Safety Reports Series

No. 41

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

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

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Printed by the IAEA in Austria
March 2005
STI/PUB/1209

SAFETY REPORTS SERIES No. 41

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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2005

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http://www.iaea.org/Publications/index.html

IAEA Library Cataloguing in Publication Data

Safety of new and existing research reactor facilities in relation to external events. — Vienna : International Atomic Energy Agency, 2005.
p.; 24 cm. — (Safety reports series, ISSN 1020–6450 ; no. 41)
STI/PUB/1209
ISBN 92–0–111704–3
Includes bibliographical references.

1. Nuclear reactors — Safety measures. 2. Nuclear reactors — Design and construction. 3. Nuclear engineering — Safety measures. I. International Atomic Energy Agency. II. Series.

IAEAL 05–00384

FOREWORD

In recent years, concerns about the safety of research reactors have been growing, mainly due to the ageing of these installations around the world. In 2000, the General Conference of the IAEA requested the Secretariat to improve efforts to assist Member States in safety activities that focus on the design of new facilities and the re-evaluation of existing ones.

The IAEA singled out evaluation of the structural and mechanical performance of such facilities in relation to external events as a key area needing further investigation, and considerable emphasis was given to this topic in the review of IAEA Safety Standards Series publications on research reactors.

The IAEA has issued a number of technical publications that document the experience of Member States in this area. To complement these publications, a Safety Report was also planned to compile recommendations and examples of good practices in the design and re-evaluation of research reactors. With the assistance of a number of structural and design experts, the IAEA has completed the first step in fulfilling the directive of the General Conference. The present publication provides insights, guidance and a framework for Member States to conduct a realistic safety assessment using a graded approach based on the radiological hazard that the facility represents to the environment, the public and workers. It is hoped that this publication will stimulate the preparation of national guidelines for the design and re-evaluation of research reactors in Member States.

The IAEA officer responsible for this publication was P. Contri of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Siting and design of research reactors in relation to external events is not an established discipline in IAEA Member States. In some cases, provisions for conventional industrial design are applied, while in others standards and regulations pertaining to nuclear power plants (NPPs) are considered. One of the main reasons for such broadly differentiated approaches is the difficulty associated with the categorization of research reactors. The research, experiments and production activities that are carried out in research reactors lead to very different safety cases and plant layouts.

The safety objectives for research reactors are defined in Ref. [1], which is complemented by two Safety Guides [2–3]. These Safety Guides provide recommendations to Member States and support the IAEA as it reviews the safety of research reactor facilities.

However, it was recognized that in many cases detailed safety requirements were lacking, particularly in relation to the radiological hazard posed by research reactor facilities to the environment, the public and workers as a consequence of external events. This fact sometimes compelled designers to adopt the most demanding safety criteria (typical for the design of NPPs) in the design against external events to avoid very complicated safety cases in support of less stringent safety approaches. Therefore it was determined that a Safety Report was needed to address the peculiarities of the different kinds of research facilities and to provide a consistent framework for the evaluation of their safety. This publication would set out a graded approach, i.e. a suitable gradation between the safety requirements developed for NPPs and the requirements for facilities with conventional industrial risk. The concept of a graded approach is established in Refs [1, 4], and is now practiced in many Member States. The graded approach provides the general safety framework for both the implementation and review of design/re-evaluation projects, even if the safety classification of structures, equipment and components and the design methodologies still differ between countries.

IAEA safety publications that consider hazards for a broader spectrum of nuclear installations and are relevant to the present publication are Ref. [5], which covers the design of nuclear facilities other than NPPs, Ref. [6], which covers siting, and Ref. [7], which covers quality assurance (QA) for nuclear installations. Reference [8], a report on the seismic re-evaluation of NPPs and Ref. [9], which covers seismic hazards for NPPs, are also of interest. IAEA technical publications that have been widely used in recent years have addressed only isolated aspects of the design of research reactors. A key technical publication[1] that focuses on seismic considerations has been used in many countries to provide the contractual basis for the design of some research reactors. The successor to that publication [5] presents simplified siting and design methods aimed at minimizing the need for sophisticated calculations, emphasizing the importance of construction and structural detailing since the design phase. In addition, it proposes an approach for seismic safety evaluation that is an alternative to the complex methodologies associated with NPP analysis and design. Other IAEA technical publications [10–14] provide examples of practices in various Member States and contain the preliminary formulation of a graded approach, which is examined in detail in the present publication.

1.2. OBJECTIVE

The main purpose of this publication is to provide guidance for conducting a safety evaluation of new and existing research reactors in relation to the hazards posed by external events, consistent with the general safety requirements set forth in Refs [1, 4].

This publication is based on the experience available in Member States in evaluating the safety of research reactors, and provides a coherent framework for the application of a graded approach to design safety. It is intended for use by regulatory bodies and organizations in charge of the safety assessment of research reactors, by designers and by contractors. It provides a technical basis for the safety aspects of self-assessment, in line with the IAEA safety requirements. The framework presented here can be used for the development either of site-specific or plant-specific guidelines for the actual conduct of the design and safety assessment.

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

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory, IAEA-TECDOC-348, IAEA, Vienna (1985). (Superseded by Ref. [5].)

1.3. SCOPE

From the standpoint of safety of the public and workers, research reactors are very difficult to categorize due to their very broad range of application (e.g. training, research and isotope production), their design characteristics and their power. This publication develops a categorization initially based on the radioactive inventory of research reactors. It is intended to apply generally to any type of research reactor or other types of nuclear facilities other than large nuclear reactor facilities.

Other facilities may be located at the sites of research reactors, namely:

(a) Laboratories for isotope production, industrial applications (e.g. non-destructive testing) and fuel production;
(b) Radioactive material storage and waste treatment facilities;
(c) Power supply facilities (e.g. diesel generators);
(d) Other auxiliary structures and facilities (e.g. cooling towers, tanks, isotope transfer tunnels);
(e) Storage for fresh and spent fuel elements.

Most of these facilities can be evaluated with the approach put forward in this publication, even if special care has to be applied in the analysis of special equipment such as hot cells and fuel production devices, for which dedicated requirements may need to be developed.

The external events considered in this report include both natural hazards and human induced hazards from sources external to the site or external to the safety related buildings. Explicit reference is made to the most common external event scenarios considered in the design of research reactors (earthquake, wind, precipitation (snow, rain, hail), flood, explosions and aircraft crash, external fire), for which special recommendations are provided. However, the approach to the safety evaluation discussed in the present publication can be applied to any scenario included in the facility's safety analysis report.

This publication addresses both the design of new facilities and the re-evaluation of existing ones. A re-evaluation can be required due to a comprehensive plant modification (e.g. for accommodation of new experiments), a periodic hazard re-evaluation or the modification of the licensing requirements by the regulatory body. The present publication provides suggestions and guidelines for all these cases.

1.4. STRUCTURE

This publication addresses the following main areas:

(a) Development of a detailed categorization of the radiological hazard posed by the facility to workers, the public and the environment preliminary to the selection of suitable methods and procedures for evaluation of the facility site, its design and safe operation (Sections 2–4);
(b) Execution of site investigations and development of a site-specific hazard analysis (Sections 5 and 6);
(c) Development of structural design and/or re-evaluation of safety classified structures, systems and components installed in hazardous facilities (Section 7);
(d) Evaluation of the impact on the environment of a radioactive dispersion from the reactor facility following an accident (Section 8);
(e) Development of suitable measures for monitoring, alerting, event management, post-event inspection and implemention of the emergency plan in relation to external initiating events (Sections 9–11);
(f) Development of a QA system for the activities discussed in the present publication (Section 12).

Three appendices provide examples of feedback from operating experience in past years in support of the safety approach proposed in the present publication, along with some examples of application that are considered useful for a practical and unequivocal implementation of the proposed approach.

2. SAFETY CONCEPTS IN SITING AND DESIGN

2.1. GENERAL

This section includes some basic safety concepts developed with the aim of grading the design criteria 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. In the following, reference is made to the basic steps of the safety assessment process, as described in Ref. [15].

The following reference plant states and operational modes have to be considered in the safety assessment of the plant:

(a) Normal operation;
(b) Anticipated operational occurrences;
(c) Accident conditions;
(d) Beyond design basis accidents;
(e) Long term shut down behaviour with or without the need for active cooling of radioactive material;
(f) Refuelling, maintenance;
(g) Storage and processing of radioactive material and waste.

Postulated initiating events are selected as suggested in Ref. [1]. Special care has to be taken in the identification of the postulated initiating events relevant to the external event scenarios:

(1) Accident sequences initiated by external events (postulated initiating events);
(2) Accident sequences initiated by postulated initiating events with a significant probability of being contemporaneous with external events, even though they are not correlated.

2.2. SAFETY OBJECTIVES FOR RESEARCH REACTORS

The safety objectives for research reactors which are vulnerable to external events [16] define the acceptable radiological consequences to workers and the public under accident conditions. Other safety consequences, such as chemical hazards posed by research reactors, do not fall within the scope of this publication according to the relevant criteria for nuclear installations set forth in Refs [1, 4].

(a) Normal operation should lead neither to effective doses to workers higher than a mean of 20 mSv/a over 5 years, nor more than 50 mSv over a given year. As far as the public is concerned, normal operation should not lead to yearly doses above 1 mSv/a (5 mSv/a are allowed under special conditions).
(b) Design basis accidents should have a probability of occurrence of less than 10^{-4}/a. In terms of doses, design basis accidents are subject to the same requirements as normal operation.

(c) Beyond design basis accidents should have a probability of occurrence lower than 10^{-6}/a. They should not lead to effective doses to workers and the public that are higher than 10–50 mSv/a.

Concerning the 'ALARA' process for the optimization of radiation doses, see the recommendations set forth by the IAEA in Ref. [16].

2.3. TECHNICAL SAFETY OBJECTIVES FOR RESEARCH REACTORS

Technical safety objectives have to be developed for any research reactor with reference to the three main safety functions that also need to be ensured for external event scenarios [1]:

(1) Reactivity control during and after the external event[2] allowing, either automatically or through operator action, the power of the research reactor to be reduced to a sufficiently low level to maintain a suitable margin to deal with later events or an evolution in the emergency. Redundancy and diversity in the reactor reactivity control system should be demonstrated.
(2) Cooling of radioactive material after the external event should also be possible with dedicated and reliable systems when necessary, though often for research reactors natural convection or heat accumulation in the coolant is sufficient. Whenever needed, redundancy of devices for establishing the natural convection should be considered.
(3) Confinement of radioactive material and protection of workers, the public and the environment against irradiation should be provided within prescribed limits.

For each selected postulated initiating event, a list of safety related structures, systems and components needs to be developed and safety requirements for these items established. Spatial and other possibilities of interaction between items have to be examined since an external event can alter the

[2] The analysis should consider the duration of the event and the time needed to return to a normal situation. In case the facility is not started again, the total duration of the event corresponds to the time needed to come to a new, stable and sustainable situation. The safety analysis should also consider this scenario.

behaviour of a great number of items simultaneously.[3] In particular, external events could induce chemical or biological hazards that might result in consequences for safety such as reductions in personnel availability, limitations on transportation and restriction of access.

Table 1 illustrates some of the challenges posed by different external events to the basic safety functions of a research reactor. A special column (the sixth) highlights the need for additional protection features against beyond design basis events when the development of the external event can induce cliff edge effects in the response of the facility.

The technical safety objectives aim at preventing accidents in research reactors and mitigating their consequences if they occur. It is necessary to show that any radiological consequence would be below prescribed limits, with a high level of confidence and for all design basis accidents. For research reactors this means the following:

(1) Shutting down the reactor when it is subjected to an extreme external event (reactivity control) and maintaining the reactor in a safe shutdown condition;
(2) Removal of residual heat over an extended period of time (cooling of radioactive material);
(3) Preventing radioactive releases or maintaining releases below the limits established for accident conditions (confinement);
(4) Avoiding any failure of structures, systems or components which could directly or indirectly cause accident conditions as a consequence of an external event, particularly with respect to reactivity control, cooling of radioactive material and confinement;
(5) Monitoring of the critical reactor parameters during and after an external event, in particular the reactivity;
(6) Monitoring the radiological dispersion parameters;
(7) Guaranteeing access and evacuation to the operating personnel in charge of the above functions (e.g. ventilation in the control room), communication (both among personnel and with the outside world), and alarm (for implementation of the emergency measures, both on-site and off-site).

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[3] For example: (1) operator access to safety systems may be impaired by difficult conditions at the site, e.g. heavy snow, flood; and (2) internal flooding can originate as a result of leakage from or failure of fluid storage tanks, including non-safety-related tanks.

TABLE 1. SAFETY FUNCTIONS AND EXTERNAL EVENTS

External event	Reactivity control	Cooling of radioactive material	Confinement	Common mode on a wide area[a]	‘Cliff edge effect’[b] on the consequences (except in the case of building collapse)
			Human induced		
Aircraft crash (P) (Y/N)	(✓)	(✓)	✓		
Industrial accident (D) (Y/N)	x	x	✓		
Transport accident					
— Road (D) (Y/N)	x	x	✓	Limited, mainly dealing with explosions, if any	No
— Rail (D) (Y/N)	x	x	✓		
— Water route (D) (Y/N)	x	✓	✓		
— Pipelines (D) (Y/N)	x	x	✓		
Dam rupture (P) (Y/N)	x	✓	✓	Yes	
			Natural phenomena		
Geological hazards, i.e. landslide, avalanche (D) (Y/N)	Can be excluded by site screening or hazard monitoring			Yes	No
Flooding (P) (DB)	x	✓	✓	Yes	Yes, site overflow
Earthquake (P) (DB)	✓	✓	✓	Yes	No
		Extreme meteorological conditions			
Wind (P) (DB)	[c]	(✓)	✓	Yes	No
Tornado (P) (Y/N)	[c]	(✓)	✓	Yes	No
Snow (P) (DB)	[c]	(✓)	✓	Yes	No
Icy conditions (P) (DB)	[c]	✓	x	Yes	No

TABLE 1. SAFETY FUNCTIONS AND EXTERNAL EVENTS (cont.)

External event	Reactivity control	Cooling of radioactive material	Confinement	Common mode on a wide area[a]	‘Cliff edge effect’[b] on the consequences (except in the case of building collapse)
Lightning (DB)	(✓)	(✓)	(✓)[d]	Yes	No
Forest fire (P) (Y/N)	[c]	(✓)	(✓)[d]	Yes	No

(P): The possibility of a screening criterion based on probabilistic evaluations.

(D): The possibility of a screening criterion based on deterministic evaluations (distance, etc.).

(Y/N): The event may be included or not in the design basis, according to the site evaluation.

(DB): The event is usually included in the design basis, at least with a minimum deterministic value, also to envelope effects from other sources not explicitly evaluated.

✓: The function is directly challenged.

(✓): The function is only indirectly challenged.

x: The function is usually not directly affected (only in case of building collapse or loss of containment).

[a] Common mode on a wide area usually indicates possible loss of site infrastructures such as power distribution, communication, water cooling, lubrication, remote surveillance. Therefore common mode failure could greatly impact defence in depth, also through the possibility of implementation of the emergency plan.

[b] In some external scenarios, the development of the consequences for the facility is not proportional to the growth of the load. In these cases a cliff edge effect is recorded, i.e. 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, where as soon as the water is higher than the protection, the whole site is flooded to the maximum level.

[c] The reactor should be shut down prior to the extreme event if enough warning time is available. The combination of engineering features and operational/ administrative measures is discussed below.

[d] For example, loss of confinement through loss of power (lightning), or smoke plugging the filtration system (fire), or many others.

2.4. CATEGORIZATION OF FACILITIES

Once the postulated initiating events and the safety objectives have been defined, the safety analysis of the facility can be developed. The first step is the categorization of the hazard that a facility poses to workers, the public and the environment. In the following, the term ‘facility’ includes all the structures, systems and components at the facility.

In general, the probability that external events will generate a radiological consequence depends on the characteristics of both the source (facility use, layout, design, operation) and the initiating events, such as:

(a) The amount, type and status of radioactive inventory at the site (e.g. solid, liquid, gas and vapour processed or stored);
(b) The intrinsic reliability and hazard associated with the chemical and physical processes taking place in the facility (e.g. processes involving transportation of hazardous substances might pose a greater hazard than when fuel is not moved);
(c) The installed thermal power of the facility (not only output);
(d) The configuration of the facility for different kinds of production;
(e) The concentration and number of radioactive sources at the plant (e.g. in the reactor core, reactor pool, irradiation facility waste or material storage);
(f) Whether the facility is designed for experiments and research (such activities have an intrinsic unpredictability associated with them) or might be subjected to configuration and layout changes (such as activities related to the development of new products);
(g) The need for active safety systems to cope with mitigation of postulated accidents, the number of engineering features installed for preventing and mitigating serious consequences from accidents;
(h) The possibility of installing warning systems able to promptly detect the potential unfavourable development of an event (e.g. events with slow development as opposed to aircraft crashes);
(i) The characteristics of the process or of the engineering features which might show a cliff edge effect in the event of an accident, with no possibility of preventing degeneration into radiological consequences;
(j) The characteristics and nature of the external events challenging the facility (e.g. wind and explosion have a high potential for dispersion, while earthquake and aircraft crash in the absence of fire have a lesser potential for dispersion of radioactive material);
(k) The environmental characteristics of the site relevant to dispersion (e.g. windy area, coastal site);

(l) The ease of implementation of emergency planning in relation to the event, i.e. access to the site, availability of evacuation routes, time delay between the accident and releases;
(m) The potential for long term effects in the event of contamination (long lived radionuclides, persistent effect on the environment);
(n) The number of people potentially affected by an accident at the facility;
(o) The potential for off-site or on-site radioactive contamination.

Therefore, a general evaluation of the risk associated with a research reactor is difficult due to the high number of variables, and depends on the specific layout. In general a reasonable and reliable risk classification can be made only on a case by case basis, possibly after a detailed probabilistic safety assessment (PSA), which is usually not available at the design stage. In the framework of this publication, which is mainly oriented towards an identified group of facilities and risks, the issues listed above could be interpreted as criteria for such risk classification, driving the final evaluation of the risk associated with the facility, ranging from a minimum risk (conventional buildings) to the highest values (NPPs).

A reasonable and much simplified approach could entail a reduction in the number of the criteria described above, as for most research reactors the application of the criteria for facility classification shows a strong correlation between the risk associated with the facility and its installed power or radioactive inventory. This correlation might simplify the classification process at the beginning of the design. Clearly such assumptions have to be assessed in the safety assessment phase and justified in the safety analysis report. Table 2 provides an example of such a simplified approach based on the power rating of a research reactor or on the quantity and form of radioactive material (source term) in the facility. In the case of conflict, the most stringent criteria can be applied.

For the purposes of this publication, facilities with power or radioactive inventory higher than category 1 (hazard category 1) can be regarded as NPPs. Even if the categorization is based only on the power (source term), there are criteria for up or downgrading the category:

(1) 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, the category defined by the power can be decreased by one. The same can be done for pool type reactors if the cladding material of the fuel is stainless steel or zirconium alloy.
(2) If the reactor is categorized as hazard category 2, hazard category 3 or hazard category 4, the category which hinges on the power can be

TABLE 2. EXAMPLE OF HAZARD CATEGORIZATION FOR RESEARCH REACTORS

Hazard category of the facility	Power rating (MW)	Inventory (TBq (10^{12} Bq) (I))	
		γ[a]	α[b]
1 high	$10 \leq P < 100$	$I > 2 \times 10^6$	$I > 10$
2 medium	$2 \leq P < 10$	$4 \times 10^5 < I < 2 \times 10^6$	$2 < I < 10$
3 low	$0.1 \leq P < 2$	$4 \times 10^4 < I < 4 \times 10^5$	$0.2 < I < 2$
4 very low (special risk)	$P \leq 0.1$	$I < 4 \times 10^4$	$I < 0.2$

[a] These values correspond to normalized values from a 20 MW (U_3Si_2) equilibrium core.
[b] These values correspond to approximately 207 irradiation days. With this assumption, a 10 MW reactor should have about 10 TBq (exactly 7.48 TBq).

increased by one under one or more of the following conditions: the reactor is associated with high temperature and pressure experiments, or it contains fuelled experiments. The same applies for prototype power reactors.

(3) Category 4 is intended to include facilities in which the core cannot melt and therefore the source term for radioactive dispersion into the environment is particularly small. Hazard category 4 facilities can be regarded as industrial installations at special risk and therefore they are not discussed in this publication. Category 3 research reactors can be downgraded to category 4 if they show such an intrinsic feature in relation to the core melting.

While the initial hazard categorization of research reactor facilities is primarily a function of power rating and radioactive inventory, it can also be affected by site characteristics. A research reactor facility that would normally be classified as a hazard category 1 facility if located several kilometres from residential or industrial facilities may be upgraded to hazard category 2 if it is located in a built up area, and a hazard category 2 facility may be categorized as hazard category 3 for similar reasons. This approach is applied in some countries where the hazard categorization is based simply on the need for emergency systems in the case of an extreme event to meet the safety requirements. However, this approach is banned in other countries where the hazard is associated only with the facility and the design is independent of external conditions such as population density in the surroundings. Any categorization approach should be evaluated with care and its outcome judged in the framework of a risk comparison approach, as described in Ref. [17].

2.5. SAFETY CLASSIFICATION AND EXTERNAL EVENT CATEGORIZATION FOR STRUCTURES, SYSTEMS AND COMPONENTS

A safety classification should be developed for structures, systems and components, as set forth in Ref. [1], as the second essential step in the safety analysis. Many criteria can be applied in this process. A possible approach, which appears in Ref. [4], is particularly suitable to be applied in a graded approach to safety. It foresees development of a safety classification as a function of the unmitigated radiological consequence that would result to workers, the public and the environment from the failure of the structures, systems and components to fulfil their required safety functions. This approach provides an alternative to the classical deterministic approach based on the safety function of each component and is mainly based on analysis of the consequences of any failure affecting the structures, systems and components. The main advantages of focusing on radiological consequences are more realistic modelling of the risk associated with the components (definitely closer to any external event PSA approach), and a drastic reduction of the number of safety classified items. Conversely, the approach needs a little more information on the safety analysis of the plant, the consequences of component failure, and relevant probabilities (with an associated ‘quality index’).

Following this approach, safety classes may be developed as in the following, with reference to the safety analysis of all postulated initiating events:

(a) Safety class 1 is specified for a structure, system or component when the postulated failure (unmitigated) of the structure, system or component to perform its required safety function would result in an unacceptable release of radiation to the environment, the public and workers in any plant state. Safety class 1 structures, systems and components are usually located in hazard category 1 facilities. However, this classification can also be used for classification of confinement barriers in hazard category 2 facilities, according to the defence in depth approach applied to the facility.
(b) Safety class 2 is specified for a structure, system or component when the postulated failure (unmitigated) of the structure, system or component to perform its safety function would result in an unacceptable release of

radiation to the environment within the site boundary or to workers in any plant state. Safety class 2 structures, systems and components are usually located in hazard category 2 facilities, but may also be applied in hazard category 1 facilities to classify structures, systems and components when the unmitigated release is a small fraction (20% or less) of the safety class 1 release limit. Safety class 2 may also be used for the confinement barriers in hazard category 3 facilities.

(c) Safety class 3 is specified for structures, systems and components of hazard category 3 facilities, except the confinement, but may also be applied in hazard category 1 and hazard category 2 facilities to classify structures, systems and components when the unmitigated release is a small fraction of the release limit for safety classes 1 and 2.

(d) Safety class conventional risk is specified for structures, systems and components when there is no radiological consequence of the failure of a safety function.

It has to be noted that an unmitigated failure by itself may not lead to a release. However, the safety classification should consider all the scenarios foreseen by the postulated initiating events. The emergency systems are a typical example. Their failure may lead to a release if there is a coincident failure of other structures, systems or components which require proper functioning of the safety systems.

In addition to the safety classification, an external event categorization may be useful to drive a rational design process [14]. External event categories 1, 2, 3 and conventional risk may be identified using the same unmitigated release criteria as for the respective safety classes, but with reference only to the selected postulated initiating events for external events.

Further comments on the interaction of the hazard category, safety class and external event category follow:

(1) The hazard category is a qualitative measure of the hazard posed by the facility. It should be a result of the safety classification of the components, but actually it accounts for many other issues, not always made explicit, and therefore engineering practice prefers that it be assigned first.

(2) The safety class is a quantitative measure of how the radiological hazard is distributed at the facility. In principle, a single very hazardous component or many low hazard components might result in assignment of the same hazard category to a facility.

(3) The external event category is a quantitative index of how much the external events can trigger hazardous consequences at the plant. Equipment with a high safety classification may not be exposed to or

TABLE 3. SAFETY CLASSES AND EXTERNAL EVENT CATEGORIES CONTRASTED

Classification of structure, system or component		Compatible facility hazard category
Safety class	Compatible external event category	
1	External event category 1, 2, 3, or conventional risk	Mainly 1, but also 2 (for defence in depth barriers)[a]
2	External event category 2, 3, or conventional risk	Mainly 2, but also 1 (for low release) and 3 (for defence in depth barriers)[a]
3	External event category 3, or conventional risk	Mainly 3, but also 1 or 2 (for low release) and conventional risk (for defence in depth barriers)[a]
Conventional risk	External event category conventional risk	All

[a] See Section 2.8 for the categorization of defence in depth barriers.

affected by external events and therefore its external event category may be quite low.

(4) The external event categorization does not require the safety classification to be developed as described above. Both the external event categorization and the safety classification can be developed following a different logic. An external event categorization can also be developed independently of a safety classification. The proposal presented here is particularly suitable for a graded approach to safety.

As a consequence of such assumptions, the interaction between safety classes and external event categorizations is shown in Table 3. For items that interact (through mechanical or chemical interaction, induced fire, flood, electromagnetic interference), the category of the impacting item may be the same or lower than that of the impacted item. A detailed analysis can be carried out to evaluate both the expected consequences of the impact and the joint probability that failure of the impacting item is going to induce failure in the impacted one. If the probability of a radiological consequence from the interaction is significant, the impacting item can have the same category as the impacted item; if either the consequence of the interaction is negligible or the probability of impact is too low, the category of the impacting item may be lower than the category of the impacted item.

Beyond the general list provided in Annex I of Ref. [1], a more detailed description of the typical safety systems and relevant safety functions of a research reactor in relation to external events is provided in Table 4. These systems are typically external event category categorized systems.

2.6. PERFORMANCE GOALS

The contribution of each structure, system or component to the facility hazard can be measured through the probabilistic concept of the performance goal. The performance goal for a structure, system or component in relation to a specific external event is defined as the probability of failure (P_F) of the structure, system or component to perform its required safety function in the case of that external event. The performance goal for an external event may be lower than the performance goal for internal accidents.

The probability of failure of structures, systems or components as the result of external events is computed as the product of the full range hazard curve of the external events convoluted with the derivative of the fragility of the structure, system or component under consideration, as shown in Section 4. The fragility of structures, systems and components is defined as the cumulative conditional P_F (unacceptable performance) versus the selected hazard parameter. The hazard parameter is typically represented by factors such as the peak ground acceleration (PGA) for earthquakes, the water depth for floods and the maximum wind speed for winds.

Typical values for performance goals for research reactors are presented in Table 5 in relation to generic external event scenarios. For comparison, it should be remembered that performance goals for NPP components in the highest safety class are normally established at a P_F of 10^{-6}/a. In principle, the performance goal could also be a function of the hazard category of the facility, but for this publication it is only a function of the external event category.

TABLE 4. TYPICAL STRUCTURES AND SYSTEMS OF A RESEARCH REACTOR TO BE CONSIDERED DURING THE DESIGN OR RE-EVALUATION FOR WITHSTANDING EXTERNAL EVENTS

Item	Identification of the safety function of the item	Effect of the loss of safety function of the item
Reactor building	Structural integrity[a], stability[b]	Damage to the reactor, and status and control system
Reactor pool with or without pool lining or tank	Structural integrity and leaktightness[c], stability	Inability to maintain reactor pool water level
Control building	Structural integrity, stability	Inability to monitor and control safety activities
Ventilation stack	Stability	Damage to items important to safety
Shielding structures, protection dams	Structural integrity, stability	Loss of shielding or protection
Reactor vessel and reactor internals, or reactor block	Structural integrity, stability	Core damage
Control rod drive mechanism	Functionality[d]	Core damage
Reactor scram system	Functionality	Core damage
Reactor cooling system	Structural integrity, functionality (when required)	Core damage
Second shutdown system	Structural integrity, functionality	Insufficient shutdown margin for selected reactor types
Effluent filtration system	Structural integrity, functionality	Greater radioactive releases
Emergency power supply	Functionality	Inability to perform safety functions
Safety significant instrumentation and control, and safety protection systems	Functionality	Inability to perform the safety function

[a] Structural integrity means that the structures, systems and components will continue to maintain their geometry and transfer load.
[b] Stability means that the structures, systems or components will not collapse.
[c] Leaktightness means that the structures, systems or components will maintain fluid inventory under acceptable limits.
[d] Functionality means that the structures, systems or components will continue to perform their required safety functions during and following an external event.

TABLE 5. PERFORMANCE GOAL AND EXTERNAL EVENT CATEGORY

Hazard category of the facility	External event category 1	External event category 2	External event category 3
1	10^{-5}/a	10^{-4}/a	10^{-3}/a
2	10^{-5}/a (only for the barriers, if needed)	10^{-4}/a	10^{-3}/a
3	See note	10^{-4}/a (only for the barriers, if needed)	10^{-3}/a (industrial installations)
4 (special risk)	See note	See note	See note

Note: These facilities cannot host components in this external event category. See Section 2.5.

2.7. DESIGN CLASS FOR STRUCTURES, SYSTEMS AND COMPONENTS

A design class for structures, systems and components can be defined as the level of safety margin[4] (i.e. the inverse of the reliability[5] to perform the assigned safety function) that can be used in the design/qualification of a structure, system or component. It can be evaluated according to the external event probability of exceedance[6] and the performance goal associated with the structure, system or component.

The design of external event class 1 and 2 structures, systems and components typically makes use of procedures (but not necessarily load levels) developed for NPPs. Design of external event class 3 structures, systems and

[4] Safety margin means:
- The availability of one or more lines of defence, before radionuclides can be released to the environment, which is demonstrated by a capacity divided by a demand value greater than 1.0,
- The capability of detection — such as the ‘leak before break’ concept — which would allow preventive measures to be taken in time, or
- The availability of mitigation measures.

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

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

components is typically based on procedures defined in codes and standards for hazardous conventional industrial facilities. The application of different design/ qualification codes implies the use of different reliability levels and therefore different failure probabilities. However, this level may be embedded and hidden in the codes. Therefore, a detailed analysis of the assumptions implicit in the design classes has to be carried out with the aid of Table 6 to check the applicability of any design standard and code to the framework of interest.

Table 6 gives an example of how a design class might be evaluated in a simplified way as a ratio between the performance goal and the probability of exceedance for the external event (P(EE)). This table also provides an acceptable range for the values of the design class and P(EE). In fact, in principle, all possible combinations meeting the performance goal value should be acceptable, but engineering practice limits the range of design class and P(EE) to one or two orders of magnitude (see Section 7.3 for further details).

The selection of the P(EE) should follow the considerations of Section 2.4 that some events show higher destructive potential, or the potential for common cause failures, and therefore their return period may be longer. However, physical considerations may also affect the choice of P(EE). For some events, 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). A feasible choice for P(EE) is therefore suggested in the following; the values have to be interpreted as minima in order to reliably estimate the associated physical description of the external event scenario.

— Earthquake: 10^{-3}–10^{-4}/a;
— Straight wind: 10^{-3}/a;
— Rotating wind: 10^{-5}/a;
— Flood: 10^{-4}/a;
— Human induced events: 10^{-5}/a.

2.8. APPLICATION OF THE DEFENCE IN DEPTH CONCEPT

The defence in depth concept should be used in the analysis and design of new research reactors and in the re-evaluation of existing research reactors. It aims at providing the required safety functions with a suitable level of reliability, according to Refs [1–5, 8–12, 15–18]. The definition of defence in depth used in this report is that given in Ref. [4].

TABLE 6. DESIGN CLASS DETERMINATION

Performance goal	P(EE) = 10^{-3}/a	P(EE) = 10^{-2}/a
10^{-5}/a	Design class 2 10^{-2}/a – 10^{-3}/a	Design class 1 10^{-3}/a – 10^{-4}/a
10^{-4}/a	Design class 3 10^{-1}/a – 10^{-2}/a	Design class 2 10^{-2}/a – 10^{-3}/a
10^{-3}/a	Design class 4 codes and standards for conventional risk facilities	Design class 4 codes and standards for conventional risk facilities

The external event category 1 and 2 structure, system or component "should be soundly and conservatively" constructed[7], evaluated, procured, operated and maintained "in accordance with appropriate quality levels and engineering practices, such as the application of redundancy, independence and diversity".

Defence in depth aims at a balance between two major aspects of the safety approach, namely:

(1) Detection of deviation from normal operation, as the external event might induce unavailability of safety systems, remote control and surveillance systems;
(2) Mitigation of significant events to ensure minor consequences for any postulated accident. Passive external event category 1 and 2 structures, systems and components are preferable to address this concern.[8]

While the procedure proposed in this publication uses the performance goal as a measure of the required reliability of any structure, system or component (including safety barriers), 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

[7] Construction includes compliance with administrative requirements, documentation, material selection and qualification, design, fabrication installation and commissioning.

[8] Passive structures, systems and components are those whose functioning does not depend on external input (structural items, shielding, etc.).

number of levels and their reliability are a function of the facility hazard categorization, and have to be seen as an additional 'robustness' applied to the design.

It must be noted that, according to a strict application of Ref. [1], there should always be five defence in depth levels and the need for systems at any defence in depth level should be defined in connection with the safety analysis of the plant and therefore with the safety classification of its structures, systems and components. However, many safety issues have to be considered at a research reactor, not always explicitly correlated with component failures, such as most of the items listed in Section 2.4, which are part of the hazard categorization of the facility. Therefore the recommended global approach tries to synthesize them and develop a comprehensive proposal.

In general, it is the safety analysis of the facility that supports the need for dedicated systems at any defence in depth level. For example, for 'small' research reactors the postulation of design basis accidents may not lead to unacceptable releases and therefore the third level of defence in depth may not be needed. Table 7 was developed on the basis of engineering experience to simplify the application of the defence in depth approach to research reactors. Some alternative solutions, which can be considered to be equivalent on the basis of probabilistic considerations, are shown in the table. Such a proposal should always be agreed to with the national safety authority.

Barriers of the 'a' type can be designed with high reliability ('a+', the relevant performance goal, shows an extra order of magnitude) or with a low margin ('a' type structures, systems and components are designed/qualified according industrial standards).

Barriers of the 'b' type represent administrative measures and operational procedures. A typical 'b' barrier is emergency planning.

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.

Application of the defence in depth approach to a research reactor in case of external events entails certain clarifications:

(a) Protection of the facility against external events is always part of the first level of defence, as defined in Ref. [1], and therefore has to be established through robust and reliable design.
(b) Robust design has to be understood as being of high quality and low sensitivity to variation in design parameters. It is usually achieved by means of a high connectivity layout, detailing of joints, consideration of

TABLE 7. LEVELS OF DEFENCE IN DEPTH

Lines of defence and design requirement for systems	Grade of safety function items				
	Hazard category 1		Hazard category 2		Hazard category 3
Standard and codes to be used for structure, system and component qualification and QA	Nuclear graded components and materials		Nuclear graded components and materials	Conventional industrial components	Conventional industrial components
First level of defence (robust design)	Needed, with single failure applied to the active systems		Needed, with single failure applied to the active systems	Needed, with single failure applied to the active systems	Needed, with single failure applied to the active systems
Additional level of defence ‘a’	Needed, with single failure	Or equivalent Over design ‘a^+’ (one order of magnitude less in performance goal)	Needed, with single failure	Needed ‘a^-’	For operating purposes only
Additional level of defence ‘a’	Needed, with single failure		Not needed	Needed ‘a^-’	Not needed
Additional level of defence ‘b’	Administrative and/or operational procedures				
Required additional levels of defence	2a + b	$a^+ + b$	a + b	$2a^- + b$	$(a^-) + b$

beyond design basis events, high conservatism, and is demonstrably conservative.

(c) Some safety systems and barriers needed for levels of defence in depth higher than one (i.e. to prevent any 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 nonetheless guaranteed, particularly for prevention of any interaction with external event items that have been categorized (as is true for the containment, which has to be designed to withstand external events even though its function has to do with an internal accident).

(d) Barriers and levels should provide adequate reliability. Single failure criteria should be applied to safety systems. Passive barriers may represent exceptions in the sense explained in Ref. [4]. Special attention should be given to external events with respect to both common mode effects on structures, systems and components in the same facility and on different facilities at the same site.[9] Provisions for implementation of this criterion are provided in Ref. [19].

(e) Beyond design basis capacity for external events is usually specified in the design of the plant. 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 at least achieve a safe shutdown state.

(f) 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 and hindrance of access by rescue teams, special attention has to be paid to analysis of the implementation of the emergency procedures during and after an external event.

(g) Hazard category 4 facilities require only a robust design and the implementation of emergency planning.

2.9. RE-EVALUATION ISSUES

For re-evaluation of an existing facility in relation to external events, the basic approach set forth in this publication for the design of a new research reactor may be applied in its entirety. However, there is a lack of general agreement among Member States concerning some aspects of the procedures. These are identified below and complemented with a suggestion for consistent use.

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

(a) *Definition of the external event hazard:* In many cases a re-evaluation of the external event hazard triggers the re-evaluation process carried out at the request of the regulatory body. The external event return period has to be estimated to allow use of the procedure proposed in this publication.
(b) *Definition of the external event categories:* A reduced set of structures, systems and components to be categorized may be identified, usually associated with only one safe shutdown path, with redundancy. Therefore, some emergency systems needed to mitigate the effects of sequences initiated by internal accidents may not be categorized for external events.
(c) *Definition of the performance goal:* For re-evaluation of an existing facility, a performance goal a factor between 2 and 10 higher than that for a new design is usually accepted.
(d) *Reference plant status:* For re-evaluation, a limitation on the operational status is usually allowed (i.e. only normal operation, no consideration for outage or fuel loading).
(e) *Material capacities:* Actual capacities of the materials, which include ageing randomness effects, are allowed for re-evaluation where the design usually refers to specified minimum or code values.

2.10. PERIODIC SAFETY REVIEW

According to Ref. [1], a periodic safety review should be conducted to assess whether or not the safety performance of the research reactor meets the applicable safety requirements, to account for external event hazard changes and research reactor configuration changes. The method discussed in Section 4 for the overall safety review may be used to this end.

3. GENERAL APPROACH TO SITING AND DESIGN

Two ideal approaches may be used for siting and design of a nuclear installation, either a full deterministic approach or a full probabilistic approach. However, both approaches have drawbacks:

(a) The full deterministic approach does not include risk evaluation. A full deterministic approach would make ‘grading’ according to the hazard posed by the facility to the environment difficult and would necessitate the use of a very high degree of conservatism with no clear quantification of the overall safety margin.
(b) The full probabilistic approach is affected by great uncertainties in the definition of the design basis parameters and their statistical distributions. This is why PSA is not recognized as a reliable basis for design (it has only a confirmatory role) but is extensively used in external event hazard evaluation.
(c) Analysis of the dispersion of radioactive material, which is needed to check the safety objectives in both approaches, is affected by uncertainties, mainly related to the definition of the source term and the simulation of the leak path.
(d) The feedback from PSA and radioactive dispersion analysis to the design may not be straightforward and therefore a return loop for resultant optimization may be difficult.

In conclusion, the engineering practice for research reactor design suggests that neither approach may be completely reliable nor effective, and therefore the following mixed approach is proposed, in agreement with Ref. [1]. This proposal is more in line with current practice in Member States, where a simplified risk informed probabilistic approach is used to clarify the use of a more code oriented deterministic design.

(1) The basic safety objectives are defined in terms of a probabilistic target for radiological doses to workers, the public and the environment (see Section 2.1);
(2) The external event hazard is evaluated on a probabilistic basis;
(3) The component fragilities are evaluated on a probabilistic basis, but a preliminary screening of the high confidence of low probability of failure (HCLP$_F$) value may turn their evaluation into a simplified equivalent deterministic procedure;
(4) The number of levels in the defence in depth framework is selected deterministically according to the hazard classification;
(5) The item classification is carried out on the basis of unmitigated release following a failure;
(6) Both the site parameter evaluation and the design are carried out in a deterministic manner with some conservatism;

(7) Level 1, 2 and 3 PSAs[10] and an analysis of the dispersion of radioactive material are carried out only at the end of the design, as a final confirmatory assessment.

The proposed sequence relies on some degree of conservatism in classification and design in order to avoid any further iteration on the design as a consequence of the radioactive dispersion analysis. The approach is more straightforward, even if it relies on engineering practice in the choice of the conservatism level to avoid loop back from the radiation doses to siting and design of the facility.

This methodological approach is also accompanied by measures to control the safety margin embedded in the deterministic procedures for site investigation and design and the level of conservatism which is meant to compensate for reduced investigative effort, simplified design methodologies, reduced long term monitoring, etc.

The safety margin and conservatism (or robustness) are validated by the engineering experience and are driven by the design class method and by a series of deterministic assumptions at all phases of the siting and design process. Details of the multi-step approach are provided in Fig. 1.

Step 1: Initial categorization of the hazard posed by the facility to the environment, the public and workers in case of an 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 both the radioactive inventory and the installed power. Final categorization is a function of excessive, unmitigated radioactive release to the public and the environment (category 1) or workers (category 2). The hazard category defines the need for:

— The level of defence in depth to be applied;
— The level of detail in the safety analysis report of the facility [2];
— The level of QA to be applied to materials, siting/design/construction/ surveillance activities and documentation [7, 12];

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[10] Three levels of PSA are generally recognized: level 1 comprises the assessment of plant failures leading to the determination of core damage frequency; level 2 includes the assessment of containment response leading, together with level 1 results, to the determination of containment release frequencies; level 3 includes the assessment of off-site consequences leading, together with the results of the level 2 analysis, to estimates of risks to the public.

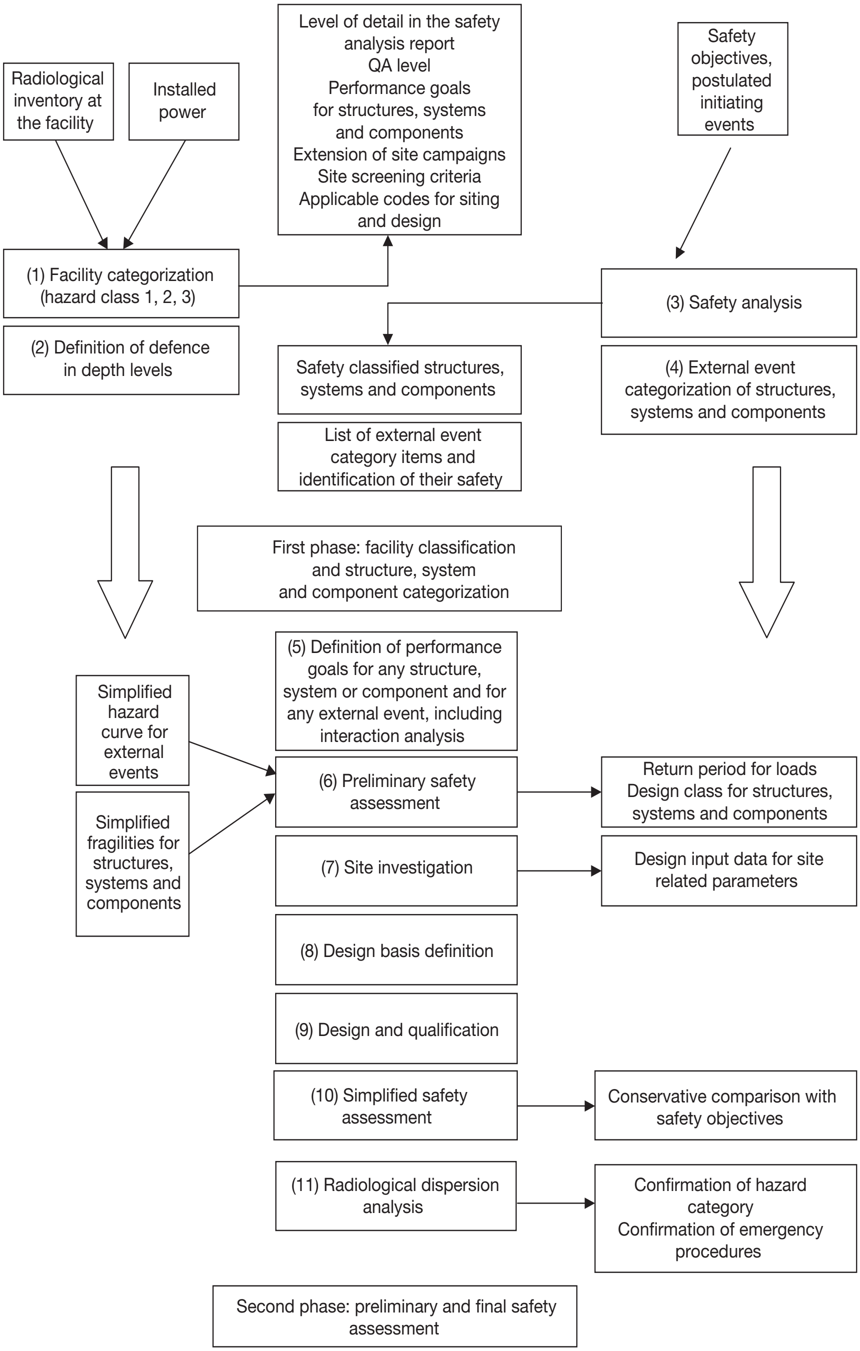

FIG. 1. Flow chart of the design safety analysis.

— The extent of plant-specific site investigation requirements, not needed for the lowest hazard category of facilities, while conservative, simplified approaches are not acceptable for category 1 facilities;
— The applicability of site screening criteria — for the lowest 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 structures, systems and components is allowed only for the lowest category.

Step 2: The safety classification of structures, systems and components reflects the internal postulated events and external events as set forth in the safety analysis of the plant (box (3) of Fig. 1). The definition of the defence in depth levels and barriers [2], the application of the single failure criterion and the assessment of the potential for common cause failures are identified in box (2) of Fig. 1 [19], bearing in mind the categorization of the facility. Next is the evaluation of the need for emergency procedures, both on and off the site. This is followed by identification of the internal events to be considered as a consequence of an external event or as contemporaneous to an external event, and therefore of the safety functions to be maintained in case of an external event (e.g. cooling of radioactive material, reactivity control, confinement).

Step 3: External event categorization (external event category) of structures, systems and components (box (4) of Fig. 1) includes identification of the safety related structures, systems and components (step 2). This categorization is affected by external events and evaluation of the radiological hazard posed to the environment, the public and workers from a failure that has not been mitigated, with the plant in any possible status (normal operation, accident). The outcome is realization of the performance goals (median values) and the associated technical requirements (behaviour limits for structures, structural or leaktight integrity, operability of equipment and components) for any structures, systems and components and for any external event (box (5) of Fig. 1). Simplified deterministic safety criteria for the systems, which mitigate external events, may be specified at this point (redundancy, diversity, quality, robustness).

Step 4: Defining the site-specific hazard level and the design class for any structure, system or component to be used in the external event design basis is subject to the performance goals assigned to any structure, system or component (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 structures, systems and components that have been categorized for external events. Simplified tables are also suggested for ease and speed of reading.

Step 5: Evaluation of the design basis reflects the hazard level defined in step 4 (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 should be carried out according to the requirements defined in step 1 (box (7) of Fig. 1).

Step 6: Design and/or qualification of structures, systems and components that have been categorized for external events reflects the design class identified in step 4 and the design basis developed in step 5 (box (9) of Fig. 1). The methodologies to be used for the design and qualification can be selected according to the facility categorization developed in step 1 (see Section 7).

Step 7: Final safety assessment of the plant and evaluation of the probability failure of the structures, systems and components that have been categorized for external events are based on the actual hazard and design methodologies used in the design/qualification (box (10) of Fig. 1). This step aims at fine tuning the engineering safety features to ensure that any structure, system or component can provide the required safety function with the required reliability (Section 4). This step replaces a full scope PSA with simplified probabilistic methodologies.

Step 8: In the analysis of the dispersion of radioactive material (box (11) of Fig. 1) the source term is selected according to the assumptions made in step 2, in terms of the functions to be maintained during an external event. The radiological doses to the environment, the public and workers in case of an external event scenario 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, respectively. In this step, the final requirements for the containment or confinement and emergency procedures are developed for accident mitigation (Section 8).

For re-evaluation of an existing facility, the approach is expected to be the same. However, major modifications in the technical details can be made 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 extent of the site investigation procedures, in the design/qualification methodologies and in the simulation of the radioactive dispersion into the environment. The use of simplified methodologies has to be adequately documented and agreed with the regulatory body. The following sections give details of the implementation of the steps defined in this section, while the appendices provide sample values and examples of application.

4. PRELIMINARY AND FINAL SAFETY ASSESSMENT

4.1. GENERAL

This section deals with the evaluation of the overall safety of a facility in relation to the basic safety objectives set forth in Sections 2.2 and 2.3. With reference to the design tasks discussed in Section 3, the proposed approach to safety assessment may be used in two steps, namely:

(1) In the selection of the siting and design procedures in relation to the performance goal assigned to structures, systems and components (preliminary evaluation) (boxes (5) and (6) of Fig. 1);
(2) In the final safety evaluation of structures, systems and components of the facility at the end of the design/qualification process (final evaluation) (box (10) of Fig. 1).

For re-evaluation of an existing facility, the proposed methodology for safety assessment may be used to confirm the adequacy of the design and to account for external event hazard modifications or research reactor configuration changes after the design and construction periods. In general, due to the complexity of a research reactor's processes and systems, the overall plant safety assessment is done by means of a PSA. The safety evaluation can use either the success path or the accident sequence method. However, when the accident scenarios corresponding to external initiating events and the relevant shutdown paths are easily identified, simpler methods than a PSA can be used in the evaluation of the overall risk associated with a research reactor.

It is preferable to evaluate the external event hazard on a probabilistic basis. The frequency of occurrence of the parameters describing the severity of the external hazard (such as earthquake ground acceleration, wind speed, water elevation) is estimated by probabilistic methods. Statistical parameters used for extreme events include return period and annual probability of exceedance. The hazard from other 'rare' external events such as accidental aircraft crashes or explosions reflects the frequency of occurrence of an event with postulated characteristics (quantity of explosive material, weight and velocity of the missile, etc.), as proper statistics may not be available for the area of interest. Performance goals depend on the external event categorization as defined in Section 2. For practical use they can be approximated by deriving the product (for continuous hazard levels it can be the convolution) between the annual probability of exceedance of an external event and the P_F induced by that specific external event. Probability values for performance goals apply individually to each external event. Hence, the performance goal established for a single event will generally be lower than the goal established for the total of all the events.

In general, different combinations of hazard level and design class can be used to achieve the performance goal. Once a suitable combination of hazard level and design class is selected, siting, design and qualification of structures, systems and components can take place, as in the following sequence:

(1) Development of a site dependent median (if mean values are chosen, a target probability one order of magnitude lower can be chosen) external event hazard definition in terms of the probability of exceedance for the annual frequency;
(2) Evaluation of (structural or functional) demand associated with the external event hazard;
(3) Evaluation of the research reactor facility's response, dominant failure modes, as well as structure, system and component fragilities;
(4) Evaluation of the damage states for the facility or success paths associated with each external event;
(5) Evaluation of the uncertainties associated with external event demand;
(6) Comparison of the P_F for any structure, system or component with the assigned performance goal for each external event included in the design basis.

At the end, as confirmation that the basic safety objectives for the facility have been met, a safety assessment for the facility can be carried out based on the plant safety analysis, as described in the following. This assessment is expected to be included in the plant's safety analysis report. A simplified sequence for calculation of the P_F is shown in Fig. 2.

4.2. EVALUATION OF THE CAPACITIES OF STRUCTURES, SYSTEMS AND COMPONENTS

Based on the performance goal assigned to every structure, system and component by the facility and component categorization, an external event hazard level has to be selected for the design/re-evaluation process. The information available at the beginning of the design process usually includes neither a full scope hazard curve nor fragility curves for the structures, systems and components that still have to be designed/qualified. Therefore a simplified approach is needed for the preliminary selection of hazard level and design class.

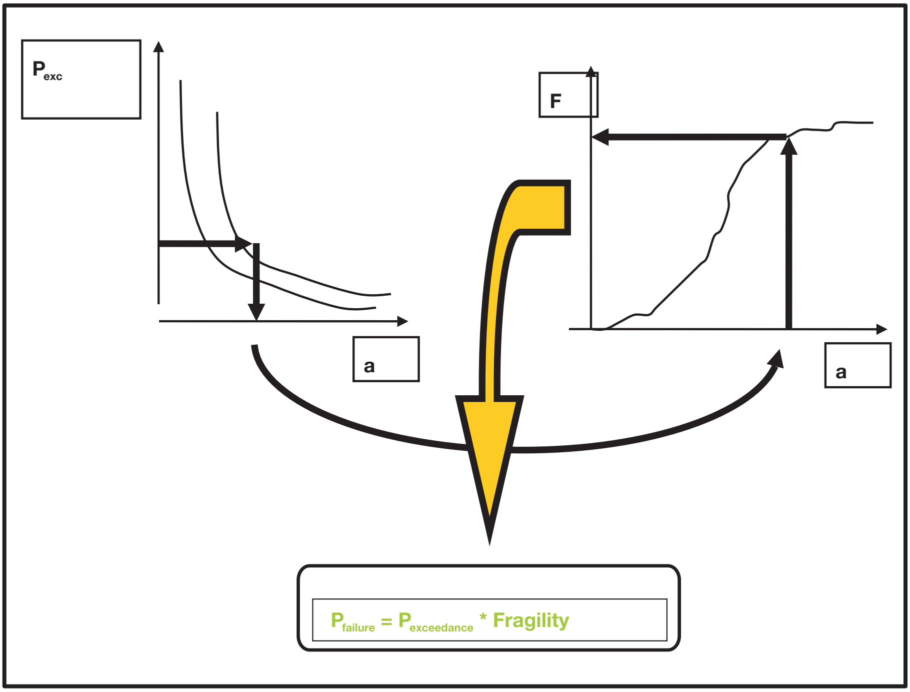

FIG. 2. Simplified sequence in the calculation of the P_F *(*P_{exc} *= yearly probability of exceedance of parameter a representing the external event intensity,* F *= the normalized cumulative* P_F *of a component when subjected to an external action of intensity* a*, also called 'fragility').*

The fragility of a structure, system or component is defined as the conditional P_F (unacceptable performance) versus the selected hazard parameter. It is usually acceptable to assume that the component failure is lognormally distributed. The fragility can be expressed in a simplified way either through its median capacity $C_{50\%}$ or $HCLP_F$[11] capacity and variability parameter β. For preliminary estimation, β = 0.3 can be used for the entire facility (Fig. 3).

The unacceptable performance or the P_F of a generic component is the result of the convolution of the median hazard curve and median fragility for any structure, system or component and for any external event:

[11] $HCLP_F$ is the acronym for a high confidence (95%) of a low probability of failure (5%), or 50% confidence with a probability of failure of 1%.

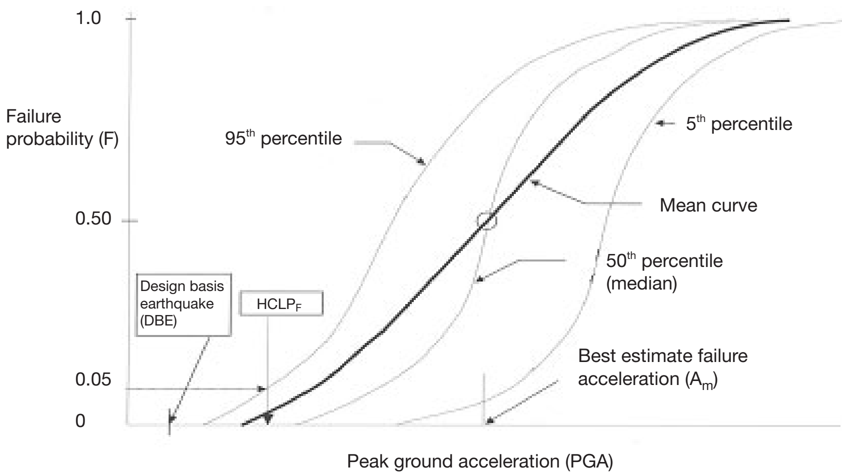

FIG. 3. Schematic representation of a fragility curve (for seismic fragility).

$$P_F = \int_0^{\infty} H(a)\left(\frac{\mathrm{d}P_F(a)}{da}\right)da \tag{1}$$

where H is the hazard curve, $P_F(a)$ is the fragility and a is the selected hazard parameter.

If the hazard is defined by a single value describing the frequency of occurrence of the external event with postulated intensity parameters (aircraft crash, explosions, etc.), Eq. (1) may be approximated by:

$$P_F = \text{Hazard level} \times \text{component fragility (at the same level)} \tag{2}$$

Equation (1) could first be used to check if the external event hazard and the component fragility are consistent with the performance goal. This means that the calculated total P_F corresponding to an accident sequence (damage state) as result of the loads induced by the external event has to be less than the performance goal:

$$P_F \leq \text{performance goal} \tag{3}$$

An example of the estimation of the total P_F associated with an earthquake for both an individual structure, system or component and for the whole facility is presented in Appendix II. Fragility evaluation can follow one of the following methods:

(a) Evaluation based on earthquake experience data;
(b) Evaluation based on generic seismic testing data;
(c) Equipment-specific qualification.

Further information on the evaluation of fragility can be found in the references. Special care has to be taken as to the method selected, as each method requires engineering judgement and can significantly affect the reliability of the final result.

In most cases a simplified fragility evaluation is sufficient to estimate the P_F. The calculation of a single value $HCLP_F$ is used to screen out rugged structures, systems and components. This is dealt with further in Appendix II.

4.3. PRELIMINARY ESTIMATION OF THE HAZARD LEVEL AND DESIGN CLASS CONSISTENT WITH THE PERFORMANCE GOAL

For a preliminary selection of a convenient pair of hazard level and design class values for any structure, system or component, given its performance goal, the following steps can be followed. In other words, the convolution of the external event hazard curve and the fragility curve of the component (evaluated according to Eq. (1)) must give a total P_F that is less than the performance goal (evaluated according to Eq. (3)). On such a basis, the combination of hazard level and design criteria can be defined before the design process is started, with reasonable confidence that the final design will meet the safety objectives.

Step 1: Define an approximate external event hazard curve applicable for the site. This step requires a site-specific investigation as described in Sections 5 and 6.

Step 2: Use Tables 4 and 5 to select the applicable performance goal for the hazard category of the research reactor facility.

Step 3: Use Table 6 to select the required fragility of any structure, system or component for a first trial on the basis of the applicable design class and the associated hazard level.

Step 4: Check if the candidate median capacity/fragility convolved with the hazard curve produces a total P_F that is less than the performance goal (check Eq. (3)).

Step 5: Perform iterations, modifying fragility parameters until Eq. (3) is checked. The result will be the target fragility (target fragility represents a point on the fragility curve) and the corresponding design class.

In the selection of the design class, a trade-off has to be reached between a combination of a low probability hazard with a high P_F of the component and a combination of a high probability hazard with a low P_F. Many considerations can influence this choice, namely:

(a) Low probability hazards require sophisticated extrapolation methods and are affected by high levels of uncertainty. However, they provide a physical quantification of the hazard curve behaviour, particularly when the phenomena have a non-stationary trend.

(b) Low safety factors (high P_F) allow the application of conventional codes for structure, system and component design and qualification, with great cost savings and high reliability. However, this choice implies the use of low probability hazards with the drawbacks highlighted above.

In conclusion, any combination that meets the performance goal is acceptable, but a site specific global evaluation may indicate that one approach or the other will obtain the most reliable results for the final design. One practical approach consists of carrying out a preliminary analysis of the fragilities of the safety related structures, systems and components and making a final decision after a precise evaluation of the effort required for design and qualification using either option.

4.4. FINAL EVALUATION OF PLANT SAFETY

After completion of the design process, the failure probability both at the structure, system and component level and at the facility level is evaluated and an overall performance evaluation carried out. At the end of the design process, the external event hazard is available in detail. The performance evaluation in relation to external events includes the following steps:

Step 1: Selection of the success path and/or accident sequences to be considered in the safety analysis.

Step 2: Evaluation of individual structure, system and component fragilities on the basis of the design documentation. A simplified evaluation can be carried out on the basis of the design review and design code analysis. The outcome of this evaluation is the median capacity and associated variability. The capacity variability can be evaluated using generic data and engineering judgement. The median capacity and total variability define the fragility curve for individual structures, systems and components.

Step 3: Use Eq. (1) to calculate the failure probability and compare P_F with the performance goal (check Eq. (3)) for any structure, system or component. Combine individual structure, system and component fragilities to derive the overall fragility for the research reactor (an example of seismic fragility is given in Appendix 2). Weak links can be identified and design changes recommended.

Step 4: Calculate the release arising from structure, system or component failures consistent with the safety analysis. Evaluate the global probability of release at the facility level, for comparison with basic safety requirements.

5. SITE INVESTIGATION

5.1. GENERAL

The investigation of the site should cover all disciplines affecting site safety, namely geology, seismology, geotechnics, hydrology, meteorology, marine environment, human development plans, industrial installations, communications, and traffic on waterways, by rail, road and air [1].

The evaluation of the site hazard for external events can in general follow the IAEA recommendations for siting and design of NPPs. Many IAEA publications address siting and design of NPPs in relation to external events [4, 6, 9, 15, 20–25]. However, the site selection process can consider more restrictive site exclusion criteria than those described in general terms in Ref. [1], as a compromise with the investment required for the design, construction and operation of the facility. In this sense, some events which are difficult or expensive to protect the facility from may be used as site screening criteria, such as accidental aircraft crash (low probability, thick shielding and special equipment qualification would be required at facilities without a confinement), accidental explosions (blast resistant structures, would be required), flooding (site protection engineering structures would have to be built and maintained), etc. In fact, in low power research reactors, the internal accident scenarios usually do not entail great demand on the structures as they would for NPPs where, for example, a containment is normally part of the design. Therefore, protection from external events would add difficult to fulfil requirements to the design which might be incompatible with a rational design approach.

A graded approach to the siting and design of research rectors may be used in accordance with their hazard classification [1]. In particular:

(a) The safety margin in the design has to be easily proven, even in cases where codes are applied that differ from the code for NPP design;
(b) An adequate level of conservatism has to be guaranteed to compensate for reduced site hazard analysis, site investigation campaigns and simplified analysis methods, in accordance with the main objectives of the present publication.

The following sections provide information on how such grading can be applied to the siting phase. Additional information is available in Ref. [5].

5.2. EVALUATION OF SITE CHARACTERISTICS

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

Hazard category 1:

The extent of the survey is based on regional studies, with the requirements for data quality and quantity set forth in Refs [1, 6]. The investigation is site-specific and covers an area within an approximately 50 km radius of the site. This area may be extended to compensate for lack of data in the time record (see Section 6.2). It may be smaller if the area is not populated and possible causes of events do not exist. The record length to be considered for site-specific evaluation is chosen with reference to the return period selected for the design basis. Appropriate extrapolation techniques have to be applied and validated. The projected growth of population around the site during the lifetime of the facility is evaluated.

Hazard category 2:

The site-specific investigation confirms the hazards defined in the national building codes at the regional level. The investigation is carried out within a 20 km radius of the site. This radius may be smaller if the area is not populated and possible causes of events do not exist. Data record length and extrapolation methods are the same as for hazard category 1. The facility's lifetime is compatible with the projected population growth around the site.

Hazard category 3:

The survey's extent will be based on expert judgement to be used as confirmation of the hazard proposed in the national building codes. The following sections deal with general considerations on site evaluation for hazard category 1 and 2 facilities. For hazard category 3 facilities, site evaluation may correspond to the national practice for a conventional risk facility.

5.2.1. Geography

The following is generally evaluated when the location of a research reactor is being considered:

(a) Research reactor location, size of the site;
(b) Relative elevation in relation to surrounding areas;
(c) Location of nearby public places such as residential areas (for both permanent and temporary stays), industrial areas, commercial areas, educational areas, military areas, sports and recreational areas, airports, and harbours, with their sizes and relative distances;
(d) Location of transport routes;
(e) Location of emergency facilities and evacuation routes relative to the research reactor (fire brigade, hospital, police);
(f) Location of electrical and communication lines, gas and oil pipes, main drinking water pipes;
(g) Environmental conditions (forest, pastures, cultivated land);
(h) Vegetation (species and density);
(i) Fauna (species and density).

5.2.2. Demography

For the geographical areas identified in this section, the demography evaluation involves analysis of the following:

(a) The present population in the research reactor facility;
(b) The projected growth of population in the research reactor facility as a result of future expansion projects;
(c) The current population in the vicinity of the research reactor (both stable and transient);
(d) The projected growth of population in the vicinity of the research reactor (including all categories of population);

(e) Daily (work hours), weekly and yearly population movements resulting from the activities of the research reactor;
(f) Daily (work hours), weekly and yearly population movements resulting from activities in the vicinity of the research reactor;
(g) Age distribution (if a major deviation from average national percentages is expected).

5.2.3. Nearby facilities

A detailed analysis can include:

(a) The types of activities related to the use and/or transportation of hazardous materials;
(b) Identification of probable accidents in the facilities and the extent of the affected areas;
(c) Identification of storage facilities for hazardous materials;
(d) The location of water and food storage facilities;
(e) The location of potential sources of accident conditions (other hazardous facilities, dams);
(f) Identification of military areas with explosive inventory and target practice activities with explosives or artillery.

5.2.4. Transportation routes

Nearby transport accidents may affect the research reactor through overpressure following explosions, fires, missile impact, or attack with toxic materials. The analysis can include the following:

(a) Air transport:
   - Aircraft traffic and proximity to nearby (15 km) airports, if any, types and numbers of aircraft (civil, military, sizes, movements);
   - Restricted airspace;
   - Flight patterns;
   - Airport availability in case of accident.
(b) Road transportation:
   - Traffic volumes as a function of time;
   - Identification of roads used for the transport of hazardous materials;
   - Transport routes for chemical and fuel materials inside and outside the research reactor facility;
   - Average and maximum vehicle size.

(c) Rail and water transportation:
   - Traffic volumes;
   - Identification of the transport of hazardous materials;
   - Sizes and traffic of trains and vessels in nearby train stations and harbours, and the availability of these hubs in case of accident.

5.2.5. Hazards from site services

The analysis of site services can include those facilities that could pose a hazard to the research reactor (such as the failure of a large gas pipeline) and also the availability of those services necessary to the research reactor's safety systems (such as water supply).

The survey can include the following:

(a) The water supply system (inside and outside the facility):
   - Potable water system;
   - Fire system;
   - Wastewater system;
   - Water drainage system;
   - Dedicated water supply to the research reactor (cooling towers, fire protection).

(b) The electricity supply:
   - Location of external electrical lines and distribution facilities;
   - Analysis of the consequences of failure of the electrical distribution in case of an accident at the research reactor;
   - A dedicated power supply to the research reactor.

(c) Others:
   - Gas and/or oil lines and storage facilities near the site;
   - Communications lines and towers.

5.2.6. Geological and geotechnical data

Geological and geotechnical investigations at the site are typically carried out with the following objectives:

(a) Assessment of the possible geological or geotechnical hazard from surface rupture due to faulting, liquefaction, collapse and slope stability;
(b) Evaluation of the soil characteristics to achieve a reasonable soil, for example for seismic wave propagation studies;
(c) Evaluation of geotechnical parameters to be used in the design of the foundation and seismic design of the facility.

The number of geotechnical investigations to be carried out can be based on the potential consequences of the site related hazards. For all classes of research reactors, soil characterization may involve borehole drillings in sufficient number and depth, depending on soil conditions. At least one borehole should be drilled for every safety related building at the site. However, drilling may not be necessary for competent rock sites where the rock formation continues to a sufficient depth.

It is recommended that the soil profile be physically identified (e.g. through drilling) to a depth equal to at least one half of the maximum foundation depth. The depth to firm bearing strata may also be determined using boreholes and/or geophysical methods.

For hazard category 1 and 2 research reactors, dynamic characteristics of the soil profile may be determined by means of cross holes or geophysical methods up to the base rock or to a depth of at least 1.5 times the maximum foundation depth. The dynamic characteristics of the soil material for each layer may include:

(1) The type of material;
(2) The layer thickness;
(3) The shear wave velocity (low strain);
(4) The density;
(5) The Poisson ratio;
(6) Material damping (low strain);
(7) Material characteristic curves g–γ (shear modulus–strain) and d–γ (damping ratio–strain).

Curves g–γ and d–γ are necessary for soft soil conditions and for hazard category 1 and 2 research reactors. The soil characteristic curves can be determined by laboratory test or by generic curves which are available in the technical literature [26]. For hazard category 3 research reactors, standard curves can be used from the technical literature for the identified soil conditions.

In parallel with the foundation investigation and the use of available geological or geotechnical data, studies may be carried out at the site to assess possible hazards which could result in permanent soil deformation (including surface rupture, liquefaction, collapse, slope instability). If these investigations indicate potential consequences from such hazards, further studies may be necessary or the site has to be rejected.

5.2.7. Seismology

An understanding of the regional tectonics should be developed as a basis for site screening (in case of fault ruptures) and for detailed site evaluation. Tectonics and seismological data should be correlated with the geological database, both at the regional level and the site area, in order to gain a complete understanding of potential source mechanisms.

A seismological catalogue has to be recovered for the region. In addition, complementary information should be gathered on local faults (in the vicinity of the site), to be cross-checked with palaeoseismological observations. This evidence is particularly important in contractional (reverse faulting) regions, where it is common to have no background events between major earthquakes and clustered activity is recognized to be a common phenomenon in plate interior regions. Thus, a palaeoseismological approach (which is more often used in plate boundary regions) is even more critical to accurate seismic evaluation in plate interior regions. In case of difficulties in the characterization of the activity of local faults, a micro-earthquake network should be installed at the site and operated for some years.

5.2.8. Meteorology

The meteorological survey covers the evaluation of:

(a) Wind speed and dominant directions;
(b) Severe winds (tornadoes, hurricanes, typhoons, cyclones, etc.);
(c) Normal and extreme precipitation (snow, rain, ice);
(d) Barometric pressure;
(e) Evaporation;
(f) Atmospheric dispersion;
(g) Temperature (normal and extreme);
(h) Flooding (storm surge) induced by extreme wind;
(i) Lightning.

Local precipitation in the form of rain, snow or ice can have a direct bearing on the design of safety related structures, particularly roofs, external cables and towers. The following data are usually collected or extrapolated:

— Maximum recorded precipitation in 6 and 24 hour periods;
— Projections of the amounts of precipitation for 50, 100 and 1000 year return periods.

Further details about meteorological induced events are discussed in Ref. [22].

5.2.9. Hydrology

Surface hydrology is relevant to safety because it may represent a pathway for the dispersion of radioactive material and is also a possible source of flooding. It may affect the foundation bearing capacity as well. The survey can include:

(a) Surface hydrology:
  — Location of rivers and lakes, elevation, volume, flow rates and drainage ratio;
  — Relative location of the research reactor in relation to surface water;
  — Locations of dams, with their capacity and probable drainage in case of failure;
  — History of floods;
  — Marine environment.

(b) Groundwater hydrology:
  — Evaluation of water level, underground flow and flow directions;
  — Location of aquifer systems;
  — Evaluation of soil permeability.

Further details about surface hydrology and induced events are discussed in Ref. [22].

For rivers that could be potential sources of site flooding, the potential for flooding can be characterized by collecting the following information:

(1) The location and elevation of the rivers closest to the site;
(2) Historical records of stream flow data (maximum yearly peak discharge and stage elevation) with recording location;
(3) The maximum flood level that may be expected from a combination of the most critical meteorological and hydrological conditions;
(4) Characterization of the geometric and hydraulic properties of the channel closest to the site. The geometric properties of the channel include Manning's roughness coefficient and top width elevation tables for cross-sections, and stream bed slope;
(5) The presence of bridges or natural river flow constrictions which could cause flooding due to ice or debris jams.

In the case of rivers for which no peak discharge records are available, the following information can be gathered:

(i) Characteristics of the watershed basins;
(ii) Properties of the drainage basins, including topographic maps of the basin and land cover maps.

5.2.10. Baseline environmental radioactivity

The purpose of establishing the baseline environmental radioactivity is to allow comparison with future environmental radioactivity surveys at the site. The survey may be developed on the basis of air, water, soil and biological samples.

The area to be surveyed can be the local facility site for hazard category 3 and within a 20 km radius for hazard category 2. For hazard category 1, an area within a 50 km radius of the site is suggested unless highly populated areas are present near the facility. In this case, the survey can include the areas likely to be affected by a release of radioactive material.

6. EVALUATION OF EXTERNAL HAZARDS

6.1. HAZARD SCREENING FOR EXTERNAL EVENTS

A preliminary screening of the external events to be considered in the design of a research reactor can be done on the basis of a detailed hazard categorization. For facilities in the lowest category (hazard category 3) some extreme scenarios (aircraft crash, blast loads, tornado) may be screened out either by virtue of the selection of the site or on the basis of a low probability of occurrence.

The screening process can then consider the potential for off-site and on-site consequences induced by the external event. Table 8 presents possible consequences of external events to be analysed for screening.

TABLE 8. POSSIBLE CONSEQUENCES OF EXTERNAL EVENTS ON RESEARCH REACTORS

External event	Off-site damage	Safety function affected	Severity of potential damage and warning
Geological and geotechnical hazard	Loss of off-site power and other utilities Loss of access roads	Structural stability and integrity Reactor could be critical or in shutdown state	High Sudden action without warning
Earthquake	Loss of off-site power, communications and other utilities	Reactivity control Heat removal systems Confinement systems Structure, system and component integrity and operability Reactor could be critical or in shutdown state	High No warning
Liquefaction (produced by earthquakes)	Loss of off-site power and other utilities Loss of access roads	Structural stability Reactor could be critical or in shutdown state	High No warning
Landslides and avalanches	Loss of off-site power and other utilities Loss of access roads	Structural stability Reactor could be critical or in shutdown state	High No warning
Extreme wind	Loss of off-site power	Partial structural integrity Reactor in shutdown state	Moderate Monitoring system warning
Extreme rain, snow and ice	Loss of off-site power	Structural integrity Reactor in shutdown state	Low Monitoring system warning
Flooding	Loss of off-site power and other utilities	Reactivity control Confinement system Reactor in shutdown state	High to moderate Monitoring system warning
Abrasive dust and sandstorm		Ventilation system Reactor in shutdown state	Low Monitoring system warning
Lightning	Loss of off-site power	Reactor could be critical or in shutdown state	Low No warning

TABLE 8. POSSIBLE CONSEQUENCES OF EXTERNAL EVENTS ON RESEARCH REACTORS (cont.)

External event	Off-site damage	Safety function affected	Severity of potential damage and warning
External fire		Ventilation system Reactor in shutdown state	Low Monitoring system warning
External off-site explosion	Loss of access roads, off-site power	Structural integrity Reactor could be critical or in shutdown state	Moderate No warning
Aircraft crash	Local damage	Structure, system and component integrity and operability Reactor could be critical or in shutdown state	High to moderate No warning
Release of hazardous liquids/gas from off-site and on-site storage	Affects the personnel of the research reactor	Reactor could be critical or in shutdown state	Low Monitoring system warning
Electromagnetic interference from off and on the site		Reactivity control Reactor could be critical or in shutdown state	Low Monitoring system warning

6.2. DESIGN BASIS FOR EXTERNAL EVENTS

6.2.1. Earthquakes

For hazard category 1 and 2 facilities, if instrument data, location of seismogenic sources and zones, attenuation relationships and maps of hazards within the region are available for the region surrounding the site, site-specific design response spectra (including site effects) can be generated either by using the envelope of response spectra (for 5% damping) calculated from recorded data (then extrapolated to the required return period) or by using hazard maps that have been developed for this purpose using such data (applying an appropriate safety margin).

It is advisable to use the IAEA Safety Guides on the siting of NPPs [6, 9] for this, particularly for evaluating the sufficiency and reliability of available data. This approach is equivalent to the application of the safety margin required for NPP siting. Appropriate simplifications of this conservative approach, if approved by the competent regulatory body, might lead to a reduction of such a safety margin, for example according to the facility hazard categorization described in Section 2, through reduction of the reference return period.

In the event that instrument data are not available for the region surrounding the existing or planned facility, the design basis ground motion can be conservatively evaluated on the basis of the maximum historical intensity in an area broader than the region. For this evaluation, the following procedures might be applied for hazard category 1 and 2 facilities, provided that the region surrounding the site shows a reasonable uniformity from the seismotectonic point of view:

(a) Consider a reference zone within a radius of at least 100 kilometres of the site. A larger radius of up to 200 kilometres can be considered when data are lacking and there is low seismicity.
(b) Use available publications and catalogues to establish the maximum observed intensity in this area and apply this to the site. The information should cover as much historical data as possible, but a minimum of 100 years [8].

In the case of intraplate regions, a model for diffuse seismicity should be developed to complement the extrapolation of historical strong motion data. The methods described in Ref. [9] are recommended. Hazard category 3 facilities may be assessed using national seismic codes confirmed by local evidence.

A minimum value of design free-field acceleration for firm bearing strata can be assigned.[12] These assigned values have to be compatible with the seismic provisions of the national building design codes currently adopted by several countries. However, intermediate design accelerations can be assigned on the basis of detailed analyses of the data.

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

6.2.2. Potential for seismic liquefaction

The geotechnical investigation aims at assessing the potential consequences of any liquefaction and soil strength loss, including estimation of differential settlement, lateral movement or reduction in foundation soil bearing capacity, and can discuss mitigation measures. Such measures can be considered in the design of the structure and can include, but are not limited to, ground stabilization, selection of appropriate foundation types and depths, selection of appropriate structural systems to accommodate anticipated displacement, or any combination of these measures.

The potential for liquefaction and soil strength loss can be evaluated for site PGAs, magnitudes, and source characteristics consistent with the design earthquake ground motions. The determination of PGAs is allowed on the basis of a site-specific study taking into account soil amplification effects or, in the absence of such a study, PGAs can be assumed to be equal to $S_{DS}/2.5$ where S_{DS} is the spectral peak acceleration. Procedures for evaluation of the liquefaction hazard are set forth in Refs [5, 25]. If these investigations indicate a high liquefaction hazard, either further studies are necessary or the site is rejected.

6.2.3. Extreme precipitation (snow, rain, ice)

For hazard category 1 and 2 research reactors, the design value for precipitation can be based on the probability of exceedance compatible with the performance goals assigned to structures, systems and components. 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. If no data are available, the load can be taken from the national building code for hazardous facilities, multiplied by a factor of 2 for safety class 1 structures, systems and components and by 1.5 for safety class 2 structures, systems and components. The national building code for hazardous facilities may be used for hazard category 3 research reactors and their structures, systems and components.

6.2.4. 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. Data from monitoring typically cover at least 10 continuous years of annual extreme wind speed records. The type of wind speed recorded over time has to be specified (e.g. average, 10 min peak, 3 s gust, etc.) so that a proper gust factor can be defined in converting wind velocities to wind pressure loads. Anemometers located in flat, open terrain can be used to record wind speeds. The elevations at which wind speeds are recorded can be 10 m above ground. If the last two conditions are not met, the recorded wind speeds can be corrected using accepted wind boundary layer conversion methods. Data from on-site stations for which fewer than ten years of records exist can be used if there are a sufficient number of historical records from nearby stations within the same topographic and wind region (stations close to but separated from the site by mountain ranges do not qualify). Either way, the extreme value may not be lower than the value provided by the national building code.

For hazard category 1 and 2 research reactors, the design basis wind can be evaluated on the basis of the selected probability of exceedance for the external event hazard according to the performance goal assigned to their safety class 1 and 2 structures, systems and components. For more sophisticated investigations and analysis, further guidance is provided in Ref. [22].

For hazard category 3 facilities, the design basis wind can be taken from the national building codes for hazardous facilities. National building codes typically give design basis wind velocities and pressure distributions, including variation with the height above ground and relative values with respect to the building geometry. These assumptions may be applied to the research reactor design provided that 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.

6.2.5. 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, at both river or coastal sites. If it is not possible to select dry sites, all safety related components must be constructed at an altitude above the reference level of a flood, which can be determined using the methods given below. Because of its special consequences to nuclear criticality and other consequences to electrical equipment, the presence of free and unwanted water in a nuclear facility has to be fully controlled and should preferably be excluded.

6.2.5.1. River sites

The boundaries of the region to be investigated for the 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 carried out 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) By means of empirical formulas which have been developed for various parts of the world, giving a relationship between drainage basin parameters and potential flood characteristics;
(2) By use of extrapolated hazard curves, based on series of maximum annual flows, which can be used for evaluating the reference flood; these hazard curves can be derived from the available data, taking into account random components, trends and jumps. If properly used, this method may allow a reasonable evaluation of a reference flood.

Results evaluated for hazard category 1 and 2 research reactors should not be less than any recorded historical occurrence. Based on the reference flood flow, a reference level can be obtained with appropriate hydraulic formulas that 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 of the site can be evaluated by assuming contemporaneous failure of all dams on the same stream.

6.2.5.2. 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 subjected to tropical storm effects (typhoon, hurricane, cyclone) or if there is a history of tsunamis, historical data on the phenomena have to be collected. An analysis of available data can give a good indication of the maximum flood level at the site and, with an adequate margin, provides the minimum altitude for dry sites. Further guidance on a more sophisticated method is provided in Ref. [21].

If flooding of the site is not precluded, then the design water load suggested in applicable national building codes may be used for design purposes. In the absence of such codes, analytical models which include both hydrostatic and hydrodynamic loads on safety related structures, systems and components may be used.

6.2.6. Rotating wind

Tornado, hurricane, typhoon and cyclone winds are violently rotating winds which can reach speeds in excess of some hundreds of kilometres per hour. High probability rotating wind sites are those where rotating wind velocities exceed extreme straight winds at a 10^{-4}/a probability of exceedance. Moderate probability rotating wind sites are those where rotating wind velocities exceed extreme straight winds at the 10^{-5}/a probability of exceedance level. Low probability rotating wind sites are those where rotating winds exceed straight velocities at the 10^{-6}/a probability of exceedance level. Rotating winds can be excluded from the design basis if the rotating wind probability of exceedance is less than the probability of exceedance for the selected external event. Tornadoes do not have to be considered for sites with hazard category 3 research reactors unless they are included in national building code requirements. For hazard category 1 and 2 research reactor sites for which no site-specific, up to date probabilistic analysis has been performed for a tornado wind hazard, the following data can be collected for rotating wind striking within 300 kilometres of the site:

— Rotation track (latitude and longitude);
— Intensity;
— Length and width.

Particular consideration has to be given to evaluation of:

— The sudden pressure drop which accompanies the passage of the centre of a tornado;
— The impact of wind generated missiles on the facility's structures and equipment.

6.2.7. Wind-borne missiles

When wind-borne missiles are likely to affect the site, two types of missile have to be considered in design, penetrating and impacting. Penetrating missiles typically have relatively high velocities, 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 [27].

6.2.8. Accidental chemical explosions

The site should be located in an area where the effects of explosions are not significant. A complete study of surrounding industrial activity and transportation by road, river, sea, train or pipeline has to be made in order to identify the chemical nature, geographical position, quantity, frequency of occurrence, storage or transport conditions (eventually protection against explosion) of potential explosive material to be accounted for or not, depending on the protection in the building design. As far as the following approach presents significant safety margins, focalization effects may be ignored.

For fixed or mobile sources of explosions, or for sources of hazardous cloud, the distance from the source of an explosion can be evaluated deterministically or probabilistically with the method given in Ref. [20]. If it is not possible to locate the plant in an area where the risk is not significant, the plant should be protected against these events.

The design can follow 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:

(a) An equivalent TNT mass ratio is applied to the mass of explosive product and gives the equivalent mass of TNT for its explosive effects.
(b) A coefficient for gaseous conditions that defines the ratio of the total mass present in the storage or transport that is involved in the explosion, depending on storage or transport conditions. If a more rigorous estimate is not made, this ratio is taken to be equal to 20% for hydrocarbons.

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 carried out or the site should be rejected.

6.2.9. Accidental aircraft crash

The research reactor may be located in an area where the risk of aircraft crash is not significant. In agreement with the basic principles discussed in Ref. [20], two approaches are recommended:

(1) The safe distance from the airport can be evaluated with the formulas given in Ref. [20].
(2) A probabilistic approach can be used. The probability of an aircraft hitting sensitive parts of an installation is correlated with the size of the installation.

Obtaining fundamental data for aircraft crash protection calls for an extensive knowledge of air traffic in the vicinity of the facility. Basic studies can be undertaken to determine the following parameters:

(i) The presence of an airfield in the vicinity of the facility site;
(ii) The probability of impact per flight from statistical data in the entire country concerned or in the smallest possible area, including the site;
(iii) The number of flights per year;
(iv) The mass and impact characteristics of the different types of aircraft;
(v) The speed of the aircraft upon impact.

From all these parameters, for each aircraft category, a probability of aircraft impact per unit surface and per year can be derived. From its geometry, a virtual area of the facility is 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 is evaluated as the product of the probability of impact per unit surface and per year, multiplied by the virtual surface of the facility. The need for aircraft crash protection depends on the probability of a crash for each category. In general, if this probability is higher than 10^{-5}/a, the facility's design should consider the impact characteristics corresponding to their category.

6.2.10. Malevolent vehicle intrusion and explosion

In some Member States there is a need to evaluate the adequacy of the design of structures, systems and components of safety classes 1 to 3 in hazard category 1 to 3 facilities. This is usually done by installing physical barriers to vehicle intrusion, and providing sufficient stand-off distance to ensure the required performance. Reference [28] provides practical examples of this.

6.3. EVOLUTION OF HAZARDS OVER TIME

Sufficient margin in the design basis values can be included to accommodate evolution with time of the input data, or to take experience feedback into account. There is a balance between the extra cost linked to the overestimation of such parameters in the design phase and the hypothetical cost of future retrofitting of the research reactor to accommodate the evolution.

Evolution of air traffic has to be anticipated, as well as the evolution of infrastructure such as the introduction of dam equipment on an undammed river or highway construction in valleys, which then leads to modification of the flooding parameter. An anticipation of the evolution of human activity should also be taken into account in the design parameters.

7. DESIGN, QUALIFICATION AND RE-EVALUATION

7.1. GENERAL

The design and re-evaluation process for structures, systems and components of research reactors in relation to external events consists of the following steps:

(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 structure, system and component;
(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 with the given external event);
(d) Selection of acceptable design or re-evaluation approaches (for each structure, system and component and each external event), among the following:
    — Qualification by analysis: use of code based stress and strength analysis (‘A’ in Table 9);
    — Qualification by testing (‘T’ in Table 9);
    — Qualification by experience (‘E’ in Table 9);

- Qualification by special investigation when A, T or E are not applicable ('S' in Table 9), 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 structure, system and component 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 as described in Section 7.4;
   - Identification of load combinations to be considered as described in Section 7.5;
   - Qualification by analysis as described in Section 7.6, consisting of the following steps:
     - 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, as described in Section 7.7;
   - Qualification by experience, as described in Section 7.8.

7.2. SELECTION OF ACCEPTABLE DESIGN AND RE-EVALUATION APPROACHES

Table 9 summarizes the general methods, as defined in Section 7.1, for the selection of acceptable design and re-evaluation approaches for structures and equipment of research reactors and similar facilities in relation to external events.

7.3. SELECTION OF ACCEPTABLE CODES (STANDARDS)

According to the procedures followed for hazard evaluation, and in agreement with the design classification, the codes (standards) to be applied for design and re-evaluation can be selected in accordance with Tables 6 and 10. It has to be noted that the recommendations in Table 10 should be assessed against the values in Table 6.

TABLE 9. SUMMARY OF ACCEPTABLE DESIGN AND RE-EVALUATION APPROACHES FOR STRUCTURES, SYSTEMS AND COMPONENTS OF RESEARCH REACTORS AND SIMILAR FACILITIES IN RELATION TO EXTERNAL EVENTS

External event		Acceptable design and re-evaluation methods (as defined in Section 7.1)
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 sandstorms	S
	Landslides and avalanches	S
	Lightning	Design rules/provisions only
	Volcanic activity	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 above events as a result of a common initiating event	A, T, S and/or design rules/provisions

TABLE 10. SELECTION OF ACCEPTABLE CODES (STANDARDS) BASED ON THE DESIGN CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Design class of the structure, system or component	Codes (standards) for design	Codes (standards) for re-evaluation
1	Nuclear codes	Nuclear codes plus best up to date engineering knowledge and experience
2	Nuclear codes	Nuclear codes plus best up to date engineering knowledge and experience
	or	or
3	conventional non-nuclear codes	Conventional non-nuclear codes plus best up to date engineering knowledge and experience
4	Conventional non-nuclear codes	Usually not needed

7.4. SELECTION OF AN APPROPRIATE DESIGN AND RE-EVALUATION METHOD

The selection of an appropriate design and re-evaluation method can be based on a clear understanding of safety functions assigned to each structure, system and component, its potential failure modes and relevant acceptance criteria (e.g. integrity, stability, operability). Equipment and components whose active safety functions are required should be designed so as to ensure their operability during and/or after an external event. The design margin for structures, systems and components 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 codes or standards (see Table 10).

Deterministic methods are typically used for design and re-evaluation. 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 design procedure can be associated with some additional conservatism. Procedures to deal with it are available in Ref. [5]. For re-evaluation, more realistic and less conservative (based on the best current engineering knowledge) material strength and stiffness characteristics, damping values, inelastic energy absorption factors and structural modelling can be used. An 'easy fix' programme may also be implemented for upgrading already existing facilities with the aim of minimizing investment costs while maximizing the increase in safety margin. An adequate level of conservatism needs to be guaranteed for simplified design and re-evaluation methods. The designer is usually asked to demonstrate such conservatism.

Most of the available engineering procedures deal with the seismic qualification of structures, systems and components. Examples of qualification procedures in relation to other external events can be taken from the NPP 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 should be provided.

7.5. LOAD COMBINATIONS AND LOAD FACTORS

7.5.1. General considerations

The load combinations for external events and the corresponding load factors can be taken from the applicable standards and codes (see Table 10).

The facility process 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 accident conditions.

Loads L2 and L3 are usually included in load combinations for external events if they are concurrent with the particular external event, i.e. 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. For most external events, loads L2 and L3 are unlikely.

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

7.5.2. Earthquakes

Table 11 shows typical seismic load combinations and load factors that may be used for design class 1 and 2 structures, systems and components.

TABLE 11. TYPICAL LOAD COMBINATIONS AND LOAD FACTORS TO BE USED FOR STRUCTURES, SYSTEMS AND COMPONENTS

Structure, system or component	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 SL
Non-bearing building structures	1.0 D + 1.0 L + 1.0 SL
Passive and active equipment components	1.0 D + 1.0 L + 1.0 P + 1.0 SL
Equipment nozzles, pipe flange/threaded connections	1.0 D + 1.0 L + 1.0 T + 1.0 P + 1.0 SL
Equipment supports and their anchoring	1.0 D + 1.0 L + 1.0 T + 1.0 SL
Pipes	1.0 D + 1.0 L + 1.0 P + 1.0 SL
Pipe supports and their anchoring	1.0 D + 1.0 L + 1.0 T + 1.0 SL
Cable structures, supporting platforms, etc. (including their anchoring)	1.0 D + 1.0 L + 1.0 T + 1.0 SL

[a] D = Dead load.
L = Live load under operating conditions (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 normal operating conditions (if any).
SL = Seismic load (inertia effect combined with seismic anchor movement, if any, using the square root of sum of the squares rule).
SRSS = Square root of square sum.

[b] Temperature load effects are typically considered in combination with earthquake load on structures but are not so considered for evaluation of 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.

7.5.3. Aircraft crashes

Evaluation of the effects of an aircraft crash can generally include:

(a) Global bending and shear effects on the affected structures ('overall missile effects');
(b) 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;
(c) Localized effects including penetration, perforation, scabbing and spalling, by primary and secondary missiles ('local effects');
(d) The effects of fuel fires and possibly explosion on structural components, 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 the building can provide some degree of protection.

Therefore the analysis can consider that an impact can take place anywhere in the building (peripheral walls and roof) and that the flying object can travel in any direction inside the building. In principle, all exposed structural elements are checked against all the mechanisms discussed above. Moreover, definition of the impacting object is usually very difficult and can consider a wide variety of aircraft, helicopters, missiles, etc.

For local and global analysis, the load combination for local stress/strain analysis may be (it is a typical 'beyond design basis combination'):

1.0 normal loads (dead + live) + 1.0 aircraft crash loads

Aircraft or missile fuel access into the facility and its effects can be specifically analysed, applying the criteria for explosion and fire.

7.6. QUALIFICATION BY ANALYSIS

7.6.1. Evaluation of the external event demand for structures, systems and components for specified load combinations

It is common engineering practice to determine the demand for an analysed structure, system or component and for a specified load combination based on the assumption that the structure, system or component 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 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, non-linear plastic analysis is widely used. A generic reference is provided in Ref. [5].

7.6.2. Capacity determination for qualified structures, systems and components

For design purposes, the capacity determination of analysed structures, systems and components is based on the limits (stress and strength for materials and other appropriate characteristics) as given in the selected standards and codes (Table 10) relative to all potential critical failure modes for the analysed item. These limits are the same as those adopted by these standards and codes and by 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 for the evaluation of the failure for structures, systems and components. The design stress limits required by design codes for conventional risk facilities for normal loads such as dead load, live load, operating pressure, etc., vary between one half and two thirds of the yield stress of the material with a resulting median P_F of about 10^{-4}/a, corresponding to the design load. Occasional or extreme loads, which typically have a probability of exceedance 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 P_F in the range of 5×10^{-3}/a to 10^{-2}/a, assuming that 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 structure, system or component may be based on the 95% exceedance of actual material strength limits. If such test data are not available, the corresponding limits from the selected standards and codes (Table 10) are used if properly verified by in situ investigations. Additional details for the seismic case are provided in Ref. [8].

7.6.3. Comparison of demand with capacity

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

$$(D_{\mathrm{NOC}} + D_{\mathrm{ANOC}} + D_{\mathrm{AC}} + D_{\mathrm{EE}}) \leq C \tag{4}$$

where

D_{NOC} is the demand on the structure, system or component in normal operation and normal environmental conditions (concurrent with the given external event);

D_{ANOC} is the demand on the structure, system or component due to an anticipated operational occurrence (if any, concurrent with the given external event);

D_{AC} is the demand on the structure, system or component due to accident conditions (if any, concurrent with the given external event);

D_{EE} is the demand on the structure, system or component due to a particular external event (or due to the effect of a rational combination of several external events resulting from the common initiating event);

C is the capacity of the structure, system or component.

For earthquakes, assuming that the structure, system or component behaves in a linear elastic manner, the general acceptance criterion would be:

$$D_{\mathrm{EE}} = D_E = [(D_{E,i}/k_D)^2 + (D_{E,a} \times k_{D,\mathrm{tot}})^2]^{½} \tag{5}$$

where *demand* means strength demand and

$$D_{\mathrm{EE}} = D_E = [(D_{E,i} \times k_D)^2 + (D_{E,a})^2]^{½} \tag{6}$$

where *demand* means displacement demand and

$D_{E,I}$ is the demand on the structure, system or component 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);

$D_{E,a}$ is the demand of the structure, system or component due to the anchor movement effect of an earthquake event (if any);

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

$k_{D,g}$ is the 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 (sample values are provided in Appendix III);

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

For application of Eq. (6) the following applies:

(a) 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 11).

(b) The inelastic energy absorption factors can be applied only when the seismic response of the structure, system or component is calculated in a linear elastic manner.

Nearly all structures, systems and components 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 estimate of the seismic failure margin. Limited inelastic behaviour is usually permissible for those facilities with adequate design details, making ductile response possible, or for those facilities with redundant lateral load paths. For design class 3 structures, systems and components, 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 been proven to strongly influence the results of the seismic analyses of structures, systems and components. Because of the engineering judgement required in the definition of their value, recommended values are provided in Appendix III. Reference [5] provides typical earthquake design provisions and proper structural details that apply to research reactors and comparable facilities.

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 is meant 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 of the plant protection are provided in Ref. [5].

7.7. 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 they are subjected to other external events.

Test data can be tested and processed on the basis of the corresponding nuclear or industrial standards [29–35]. The types of testing can be summarized as follows:

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

The qualification test programme may include the following elements:

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

The test 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 such effects 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 (such as nozzle loads) [23].

The test results should 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, respectively. References [33–36] provide further details on procedures and evaluation of test results for seismic testing of equipment.

7.8. QUALIFICATION ON THE BASIS OF EXPERIENCE

Currently, qualification methods based on experience are available primarily for seismic design and seismic re-evaluation of equipment [37–39]. Earthquake experience methods are simple and efficient tools to verify the seismic adequacy of selected mechanical, electrical and instrumentation and control equipment classes. Earthquake experience methods are also used to verify the seismic adequacy of piping, anchoring of piping supports and masonry walls, and to check potential seismic interactions. These methods are primarily screening and walkdown procedures and are summarized in Appendix III. Some of them involve establishing the similarity of candidate items to reference items. Similarity requires both the following basic conditions:

(a) The seismic input to be considered in the qualification of the candidate item envelops the reference or design requirements for that item;
(b) The seismic input used in the qualification of the reference item equals or exceeds that required for the candidate item.

Similarity also requires that the physical and support conditions, the functional characteristics for active items and the requirements of the candidate item closely resemble those of the reference item. If an item of equipment is classified as an outlier (i.e. it does not meet minimum capacity requirements or these are unknown), more rigorous approaches such as testing on the 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 similar facilities in any location with a PGA that does not exceed 0.33 g. For higher design basis values, the engineering experience is considered not to be developed enough to provide a basis for qualification, and other approaches should be applied.

Relays, switches, transmitters and similar electronic devices installed in research reactors may be significantly different from those considered in these methods. Therefore, it is recommended that their seismic functionality be verified, if necessary, by testing.

7.9. AGEING

Ageing effects in research reactors are considered by means of the following:

(a) Appropriate provisions during design (this should focus mainly on appropriate selection of material and development of technical specifications for periodic inspections);
(b) Surveillance and testing to assess the degradation of structures, systems and components;
(c) Development of a preventive maintenance programme;
(d) Optimization of operating conditions;
(e) Management of repairs, and replacement or refurbishment of structures, systems and components.

Further details on ageing aspects in research reactors are given in Ref. [10]. For existing facilities, the as-is conditions should be assessed. This assessment includes a review of the documentation (drawings and inspection results) and conducting site walkdowns to determine deviations from the documentation and any in-service deterioration. Material strength can be tested on-site. Corrosive actions and other ageing degradation processes may be considered. Existing facilities should be evaluated by order of priority, with the highest priority being given to those areas identified as weak links by preliminary investigations and to areas that are most important to safety.

7.10. SIMPLIFIED APPROACHES

Many simplified procedures can be used for seismic design and re-evaluation purposes in the solution of special problems; for example:

(a) Assessment of the potential for liquefaction [5, 26, 40];
(b) Assessment of soil–structure interaction [5, 29, 30];

(c) Calculation of pulling forces on anchor devices [5];
(d) Seismic resistance of pipelines with the load coefficient method [31].

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

7.11. ANCHORING OF EQUIPMENT

The lack of anchoring or inadequate anchoring has been a significant cause of failure of equipment to function properly during and after external events. Earthquakes especially have demonstrated that equipment components can slide, overturn or move excessively when not properly anchored.

Verification of equipment anchoring relies on a combination of inspections, calculations and engineering judgement. Inspections consist of measurements and visual evaluation of the equipment and its anchoring, supplemented by plant documentation and drawings. Calculations can be used to compare the anchoring capacity to the corresponding loading (demand) imposed upon the anchoring. Engineering judgement also plays an important part in the evaluation of equipment anchoring.

Various combinations of inspections, calculations and engineering judgement can be used to verify the adequacy of equipment anchoring. The responsible engineer may select the appropriate combination of assessment methods for each anchoring installation, based on the information available in the design documentation or from the walkdown. For example, a simple hand calculation may be sufficient for a pump that has only a few and very rugged anchor bolts in a symmetrical pattern. At other times it may be advisable to use computer codes that are especially tailored to equipment anchoring to determine the loads applied to multi-cabinet equipment, particularly if the anchoring of concern is not symmetrical.

Generally, the four main steps for evaluating the adequacy of equipment anchoring include:

(1) Inspection of the anchoring installation (for already existing equipment);
(2) Determination of the anchoring capacity;
(3) Determination of anchoring demand;
(4) Comparison of capacity to demand.

It is not necessary to perform the above steps in the given order. A trade-off between alternative approaches can 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 anchoring verification are available in Refs [38, 39, 41, 42].

7.12. INTERACTIONS

7.12.1. Seismic interactions

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

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

(a) Proximity (impacts of adjacent equipment or structures on safety related equipment due to their relative motion during an earthquake);
(b) Structural failure and falling of overhead or adjacent structures, systems and components;
(c) Flexibility of attached lines and cables;
(d) Flooding due to earthquake induced failures of tanks or vessels;
(e) Fire induced by earthquake induced failures;
(f) 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 given in Refs [43–46].

7.12.2. Other non-seismic interactions

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

7.13. SUMMARY OF THE SITING AND DESIGN PROCESS

Table 12 summarizes the grading assumptions discussed in the previous sections.

TABLE 12. SUMMARY OF GRADING ASSUMPTIONS

Type of facility/graded item[a]	Reference	Hazard category 1	Hazard category 2	Hazard category 3	Hazard category 4 (special risk)[a]	Conventional risk
Definition	[4]	$10 \leq P < 100$ MW	$2 \leq P < 10$ MW	$0.1 \leq P < 2$ MW	$P \leq 0.1$ MW	No radiation inventory
Safety analysis report content	[18]	[2]	[2]	[2]	[2]	/
QA level	[7, 48]	[7, 12, 48]	[7, 12, 48]	[7, 12, 48]	[7, 12]	—
Levels of defence in depth	5	1 + 2a + b	1 + a + b	$1 + a^- + b$	1 + b	1
Performance goal for structures, systems and components	10^{-5}/a	External event category 1: 10^{-5}/a External event category 2: 10^{-4}/a External event category 3: 10^{-3}/a External event category R: 5×10^{-3}/a	External event category 2: 10^{-4}/a External event category 3: 10^{-3}/a External event category R: 5×10^{-3}/a	External event category 3: 10^{-3}/a External event category R: 5×10^{-3}/a	External event category R: 5×10^{-3}/a	External event category R: 5×10^{-3}/a
Additional site screening criteria				No ACC	No ACC	
Extension of the siting campaign	[6]	Regional (50 km) + site-specific	Regional (20 km) + site-specific	IBC[b] + expert judgement on the site conditions	IBC	IBC
Applicable codes for design	Nuclear	Nuclear or conventional, according to design class	Nuclear or conventional, according to design class	Nuclear or conventional, according to design class	IBC	IBC
Probability of exceedance for external event	10^{-4}/a–10^{-5}/a [6]	10^{-3}/a–10^{-2}/a	10^{-3}/a–10^{-2}/a	10^{-3}/a–10^{-2}/a	10^{-3}/a–10^{-2}/a	10^{-2}/a

TABLE 12. SUMMARY OF GRADING ASSUMPTIONS (cont.)

Type of facility/graded item[a]	Reference	Hazard category 1	Hazard category 2	Hazard category 3	Hazard category 4 (special risk)[a]	Conventional risk
Geotechnical investigations	[51]	[51]	[51]	Soil dynamic properties from the literature, at least 1 bore hole per building	Soil dynamic properties from the literature, at least 1 bore hole per building	IBC
Seismic hazard	[9]	[9] or conservative assumptions	[9] or conservative assumptions	[9] or conservative assumptions	[9] or conservative assumptions	IBC
Meteorological events	[22]	[22]	[22]	IBC	IBC	IBC
Flood	[21]	[21]	[21]	IBC	IBC	IBC
Human induced	[20]	[20]	[20]	[20]	[20]	IBC
Malevolent	[28, 50]	[28, 50]	[28, 50]	[28, 50]	[28, 50]	IBC
Emerg. planning for rad dispersion	[47–52]	On-site and off-site	On-site and off-site	On-site	On-site	/

[a] Values in this column are only indicative, they are provided here for comparison with research reactors.

[b] IBC: Industrial building code.

8. DISPERSION OF RADIOACTIVE MATERIAL IN THE ENVIRONMENT

The objective of a radiological hazard evaluation is twofold:

(1) To establish the final hazard categorization of facilities and safety, and hence of the performance and design classification of the structures, systems and components according to the radiological hazard posed to the environment, individuals and the population in the event of an unmitigated accident;

(2) To define the requirements for emergency procedures and evacuation of the population living in the surrounding areas.

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 the fuel can be evaluated on the basis of the percentage of core melting, as defined in the reactor safety analysis. Realistic analyses can consider all the uncertainties affecting the results and therefore avoid too much credit being allocated to assumptions regarding an unmelted core in case of an accident. Particularly, the assumptions on core melting or loss of fuel barrier integrity can consider some important aspects of the accident scenario being induced by external events, such as the possibility of debris falling onto the core and preventing natural convection and power density.

A release to the atmosphere, water or groundwater for option (2) above can be evaluated on the basis of additional hypotheses (see Ref. [47]):

(i) The absorption in the pool water, provided the presence of water in the pool is guaranteed by a robust tank design;
(ii) The filtering effect of the confinement, to be consistent with the hypothesis of damage to the confinement from external events which are expected to initiate the accident sequence; realistic assumptions can exclude limit values and allocation of excessive credit for the passive confinement features;
(iii) The presence of additional safety features designed to mitigate accidents, provided their design basis could guarantee their operability during and after external events which are expected to initiate the accident sequence.

A simulation of the propagation of radioactive material in air may require some hypotheses on the topography of the site, air turbulence, humidity and direction of prevailing 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.

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

Simulation of the propagation of radioactive material through groundwater can be based on the analysis of the groundwater flow, its configuration, 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 greatly influence the final assessment of the radiological doses to the population and therefore on the hazard categorization for the whole facility. However, while it is accepted that 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 in the design of the facility. In particular, the uncertainties that affect other important contributors to the analysis should be considered for a realistic, global categorization of the facility.

9. MONITORING

The decision to install monitoring instrumentation and to safety classify it is usually taken on the basis of the relevance of the external event hazard for system design and, in general, on the basis of the instrumentation's significance for the plant's emergency procedures. Seismic monitoring and automatic scram systems, when installed, need to be properly classified for safety and adequate redundancy according to their objectives.

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

(a) To confirm a site hazard in relation to the scenarios which proved to be relevant for plant safety. In this instance, the purpose of monitoring is to detect site hazards — the data are analysed in the framework of periodic safety reviews of the facility.
(b) To enable the operator to take appropriate action during significant external events. When practicable and according to the characteristics of the event (e.g. development time, possibility of forecasting), environmental monitoring is designed, installed and operated to provide adequate warning signals for emergency operator actions during relatively slowly developing external events, and to support operator actions after the event. 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 to plant safety should be documented and reported. An extensive plant inspection after the occurrence of an external event either close to the design basis external event or significant to plant safety should be carried out to assess the behaviour and consequences on structures, systems and components against their safety classification, accessibility and representativeness for all items of the external event category.

10. AUTOMATIC SCRAM AND POST-EVENT OPERATOR ACTION

For research reactors, consideration is given to automatic actions to attain a safe state in the case of an external event when these actions are compatible with the speed of development of the external events. The facility should have protection capabilities in all operating modes and conditions. The systems in charge of this are considered safety related and 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 [5] provides information on automatic seismic trip systems for NPPs and other facilities.

After the development of an extreme external event and after the operator has taken immediate action, 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.

11. EMERGENCY PROCEDURES

Some Member States require off-site emergency procedures[13] for research reactors that are independent of the dose limits arising from deterministically postulated accidents and that may be related to both the type of reactor and its power. In other Member States the decision for setting in motion off-site emergency procedures hinges on the individual dose or the dose to the population after an accident. The upper limit of the source term may be determined on a case by case basis in order to decide if an emergency plan has to be established. In addition, if an off-site emergency plan is established, the drawing up of the plan may be used as an opportunity to define important parameters for the emergency procedures themselves. One possible approach is to consider these source terms and take into consideration only those engineering safety features indicated by the regulatory body. The emergency plan would then be extended to the point at which the doses are lower than the emergency reference level [11].

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

(a) The inherent safety of hazard category 3 research reactors prevents significant exposure of the public in the event of many postulated accidents. For reactors of this group, it can be demonstrated that there is no need for off-site emergency procedures. However, local or on-site emergency procedures will be required to protect personnel at the facility in case of accidents.

(b) Because of the inherent features of hazard category 2 research reactors, fuel melting and any significant release of radioactive material should prove unlikely for all accidents, including seismic and other external events (e.g. sufficient water will always remain in the core for fuel cooling and, in general, releases from the core will be very small). Therefore, emergency procedures are not normally required. If fuel melting or any

[13] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS OFFICE FOR THE CO-ORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-R-2, IAEA, Vienna (2002).

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 at the reactor and possibly in a limited zone around the reactor are required.

(c) For hazard category 1 research reactors, the potential for fuel damage and fission product release is related to the adequacy of the shutdown heat removal system. The requirement for and extent of emergency procedures has to be established on a case by case basis.

12. QUALITY ASSURANCE

A QA system compatible with the initiation of site evaluation activities should be established as soon as possible [7, 48]. The programme should cover all the major tasks involved in site evaluation, design, monitoring and operation of the facility, but also the transfer of data between these tasks.

Due to the frequent subcontracting of the different tasks to different contractors, a unique project, plant or facility-specific QA system is put in place for hazard category 1 and hazard category 2 facility projects, as well as for hazard category 3 facilities if such facilities contain structures, systems and components that are of a safety class higher than 3. The requirements set forth in Refs [7, 48] may be used for the QA system for a research reactor.

The methodologies for design and/or assessment of existing facilities, which are developed on the basis of this publication, should be adequately validated. Special validation effort and QA activity should be spent on the qualification procedures based on engineering experience, due to the intrinsic need for trained personnel to follow prescribed procedures.

Appendix I

EXPERIENCE FEEDBACK

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

These external events have been selected because their characteristics reflect the safety approach proposed in the present publication. Short sections deal with the lesson(s) learned, summarizing each of the reported events in terms of the content of this publication. More comprehensive reviews may be found, for example, in Ref. [54].

I.1. UNCONTROLLED REACTION TRIGGERED BY EXTERNAL EVENTS

A recent survey performed by the Los Alamos National Laboratory in the USA [55] identified 38 accidents involving a situation of uncontrolled power in US research reactors in recent years, with a total number of fission events up to 10^{20}. Twenty-two events resulted in injuries and/or fatalities to workers and the public. The survey did not consider malfunctions of both the cooling system for the radioactive material and the confinement system.

Lesson learned:

A safety analysis can carefully identify whether external events can be initiators of accidents involving the control of a reaction, with potential releases of radioactive material into the environment.

I.2. IMPLEMENTATION OF EMERGENCY MEASURES

In 1999 a bush fire developed around the Hanford Laboratories in the USA [56]. The fire was quickly spread by strong wind and evacuation of the site was difficult.

A forest fire developed around a nuclear facility at Cadarache, France [54]. Fire fighting aircraft were asked, as a first priority, to protect private properties outside the plant perimeter threatened by the fire instead of protecting the nuclear facilities. The nuclear facilities where affected by heavy smoke, with impairment of operator actions. The fire fighting aircraft posed an additional hazard to the plant with low flights over the site in hazardous flying conditions.

Lessons learned:

If external events affect nuclear and non-nuclear facilities located in the same area, priority should be given to 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 should be given to the hazard posed by fire fighting airplanes in the assessment of the interaction of different external hazards.

I.3. HUMAN ACTIVITIES IN THE VICINITY OF FACILITIES

At a site in France, in the late 1980s, general maintenance activities (gardening) caused a disturbance to the ventilation system of an intermediate storage facility due to clogging of the air inlet of the ventilation systems.

Lesson learned:

Analysis of human activities should not be limited to the industrial environment surrounding the site, but should also consider all the regular activities at the site or in the vicinity of the research reactor.

I.4. HAZARD EVALUATION

An external event, flooding of the site of the Le Blayais (France) NPP, was recorded in December 1999 [54].

Lesson learned:

Natural phenomena such as flooding should be carefully investigated for long periods of time as the high water level might vary over decades. Drainage after construction could require disproportionate remediation work. Checking the history of the water table is also valid for liquefaction evaluation, where the level of the water in the ground is of major importance.

A number of other external events resulted in significant damage to parts of the facilities as a consequence of inadequate evaluation during design:

(a) In 1990, the load of snow on the roof of the auxiliary building in the Super-Phénix plant (France) caused the roof to collapse. The snow load exceeded the values defined for design in the building code, but was less than the values that were used for a safety related analysis of the plant's structure.
(b) Icy conditions at a plant led to the unavailability of the heat removal system at Chinon, France, in 1987.
(c) Snow, water, fire and tornadoes have in many cases shown the potential to damage structures used to store documentation and files containing vital safety related information on the design of a plant.

Lesson learned:

The design basis should be derived from an accurate evaluation of the external hazard, with strong reference to the safety functions required of the affected structures, systems and components and the potential for radiological consequences from interactions with non-safety related items.

Appendix II

EXAMPLE OF EVALUATION OF THE OVERALL SAFETY MARGIN

The simplified hybrid method is proposed for evaluation of the seismic safety of existing research reactor facilities or the seismic design review of new ones. This method is especially effective for research reactors, where the success path or the plant damage state cut set can be determined with less effort than for an NPP.

The hybrid method is a combination of the seismic probabilistic risk assessment (SPRA) and the seismic margin assessment (SMA) methods. The hybrid method combines the advantages of SPRA with the simplicity of SMA. The background of the hybrid method is described in Ref. [57].

The main steps in the hybrid method are as follows:

(a) Estimation of the simplified mean hazard curve.
(b) Selection of the primary and secondary success paths or research reactor damage state cut sets (list of structure, system and component items to be qualified against external events), seismic walkdown and CDFM-HCLP$_F$ (i.e. evaluated with the conservative deterministic failure method defined in Refs [58, 59]) calculation for the selected structures, systems and components.
(c) Evaluation of variability parameters β and calculation of the plant HCLP$_F$ or damage state fragility.
(d) Estimation of the damage state risk P_F.

Estimation of hazard:

The seismic hazard curves may be assumed to be close to linear when plotted on a log–log scale. Thus, over at least any tenfold difference in exceedance frequencies, such hazard curves may be approximated by:

$$K(a) = K_I a^{-K} \tag{7}$$

where K(a) is the annual frequency of exceedance of ground motion level a, K_I is the appropriate constant, and K_H is a slope parameter defined by:

$$K_H = \frac{1}{\log(A_R)} \tag{8}$$

where A_R is the ratio of ground motions corresponding to a tenfold reduction in exceedance frequency. A_R typically ranges between 2 and 5.

Evaluation of HCLP$_F$:

Determine the component HCLP$_F$, for example with the 'CDFM method', as described in Refs [58, 59].

Estimation of variability:

For structures and major passive mechanical components mounted on the ground or at low elevations within structures, the typical range for β is 0.3–0.5. For active components mounted at high elevations in structures, the typical range for β is 0.4–0.6. Note that overestimating β is not conservative because it increases $C_{50\%}$. Limited calculations or published data for similar components can be used for β estimation if the 'as-built' and 'as-operated' conditions comply with the caveats listed in Refs [58, 59].

Determination of the seismic risk P_F for the structure, system or component:

The relationship between CDFM-HCLP$_F$ capacity (C_{CDFM}) and median capacity ($C_{50\%}$) is:

$$C_{50\%} = C_{\mathrm{CDFM}}\, e^{2.326\beta} \tag{9}$$

Estimate 10% conditional P_{F} capacity $C_{10\%}$ from:

$$C_{10\%} = F_\beta\, C_{\mathrm{HCL}} P_F \tag{10}$$

$$F_\beta = e^{1.044\beta}$$

Determine hazard exceedance frequency $H_{10\%}$ that corresponds to $C_{10\%}$ from the hazard curve.

If the fragility curve $P_{F/a}$ is log-normally distributed and the hazard curve is defined by Eq. (7), a rigorous closed form solution exists for the seismic risk equation:

$$P_F = \int_0^\infty H(a)\left(\frac{dP_{F/a}}{da}\right)da \tag{11}$$

$$P_F = HF_{50\%}^{K_H} e^{\alpha} \tag{12}$$

$$F_{50\%} = \frac{C_{50\%}}{C_H} \tag{13}$$

$$\alpha = \frac{(K_H \beta)^2}{2} \tag{14}$$

where H is any reference exceedance frequency, C_H is the ground motion level that corresponds to this exceedance frequency H from the seismic hazard curve, $C_{50\%}$ is the median fragility and β is the logarithmic standard deviation of the fragility.

Next a specific hazard exceedance frequency $H_{10\%}$ is substituted for H, where $H_{10\%}$ is defined at the ground motion corresponding to 10% conditional P_F. Thus:

$$F_{50\%} = \frac{C_{50\%}}{C_{10\%}} = e^{1.282\beta} \tag{15}$$

from which:

$$P_F / H_{10\%} = e^{-h_\beta} \tag{16}$$

$$h_\beta = 1.282(K_H \beta) - 0.5(K_H \beta)^2 \tag{17}$$

$$P_F = H_{10\%} e^{-(1.282(K_H \beta - 0.5(K_H \beta)^2))} \tag{18}$$

Over the most common A_R range, the following relationship for P_F is obtained:

$$P_F = 0.5\, H_{10\%} \tag{19}$$

For $\beta = 0.4$, Eq. (18) can be used for A_R from 1.6 to 5. However, this range can essentially cover any hazard curve of interest.

As an alternative, Eq. (11) can be solved with high accuracy using a numerical integration algorithm (see Fig. 4). First the fragility curve is generated using $C_{50\%}$ and the β parameters, and then each fragility is integrated according to Eq. (11) with selected hazard curves. The seismic hazard curve for a given site or region is defined numerically as shown in Fig. 5.

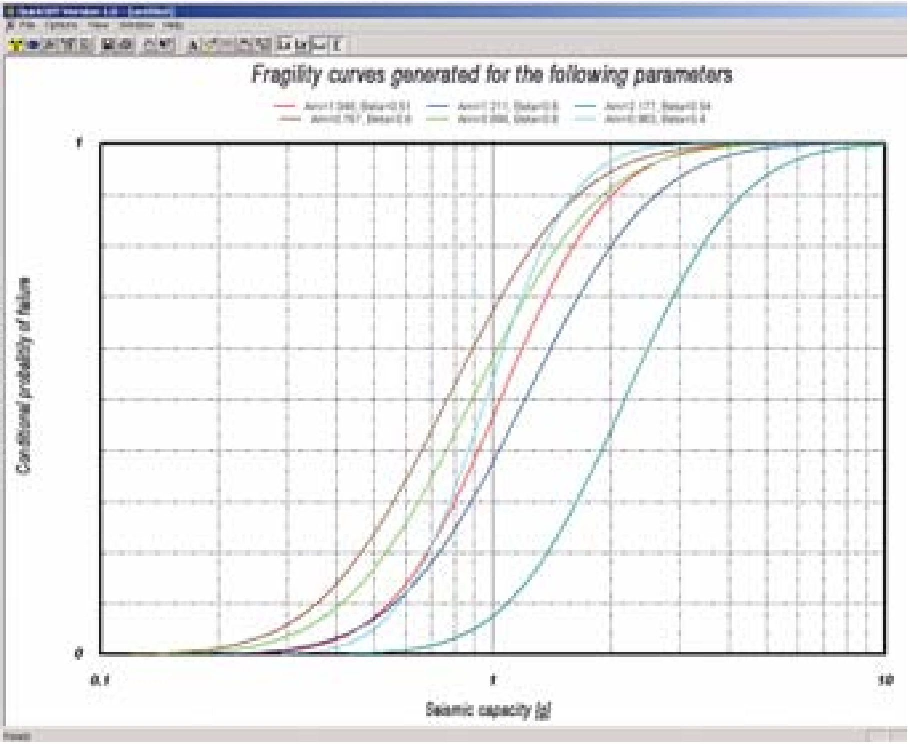

FIG. 4. Example of fragility curves generated based on $C_{50\%}$ *and* β *from Table 13.*

Table 13 presents the total P_F calculated using the above equations assuming that for the given structure, system and component items A, B, C, D, E and F the C_{CDFM} is calculated using the SMA method and then the variability values are estimated based on engineering judgement and published data.

Estimation of the overall P_F of the facility using the simplified hybrid method

Assuming that two success paths, SP1 and SP2, have been evaluated using CDFM, the damage state will occur when success paths SP1 and SP2 both fail.

$$
\begin{aligned}
&DS = SP1ISP2 \\
&SP1 = AYBYC \qquad (20) \\
&SP2 = DYEYF
\end{aligned}
$$

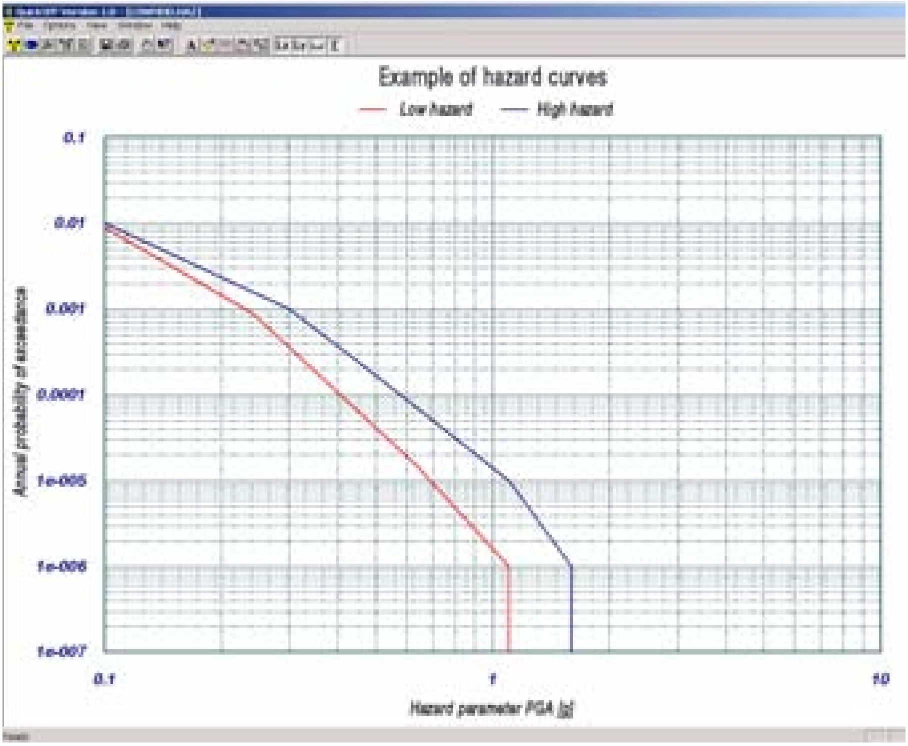

FIG. 5. Example of hazard curves.

TABLE 13. EXAMPLE OF CALCULATION OF P_F FOR INDIVIDUAL COMPONENTS

Structure system or component	C_{CDFM} [g]	β	$C_{50\%}$ [g]	$C_{10\%}$ [g]	$H_{10\%}$	P_F Eq. (12)	P_F Eq. (5)
A	0.32	0.51	1.048	0.545	2.25×10^4	9.93×10^5	6.06×10^5
B	0.30	0.60	1.211	0.561	1.89×10^4	8.34×10^5	6.11×10^5
C	0.62	0.54	2.177	1.089	1.32×10^6	6.98×10^7	4.55×10^6
D	0.19	0.60	0.767	0.355	6.44×10^4	2.84×10^4	2.42×10^4
E	0.22	0.60	0.888	0.412	5.20×10^4	2.29×10^4	21.60×10^4
F	0.38	0.40	0.963	0.577	1.54×10^4	6.82×10^5	4.48×10^5

Note: The C_{CDFM} column is calculated using the SMA method for each item. The β column is estimated on the basis of generic data, design review, field data and engineering judgement. The rest of the columns are calculated using the low hazard curve given in Fig. 4. Equation (11) is integrated via numerical integration [60].

$$
\begin{aligned}
&P_{F(\mathrm{DS})} = P_{\mathrm{F\,(SP1)}} P_{\mathrm{F(SP2)}} \\
&P_{F(\mathrm{SP1})} = P_{F(\mathrm{A})} + (1 - P_{F(\mathrm{A})})[P_{F(\mathrm{B})} + (1 - P_{F(\mathrm{B})})P_{F(\mathrm{C})}] \\
&P_{F(\mathrm{SP2})} = P_{F(\mathrm{D})} + (1 - P_{F(\mathrm{D})})[P_{F(\mathrm{E})} + (1 - P_{F(\mathrm{E})})P_{F(\mathrm{F})}]
\end{aligned} \tag{21}
$$

Equation (20) shows how the individual component fragilities are combined to obtain the plant damage state fragility parameter using the plant damage state cut set. The above calculations have to be done for at least 10 to 20 acceleration values within the acceleration range of interest.

A simplified approach is to combine the HCLP_F capacities for the individual structure, system or component with the HCLP_F max/min method to estimate the damage state HCLP_F capacity. This approach can be applied to the fault tree diagram. The minimum HCLP_F value can be chosen when there is an AND gate and the maximum value when there is an OR gate.

Because of the convolution, the damage state fragility curve has a lower β value than the individual component fragility curves. It is advisable to use $\beta = 0.3$ for the damage state variability. The fragility of the overall facility can be calculated either by using Eq. (18) or by direct integration of Eq. (11) using a computer code.

Appendix III

SUGGESTED VALUES FOR CRITICAL PARAMETERS AND REFERENCE METHODS FOR QUALIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

TABLE 14. VALUES FOR THE GLOBAL INELASTIC ENERGY ABSORPTION FACTOR $k_{D,g}$

Structural system	Global inelastic energy absorption factor $k_{D,g}$ (design/re-evaluation)		
	Design class 1	Design class 2	Design class 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 rather conservative regarding the fact that they should respect a large variety of national and international design/re-evaluation methods, as well as a large variety of structural systems with non-uniform quality of performance in different countries.

TABLE 15. 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

TABLE 15. VALUES FOR THE LOCAL INELASTIC ENERGY ABSORPTION FACTOR $k_{D,l}$ (cont.)

Structural element, equipment (failure mode)	Local inelastic energy absorption factor $k_{D,l}$ (design/re-evaluation)
Equipment components and pipes	
— Equipment components which should remain functional	1.00/1.00
— Equipment components where a brittle failure mode dominates (i.e. 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)	1.00 /1.50
— Welded equipment nozzles	1.00/1.25
— Threaded pipe connections	1.00/1.00
— 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 anchoring (brittle failure mode)	1.00/1.00
— Equipment supports and their anchoring (ductile failure mode)	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 16. RECOMMENDED DAMPING VALUES FOR SEISMIC ANALYSES

Structure, system or component	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

[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 the shear wave velocity 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 17. SOME SEISMIC EXPERIENCE BASED METHODS

Method	Items to be verified	Reference (public domain)
DOE procedure	Selected mechanical and electrical equipment classes	[39]
	Cable supporting structures	
	Anchoring of equipment	
	Non-bearing brick walls	
	Architectural details	
	Seismic interactions	
GIP procedure	Selected mechanical and electrical equipment classes	[37]
	Cable supporting structures	
	Anchoring of equipment	
	Seismic interactions	[38]
LLNL procedure	Selected mechanical and electrical equipment classes	
	Anchoring of equipment	[43]
	Seismic interactions	[44]
Stevenson procedure	Pipelines (limited scope)	[45]
Antaki procedure	Pipelines (limited scope)	[46]

Note: A critical comparison among these methods and other available methods is available in Ref. [53].

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DEFINITIONS

For general terminology concerning safety concepts, see Refs [19, 50]. The definitions were compiled solely for the purpose of the present publication. The list does not represent a consensus or an endorsement by the IAEA.

conservatism. An additional margin, to be added to the safety margin, to be used to compensate for simplified approaches in siting, design or assessment.

conventional code. For design/qualification of structures, systems and components: design standards with specifications for safety margins and QA principles considered acceptable for conventional risk installations.

design class. For design class 1, 2, 3 structures, systems and components, the design class represents the safety margin that can be used in design/ qualification.

external event. Events unconnected with the operation of a facility or activity which could have an effect on the safety of the facility or activity.

external event category. Three categories exist: external event category 1, 2 and 3. For structures, systems and components and for external events an external event category may be identified with reference to the unmitigated radiological consequences that result from the failure to fulfil the required safety function in case of an external hazard. The criteria that are used for the external event category are the same as for the respective safety classes, but with reference only to a specific external event postulated initiating event chosen for the plant design.

external event hazard. Probability of exceedance, as a function of selected representative parameters for an external event.

failure. Inhability of a structure, system or component to function within acceptance criteria.

fragility. The conditional P_F (unacceptable performance) versus the selected hazard parameter. It is usually acceptable to assume that the component failure is log-normally distributed. Fragility can be expressed in terms of median capacity $C_{50\%}$ or $HCLP_F$ capacity and variability parameter β. For preliminary estimation, $\beta = 0.3$ can be used for the entire facility.

hazard category. Three categories exist: hazard category 1, hazard category 2 and hazard category 3. Facilities may be categorized according to the hazard they pose to workers, the public and the environment. In the present publication only the radiological risk is considered. In a simplified way, hazard categorization of research reactor facilities may be expressed as a function of the power rating and radioactive inventory. It can also be affected by siting characteristics.

HCLP$_F$. A value for the selected parameter representing an external event, chosen on a median capacity curve A_m of a structure, system or component, with e_R and e_U random variables with unit medians, representing the inherent randomness of the median and the uncertainty in the median value respectively. Assuming that both e_R and e_U are log-normally distributed with logarithmic standard deviations β_R and β_U, respectively, the variables A_m, β_R and β_U determine a family of fragility curves representing various levels of confidence. The point on the 95% confidence curve that corresponds to a 5% P_F is commonly referred to as the high confidence of a low probability of failure ($HCLP_F$) value, therefore:

$$HCLPF = A_m e^{-1.65\left(\beta_R + \beta_U\right)}$$

nuclear code. For design/qualification of structures, systems and components: design/qualification standards with specifications for safety margin and QA principles more restrictive than those applied for conventional risk industrial installations, aiming at providing a higher level of reliability to the design (lower probability of consequences to workers, the public and the environment).

operational mode. Plant status when activities are performed to achieve the purpose for which a facility was constructed.

performance goal. The performance goal for a structure, system or component and for an external event is the mean P_F of the structure, system or component to perform its required safety function if that external event were to occur. Performance goals applied to external events are computed as the product of the full range of external events convoluted with the derivative of the fragility of the structure, system or component under consideration.

postulated initiating event. An event identified during design as capable of leading to anticipated operational occurrences or accident conditions.

plant status. Normal operation, anticipated operational occurrences, design basis accident, beyond design basis accident.

return period. The average time between consecutive events of the same or greater severity (e.g. earthquakes with maximum ground acceleration of 0.1 g or greater). It should be emphasized that the return period is only an average duration between events and should not be interpreted as the actual time between a small number of occurrences, which could be highly variable. A given event of return period T is equally likely to occur any year. Thus the probability of that event being exceeded in any one year is $1/T$. The annual probability of exceedance 'p' of an event is the reciprocal of the return period of that event (i.e. $p = 1/T$).

safety classification. Three classes (safety class 1, safety class 2 and safety class 3) exist for structures, systems and components. Structures, systems and components may be classified as a function of the unmitigated radiological consequence that would result to workers, the public and the environment from their failure to fulfil their required safety function for all selected PIEs (both internal and external).

safety margin. A measure of the reserve capacity of structures, systems and components beyond their design conditions, in relation to the assigned safety functions.

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