4.

In particular, the definition of performance goal and external event category drives an informed choice of the standard to be used for design and qualification, between ‘nuclear grade’ and ‘conventional’.

The use of nuclear grade standards is recommended for higher class SSCs in combination with design basis values chosen at a lower probability of occurrence.

6.4. SELECTION OF THE APPROPRIATE DESIGN METHOD

The selection of an appropriate design and re-evaluation method can be based on a clear understanding of safety functions assigned to each SSC, their potential failure modes and relevant acceptance criteria (e.g. integrity, stability, operability).

Equipment and components whose active safety functions are required have to be designed to ensure their operability during and/or subsequent to the considered external event.

The design margin for SSCs subjected to external events is usually at least the same as those design margins that are adopted in related design practices for extreme events as specified by the corresponding acceptable codes or standards (see Table 8).

Deterministic methods are typically used for design and re-evaluation purposes. In load factor design (limit state design), the behaviour limits and design margins are defined by variable load factors with set limits on stress, strain or deformation. This is in contrast to working stress design (allowable stress design) where the variable behaviour limits and design margins are applied to stress, strain or deformation for a fixed set of loads. Increasing allowable stress, strain or deformation has the same effect as reducing load factors and design margins in linear systems.

The choice of the design procedure can be associated with some additional conservatism: procedures to deal with it are available in IAEA Safety Standards Series No. NS-G-1.5, External Events Excluding Earthquakes in the Design of Nuclear Power Plants [26].

Most of the available engineering procedures deal with the seismic qualification of SSCs. Examples of qualification procedures in relation to other external events can be taken from the nuclear power plant engineering community.

Suitable coefficients can be applied to the results to compensate for the level of conservatism associated with any calculation methods. An adequate validation of such coefficients has to be provided.

7. DESIGN AND QUALIFICATION OF SSCs

7.1. MODELLING THE SCENARIOS

7.1.1. General considerations

The external event scenarios to be used for the safety assessment of SSCs have to be converted into loads and load combinations, in consideration of their complex nature and of their correlation with internal accident scenarios at the research reactor.

Load combinations for external events and the corresponding load factors have to be in accordance with the applicable standards and codes (see Table 9).

The facility states and ambient loads are typically grouped as follows:

- L1: Loads due to normal operation and/or normal ambient conditions.
- L2: Additional loads due to anticipated operational and/or anticipated ambient conditions.
- L3: Additional loads due to accidental conditions.

Loads L2 and L3 are usually included in load combinations for external events if they are concurrent with the particular external event, that is, if they are caused by the external event or if they have a high probability of coinciding with this particular external event. They can be identified on the basis of probabilistic considerations.

Suitable combinations of loads and scenarios have to also be applied when assessing the highest levels of the defence in depth, namely the emergency planning; site access and evacuation represent a real concern in scenarios triggered by severe external events.

In general, the load combinations can follow the practice suggested in standard building codes. Only seismic and impact loads may follow a different treatment, as explained in the following sections.

Special care has to be taken in modelling scenarios which may induce the cliff-edge effects, such as flooding. Different acceptance criteria and safety margins have to be applied in those cases to account for the sudden increase in risk.

7.1.2. Earthquakes

Table 9 shows typical seismic load combinations and load factors that may be used for SSCs of design classes 1 and 2.

TABLE 9. TYPICAL LOAD COMBINATIONS AND LOAD FACTORS TO BE USED FOR SSCs

SSC	Seismic load combinations and load factors[a, b, c]
Bearing concrete and steel building structures	1.0 D + 1.0 L + 1.0 T + 1.0 S
Non-bearing building structures	1.0 D + 1.0 L + 1.0 S
Passive and active equipment components	1.0 D + 1.0 L + 1.0 P + 1.0 S
Equipment nozzles, pipe flange/threaded connections	1.0 D + 1.0 L + 1.0 T + 1.0 P + 1.0 S
Equipment supports and their anchorage	1.0 D + 1.0 L + 1.0 T + 1.0 S
Pipes	1.0 D + 1.0 L + 1.0 P + 1.0 S
Pipe supports and their anchorage	1.0 D + 1.0 L + 1.0 T + 1.0 S
Cable structures, supporting platforms, etc. (including their anchorage)	1.0 D + 1.0 L + 1.0 T + 1.0 S

Notes: [a] D = Dead load;
L = Live load under NOC (the part of the live load that is applicable at the time of an earthquake);
T = Temperature load, including temperature gradients and due to restrained free temperature displacement under NOC (if any);
P = Internal pressure under NOC (if any);
S = Seismic load (inertia effect combined with seismic anchor movement, if any, using the square root of the sum of squares rule);
NOC = Normal operational (and environmental) conditions.

[b] Temperature load effects are typically considered in combination with earthquake load effects on structures but are not so considered in evaluating mechanical systems and components. Loads due to restrained free temperature displacement and seismic anchor movement are not considered for pipes themselves but are considered for component nozzles, pipe supports and the supporting structures.

[c] Acceptance criteria or capacity as defined in the applicable code or standard.

7.1.3. Aircraft crashes

Evaluation of the effects of an aircraft crash can include:

— Global bending and shear effects on the affected structures (‘overall missile effects’) and global overturning;
— Induced vibrations on structural members and safety related equipment (‘global effects’), particularly when safety related items are located close to the external perimeter of the structures;

— Localized effects. including penetration, perforation, scabbing and spalling, by primary and secondary missiles ('local effects');
— The effects of fuel fires and possible explosions on structural members as well as exposed safety related equipment (ventilation system, containment openings, air baffles).

In general, research reactors do not show a distributed resistance to a crash, as they are built with steel and concrete frame structures; only continuous concrete walls at the external boundary of a building can provide some degree of protection.

The location of an impact is potentially anywhere on a building (peripheral walls and roof) and a flying object can travel in any direction inside a building. In principle, all exposed structural elements are checked against all mechanisms discussed above. Moreover, the definition of the impacting object is usually very difficult to determine and could be any of a wide variety of aircraft, helicopters, missiles and the like.

For local and global analysis, the load combination for local stress/strain analysis is typically:

$$1.0 \text{ Normal loads (dead + live)} + 1.0 \text{ Aircraft crash loads} \tag{1}$$

Concerning the effects of aircraft or missile fuel, a dedicated analysis can be carried out to review fuel access into the facility; the criteria described for explosion and fire may be applied here.

7.2. QUALIFICATION BY ANALYSIS

7.2.1. Evaluation of the external event demand for SSCs for specified load combinations

It is common engineering practice to determine the demand for an analysed SSC and for a specified load combination based on the assumption that the SSC behaves in a linear elastic manner. In such a case, the principle of superposition is applicable. When plastic behaviours are significant, the ductility (i.e. the ability to strain beyond the elastic limit) model still allows for linear modelling, provided suitable correction factors are applied (typically, the inelastic energy absorption factors). In other cases, such as the analysis of the response of civil structures that are subjected to high impact loads, the non-linear plastic analysis is widely used [21].

7.2.2. Capacity determination for qualified SSCs

For design purposes, the capacity determination of analysed SSCs is based on the limits (stress and strength for materials and other appropriate characteristics) in the selected codes and standards (Table 8) relative to all potential critical failure modes for the analysed item. These limits are the same as those employed in related engineering practices for extreme load combinations.

If the safety function is associated with a structural failure, the reference behaviour limit in terms of factors such as stress and strain needs to be defined in evaluating the failure for SSCs. The design stress limits required by design codes for conventional risk facilities for normal loads such as dead, live load and operating pressure vary between one half to two thirds of the yield stress of the material, with a resulting median probability of failure of about 10^{-4}/a, corresponding to the design load. Occasional or extreme loads, which typically have a probability of exceeding in the range of 10^{-1}/a to 10^{-2}/a, have allowable stresses increased by between 20% and 33% and conditional probabilities of failure in the range of 2×10^{-4}/a to 10^{-3}/a.

For structures, the limiting behaviour levels are at yield or approximately 1.2 times yield, which give a probability of failure in the range of 5×10^{-3}/a to 10^{-2}/a, assuming stresses have been computed elastically. For mechanical components, higher stress levels are typically allowed up to twice the yield or 70% of the ultimate stress. However, there is some conservatism in the analysis such that the failure probability ranges between 10^{-2}/a and 5×10^{-2}/a, with the fragilities expressed as median capacities.

For re-evaluation purposes, the capacity determination of an analysed SSC may be based on the 95% exceeding of actual material strength limits. If such test data are not available, the corresponding limits from the selected codes and standards (Table 8) are used, if properly verified by in situ investigations. Additional details are provided in IAEA Safety Standards Series No. NS-G-2.13, Evaluation of Seismic Safety for Existing Nuclear Installations [7] for the seismic case.

7.2.3. Comparison of demand with capacity

The general acceptance criterion for a comparison of demand with capacity can be written as follows:

$$(D_{NOC} + D_{ANOC} + D_{AC} + D_{EE}) \leq C \tag{2}$$

where

D_{NOC} = Demand on the SSC due to the effect of normal operation and normal environmental conditions (concurrent with the given external event);

D_{ANOC} = Demand on the SSC due to the effect of anticipated operational occurrence (if any, concurrent with the given external event);

D_{AC} = Demand on the SSC due to the effect of accident conditions (if any, concurrent with the given external event);

D_{EE} = Demand of the SSC due to the effect of a particular external event (or due to the effect of a rational combination of several external events resulting from the common initiating event);

C = Capacity of the SSC.

For earthquakes, and assuming that the SSC behaves in a linear elastic manner, the general acceptance criterion would be:

$$D_{EE} = D_E = [(D_{E,i} / k_D)^2 + (D_{E,a} \times k_{D,tot})^2]^{½} \tag{3}$$

where *demand* means strength demand, and

$$D_{EE} = D_E = [(D_{E,i} \times k_D)^2 + (D_{E,a})^2]^{½} \tag{4}$$

where *demand* means displacement demand, and

$D_{E,i}$ = Demand of the SSC due to the inertia effect of an earthquake event (or due to a combination of the inertia effect of an earthquake with other seismic induced effects);

$D_{E,a}$ = Demand of the SSC due to the anchor movement effect of an earthquake event (if any);

$k_{D,tot}$ = $\mathrm{k_{D,g}} \times \mathrm{k_{D,l}}$ = Total inelastic energy absorption factor (ductility factor);

$k_{D,g}$ = Global inelastic energy absorption factor, which relates to the overall response of a structural system such as a space frame, a planar frame, a load bearing shear wall, a non–load bearing shear wall (example values are provided in Appendix III);

$k_{D,l}$ = Local inelastic energy absorption factor, which relates to the local, member, or element ductility associated with columns, beams, bracing members, equipment components (example values are provided in Appendix III).

Notes: To determine the demand D_{NOC}, D_{ANOC} and D_{AC}, the rules and provisions of the selected codes (standards) are to be used (see Table 8).
The inelastic energy absorption factors can be applied only when the seismic response of the SSC is calculated in a linear elastic manner.

Nearly all SSCs exhibit at least some ductility (i.e. the ability to strain beyond the elastic limit) before failure or even significant damage. Because of the limited energy content and oscillatory nature of earthquake ground motion, this energy absorption is highly beneficial in increasing the seismic margin against failure. Ignoring this effect will usually lead to an unrealistically low estimation of the seismic failure margin. Limited inelastic behaviour is usually permissible for those facilities with adequate design details such that ductile response is possible or for those facilities with redundant lateral load paths. For SSCs of design class 3, when the seismic input is considered in accordance with the conventional non-nuclear codes or standards, the designer needs to verify whether the global ductility is not latently considered, for instance by some reduction factors applied directly to the seismic input.

Damping values have a proven high influence on the results of the seismic analyses of SSCs. Because of the engineering judgement required in defining their value, suggested values are provided in Appendix III.

References [26, 33] provide typical earthquake design provisions and proper structural detailing that apply to research reactors and comparable facilities.

As an alternative to the specific methodologies presented above, many simplified procedures can be used for seismic design and re-evaluation purposes in the solution of special problems, such as:

- The assessment of the potential for liquefaction [26, 34–35];
- The assessment of soil–structure interaction [26, 36];
- The calculation of pulling forces on anchor devices [26];
- The seismic resistance of pipelines with the load coefficient method [37];
- The evaluation of sloshing effects in large free surface pools and tanks [24, 27].

However, any simplified approach needs to be validated for the application of interest, as it is usually heavily dependent on engineering judgement.

For aircraft crashes, the acceptance criteria for the stress–strain fields induced in a structural element depend on the safety function assigned to each structural element. For local design, if the only function of the element is to stop the aircraft and maintain the global stability of the building, it may be designed with plastic excursions of reinforced bars reaching a tensile deformation of $\varepsilon = 2\%$.

If the structural element supports equipment that has to guarantee a safety function, the tensile plastic excursions can be limited to $\varepsilon = 1\%$ deformation. In both previous cases, namely local and global design, the acceptance criterion for concrete in compression can be $\varepsilon = 0.35\%$.

If the element has a tightness function, no plastic excursion can be allowed, and elastic behaviour has to be guaranteed. In this case, however, it is more convenient to design a shielding structure able to protect the safety related buildings.

Detailed methodologies for structural design are provided in Refs [21, 26, 27].

7.3. QUALIFICATION BY TESTING

Qualification by testing is primarily used to verify the seismic adequacy of equipment components and, in some cases, the seismic adequacy of specific building structures. Qualification by testing may also be used as a special investigative tool to verify the real capacity of structures and equipment when subjected to other external events.

The testing and processing of test data can be performed on the basis of the corresponding nuclear or industrial standards [38–43].

The types of testing can be summarized as follows:

— Type approval test (fragility test);
— Acceptance test (proof test);
— Characteristic test (e.g. dynamic characteristic test);
— Code verification test (generic verification of analytical procedures).

The qualification test programme may include the following elements:

— Determining the test sequence, test loads and acceptance criteria;
— Determining mounting conditions;
— Determining environmental and operating conditions (e.g. pressure temperature, voltage);
— Monitoring the output response and performance of the tested item during the test;
— Demonstrating the operability of the tested item (when required) and preparing the test documentation.

The testing procedure needs to be based on subjecting the item to conservatively derived test conditions in order to produce effects at least as severe as those of the design basis, concurrent with other operating or design conditions. Account needs to be taken of effects such as radiation and ageing, or other conditions that may affect the characteristics of the tested item during its in-service life. Caution is needed to take into account the external mechanical loads acting on the tested item (e.g. nozzle loads) [29].

The test results have to show a margin of at least 40% against the failure limit for design purposes and of at least 25% against the failure limit for re-evaluation purposes. References [38–43] provide further details on procedures and evaluation of test results for seismic testing of equipment.

7.4. QUALIFICATION ON THE BASIS OF EXPERIENCE

Currently, qualification methods that are based on experience are available primarily for seismic design and seismic re-evaluation of equipment [7, 29, 44–47]. Earthquake experience methods are simple and efficient tools to verify the seismic adequacy of selected mechanical, electrical, instrumentation and control equipment classes. Earthquake experience methods are also used to verify the seismic adequacy of piping, anchorage of piping supports, masonry walls and to check potential seismic interactions. These methods are primarily screening and walkthrough procedures; they are summarized in Appendix III, Table 15. Some of them involve establishing the similarity of the item being qualified to the item that has experienced the seismic motion. Paragraphs 6.29 and 6.30 of IAEA Safety Standards Series No. NS-G-1.6, Seismic Design and Qualification for Nuclear Power Plants [29] provide the following guidance:

> “6.29. The direct seismic qualification of items by means of the use of experience from strong motion seismic events has had limited but growing application. Only in recent years have data from strong motion earthquakes generally been collected in the quality and detail necessary to provide the information necessary for direct application to individual items.
>
> “6.30. The level of seismic excitation experienced during a real earthquake by an item identical to the item being qualified should effectively envelop the seismic design motion at the item’s point of installation in the building’s structure. The item being qualified and the item that underwent the strong motion should be of the same model and type or should have the same physical characteristics and have similar support or anchorage characteristics. For active items it should be shown that the item performed

the same functions during and following the earthquake, including any aftershock effects, as would be required of items in seismic category 1 or 3."

If an item is classified as an outlier (i.e. does not meet minimum capacity requirements or is unknown), more rigorous approaches such as testing on a shaking table, a more detailed study of input data and more sophisticated analyses may be needed to verify its adequacy.

These methods may be used for all research reactors and in any location with peak ground acceleration that does not exceed 0.33g. For higher design basis values, other approaches have to be applied.

Relays, switches, transmitters and similar electronic devices installed on research reactors may be significantly different from those considered in these methods. Therefore, their seismic functionality needs to be verified, if required, by testing.

7.5. ANCHORAGE OF EQUIPMENT

Anchorage of equipment plays a significant role in its functioning, especially during and after earthquakes. Experience has shown that inadequate anchorage has been a major cause of failure of equipment to function, especially during and after earthquakes, as the equipment components can slide, overturn or move excessively when not properly anchored.

Verification of equipment anchorage relies on a combination of inspections, calculations and engineering judgement. Inspections are performed to verify that equipment anchorage is in compliance with the facility documentation and drawings by way of measurements and visual evaluations. Calculations can be performed to verify that the anchorage capacity is sufficient to the corresponding loading (demand) imposed upon the anchorage. Another important factor in the evaluation process is engineering judgement.

Use of a graded approach is suggested in selecting combinations of inspections, calculations and engineering judgement to verify the adequacy of equipment anchorage. Where the failure of equipment to function could impair the safety function of the reactor or the anchorage is complex, with much of the equipment supported by the same anchorage, such as motor control centres with non-symmetrical anchorage, it is advisable to use appropriate computer codes that are especially tailored to equipment anchorage to determine the loads. For simple symmetrical anchorage, for example, a pump with few and very rugged anchor bolts in a symmetrical pattern, a simple hand calculation may be sufficient. The responsible engineer may select the appropriate combination of assessment methods for each anchorage installation based on the information available in the design documentation or from the walkthrough.

Reference [45] states:

> "The four main steps for evaluating the seismic adequacy of equipment anchorages include:
>
> (1) Anchorage Installation Inspection
> (2) Anchorage Capacity Determination
> (3) Seismic Demand Determination
> (4) Comparison of Capacity to Demand
>
> "It is not necessary to perform the above steps in the order given. Trade-offs between different alternative approaches could affect the order in which these steps are performed."

The capacities of anchors of various types and sizes are typically given for different loadings, geometric locations and other conditions in the manufacturer's specifications and in national standards. Further details on anchorage verification are available in Refs [45–48].

7.6. INTERACTIONS

7.6.1. Seismic interactions

Seismic interactions are physical interactions of any structures, distribution systems or mechanical or electrical components with nearby items of safety related structural systems or equipment caused by an earthquake.

The seismic interaction effects that need to be considered during the design/re-evaluation process are:

- Proximity (impacts of adjacent equipment or structures on safety related equipment due to their relative motion during an earthquake);
- Structural failure and falling of overhead or adjacent SSCs;
- Flexibility of attached lines and cables;
- Flooding due to earthquake induced failures of tanks or vessels;
- Fire induced by earthquake induced failures;
- Impairment of operator actions and/or access.

Practical approaches on how to avoid such seismic interactions and how to protect items important to safety are presented in Refs [34, 43, 49–50].

7.6.2. Other non-seismic interactions

These interactions mean physical interactions of any structures, distribution systems or mechanical or electrical components with nearby items of safety related structural systems or equipment caused by other non-seismic external events. They are considered during the design or re-evaluation process, if any, and assessed through expert walkthroughs.

7.7. AGEING

Ageing issues are quite relevant for research reactor assessment, as more than 70% of the world's research reactors are currently more than 30 years old.

Moreover, research reactors often do not have a limited design life, which could be too dependent on the rate of utilization of the facility, something that is difficult to plan for.

Ageing in research reactors can be managed through the following activities, often supported by a specific ageing management programme:

- Appropriate provisions during design (this needs to focus mainly on the selection of appropriate materials and the development of technical specifications for periodic inspections);
- Surveillance and testing to assess the ageing degradation of SSCs;
- Development of a preventive maintenance programme;
- Optimization of operating conditions;
- Management of repairs, replacement or refurbishing of SSCs;
- Management of non-physical ageing such as obsolescence.

Further details on ageing aspects in research reactors are set forth in Refs [51, 52]. The content of ageing management programmes, associated monitoring systems and methodologies for the assessment are available in Ref. [52].

For existing facilities, an assessment has to be made for as-is conditions, as a starting point for any ageing assessment. Such an assessment includes a review of documentation (drawings and inspection results) and site walkthroughs to determine deviations from the documentation and any in-service deterioration. The strength of materials can be tested on-site. Corrosion and other ageing degradation processes have to be considered.

Existing facilities have to be evaluated by order of priority, with the highest priority given to those areas identified as weak links by preliminary investigations and to areas that are most important to safety.

A special issue is connected to evaluating accumulated damage in research reactors that have already experienced significant external events of intensity close to the design basis values; the rate of penetration in the plastic domain has to be evaluated as part of the post-event inspection and assessment so that the residual capacity reserve can be calculated.

8. EVALUATION OF MARGINS FOR DESIGN EXTENSION CONDITIONS

In addition to the standard design and qualification carried out according to reference acceptance criteria, the design process (and the re-evaluation of the safety of existing nuclear installations) has to consider external event scenarios for design extension conditions, in order to account for:

— Residual risk from events with intensities beyond the design basis (i.e. the ‘tail’ of the probabilistic hazard distribution);
— Inaccurate or incomplete modelling of the hazard due to insufficient data availability or knowledge of the scenario development (e.g. undetected faults in the seismic hazard);
— Unforeseen scenarios not included in the design process (i.e. scenarios from unforeseen sources).

The objective of this supplemental assessment is to calculate the margin in the design available to accommodate design extension conditions, in order to prevent the onset of cliff-edge type behaviour induced by external events and to address the uncertainties described above. The target amount of margin is very much dependent upon the nature of the external event scenario and associated risk. For example, in case of seismic margin, the general practice in Member States sets that value between 1.5 and 2 times the design conditions.

To meet this objective, external event scenarios to be used for evaluations of the items important to design extension conditions have to be developed. Some scenarios may offer physical bounding criteria while in others (e.g. seismic) the associated risk to people and the environment is probably the only approach to bound the scenario intensity.

Special modelling capabilities for complex scenarios, extending the analysis to the effects on the site and on the surrounding region, may have to be applied.

Available simulation methods (PSA, SMA, etc.) have to be applied, with due consideration to development over time and the broad variety of potential consequences. Practical guidelines for simplified calculation of the margin are provided in Appendix II.

In the margin evaluation, not only safety related issues have to be addressed (e.g. site access and evacuation, contamination of personnel and public), but also social consequences (such as mass evacuation in cases of large releases from facilities).

9. SAFETY RE-EVALUATION OF EXISTING FACILITIES

9.1. EXISTING FACILITIES

For re-evaluation of an existing facility in relation to external events, the IAEA's Safety Reports Series No. 80, Safety Reassessment for Research Reactors in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant [16], provides guidance on establishing an approach to perform a safety reassessment of research reactor facilities. The publication covers all the steps in performing a safety reassessment, focusing primarily on operating research reactors. The publication includes a review of the design basis of a research reactor facility and assessment of the design extension conditions of the facility and its site, as well as a reassessment of emergency preparedness and response. The publication also provides guidance on the application of a graded approach and recommended processes for implementing the findings of the safety reassessment.

The basic approach set forth in this publication for the design of a new research reactor may also be applied for existing facilities. The following aspects need to be considered when re-evaluating existing facilities:

(a) The definition of the external event hazard: the external event return period has to be estimated.
(b) The definition of the external event categories: a reduced set of SSCs to be categorized may be identified, usually associated with safe shutdown heat removal and confinement path with redundancy. Some emergency systems needed to mitigate the effects of sequences initiated by internal accidents might not be categorized for external events.

(c) The definition of the performance goal: a higher performance goal is usually accepted when re-evaluating an existing facility compared with that for a new design. The factor is in the range of 2 to 10.
(d) The reference facility status: for re-evaluation, a limitation on the operation state is usually allowed (i.e. only normal operation, no consideration for outage, fuel loading).
(e) Material capacities: actual capacities for the materials, which include ageing effects, are allowed for re-evaluation, where the design usually refers to specified minimum or code values.

The re-evaluation of existing facilities may use more realistic and less conservative approaches (based on the best current engineering knowledge) than those used for the design of new installations. Realistic values for material strength and stiffness characteristics, damping coefficient, inelastic energy absorption factors and structural modelling can be used for the assessment.

The most relevant difference in a re-evaluation is the identification of the SSCs to be re-evaluated: in the case of existing facilities, the assessment is limited to a subset of safety and seismically classified items, which are required to perform safety function(s) during and after an external event, bringing the facility to a safe state.

In any case, the assessment has to be based upon as-is conditions, to be determined through material testing, expert walkthrough and configuration control.

For some external event scenarios (especially flooding, meteorological and human induced scenarios), expert walkthrough is a recommended assessment process in many countries. The walkthrough approach (based upon informed expert judgment), applied to an existing facility, as compared with an analytical assessment, is considered advantageous in terms of low resource consumption. Detailed guidance is provided in Ref. [16]. An 'easy fix' programme may also be implemented for upgrading already existing facilities, with the aim of minimizing investment costs while optimizing an increase in the safety margin.

9.2. ISSUES WITH COLLOCATED FACILITIES

Research reactors are often located in centres with many facilities. New facilities are added over time and therefore interaction of facilities in the near vicinity has to be considered. These considerations include:

- Shared resources such as electrical power and water;
- Common cause failures such as flood or fire;
- Emergency arrangements for individual facilities and for the whole site;
- Additional hazards due to the presence of other facilities in the vicinity (e.g. release of toxic material);
- Nuclear or radiological hazards that may arise from accidents in one facility that affect another facility;
- Additional exposure of the workers under normal operation due to other facilities located in the vicinity;
- Exposure of people and the environment under accident conditions, especially due to a common cause failure that may challenge all the facilities at the site.

Additional guidance on the effects of a new installation on or near an existing site is provided in SSG-35 [24]. Additional information is also available in Ref. [16] on safety reassessment of research reactors due to external events affecting the whole site.

10. DISPERSION OF RADIOACTIVE MATERIAL INTO THE ENVIRONMENT

The objective of a radiological hazard evaluation is twofold:

(i) To establish the final hazard categorization of facilities, and hence, of the performance and design classification of the SSCs, according to the radiological hazard posed to the environment, individuals and population in the event of an unmitigated accident;
(ii) To define the requirements for emergency procedures and preparedness.

In a research reactor, the radiological hazard emanates from the reactor core, spent fuel stored at the reactor site in spent fuel pools and the presence of other radioactive materials such as radioisotopes and radioactive waste.

When determining the source term, all these sources have to be addressed. Paragraph 4.1 of Ref. [53] provides the following information on grouping the radiological consequences into various categories:

"— On-site consequences inside the reactor building with doses to operating staff or personnel within the building;
— On-site consequences outside the reactor building from:
  • Direct radiation from the containment;
  • Gaseous or liquid release of radioactive material from the containment;
— Off-site consequences (to members of the public) from:
  • Direct radiation from the containment;
  • Gaseous or liquid release of radioactive material from the containment to the environment."

The source term can be evaluated with reference to all the accident initiating events postulated by the safety analysis of the reactor. A potential release of radioactive material from various sources can be evaluated on the basis of the percentage of fuel melting, as determined in the reactor safety analysis. Realistic analyses can consider all the uncertainties affecting the results and therefore avoid too much credit for assumptions regarding an unmelted fuel in case of an accident. In particular, the assumptions on fuel melting or loss of fuel barrier integrity can consider some important aspects of the accident scenario induced by the external events, such as the possibility of debris falling into the core, resulting in damage to fuel, changing core configuration (re-criticality) and preventing natural convection. Further, the uncovering of fuel due to loss of water from the core and/or spent fuel pool due to an external event has also to be addressed.

A release into the atmosphere, surface water bodies or groundwater can be evaluated on the basis of additional guidance provided in Ref. [53].

For evaluation of the releases, credit can be taken for following:

— Retention in the pool water, provided the presence of the water in the pool is guaranteed by a robust pool design;
— The filtering effect of the confinement, consistent with a hypothesis of damage to the confinement for external events which are expected to initiate the accident sequence; realistic assumptions can exclude limit values and excessive credit for the passive confinement features;
— The presence of additional safety features designed for accident mitigation, provided their design basis can guarantee their operability during and after the external events which are expected to initiate the accident sequence.

When determining the exposure pathways, special features of research reactors, such as beam tubes, experimental facilities penetrating the confinement and providing release routes following an accident due to an external event, also have to be considered.

A simulation of the propagation of radioactive material into the air may need some hypotheses on the topography of the site, air turbulence, air humidity and direction of dominant winds. However, the simulation can be carried out with simplified models, for bounding analysis, or with more refined models which provide a detailed representation of the three-dimensional problem.

From the concentration of radioisotopes that have been released, a preliminary evaluation of resulting doses to workers and the public can be developed and compared with the allowable values defined by national regulations. In a more refined approach, the concentration of the released radioisotopes can be combined with the population distribution (actual and predicted) to evaluate a new dose.

The simulation of the propagation of radioactive material through groundwater can be based on an analysis of groundwater flow: configuration of the flow, flow rate and periodicity. Particular care may be needed for research reactors close to aquifers that are used for drinking water.

In general, population density and other physical parameters (winds, topography) may have a strong influence on the final assessment of radiological doses to the population and therefore on the hazard categorization for the whole facility. While a facility in a sparsely populated area or at a site with a large exclusion area is less hazardous than a facility in a town, a high degree of engineering judgement has to be used to interpret the results of the radiological simulations in order to avoid unacceptable conclusions on the design of the facility. In particular, the uncertainties that affect other important contributors to the analysis have to be considered for a realistic, global categorization of the facility.

11. MONITORING

11.1. INSTRUMENTATION

The decision to install monitoring instrumentation, and to determine its safety class, is usually taken on the basis of the relevance of the external event hazard for the system design and on the basis of the instrumentation's significance for safety actions and/or the emergency procedures at the reactor. Seismic monitoring and automatic scram systems, when installed, are properly safety classified and adequate redundancy is provided according to their objectives.

In general, monitoring systems are installed at the site with the following objectives:

- To confirm a site hazard in relation to the scenarios that is relevant to reactor safety. In this instance, the purpose of monitoring is to detect site hazards and the data are analysed in the framework of periodic safety reviews of the facility.
- To enable the operator to take appropriate action in significant external events. When practicable and according to the characteristics of the event (e.g. development time, possibility of predicting), monitoring is designed, installed and operated to provide adequate warning signals for emergency operator actions for external events that are relatively slow to develop and to support operator actions after the event. In this case, guidelines for emergency operator action can also be developed.

Such systems include sensors at the site, in the structure and in some critical equipment.

The occurrence of external events significant for reactor safety has to be documented and reported. An extensive inspection after the occurrence of an external event either close to the design basis external event or significant for reactor safety has to be implemented in order to assess the behaviour and consequences on SSCs against their safety classification, accessibility and representativeness for all items of the external event category. Further guidance is provided in Refs [21, 54].

11.2. AUTOMATIC SCRAM

For research reactors, consideration is given to automatic actions to attain a safe state in case of an external event, when these actions are compatible with the speed of development of the external events. The facility needs to have protection capabilities in all operating modes and conditions. The systems in charge of this function are considered safety related and are consequently categorized for external events. In particular, operational limits and conditions of a seismic scram system, including surveillance tests and intervals, are based on the safety analysis for seismic events. Reference [21] provides information on automatic seismic trip systems for nuclear facilities.

11.3. POST-EVENT ACTIONS

After the development of an extreme external event and after the operator has taken immediate actions to place the reactor into a safe shut down state, a decision needs to be taken on restoring operation. Dedicated procedures are developed which set out the roles, responsibilities (which in some cases are subject to approval of the regulatory body) and a list of systems to be inspected prior to operation. Reference [21] provides information on a post-earthquake action programme for nuclear facilities.

12. EMERGENCY PROCEDURES

Research reactors may be collocated along with other nuclear/industrial/educational facilities. An external event is likely to affect all these facilities simultaneously to varying degrees. The effect of external event(s) has to include neighbouring facilities and their impact on the research reactor. An external event may be associated with other related or unrelated event(s) based on site characteristics and other influencing factors such as earthquake–tsunami, sandstorm–bad visibility, lightning–fire. An external event by itself may not pose a threat but may trigger another dangerous situation such as internal flooding, power failure or fire. Emergency preparedness has to take all these factors into account.

The ability to shut down, cool the fuel and confine radioactivity will be the key factor in determining the extent of the emergency following an external event.

The source term for the purpose of emergency preparedness may be derived based on guidelines provided in Section 11 and Ref. [53]. The class of emergency (e.g. alert, facility emergency, on-site or off-site emergency) will depend on the damage caused by the external event or its potential to cause damage. The emergency response can be decided based on the symptoms, observable parameters, status of reactor safety systems or conditions external to the reactor. Further guidance on dealing with a nuclear or radiological emergency at research reactors is available in Ref. [55]. Various operational intervention levels following an emergency are also described in this document.

Bearing in mind the hazard categorization described in Section 3, the following distinctions apply for emergency procedures:

— For HC-3 research reactors, their inherent safety prevents significant exposure of the public in the event of extensive postulated accidents. For reactors of this group, there is no need for off-site emergency procedures. However, local or on-site emergency procedures will be required to protect personnel in the facility in case of accidents.

- For HC-2 research reactors, fuel melting and any important release of radioactive material has to be proven to be non-credible for all accidents, including seismic and other external events, because of the inherent features of these reactors (e.g. assuring that sufficient water will always remain in the core for fuel cooling and that releases from the core are very small). Therefore, off-site emergency procedures are not normally required. If fuel melting or any significant release of radioactive material is considered possible, the feasibility of an emergency plan near the reactor needs to be demonstrated. On-site emergency procedures to protect personnel in the reactor and possibly in a limited zone around the reactor are required.
- For HC-1 research reactors, the potential for fuel damage and fission product release has to be analysed for all credible initiating events that could lead to an accident and the requirement for and extent of emergency procedures has to be established on a case by case basis. Such analysis has to include on-site experimental facilities and isotope production and handling facilities. The potential for release of radioactive material depends on the specific design features of the reactor, such as a shutdown heat removal system and containment. Generally, on-site and off-site emergency procedures are required [56].

13. MANAGEMENT SYSTEM FOR SITE EVALUATION AND DESIGN

The site evaluation and design organizations have to establish and implement a management system for ensuring that all safety requirements established for the site evaluation and design of the research reactor are considered and implemented in all phases of the site evaluation and design process [18, 19, 25].

The application of management system requirements may be graded so as to deploy appropriate resources. Grading of management system requirements may be applied to the products and activities of each process [57].

The work has to be structured and interpreted as a set of interacting processes; all individuals involved have to contribute to achieving safety and quality objectives.

A project work plan has to be prepared prior to, and as a basis for, the execution of a site evaluation and design project. The work plan has to convey the complete set of general requirements for the project, including applicable regulatory requirements. In addition to general requirements, the work plan has to delineate the following specific elements: personnel and their responsibilities; work breakdown and project tasks; schedule and milestones; and deliverables and reports.

The project plan has to be established and implemented under the facility management system to cover all activities for site evaluation and design.

The results of the evaluations of external event hazards need to be documented appropriately. The safety analysis report covers site related issues, including external event hazards. It is important that the external event hazards identified and evaluated in the site chapter of the safety analysis report are included in the design process and documented in the respective chapters of the safety analysis report.

Appendix I

EXPERIENCE FEEDBACK

This appendix provides a short description of recent external events which have challenged the safety of some facilities (research reactors and auxiliary installations) in recent years. They are grouped according to the type of challenge they posed.

These external events have been selected for their characteristics, which correspond with the safety approach proposed in the present publication. A short 'lessons learned' encapsulates each of the reported events. More comprehensive reviews may be found in Ref. [58].

I.1. UNCONTROLLED REACTION TRIGGERED BY EXTERNAL EVENTS

A survey performed by the Los Alamos National Laboratory in the United States of America [59] identified 38 accidents with uncontrolled power situations in US research reactors, with 1020 fission events. Twenty-two events resulted in injuries and/or fatalities to workers and the public. The survey did not consider malfunctions of either the cooling system or the confinement system.

Lesson learned: A safety analysis can identify whether external events can initiate an accident involving the control of a reaction, with potential releases of radioactive material into the environment; such an analysis has therefore to be conducted.

I.2. IMPLEMENTATION OF EMERGENCY MEASURES

A brushfire developed in 1999 around Hanford Laboratories in Washington State in the USA [60]. Strong winds caused the fire to spread quickly, making site evacuation difficult.

A forest fire developed around a nuclear facility at Cadarache, in France [58]. Firefighting aircraft were asked as a first priority to protect private properties threatened by the fire outside the facility fence over protecting the nuclear facilities. The nuclear facilities were affected by heavy smoke, with impairment of operator actions. The firefighting aircraft posed an additional hazard to the facility with low flights over the site in hazardous flight conditions.

Lessons learned:

— If external events affect nuclear and non-nuclear facilities located at the same site, priority has to be given to the implementation of emergency measures to protect the nuclear facilities from a challenge that may have radiological consequences.
— In areas sensitive to forest fires, special attention to the hazard posed by firefighting aircraft needs to be paid in assessing the combination of different external event hazards.

I.3. HUMAN ACTIVITIES IN THE VICINITY OF FACILITIES

At a site in France in the late 1980s, general maintenance activities (gardening) disturbed the ventilation system of an intermediate storage facility by clogging the system's air inlet.

Lesson learned: The analysis of human activities need not be limited to the industrial environment surrounding the site, but has also to consider all the regular activities conducted at the site or in the vicinity of the research reactor.

I.4. STRUCTURAL DAMAGE

An earthquake occurred on 27 May 2006 in Yogyakarta, central Java, Indonesia, with a magnitude of 5.7 and duration of about 57 seconds. At that time, the Kartini reactor was in shutdown state. The integrity of the reactor and its components was preserved, but four outer columns of the reactor building suffered spalling at the bottom and the wall cracked in several places.

Lessons learned:

— A programme to reinforce the structure of the Kartini reactor building had to be prepared and implemented.
— The reactor's emergency preparedness procedure needed to consider widespread aspects, including externally induced events.
— The actuation of the reactor's protection system, using input from a seismic signal, had to be considered.

I.5. FUKUSHIMA DAIICHI NUCLEAR POWER PLANT ACCIDENT

The Great East Japan Earthquake occurred on 11 March 2011. It was caused by a sudden release of energy at the interface where the Pacific tectonic plate forces its way under the North American tectonic plate. A section of the Earth's crust, estimated to be about 500 km in length and 200 km wide, was ruptured, causing a massive earthquake with a magnitude of 9.0 on the Richter scale and a tsunami which struck a wide area of coastal Japan, including the northeast coast, where several waves reached heights of more than ten metres. The earthquake and tsunami caused great loss of life and widespread devastation in Japan.

At the Fukushima Daiichi nuclear power plant, operated by the Tokyo Electric Power Company (TEPCO), the earthquake caused damage to the electric power supply lines to the site, and the tsunami caused substantial destruction of the operational and safety infrastructure on the site [61]. The combined effect led to the loss of off-site and on-site electrical power. This resulted in the loss of the cooling function at the three operating reactor units as well as at the spent fuel pools. The four other nuclear power plants along the coast were also affected to differing degrees by the earthquake and tsunami.

Despite the efforts of the operators at the Fukushima Daiichi nuclear power plant to maintain control, the reactor cores in Units 1, 2 and 3 overheated, the nuclear fuel melted and the three containment vessels were breached. Hydrogen was released from the reactor pressure vessels, leading to explosions inside the reactor buildings in Units 1, 3 and 4 that damaged structures and equipment and injured personnel. Radionuclides were released from the plant to the atmosphere and were deposited on land and on the ocean. There were also direct releases into the sea.

This resulted in an unprecedented major off-site emergency situation, with evacuation of people within a 20 km radius, restrictions on food and water consumption and rehabilitation issues. However, all operating reactor units at these plants were safely shut down.

Lessons learned: Several lessons have been learned from the event, many of which are also applicable to research reactors, that emphasize the need for:

— Periodic safety re-evaluation and implementation of resulting corrective actions; consideration of combined natural hazards and their site-wide effect; use of both national and international operating experience; strengthening of the implementation of the defence in depth concept, in particular, the independence of each level; operability of instrumentation and control systems that are necessary during design extension conditions; robust and reliable residual heat removal systems; and a reliable confinement function.

— An up to date accident management system; inclusion of postulated severe accident conditions in training, exercises and drills; effective regulatory oversight and a strong safety culture; and a systemic approach to safety that considers the interactions among human, organizational and technical factors.

Appendix II

EXAMPLE OF SEISMIC SAFETY EVALUATION USING A GRADED APPROACH (PERFORMANCE BASED APPROACH)

II.1. USE OF THE GRADED APPROACH FOR THE NEW DESIGN

The following steps are indicative of the actions to be pursued in implementing the proposed methodology:

(1) Selecting the facility hazard category (HC) and associated performance goal, expressed in probability of failure (PF_{-goal}), using Tables 1 and 3 (Section 2);
(2) Selecting the return period, T, corresponding to the target performance goal and design class (DC) using Table 4 (Section 2);
(3) Performing a seismic hazard analysis and defining the design basis earthquake (DBE);
(4) Conducting the seismic analysis, using selected DBE, to define seismic demand;
(5) Designing the structural elements, taking into account the seismic demand calculated in step 3, using acceptance criteria according to the selected design class.

If seismic capacity exceeds seismic demand (Capacity > Demand), the structure complies with the target PF_{-goal}.

II.2. SEISMIC SAFETY EVALUATION OF EXISTING DESIGN OF RESEARCH REACTORS

The following steps are indicative of the actions to be pursued in implementing the proposed methodology:

(1) Selecting the facility hazard category and associated performance goal, PF_{-goal}, using Tables 1 and 3 (Section 2);
(2) Selecting the return period, T, corresponding to the target performance goal, using Table 4 (Section 2);
(3) Performing a seismic hazard assessment and defining the review level earthquake (RLE), corresponding to the return period, T, and spectral shape of the ground response spectra (GRS) consistent with the site response [3];

(4) Selecting the structures, systems and components (SSCs) corresponding to the main success path 1 and to a secondary success path 2 [7];
(5) Conducting a seismic analysis, using selected RLE, to define seismic demand and using the seismic margin assessment (SMA) approach to calculate the seismic margin capacity for selected SSCs using damping values from Table 14 (Appendix III) and global inelastic energy absorption from Table 12 (Appendix III);
(6) Calculating the SMA capacity for each success path and facility seismic margin capacity and associated seismic damage state probability using local inelastic energy absorption from Table 13 (Appendix III) and seismic experience based methods from Table 15 (Appendix III);
(7) Calculating the seismic damage state probability;
(8) Comparing the seismic damage state probability with the performance goal and identifying safety improvements that could lead to a reduction of seismic damage state failure probability.

II.3. SELECTION OF THE RESEARCH REACTOR HAZARD CATEGORY AND ASSOCIATED PERFORMANCE GOAL

Hazard categorization is provided in Table 1 based on the power rating of a research reactor and radioactive inventory in the facility. Hazard categories range from HC-1 (high) to HC-4 (low). The target performance goal or corresponding probability of failure is a graded function of the research reactor hazard category, as shown in Table 3.

II.4. SELECTION OF THE RETURN PERIOD, T, FOR THE REVIEW LEVEL EARTHQUAKE

The performance goal corresponding to the selected hazard category can be achieved by selecting a combination of seismic hazard severity corresponding to return period, T, and design class from Table 4. Design classes are defined in Section 2.10. To define the seismic hazard severity function of the return period the seismic hazard curves based on a probabilistic seismic hazard assessment should be available.

II.5. EVALUATION OF SEISMIC HAZARD CURVES

If the selected return period, T, is 10 000 years, a seismic hazard assessment should be conducted following the guidelines in SSG-9 [3]. If the selected return period, T, is 1000 years, the following simplified relations can be used to determine the mean seismic hazard curve:

$$H(a) = K_i a^{-K_h} \tag{5}$$

$$K_i = \frac{a_T^{K_h}}{T} \tag{6}$$

$$a_{T=1000} = \sqrt[K_h]{T \times K_i} \tag{7}$$

where

K_i = scale factor;

$K_h = \dfrac{1}{Log(A_R)}$;

T = return period;

and A_R = slope parameter (determined by the experts in geology and seismotectonics).

For crustal earthquakes, A_R values typically range between 2.5 and 3.5 except in stable continental regions, or if the site is close (less than 25 km) to capable faults. Subduction or intraplate seismic sources also require specific assessment. The mean hazard curve is used to define the peak ground acceleration corresponding to the RLE.

II.6. SELECTION OF THE SUCCESS PATH SSCs

The success path should be that path for which it is judged easiest to demonstrate the seismic margin and one that the research reactor operators would employ to bring the research reactor to a safe state after a large earthquake and maintain it based upon procedures and training. The primary success path should be a logical success path consistent with the research reactor's operational procedures. The alternate path (if available) should involve operational SSCs different from those in the preferred path. SSCs belonging to the success path(s) define the scope for conducting a seismic margin capacity evaluation defined by a high confidence low probability of failure (HCLPF).

II.7. CONDUCT OF THE SEISMIC ANALYSES

A realistic seismic response analysis is conducted using the RLE. The result of the analysis is seismic demand for all SSCs belonging to the success path(s) to facilitate:

(a) Screening of structures and equipment;
(b) Evaluation of seismic HCLPF capacities of screened-in SSCs.

II.8. EVALUATION OF SSC CAPACITY

Seismic margin capacity is defined by the HCLPF. The methodology for evaluating the seismic margin capacity is provided in NS-G-2.13 [7] and NS-G-3.1 [28]. A preliminary HCLPF can be generic based on seismic experience. A set of inclusion/exclusion rules called ‘caveats’ need to be checked for using generic seismic capacity based seismic experience. Seismic walkthroughs for observing caveats especially related to seismic interactions/anchorage are mandatory. The use of seismic experience is documented in references provided by Table 15. HCLPF calculations are conducted for low capacity SSCs (weak links) using design scaling or full re-analysis. HCLPF is obtained using the following equation:

$$HCLPF = PGA_{RLE} \times \frac{C - NSD}{SD} \times F_{\mu} \tag{8}$$

where

PGA_{RLE} is the peak ground acceleration corresponding to the RLE;
C represents the elastic seismic capacity;
NSD is the non-seismic demand concurrent with seismic loading;
SD represents seismic demand;
and F_{μ} is the inelastic energy absorption factor.

Additional references to detailed information for conducting HCLPF calculations are provided in Table 15. Assuming that two success paths, SP1 and SP2, are selected, the seismic margin capacity (HCLPF) for SP1 and SP2 and for the damage state DS (defined by the failure of both SP1 and SP2) are calculated using the following equations:

$$\begin{aligned} HCLPF_{SP1} &= \text{Min.}\ (HCLPF_A, HCLPF_B, HCLPF_C) \\ HCLPF_{SP2} &= \text{Min.}\ (HCLPF_D, HCLPF_E, HCLPF_F) \\ HCLPF_{DS} &= \text{Max.}\ (HCLPF_{SP1}, HCLPF_{SP2}) \end{aligned} \quad (9)$$

The above equations show that success path seismic capacity ($HCLPF_{SP}$) is calculated as the minimum capacity of the SSCs that build up the success path. The research reactor damage state capacity ($HCLPF_{DS}$) is calculated as the maximum capacity between the independent success paths. The damage state capacity $HCLPF_{DS}$ means the seismic capacity of both success paths SP1 and SP2, as shown in Eq. (9).

II.9. CALCULATION OF SEISMIC FAILURE PROBABILITY

Variability estimation

For structures and major passive mechanical components mounted on the ground or low elevations within structures, the typical range for β is 0.3 to 0.5 and for active components mounted at high elevations in structures the typical range for β is 0.4 to 0.5 [62]. Note that overestimating β is unconservative because it increases the median capacity A_m. Limited calculations or published data for similar components can be used to estimate β if the ‘as-built’ and ‘as-operated’ conditions comply with the caveats listed in the references from Table 15. Composite variability parameter βc is selected in a range of 0.35 to 0.45 based on controlling SSC–HCLPF capacity (item belonging to success path with the lowest value for HCLPF).

Suggested values are provided in Table 10.

TABLE 10. RECOMMENDED β VALUES

SSC type	β_c
Structures and major passive mechanical components mounted at low elevation with structures	0.35
Active components mounted at high elevations in structures	0.45–0.50
Other SSCs	0.40

Estimation of the damage state failure probability P_F

Median capacity A_m is obtained from the HCLPF of each SSC and composed variability β selected from Table 10 based on SSC type location. The SSC seismic failure probability P_F can be calculated using the following equation:

$$P_F = \int_0^{\infty} H(a)\left(\frac{dP_{F/a}}{da}\right)da \tag{10}$$

Equation (10) can be solved with high accuracy using a numerical integration algorithm (see Figs 2, 3). First, the fragility curve is generated using A_m and the β parameters and then each fragility is integrated according to Eq. (10) with selected hazard curves. The probability of failure (*PF*) is calculated based on a close form solution:

$$P_F = K_I A_m^{-K_h} e^{0.5(Kh\beta_c)^2} \tag{11}$$

where

A_m represents the median seismic capacity;
and P_F represents the probability of failure; it should be less than the performance goal corresponding to the facility hazard category.

Assuming the SSCs A, B and C correspond to the primary success path, SP1 (A, B, C), and D, E and F correspond to the secondary success path, SP2 (D, E, F), the facility damage state (DS) will occur when both SP1 and SP2 fail.

For each success path and for the damage state the failure probability is calculated as follows:

$$P_{F(SP1)} = P_{F(A)} + (1 - P_{F(A)})[P_{F(B)} + (1 - P_{F(B)})P_{F(C)}] \tag{12}$$

$$P_{F(SP2)} = P_{F(D)} + (1 - \mathrm{P}_{\mathrm{F(D)}})[P_{F(E)} + (1 - P_{F(E)})P_{F(F)}] \tag{13}$$

$DS(SP1, SP2) = P(SP1 \cap SP2)$ if *SP1* and *SP2* are independent (14)

$DS(SP1,SP2) = P(SP1|SP2) = \dfrac{P(SP1 \cap SP2)}{P(SP2)}$ if SP1 is conditioned by SP2 (15)

II.10. ILLUSTRATIVE EXAMPLE

This example shows the steps to be performed for a seismic safety evaluation to check compliance with the seismic performance goal, PF_{-goal}, associated with the facility (research reactor) hazard category described in Tables 1 and 3 (Section 2).

Step 1:

It is assumed that the research reactor based on Table 1 is HC-3 (low hazard facility). From Table 3 the performance goal (target seismic failure probability) is taken as $PF_{-goal} = 5 \times 10^{-4}$. Using Table 4 for HC-3, the RLE having a return period T = 1000y and design class DC-2, DC-3 was selected.

Step 2:

According to the national seismic hazard map the site is located in a region with a peak ground acceleration (PGA) of a_{475} = 0.1g, corresponding to a return period of 475 years. Based on seismotectonic settings, experts in geology, seismotectonics and seismology recommend using the slope parameter A_R = 3. In many cases, this value produces conservative results. The PGA for a return period of 1000 years can be calculated based on the PGA defined for T = 475y in the national seismic code, using Eqs (5) to (7):

$$K_h = \frac{1}{\mathrm{Log}(A_R)} = \frac{1}{\mathrm{Log}(3)} = 2.096 \tag{16}$$

$$K_i = \frac{a_T^{K_h}}{T} = \frac{0.1^{2.096}}{1000} = 1.688 \times 10^{-5} \tag{17}$$

$$a_{1000} = \sqrt[2.096]{1000 \times 1.688 \times 10^{-5}} = 0.143\mathrm{g} \tag{18}$$

Finally, the mean hazard curve is defined by the following equation:

$$H(a) = 1.688 \times 10^{-5} \times a^{-2.096} \tag{19}$$

The hazard curve defined above is presented in Fig. 2 (the curve for A_R = 3). A value of 3 for A_R could be appropriate in many regions where seismicity is dominated by crustal earthquakes (except stable continental regions and those where seismicity is dominated by intra-plates or in close proximity to active/capable seismic faults).

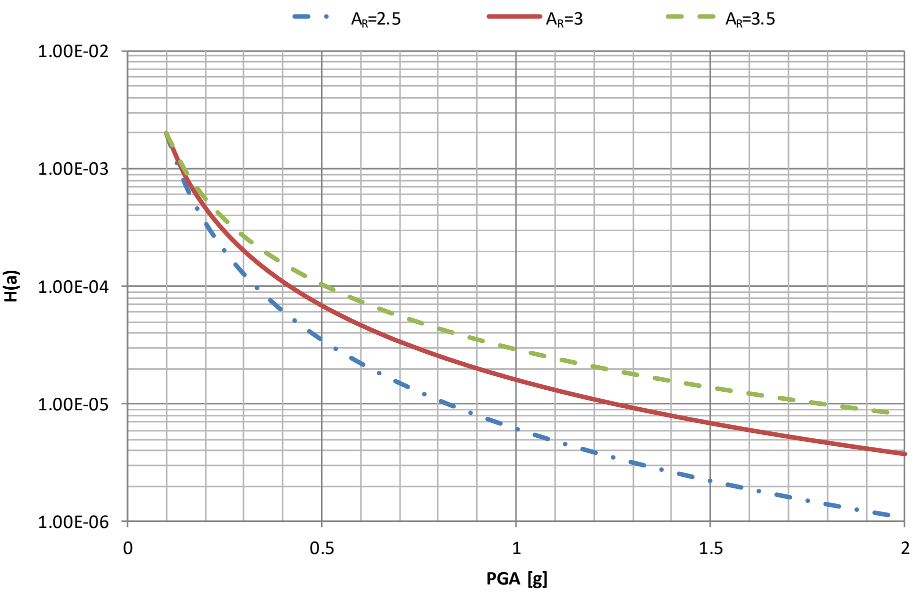

FIG. 2. Example of hazard curves for different slope parameters; A_R — slope parameter of the hazard curves, H(a) — seismic hazard, PGA — peak ground acceleration.

Step 3:

Selection of SSCs for: (a) the main success path SP1 and (b) secondary success path SP2. The methodology outlined in NS-G-2.13 [7] and its references should be followed to identify all SSCs associated with the main safety functions and their dependencies. In this example, for simplicity's sake, we assume the SSCs A, B and C correspond to the primary success path SP1 (A, B, C) and D, E and F correspond to the secondary success path SP2 (D, E, F). The facility damage state (DS) occurs when both SP1 and SP2 fail.

Step 4:

Calculation of the seismic margin capacity for: (a) each SSC, (b) each success path and (c) the damage state. The seismic margin capacity is defined by HCLPF and can be obtained using the following equation:

$$HCLPF = PGA_{RLE} \times \frac{C - NSD}{SD} \times F_{\mu} \tag{20}$$

Methodologies for calculating HCLPF are indicated in references provided in Table 15 and in NS-G-2.13 [7]. Table 13 provides values for the local inelastic energy absorption factor F_{μ}.

Using data from Table 11, the seismic margin capacities (HCLPF) for SP1, SP2 and DS are calculated, using Eq. (9):

$HCLPF_{SP1} = 0.1$
$HCLPF_{SP2} = 0.143$
$HCLPF_{DS} = Max.\ (0.1, 0.143) = 0.143g$

Step 5:

Calculation of the seismic failure probability for (a) each SSC, (b) each success path and (c) the damage state.

For each SSC, seismic fragility parameters are calculated using variability parameters from Table 10 and equations (10) and (11). Table 11 presents the total probability of failure (PF) calculated, using the above equations for the given SSC items A, B, C, D, E and F. The variability values are then estimated based on Table 10.

TABLE 11. EXAMPLE OF CALCULATION OF P_F FOR INDIVIDUAL COMPONENTS

Item	DBE	$\frac{\text{HCLPF}}{\text{DBE}}$	HCLPF [g]	HCLPF_{SP}	β	A_m [g]	P_F
A	0.1	1	0.100	$\text{HCLPF}_{\text{SP1}}$ = 0.1g	0.35	2.25×10^{-1}	5.02×10^{-4}
B	0.1	1.15	0.115		0.45	3.27×10^{-1}	2.75×10^{-4}
C	0.1	1.15	0.115		0.4	2.91×10^{-1}	3.19×10^{-4}
D	0.143	1.75	0.250	$\text{HCLPF}_{\text{SP2}}$ = 0.143g	0.45	7.11×10^{-1}	5.39×10^{-5}
E	0.143	1	0.143		0.45	4.06×10^{-1}	1.74×10^{-4}
F	0.143	1.67	0.239		0.45	6.78×10^{-1}	5.94×10^{-5}

Estimation of the overall probability of failure using the simplified hybrid method

Assuming that two success paths, SP1 and SP2, have been evaluated using a conservative deterministic failure margin, the damage state (DS) will occur when SP1 and SP2 both fail. Using calculated values for P_F from Table 11, the seismic probability of failure for SP1, SP2 and DS as well as the associated seismic margin capacities are calculated below:

$$P_{F(SP1)} = P_{F(A)} + (1 - P_{F(A)})[P_{F(B)} + (1 - P_{F(B)})P_{F(C)}] \tag{21}$$
$$P_{F(SP1)} = 1.10 \times 10^{-3}$$

$$P_{F(SP2)} = P_{F(D)} + (1 - P_{F(D)})[P_{F(E)} + (1 - P_{F(E)})P_{F(F)}] \tag{22}$$
$$P_{F(SP2)} = 2.87 \times 10^{-4}$$

$$P_{F(DS)} = Min.\ (1.10 \times 10^{-3}, 2.87 \times 10^{-4}) = 2.87 \times 10^{-4} < \mathrm{PF}_{\text{-goal}} = 5 \times 10^{-4} \tag{23}$$

In conclusion, this example illustrates a simplified method to estimate seismic failure probability for an HC-3 facility (research reactor) with the $\mathrm{PF}_{\text{-goal}} = 5 \times 10^{-4}$, and SSC capacities for SP1 and SP2, as shown in Table 11; the seismic damage state probability is $2.87 \times 10^{-4} < \mathrm{PF}_{\text{-goal}}$.

Alternatively, the seismic probability of failure can be calculated using Eqs (10) or (11), the function of the hazard curve and plant state fragility defined by the $\mathrm{HCLPF}_{\mathrm{DS}}$ and the associated variability β. For the damage/plant state fragility, a value of $\beta = 0.3$ is suggested.

For $\beta = 0.3$ and $\mathrm{HCLPF}_{\mathrm{DS}} = 0.143$ a median capacity A_m can be calculated:

$$A_m = HCLPF \times \exp(2.32\,\beta) = 0.287\mathrm{g} \tag{24}$$

Using Eq. (11) for hazard parameters $K_i = 1.688 \times 10^{-5}$, $K_h = 2.09$ (Step 2) and fragility parameters $A_m = 0.287$g and $\beta = 0.3$, the $PF(_{DS})$ obtained using Eq. (11) is 2.86×10^{-4}, which is close to the values obtained using Eq. (23).

Fragility curves corresponding to the parameters in Table 11 are presented in Fig. 3.

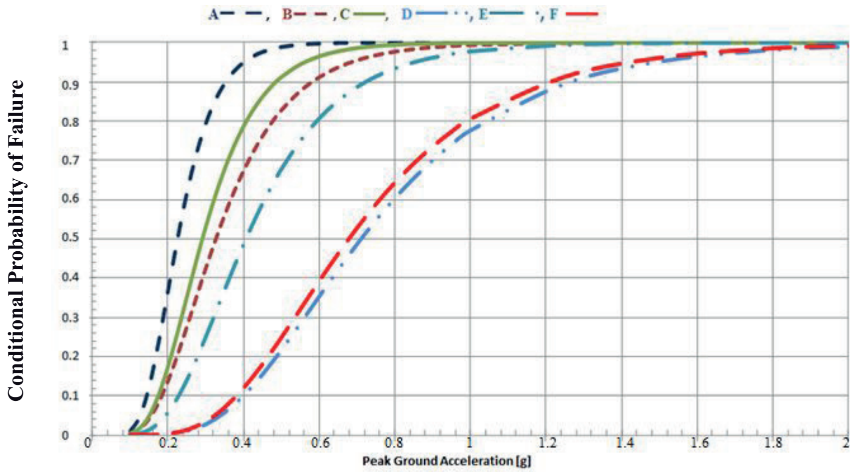

FIG. 3. Example of fragility curves generated based on A_m *and* β *from Table 11.*

Appendix III

TYPICAL VALUES FOR CRITICAL PARAMETERS AND REFERENCE METHODS FOR QUALIFICATION OF SSCs

Table 12 provides the typical values for the global inelastic energy absorption factor for design or re-evaluation of civil structural systems for different design classes. Table 13 provides the typical values for the local inelastic energy absorption factor for structural elements and equipment. These values could be used for seismic design or re-evaluation of structural systems. Table 14 provides typical damping values for seismic design or re-evaluation of SSCs in conjunction with stress levels. Table 15 provides references to some experience based methods that could be used for seismic qualification of SSCs. These methods have been developed and used in the past in different countries.

TABLE 12. VALUES FOR THE GLOBAL INELASTIC ENERGY ABSORPTION FACTOR $K_{D,G}$

Structural system	Global inelastic energy absorption factor $k_{D,G}$ (design/re-evaluation)		
	DC-1	DC-2	DC-3
Space frame			
Moment connection	1.0 / 1.50	2.00 / 3.00	4.00
Braced connection	1.0 / 1.25	1.50 / 2.00	2.00
Redundant dual connection	1.0 / 1.50	2.00 / 3.00	4.00
Shear connection	1.0 / 1.15	1.25 / 1.50	1.50
Planner frame			
Moment connection	1.0 / 1.25	1.50 / 2.00	2.00
Braced connection	1.0 / 1.15	1.25 / 1.50	1.50
Redundant dual connection	1.0 / 1.35	1.50 / 2.00	2.00
Shear connection	1.0 / 1.10	1.15 / 1.25	1.25
Load bearing reinforced concrete shear wall	1.0 / 1.25	1.50 / 2.00	2.00
Non–load bearing reinforced concrete shear wall	1.0 / 1.15	1.25 / 1.50	1.50
Load bearing reinforced masonry wall	1.0 / 1.15	1.25 / 1.50	1.50
Non–load bearing reinforced masonry wall	1.0 / 1.15	1.25 / 1.50	1.50
Non-reinforced masonry wall	1.0 / 1.00	1.00 / 1.00	1.00

Notes: Higher global inelastic energy absorption factors can be used only when properly justified.
Higher values in denominators can be used for re-evaluation purposes.
Values given in this table are conservative as they have to respect a large variety of national and international design/re-evaluation methods as well as a large variety of structural systems with a non-uniform quality of performance in different countries.

TABLE 13. VALUES FOR THE LOCAL INELASTIC ENERGY ABSORPTION FACTOR $k_{D,l}$

Structural element, equipment (failure mode)	Local inelastic energy absorption factor $k_{D,l}$ (design/re-evaluation)
Concrete	
Columns where flexure dominates	1.00 / 1.25
Columns where axial compression or shear dominates	1.00 / 1.00
Beams where flexure dominates	1.00 / 1.75
Beams where shear or tension dominates	1.00 / 1.25
Connections	1.00 / 1.00
Connections (ductile design)	1.00 / 1.25
Steel	
Columns where flexure dominates	1.00 / 1.50
Columns where axial compression or shear dominates	1.00 / 1.00
Beams where flexure dominates	1.00 / 2.00
Beams where shear or tension dominates	1.00 / 1.25
Connections	1.00 / 1.15
Connections (ductile design)	1.00 / 1.25
Concrete reinforced masonry walls	
In-plane bending	1.00 /1.75
In-plane shear	1.00 /1.50
Out-of-plane bending	1.00 /1.75
Out-of-plane shear	1.00 /1.00
Non-reinforced masonry (all)	1.00 /1.00
Equipment components and pipes	
Equipment components which have to remain functional	1.00 / 1.00
Equipment components where a brittle failure mode dominates (e.g. loss of stability)	1.00 / 1.00
Properly anchored passive equipment components with welded connections	1.00 /1.50
Welded pipelines (basic material and welds)	
Welded equipment nozzles	1.00 /1.50
Threaded pipe connections	1.00 / 1.25
Equipment components made by cast iron	1.00 / 1.00
Flanged pipe connections and flange equipment nozzles	1.00 / 1.00
Equipment supports and their anchorage (brittle failure mode)	1.00 / 1.00
Equipment supports and their anchorage (ductile failure mode)	1.00 / 1.00
	1.00 / 1.50

Notes: Higher local inelastic energy absorption factors can be used only when properly justified.
Higher values in denominators can be used for re-evaluation purposes.

TABLE 14. TYPICAL DAMPING VALUES FOR SEISMIC ANALYSES

SSC	Acceptable damping values[a]		
	Re-evaluation	Design	
		Stress level 1	Stress level 2
Structures			
Reinforced concrete structures	7%	4%	7%
Welded steel structures	5%	2%	5%
Bolted steel structures	7%	4%	7%
Non-reinforced masonry walls	5%	3%	5%
Soil–structure interaction[b]			
Horizontal and rocking modes	15%	15%	15%
Vertical modes	20%	20%	20%
Equipment			
Bolted supporting structures	7%	4%	7%
Welded supporting structures	5%	2%	5%
Pipes (all parameters and all diameters)	5%	2%	5%
Anchored mechanical components	5%	3%	5%
Electrical and I&C cabinets and panels	5%	3%	5%
Cable supporting structures[c]	5 – 10 – 15%	2 – 6 – 10%	5 – 10 – 15%
Tanks			
● impulsive mode	5%	3%	5%
● convective mode	0.5%	0.5%	0.5%

Notes: [a] Stress levels 1 and 2 mean about 50% and 100% of the bearing capacity, respectively.

[b] These values are typical and may be used for soils with a shear wave velocity of less than 1000 m/s. References [30, 31] are recommended to consider the soil–structure effects in a more exact manner.

[c] Use these three values for structures loaded by cables up to 10, 50 and 100% of their nominal capacity, respectively.

TABLE 15. SOME SEISMIC EXPERIENCE BASED METHODS

Method	Items to be verified	Reference (public domain)
DOE procedure	Selected mechanical and electrical equipment classes	[46]
	Cable supporting structures	
	Anchorage of equipment	
	Non-bearing brick walls	
	Architectural details	
	Seismic interactions	
GIP procedure	Selected mechanical and electrical equipment classes	[44]
	Cable supporting structures	[45]
	Anchorage of equipment	
	Seismic interactions	
LLNL procedure	Selected mechanical and electrical equipment classes	[33]
	Anchorage of equipment	[43]
	Seismic interactions	
Stevenson procedure	Pipelines (limited scope)	[49]
Antaki procedure	Pipelines (limited scope)	[50]

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Annex

EXAMPLE OF SITING STUDIES — OPAL REACTOR AT LUCAS HEIGHTS, AUSTRALIA

A–1. INTRODUCTION

This annex contains a summary of the siting studies performed for the Australian Nuclear Science and Technology Organisation (ANSTO) Open Pool Australian Lightwater (OPAL) research reactor at the Lucas Heights Science and Technology Centre (LHSTC), Australia. The actual results of the site studies are contained in the OPAL Safety Analysis Report (SAR), Chapter 3: Site Characteristics, which is also available on the Australian Radiation Protection and Nuclear Safety Agency (ARPANSA, the Australian nuclear regulator) website[1].

A–2. BACKGROUND

Many of the siting studies reported in the OPAL SAR were not specific to the OPAL reactor itself but were generic studies applicable to either the older High Flux Australian Reactor (HIFAR) or the LHSTC as a whole. In addition, since the site was originally established in the 1950s, many of the current siting studies tend to be evolutionary developments of previous studies, generally updated on the basis of new data (such as population data from Australian five year censuses).

A–3. LICENSING AND REGULATORY REQUIREMENTS

As is common in most Member States, the nuclear licensing process for the OPAL reactor consisted of three phases: site licence, construction licence and operating licence. With the OPAL reactor now operating, the site and construction licences are no longer applicable; ARPANSA recently reissued the operating licence (as part of an overall process to ensure consistency across all licences, not in relation to any specific OPAL-related issue). Each application to ARPANSA for a licence contained information about site characteristics, either in the form of a separate document in the case of an application for a site licence or as a chapter in the preliminary SAR (PSAR) and the SAR in the case of applications

[1] http://www.arpansa.gov.au/Regulation/opal/op_applic.cfm

for the construction and operating licences, respectively. The content and format of the PSAR and the subsequent SAR are both in accordance with the guidelines in IAEA Safety Standards Series No. SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report.

Prior to submitting an application for the site licence, ANSTO was required to prepare and submit an environmental impact statement (EIS) to the Department of Environment and Heritage (now the Department of Sustainability, Environment, Water, Population and Communities) that also contained information about the site characteristics. This EIS was prepared by specialist consultants in conjunction with, and under contract to, ANSTO who considered the environmental consequences that might arise from both normal operation and accident conditions. The environmental approval process is separate and additional to the nuclear regulatory approval process and is applicable to many major developments in Australia. The environmental approval contained 29 conditions on the construction and operation of OPAL, specifically, and on ANSTO as a whole. As well as the expected conditions, such as the need for environmental management, traffic management and storm water control plans for the construction phase, some of the conditions specific to OPAL included the following (in summarized form):

— Radioactive gaseous emissions discharged via stacks from buildings associated with radiopharmaceutical production must not increase above existing levels regardless of any future production increases.
— As part of the groundwater monitoring programme, ANSTO must establish bores at appropriate locations in the LHSTC and the buffer zone.
— ANSTO must establish a radiological site characterization or 'footprint' for the reactor site and an LHSTC/buffer zone, in general, to provide a fundamental basis for ongoing radiological monitoring programmes and the detection of radiological trends over time.
— The PSAR, to be prepared at the detailed design stage, must be subject to independent peer review.
— The assumptions used in deriving the reference accident effectively constitute design parameters for the proposed reactor and must be incorporated into the final design. In the event of changes that might render the reference accident invalid, agreement to any major design changes must be sought.

Subsequently, the site licence incorporated a number of these conditions as well as additional conditions imposed by ARPANSA specific to OPAL.

A–4. SITE CHARACTERISTICS

As identified above, each application to the Australian nuclear regulator for a licence contained information about the site characteristics. The topics covered are generally consistent across all three applications, although they became more detailed in some areas (such as seismology) as the submissions progressed. These topics are:

(1) Geography;
(2) Demography; includes current and projected population both on-site and in surrounding areas;
(3) Meteorology; includes atmospheric dispersion for input into the off-site dose analysis as well as brushfire weather in the case of OPAL;
(4) Surface hydrology; includes dams and near surface bores;
(5) Geology, soils and groundwater hydrology;
(6) Seismology;
(7) Site services;
(8) Nearby facilities;
(9) Transportation routes;
(10) Baseline environmental radioactivity.

These topics are discussed in more detail in the following subsections.

A–4.1. Geography

This topic covered the details of the LHSTC and the position of OPAL within the centre. Information included not only the expected geographical information regarding the LHSTC (as shown in Fig. A–1) but also information relating to land zoning in the vicinity of the LHSTC and actual land usage, including agricultural uses, up to 50 km away. Sources for this information included the environmental plans, development approvals and infrastructure services developed by the local Sutherland Shire Council and the New South Wales (NSW) Department of Urban Affairs and Planning, agricultural data from the NSW Department of Agriculture and other local councils and research performed by ANSTO's own Institute for Environmental Research.

A–4.2. Demography

This topic covered the demography both within the LHSTC and in the surrounding areas. The population within the LHSTC was determined by ANSTO's own records. Projections have been derived from anticipated personnel

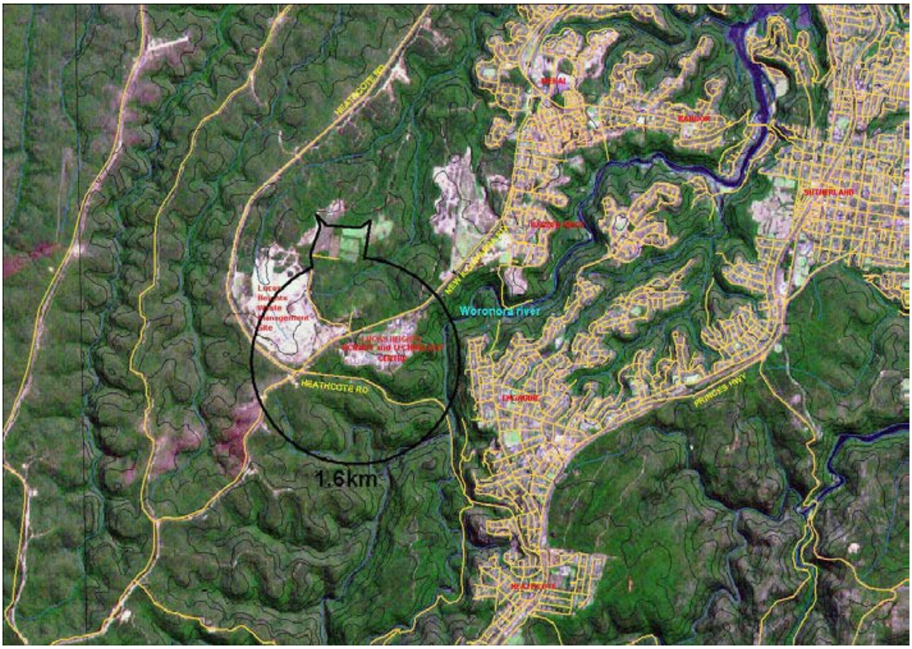

FIG. A–1. Map of the Lucas Heights area (superimposed on an aerial photo of the area) (courtesy of ANSTO).

changes arising from ANSTO's business plan and other activities that may be introduced on-site. However, since these projections are heavily dependent upon multiple external factors that may or may not come to fruition (e.g. due to funding, educational policy, commercial viability issues), a nominal LHSTC population of 1500–1600 has been assumed. There is also a 1.6 km buffer zone surrounding the LHSTC that is under the control of ANSTO and within which no permanent residences are permitted.

The principal source of information for the population in the surrounding areas was the Australian Bureau of Statistics Census data, initially from the 1996 census but subsequently updated using data from the 2001 census and the 2006 census, as appropriate. Data from the 2011 census is still being analysed and will be incorporated into the SAR at a future date. Population projections were sourced from the NSW Department of Urban Affairs and Planning and extend to 2036. Population data are documented in terms of 16 equal sectors (22.5° each) and 6 radial zones (1.6–3.2 km, 3.2–4.8 km, 4.8–10.0 km, 10.0–15.0 km, 15.0–20.0 km and 20.0–25.0 km) centred on the LHSTC.

A–4.3. Meteorology

The principal source of information regarding meteorology is ANSTO's own meteorology laboratory and weather station, located at the LHSTC, in conjunction with information from the Australian Bureau of Meteorology (BoM) to which the LHSTC meteorology laboratory and weather station also provides input. Parameters measured are:

— Temperatures at 2 m, 10 m and 49 m above ground level;
— Relative humidity;
— Wind speed and direction at 10 m and 49 m above ground level;
— Atmospheric pressure;
— Rainfall.

Since 1991, data from the LHSTC meteorology laboratory and weather station have been recorded digitally every 15 minutes, but site specific data have been collected since 1958 through the BoM and are publicly available on its web site (http://www.bom.gov.au/climate/averages/tables/cw_066078.shtml). Weather records for Sydney itself go back to the arrival of the First Fleet in Sydney Harbour in 1788 and formal weather data from the first permanent weather station established in Sydney in 1858 are available on the BoM web site.

The two main uses for the meteorological information are:

(1) To model atmospheric transport and dispersion of airborne releases of radioactive materials during normal operation and following accidents;
(2) To determine the design basis for structures, systems and components (SSCs) in relation to external and environmental loads, particularly the wind loads on the reactor building.

Much of the atmospheric modelling was done as part of the safety reassessment of HIFAR in the 1970s and 1980s. Furthermore, under highly stable atmospheric conditions, studies were performed to verify that wind flow patterns in the Woronora Valley adjacent to the LHSTC are decoupled from the flow on the plateau on which LHSTC is located. Atmospheric tracers were subsequently used to confirm this observation.

To determine the design basis for SSCs, the Australian Standard, AS-1170.2-1989, SAA Loading Code, Part 2: Wind Loads, was applied. This meant that the design basis wind speeds were calculated assuming a 5% probability of the wind speed being exceeded in a 2000 year period. This was supplemented by the HIFAR probabilistic safety analysis (PSA), which analysed the wind hazard, including tornadoes, for the LHSTC site. The analysis for the wind hazard was based on the assumption that the wind speed data fit a type I extreme value distribution (Gumbel distribution) as detailed in ANSI Standard A58.11982 and used Sydney Kingsford Smith Airport wind data that date back to 1936. A simplified approach was used to estimate the hazards associated with tornadoes with the frequency of tornado occurrences at the Lucas Heights site. This estimate compared the tornado occurrence data for the Sydney area with the tornado strike frequency on HIFAR based on the point target strike model. The tornado hazard assessment method (damage area per path length or DAPPLE) was then used to relate the tornado strike frequency to the probability of wind speed by accounting for the gradation of damage along the length and width of the tornado path in terms of mean path length. The design basis of the building and structures included not only the pressure effects associated with the wind but also the gusting effects and the pressure drop effects in the event of tornado type winds. However, the effect of the impact of missiles on the building was not considered as it is bounded by the aircraft impact.

In relation to other environmental parameters, statistical data relating to rainfall and evaporation for the years 1981 to 2002 were presented in chapter 3 of the SAR. These show that the annual rainfall across this period varied between 576 and 1482 mm, with the total annual evaporation varying between 1087 and 1462 mm for the same period. Statistical data relating to ambient wet and dry bulb temperatures for the years 1991 to 2004 were also presented in chapter 3 of the SAR, with dry bulb temperatures varying between 0.6 and 43.2°C and wet bulb temperatures varying between 0.1 and 25.2°C. The design conditions are well within the bounds of normal civil design within Australia and consistent with the Building Code of Australia (BCA). As such, they did not affect the safety design bases of the facility, although the temperatures did influence the operational design bases, particularly with respect to the secondary cooling system capacity (26.4 MW at 27ºC wet bulb temperature).

A–4.4. Surface hydrology

Due to the topography of the LHSTC and the surrounding area (Fig. A–2), flooding of the OPAL reactor site due to external areas is not a credible threat even in the event of the highest precipitation recorded. The research reactor facility is located on the top of a sandstone ridge with a higher elevation than the surrounding area and drainage in all directions. Local creek and river systems have not come anywhere near the site even under the highest historically recorded flooding.

The general topographic environment is such that no part of the LHSTC is far from a natural drainage channel in the adjacent terrain. There are no known private dams or bores that could be fed by runoff from the area surrounding the LHSTC, as determined by a review of data provided by the Department of Land and Water Conservation. As such, the dams and groundwater cannot form a possible dispersion pathway for releases from OPAL.

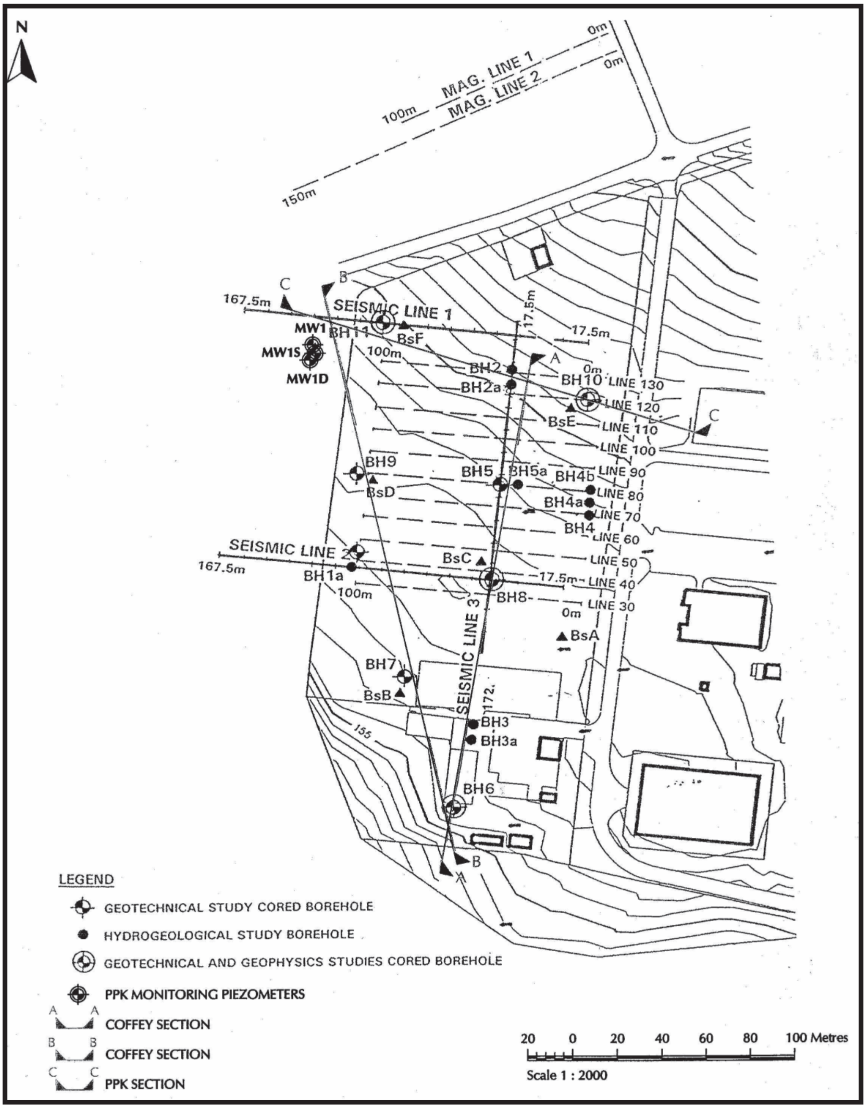

FIG. A–2. Borehole locations (courtesy of ANSTO).

A–4.5. Geology, soils and groundwater hydrology

The regional and local geology of the OPAL reactor site was studied in detail in geotechnical and geophysical investigations undertaken by specialist external consultants as was the information summarized originally in the EIS and subsequently in chapter 3 of the PSAR and SAR. The consultants supplemented their own work by referring to an extensive history of geological investigations performed in the Sydney Basin and the Illawarra in connection with mining and other tunnelling activities in the region. One such example was information sourced from a major hydrogeological investigation of land leased from ANSTO by a regional waste disposal organization.

The external consultants drilled six boreholes at various locations within the OPAL reactor site (see Fig. A–2) to a maximum depth of 45 m as part of their geotechnical investigations and five boreholes as part of their hydrological investigations. A number of the boreholes were subsequently enlarged to allow for vertical seismic shear wave profiling to be carried out and the dynamic shear modulus of the rock assessed. Core samples were used for direct measurements of unconfined compressive strength and the results compared against axial point load strength tests performed during previous site studies. Other laboratory tests included Emerson tests for erosion potential. Since the site is basically solid sandstone, with only a relatively thin layer of soil on top (less than 2 m in thickness, approaching zero in many places), liquefaction, slope stability, subsurface soil issues and other geotechnical aspects found at other sites are not relevant to this site.

Groundwater hydrology was assessed using not only the boreholes within the OPAL reactor site referred to above but also data from numerous other boreholes located within and around the LHSTC. A number of these boreholes have since been developed to form part of the LHSTC long term groundwater monitoring system. This groundwater monitoring network was designed and constructed to establish baseline groundwater conditions and to determine the nature of groundwater migrating from the site and is used for monitoring during operation of the OPAL reactor and annual reporting requirements for the entire LHSTC site.

A–4.6. Seismology

There have been extensive studies done on the seismic characteristics of the OPAL reactor site, many of which were based on earlier work that was undertaken to characterize the seismic hazard applicable to the LHSTC site, in general, and HIFAR, in particular. A publicly available report, Seismic Safety of HIFAR (http://apo.ansto.gov.au/dspace/handle/10238/189), summarized most of the work that had been performed up until 1995, the key points of which are as follows:

— The earliest deterministic studies undertaken in 1982 recommended a peak ground acceleration (PGA) of 0.13–0.18g for the Seismic Level 2 (SL-2) seismic event. The then Australian nuclear regulator subsequently proposed the use of 0.23g, based on alternative work.
— Probabilistic estimates of seismic hazard parameters were made on behalf of the Bureau of Mineral Resources in 1989 that estimated the 10 000 year return period PGA to be between 0.15g and 0.19g, depending on the depth of the earthquake assumed. This was later refined to 0.17g, based on a 14 km deep event. The value of 0.17g was agreed to by the then Australian nuclear regulator.
— The return period (RP) was modelled as:

$$RP = 1.51 \times 10^6 \times \{PGA(g)\}^{2.80}$$

which suggested an event of greater than 0.03g every 85 years. This is consistent with the historical record of an earthquake this size approximately once in 100 years in the Sydney Basin.
— Following discussions with the then Australian nuclear regulator, uncertainties due to the attenuation relationship, the source zone configuration, the magnitude recurrence relationship, graphical representations of source areas and intensity/acceleration relationship were estimated to be ±0.06g. As a result, a conservative estimate of 0.23g was agreed as applicable for Lucas Heights. The then Australian nuclear regulator subsequently advised that 0.2g would be appropriate for the safe shutdown earthquake (SSE). However, note that at that time, uncertainties were assumed to be constant, with decreasing frequency of occurrence, and this is no longer considered a suitable approach.
— Note that best current thinking at the time assumed a mean focal depth of 10km, a strong motion attenuation function as for the Australian earthquake risk study and a relation between acceleration (g) and MM intensity, I, as:

$$\mathrm{Log}(a) = I/3.1 - 1.3$$

This attenuation function is equivalent to:

$$a = 0.008\exp(1.10M_L)R^{-1.2}$$

where R is the hypocentral distance in km, a is the ground acceleration in ms^{-2}, and M_L is the Richter magnitude.

In summary, the controlling earthquake was of magnitude 5.75–6.25 on the Richter scale, with an epicentre at 15–20 km from HIFAR. The response spectrum proposed for this event was based on Carbon Canyon, with a duration of about 3 seconds.

In 1999, the Institute of Geological and Nuclear Sciences (IGNS) performed a probabilistic seismic hazard analysis of the LHSTC commissioned by the Australian Department of Industry, Science and Resources, the purpose of which was to determine the strength and recurrence of earthquake induced ground shaking and its associated spectrum. This analysis included a review of the earthquake ground shaking hazard at LHSTC using best international practice as determined from a review of current US and international procedures, including IAEA guidance. It made use of the most complete computer catalogue of New South Wales earthquakes, which is maintained by the Australian Seismological Centre. In addition, and as mentioned in relation to the site geology above, reference was made to an extensive history of geological investigations performed in the Sydney Basin and the Illawarra connected with mining and other tunnelling activities in the region as well as detailed studies performed in support of the HIFAR safety case.

In response to comments made by an IAEA peer review of the PSAR submitted as part of the application for the construction licence, an additional near field study was undertaken to characterize the area out to 5 km from the OPAL reactor site. The main geological structural features in the area of the LHSTC are shown in Fig. A–3. Fault and lineament data, compiled from rock exposures (i.e. road cuttings, natural exposures along tributaries of the Woronora River and quarry walls), were examined. Existing data were compiled from previous studies performed in 1986 and from the available consultant reports. Fieldwork was carried out following completion of the desk study that included an aerial photographic interpretation of available colour aerial photographs. The field mapping for geological reconnaissance associated with this near field study is shown in Fig. A–4.

As a result of these studies, a response spectrum was adopted for the design that was based on US Regulatory Guide 1.60 for Carbon Canyon, which utilized 5% damping and was scaled to a PGA of 0.30g, as this bounded the local response spectrum developed by IGNS for the site. This compares with HIFAR, which used the local response spectrum scaled to a PGA of 0.23g.

Following the start of excavations during the construction phase, two fault lines were discovered running directly under the reactor block (see Fig. A–5). A detailed assessment by a number of world experts determined that the last movement on the western fault was at least 5–13 million years ago, consistent with the thermochronological results of the fault being 10–35 million years old, and possibly as old as the Tasman Sea opening of 53–83 million years ago. On this basis, the faults are not considered to be potential seismic sources and do not pose a surface–fault rupture hazard. However, as a result of this assessment, and following extensive discussions with ARPANSA and other stakeholders, a revised response spectrum for the SL-2 seismic event was adopted (see Fig. A–6) that retained the same overall response spectrum but with a cut-off PGA of 0.37g.

A–5. SITE SERVICES

The OPAL reactor has generally been designed to operate independently of site services although, naturally, its location at the LHSTC enables some internal services to be backed up by site services (e.g. compressed air).

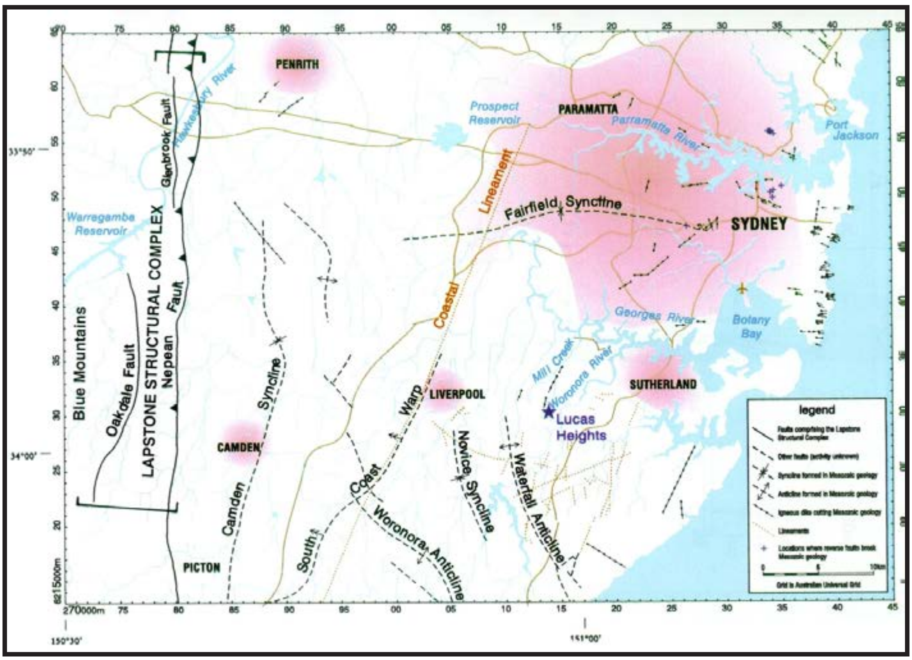

FIG. A–3. Structural features around Lucas Heights (courtesy of ANSTO).

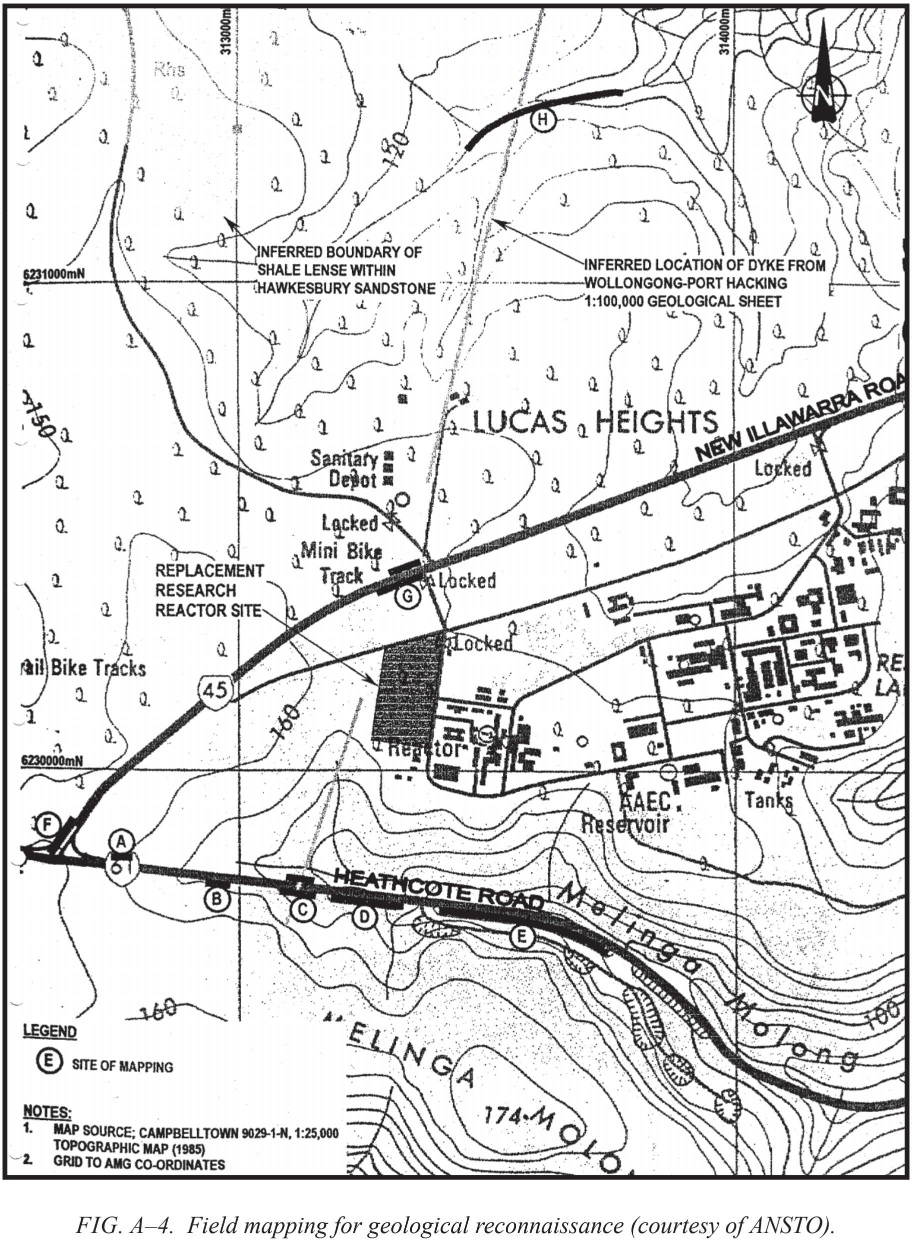

FIG. A–4. Field mapping for geological reconnaissance (courtesy of ANSTO).

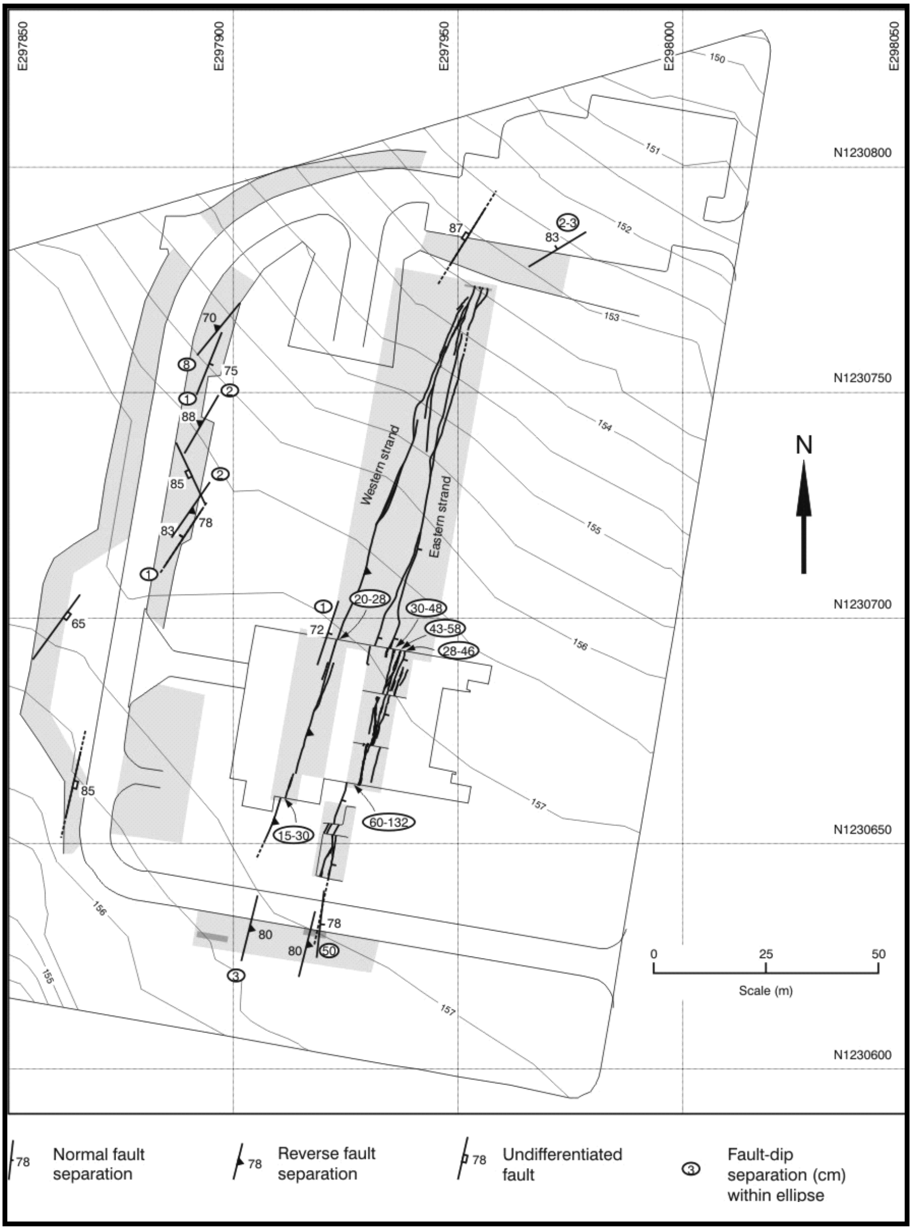

FIG. A–5. Site map showing location of faults discovered during excavation (courtesy of ANSTO).

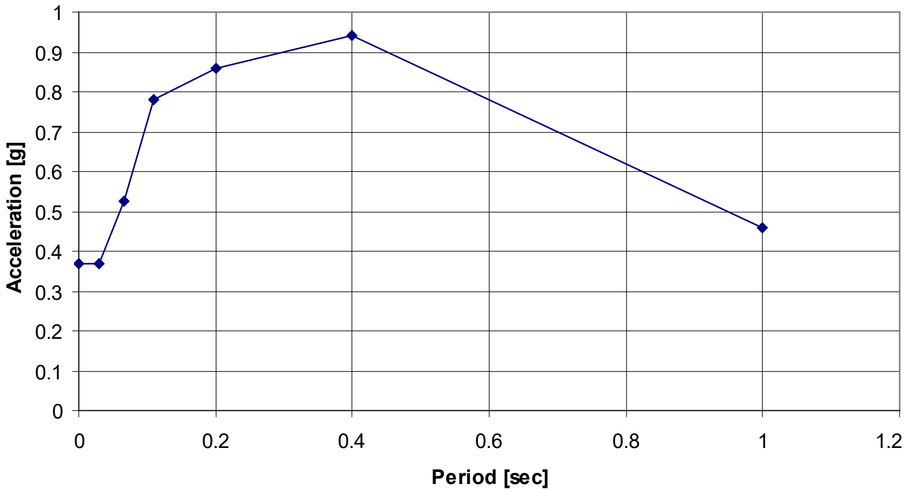

FIG. A–6. Seismic Level 2 horizontal acceleration basic ground response spectrum for the OPAL reactor (courtesy of ANSTO).

No site services are required in order to achieve and maintain the critical safety functions of sub-criticality, core cooling and containment. However, the following site services are required to maintain normal operation:

— Site water supplies due to the high water usage in the OPAL cooling towers, particularly as a result of the blowdown flow;
— In the longer term, site liquid waste handling and disposal facilities, as the OPAL reactor does not discharge liquid waste directly to the environment but instead discharges via the site system;
— Site electricity supplies, such as the OPAL reactor standby power systems (including associated diesel generators), are sized only to achieve and maintain safe shutdown, not continued power operation.

Other site services, such as telephones and computer networks, are also required for the efficient operation of the OPAL reactor as a production source for radioisotopes and for neutron beam users. However, their loss or absence does not have any adverse effect on OPAL reactor safety.

A–6. NEARBY FACILITIES

The main man-made hazards to the OPAL reactor are the other activities and facilities within the LHSTC. The safety of such activities and facilities is ensured by ANSTO's safety management system, which is applicable to all activities and facilities at the LHSTC regardless of whether they are controlled by ANSTO (some being under the control of other Commonwealth organizations). Some of the activities and facilities within the LHSTC are potentially capable of releasing minor quantities of hazardous materials that could necessitate temporary evacuation of local areas on the site and possibly lead to a temporary shutdown of the OPAL reactor. However, an accident occurring at the LHSTC that could significantly damage the OPAL reactor or its safety systems is extremely unlikely as no activities or facilities on the site have sufficient stored energy to penetrate the containment and none has the energy and proximity to cause an accident.

The list of sites for storage of dangerous goods licenced under the NSW Dangerous Goods Act 1975 indicates that there are no oil refineries, chemical plants, plastic manufacturing plants or any industrial complexes that handle large quantities of hazardous materials within an 8 km radius of Lucas Heights. The waste management centre immediately to the north of the LHSTC incorporates two electricity generating plants utilizing methane generated by the waste facility. An assessment of the hazard presented by these plants determined that the total mass of methane is very small and the effects of an explosion or fire would be negligible at the distances involved.

The LHSTC is also adjacent to a military training area run by the Australian army that includes an artillery range. There are strict administrative controls in place that ensure live fire exercises are carried out safely and the risk was assessed as part of the HIFAR PSA as being beyond the design basis (i.e. frequency $< 10^{-7}$ per annum). Since that time, the use of this artillery range has gone down somewhat, further reducing the potential for impact on the OPAL reactor.

A–7. TRANSPORTATION ROUTES

Only three modes of transportation were considered in the siting studies for the OPAL reactor: air, road and rail. Other modes, such as sea and river transport, were dismissed due to the LHSTC's distance from such transportation routes.

A light aircraft impact on the OPAL reactor was considered within the design basis as a result of ANSTO policy, even though studies performed for the HIFAR PSA indicated that any aircraft crash was beyond the design basis (i.e. frequency $< 10^{-7}$ per annum). This resulted in the OPAL reactor's unique design feature of an aircraft impact grillage covering the top half of the reactor building.

An external consultant performed an analysis of transport accidents on the road adjacent to the site (240 m from the OPAL reactor) and the nearest railway (3000 m) and identified five scenarios that were then assessed. The bounding case, involving the rupture of a liquid petroleum gas (LPG) road tanker and subsequent formation of a gas cloud and flash fire on the road 240 m from the site, was assessed to have no impact on the OPAL reactor's operation or safety.

A–8. BASELINE ENVIRONMENTAL RADIOACTIVITY

Information on measured environmental radiation at the LHSTC site and its vicinity is reported in the ANSTO annual environmental surveys, the first of which was conducted in the early 1960s. These surveys provide results of measured radioactivity and radiation levels for airborne emissions, low level liquid effluent and external radiation and, to date, all results are within the relevant discharge authorizations, which also specify the standard or guideline (e.g. WHO's Guidelines for Drinking-water Quality) against which compliance is assessed. The reports also detail the radioactivity in a range of environmental media, including environmental water samples, airborne dust, soil and marine samples, including macroalgae, barnacles and fish.

As part of the approval conditions arising from the EIS, ANSTO undertook a range of new initiatives to characterize the radiological 'footprint' at the site of the research reactor facility. A high sensitivity environmental gamma monitoring system was used, capable of measuring the background radiation characteristics of the site and isolating the comparatively very small component arising from ANSTO's operations. A mobile high sensitivity gamma detector, combined with a global positioning system, was used to map the variation in radioactivity over the reactor site to define the pre-existing radiological baseline.

ABBREVIATIONS

/a	per annum
ACI	American Concrete Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
DBE	design basis earthquake
DC	design class
DS	damage state
EEC	external event category
EIS	environmental impact statement
HC	hazard category
HCLPF	high confidence low probability of failure
HIFAR	High Flux Australian Reactor
MW	megawatt
NOC	normal operational condition
OPAL	Open Pool Australian Lightwater research reactor
PF	probability of failure
PG	performance goal
PGA	peak ground acceleration
PIE	postulated initiating event
PSA	probabilistic safety assessment
PSAR	preliminary safety analysis report
RLE	review level earthquake
SAR	safety analysis report
SMA	seismic margin assessment
SSC	structure, system and component
TBq	terabecquerel
TNT	trinitrotoluene
TRIGA	Training, Research, Isotopes, General Atomics research reactor

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