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No. 80

Safety Reassessment for Research Reactors in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant

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SAFETY REASSESSMENT FOR RESEARCH REACTORS IN THE LIGHT OF THE ACCIDENT AT THE FUKUSHIMA DAIICHI NUCLEAR POWER PLANT

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Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
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FOREWORD

This publication aims to provide practical information on the performance of safety reassessment for research reactors. The information provided is relevant for safety reassessment for all types of nuclear installation in the light of the accident that occurred in March 2011 at the Fukushima Daiichi nuclear power plant in Japan following a severe off-shore earthquake and subsequent tsunami. Flooding of the power plant and damage to equipment due to the tsunami resulted in an extended station blackout, loss of core cooling, fuel melting, hydrogen explosions and releases of radioactive material to the surrounding region, with significant contamination of the environment. The available experience from this accident will be useful for defining and implementing measures to prevent the occurrence of any accident involving a large release of radioactive material at nuclear installations, including at a research reactor, in the future.

The majority of research reactors were built to earlier safety standards, which are not fully consistent with the IAEA safety standards and with the defence in depth concept. In particular, for many research reactors the design of structures, systems and components important to safety is not in accordance with the criterion for common cause failure (i.e. the ability to withstand the failure of two or more structures, systems or components due to a single specific event or cause), and the confinement or containment buildings of several research reactors located near populated areas have deficiencies in their leaktightness. In addition, the safety analyses for many research reactors have not been updated to take into account modifications of the facilities and changes in the characteristics of their sites and site vicinity areas. These elements and the feedback from the Fukushima Daiichi accident justify a revision of the safety analysis for these facilities through the performance of a safety reassessment.

This publication provides information relevant for all steps in performing such safety reassessments for research reactors and their associated facilities, such as experimental facilities and devices, and radioisotope production facilities. Although it primarily focuses on operating research reactors, the approaches and methods provided in this publication also apply to research reactors that are in the design or construction phases, or in an extended shutdown state. The information provided is not intended to replace or supersede any of the requirements or guidance provided by the relevant IAEA safety standards but is to be used in close conjunction with them. Security topics connected with extreme external events and the emergency plans for these events are beyond the scope of this publication.

The IAEA wishes to thank all the contributors to this publication as well as the participants of the Technical Meeting on the Implications of the Fukushima Daiichi Accident on the Safety of Research Reactors (14–18 May 2012, Vienna), for the review of the first draft of this publication. The IAEA officers responsible for this publication were A.M. Shokr and H. Abou Yehia of the Division of Nuclear Installation Safety, and Y. Barnea and P. Adelfang of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The accident at the Fukushima Daiichi nuclear power plant in Japan occurred following a severe off-shore earthquake and subsequent tsunami on 11 March 2011. Flooding of the plant and damage to equipment resulted in an extended station blackout, loss of core cooling, fuel melting, hydrogen explosions and releases of radioactive material to the surrounding region, with contamination of the environment and potential long term consequences. The lessons learned from this accident will be useful for defining and implementing measures for preventing accidents involving large releases of radioactive material at nuclear installations worldwide, including research reactors.

The inventory of radioactive material, and consequently the potential hazard, associated with research reactors is much lower than that for nuclear power plants. However, the majority of research reactors worldwide were designed decades ago, and their design requirements are not fully in conformance with IAEA Safety Standard No. NS-R-4 [1]. In addition, many research reactors are located near populated areas, and for some of these the leaktightness of their confinement buildings is inadequate. These issues complicate the management of accidents that result in radioactive releases. In some other cases, the characteristics of the research reactor site and the site area and site vicinity may have changed since the facility was constructed. Not all the above mentioned issues are reflected in the safety analysis for many facilities. Thus, a revision of the safety analysis of research reactors through the performance of a safety reassessment is justified, particularly in the light of feedback from the Fukushima Daiichi accident. The priority for performance of safety reassessments needs to be decided in accordance with the potential hazard associated with each facility.

On the basis of available feedback from the accident, the topics that need to be investigated include the following: the safety requirements adopted for the design of the facility, including the seismic design of the facility; the design of the facility against flooding (from a tsunami or other cause); physical damage (from a tsunami or other cause); total loss of electrical power supply; loss of ultimate heat sink; accident management; hydrogen control; safety of spent fuel storage pools; regulations; emergency arrangements; and communication of information. Most of these topics are relevant for reassessment of the safety of research reactors when subjected to extreme external events.

1.2. OBJECTIVE

The objective of this publication is to provide a set of suggestions and methods, on the basis of current international good practices, for performing safety reassessment for research reactors, taking into consideration the available feedback from the Fukushima Daiichi accident. Information is also provided on the use of relevant IAEA safety standards in performing such a safety reassessment. This publication is intended for use by operating organizations, regulatory bodies, design organizations and other authorities involved in the safety of research reactors.

1.3. SCOPE

This publication covers all steps in the performance of safety reassessment for research reactors and associated facilities — such as experimental facilities and devices, and radioisotope production facilities — in the light of the experience from the Fukushima Daiichi accident. Although the primary focus is on operating research reactors, the information provided in this publication also applies to research reactors that are in the phases of planning, design or construction, or are in a long term shutdown state. The information provided in this publication is not intended to replace or supersede any of the requirements or guidance provided by the relevant IAEA safety standards, including those on safety analysis, evaluation of seismic and external hazards, and emergency preparedness and response in relation to research reactors. Rather, this publication is for use in conjunction with these safety standards.

The scope of the safety reassessment as described in this publication includes a review of the design basis accidents and beyond design basis accidents of the reactor facility and its site, as well as a reassessment of arrangements for preparedness for and response to an emergency resulting from such accidents. This publication also provides information on the application of a graded approach and suggested processes for the implementation of the findings of the safety reassessment.

This publication applies to research reactors having a power rating of a few hundred kilowatts up to a few tens of megawatts; it may also apply to high power research reactors having a power rating of several tens of megawatts or higher. Low risk research reactors and critical assemblies having a power rating of up to hundreds of kilowatts may need a less comprehensive safety reassessment than is outlined here, and may use a graded approach that corresponds to their potential hazards. Guidance on the use of a graded approach[1] in the application of the safety requirements for research reactors is provided in Ref. [2].

1.4. STRUCTURE

The publication is structured as follows: Section 2 discusses regulatory aspects relating to the performance of safety reassessment for research reactors in the light of the feedback from the Fukushima Daiichi accident. Sections 3–5 discuss the approaches to and methodology for performing such a reassessment: Section 3 focuses on the review of the design basis of the reactor facility and an assessment of the consequences of beyond design basis events; Section 4 focuses on reassessment of site safety; and Section 5 addresses reassessment of emergency preparedness and response. The procedure for application of a graded approach and a discussion of its use in the safety reassessment, with a focus on organizational aspects, are provided in Section 6. Section 7 suggests and describes a process for implementing the findings of the safety reassessment. The Annex provides a list of selected postulated initiating events for research reactors.

2. REGULATORY ASPECTS

The requirements on the responsibilities and functions of the regulatory body in respect of nuclear installations, including research reactors, are established in IAEA safety standards [1, 3]. These functions and responsibilities apply to the regulatory supervision of review of the implications of the Fukushima Daiichi accident for research reactors.

This publication provides an approach to and methodologies for performing safety reassessment for research reactors on the basis of the feedback from the Fukushima Daiichi accident. However, other approaches and methodologies, as required by national regulatory bodies, may also be used, provided that they result in an equivalent level of safety and that they take into consideration the use

[1] A graded approach is not used to provide relief from meeting individual safety requirements (waiving requirements). It can be applied, for example, by considering — using sound engineering judgement — the safety and operational importance of the topic, and the maturity and complexity of the area involved.

of a graded approach in the application of the safety requirements according to the potential hazards of the reactor facility.

Some States have already established regulatory requirements regarding safety reassessments and related acceptance criteria in consultation with operating organizations of research reactors. Clarity and transparency as well as effectiveness in communication (including formal communication and informal communication) between the regulatory body and the operating organization are to be ensured, including reporting of the results of the safety reassessment to the public, if this is required by national regulations.

In performing its review and assessment function, the regulatory body may require the operating organization to establish a plan for the implementation of actions identified by the safety reassessment (which could be either short term actions or long term actions). These actions can include:

— Preparation of an action plan and its submission to the regulatory body;
— Preparation of a report on the results of the safety reassessment and its submission to the regulatory body;
— Updating, as necessary, of the design and operating documentation, including the maintenance, periodic testing and inspection programme[2], and operating procedures[3];
— Revision of the training and qualification programme for reactor operating personnel[4];
— Updating of the reactor safety documents (safety analysis report, operational limits and conditions, and emergency plan and associated procedures);
— Proposals for research and development activities, as necessary, to address the identified gaps of knowledge;
— Conduct of or participation in training, drills and exercises on the performance of critical response functions, particularly in response to beyond design basis accidents.

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[2] Maintenance, periodic testing and inspection have a common objective: to ensure that the structures, systems and components function in accordance with the design intents and requirements, and in compliance with the safety analysis and operational limits and conditions. Maintenance, periodic testing and inspection may be included in a single programme and performed by the same operating personnel (see Ref. [4]).

[3] Guidance on the preparation and periodic review of operating procedures for research reactors is provided in Ref. [5].

[4] Guidance on training and qualification of operating personnel for research reactors is provided in Ref. [6].

The regulatory body can also perform specific inspections aimed at verifying the robustness of structures, systems and components (SSCs) important to safety[5], operating programmes and procedures (including maintenance, periodic testing, and inspection programmes and procedures), and emergency arrangements currently in place.

As was recognized in the early stages of the Fukushima Daiichi accident, the effective involvement of the regulatory body is essential not only in normal operating conditions but also in accident situations. Therefore, appropriate attention needs to be given to the regulatory body's ability to perform safety reviews and assessments in the case of extreme events. In this regard, as feedback from the accident, the actions that could be taken by the regulatory body include:

- Assessment of availability and adequacy of the human and financial resources necessary to perform regulatory functions;
- Review (and revision, as necessary) of the existing regulatory activities to determine whether they are adequate to verify compliance by the operating organization with new safety requirements established as a result of the Fukushima Daiichi accident;
- Review of the regulatory body's role in the event of emergency response, including the regulatory body's level of involvement during an emergency, so as to avoid interference with the prompt implementation of protective actions and other response actions on the site, and the emergency plan and associated procedures to be applied by the regulatory body;
- Ensuring that regulatory roles and responsibilities, and communication pathways and expectations for data gathering, use and retention during and after an emergency are clearly specified.

It is expected that the regulatory body will analyse the lessons learned from the Fukushima Daiichi accident and, accordingly, will proceed with the revision of existing national regulations or the development of new national regulations.

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[5] An item important to safety is an item that is part of a safety group and/or whose malfunction or failure could lead to radiation exposure of the site personnel or members of the public. Items important to safety include:

- Those SSCs whose malfunction or failure could lead to undue radiation exposure to site personnel or members of the public;
- Those SSCs that prevent anticipated operational occurrences from leading to accident conditions;
- Those features that are provided to mitigate the consequences of malfunction or failure of SSCs [7].

3. REASSESSMENT OF THE FACILITY

This section provides information on developing an approach to performing a safety reassessment of a research reactor in the light of feedback from the Fukushima Daiichi accident. This task needs to be based on the existing, approved safety analysis report. The main objective of this reassessment is to evaluate the robustness of the existing reactor protection, in terms of design features and procedures, against the impact of extreme events, with an emphasis on fulfilment of the basic safety functions[6].

Safety reassessment consists of:

— Review of the design basis of the reactor facility, as described in the safety analysis report;
— Assessment of the impact of events that are beyond the design basis of the facility (beyond design basis events), including assessment of any consequential loss of the basic safety functions and the relevance of the mitigatory actions to be taken, in order to identify the need for safety improvements in both technical and organizational aspects.

The approach to be used in the safety reassessment is essentially deterministic. However, depending on national licensing regulations, a mixed deterministic and probabilistic assessment of extreme external events may be performed [8–10].

The safety reassessment is to:

— Refer to the current status of the reactor facility as built and as operated, including all operational states of the reactor, in order to encompass existing and planned experimental facilities and experimental devices. For new research reactor projects, the assessment refers to the facility as designed and as built.
— Use the most unfavourable reactor conditions, including core configurations, that are permitted by the operational limits and conditions.

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[6] The basic safety functions are:
— Shutting down the reactor and maintaining it in a safe shutdown state for all operational states and design basis accidents;
— Providing for adequate removal of heat after shutdown, in particular from the core, including in design basis accidents;
— Confining radioactive material in order to prevent or mitigate its unplanned release to the environment [1].

— Consider the degradation of the SSCs important to safety due to ageing effects[7].
— Take into account the possible impact of failure or damage to SSCs not important to safety on SSCs important to safety, which may necessitate a detailed walkdown of the reactor facility.
— Take into account the modifications or upgrades introduced to the SSCs.
— Take into account simultaneous occurrences of more than one external event, as well as sequential and dependant events.
— Use verified and validated models and computer codes, with recognition of their limitations.

3.1. REVIEW OF THE DESIGN BASIS OF THE REACTOR FACILITY

The requirements and guidance on performing safety analyses of research reactors, including the associated experimental facilities and experimental devices, are established in IAEA safety standards [1, 8–10]. Additional guidance and examples of safety analysis methodologies for a variety of research reactor types and sizes can be found in Ref. [12].

The first phase of a safety reassessment is aimed at ensuring that the design requirements and the underlying data are valid and consistent with the current conditions of the reactor facility and its site. In the reassessment the following are reviewed:

— Postulated initiating events as well as the methodology used in the original safety analysis and its validity, including the validity of the data used at the time of the assessment;
— Design and administrative provisions, including provisions for the maintenance, periodic testing and inspection programme;
— Adequacy of the emergency arrangements for response to design basis accidents (see Section 5);
— Human resources and organizational factors, training and exercise programmes, operator qualifications and permits or licence updating, programmes for improved human performance and error reduction, etc.

If the assumptions used for the safety analysis or design basis have changed, the review needs to verify that these changes are accommodated by the existing safety analysis. Consequently, the effect on the safety margins of the SSCs

[7] Guidance on ageing management for research reactors is provided in Ref. [11].

important to safety needs to be evaluated; the associated impacts on fulfilment of a basic safety function assessed; and remediation actions and compensatory measures, as necessary, planned.

The next step of the safety reassessment is to proceed with assessment related to beyond design basis events.

3.2. ASSESSMENT OF THE CONSEQUENCES OF BEYOND DESIGN BASIS EVENTS

The second phase of the safety reassessment is an assessment of the reactor's response to beyond design basis events. This assessment is to be carried out in a systematic manner, focusing on assessment of the reactor from the perspective of defence in depth as defined in Ref. [1]. The reactor's specific vulnerabilities and, consequently, necessary safety improvements to the reactor facility and mitigatory actions have to be identified.

The assessment of the consequences of beyond design basis events covers:

- Beyond design basis events that originate from extreme events that cause damage to SSCs important to safety and challenge the fulfilment of the basic safety functions, including evaluation of safety margins[8];
- Progression of events that could lead to damage of the reactor core (and damage to spent fuel storage facilities associated with the research reactor)[9] combined with failure of SSCs important to safety;
- Adequacy of the emergency arrangements for response to beyond design basis accidents (see Section 5);
- Interaction between the reactor and the associated facilities on the site, assuming that the beyond design basis event has affected all of these facilities simultaneously (see Section 4).

Minimal combinations of SSCs and human actions that are needed to protect the reactor facility against (or mitigate the consequences of) extreme events have

[8] The safety margin is a measure of the reserve capacity of an SSC beyond the conditions for which it is designed, in relation to its assigned safety function.

[9] In some States, adoption of the 'stress test' concept necessitates calculation of 'cliff edge' values.

to be identified and documented, as well as any necessary physical improvements to the reactor facility and procedural actions. Physical modifications and altered procedures may need regulatory approval. Section 7 provides additional information on a proposed process for the implementation of such modifications.

The SSCs necessary to maintain the basic safety functions during different extreme events are identified and documented in the reactor safety analysis report. Those SSCs that are needed to maintain all or some of the basic safety functions during beyond design basis events are also identified.

The safety reassessment focuses on complete verification of the effective application of the defence in depth concept in the reactor facility following extreme internal and external events. The postulated initiating events listed in the appendix to Ref. [1], and provided in the Annex to this publication, should be reviewed in order to include those that are specific to the reactor, with emphasis on special internal and external events, in order to understand the consequences of beyond design basis events. Consideration has to be given to combinations of these events and their consequences that may be credible for the reactor site or the reactor facility.

The analysis proceeds to determine the status of those SSCs important to safety (i.e. whether they will continue to perform the intended function or will fail) that support fulfilment of the relevant basic safety functions during the course of a beyond design basis event (including simultaneous or consequential events). The operating organization may prepare a list of SSCs important to safety based on the existing approved safety analysis report, and obtain agreement of the regulatory body on the list. After this list is established, the contribution of each component to fulfilling one or more basic safety function(s) is determined. Subsequently, the operating organization needs to be able to verify the robustness of the reactor facility or to identify the missing information on SSCs important to safety. The results of this verification can be tabulated in a matrix form, as illustrated in Table 1.

The subsequent step in the analysis is to evaluate the radiological consequences of a loss of the relevant basic safety functions due to failure of the SSCs intended to perform one or more of these safety functions. These consequences are to be evaluated in terms of radiation doses to on-site personnel and the public, as well as the effect of radioactive releases to the environment.

The results obtained in the previous step can then be used to identify the beyond design basis events that necessitate further investigation. The examination is carried out if:

— The concept of defence in depth is not fully applied in the reactor facility and as a result events may lead to unacceptable radiological consequences;

TABLE 1. EXAMPLES OF SSCs FOR THE VERIFICATION OF ROBUSTNESS OF THE FACILITY FOLLOWING EXTERNAL AND INTERNAL EVENTS OR COMBINATIONS OF EVENTS

SSCs important to safety	Basic safety function affected			Result of verification of robustness						
	Reactivity control	Decay heat removal	Containment/ confinement against release of radioactive material	Earthquake	Flooding	Fire	Tornado/ hurricane	Sand blast	Heavy snow	Other (see Annex)
Reactor protection and shutdown systems: — Control rods — Second shutdown system	✓	n.a.	n.a.	✓						
	✓									
	✓									
	✓									
Core cooling system: — Primary pumps' flywheels — Flapper valves										
Emergency core cooling system										

TABLE 1. EXAMPLES OF SSCs FOR THE VERIFICATION OF ROBUSTNESS OF THE FACILITY FOLLOWING EXTERNAL AND INTERNAL EVENTS OR COMBINATIONS OF EVENTS (cont.)

SSCs important to safety	Basic safety function affected			Result of verification of robustness						
	Reactivity control	Decay heat removal	Containment/ confinement against release of radioactive material	Earthquake	Flooding	Fire	Tornado/ hurricane	Sand blast	Heavy snow	Other (see Annex)
Building and structures										
Containment or confinement: — Ventilation — Filtration system										
Electrical system: — Uninterrupted power supply — Diesel generators										
Other										

Note: n.a. — not applicable.

— Failure of the basic safety functions could lead to unacceptable consequences; or
— A potential human error could result in a failure of a basic safety function and could lead to more severe consequences.

The next step is to identify, on the basis of the results of the safety reassessment, possible preventive measures to be applied and mitigatory actions to be taken. Implementation of these actions would have to be justified, and proposed options for these actions subjected to a cost–benefit analysis. Indicative examples of possible actions are as follows:

(a) Acceptance of a failure to fulfil the basic safety functions because the radiological consequences are:
   (i) Within acceptable limits;
   (ii) Manageable with the emergency arrangements currently in place (e.g. evacuation of buildings within the reactor site); or
   (iii) Insignificant in comparison with the impact of the event on the public and the environment (e.g. a seismic event that devastates a vast area and results in thousands of casualties, while dose rates are insignificant).
(b) Bounding of the consequences of the event with a scenario that has previously been analysed in the safety analysis report and meets the acceptance criteria (e.g. an extreme seismic event may block coolant flow through the core, resulting in the same consequences as a blockage of fuel coolant channel event that has previously been analysed).
(c) Performance of additional analyses to determine the consequences of the event in order to:
   (i) Determine the time frame of the event sequence and the associated safety margin (e.g. the flooding level at which inundation of the reactor would result in the loss of a basic safety function or the time needed to prevent this loss); or
   (ii) Identify the need to strengthen the mitigatory actions by enhancing emergency response capabilities, including plans and procedures in place, provision of alternative supplies, tools, equipment, training and performance of drills and exercises for off-site response organizations and on-site response personnel to be involved in an emergency response.
(d) Enhancement of preventive measures through installation of new (or modifications or upgrades of existing) SSCs resistant to external hazards, such as:

(i) Upgrades of emergency electrical power supplies, by redundant connections to the off-site power grid, supplementary emergency power generators and/or improved back-up batteries;
(ii) Installation of seismic detectors connected to the reactor protection system and instrumentation that operate under extreme conditions;
(iii) Installation of a diverse shutdown system, such as one based on the injection of neutron absorber solution;
(iv) Modification to improve the emergency ventilation system and the associated filtration system, where provided;
(v) Provisions for recycling coolant from sub-pile rooms (containers or vessels) to the reactor pool;
(vi) Installation of passive components in the cooling systems;
(vii) Strengthening of various SSCs, especially those necessary to prevent core damage or damage to irradiation rigs in an extreme event (e.g. earthquake, flooding, tornado).

(e) Incorporation of provisions for earthquake protected and fire protected backup equipment (e.g. mobile diesel generators and pumper trucks) to maintain the basic safety functions, and associated revision of the emergency operating procedures (see Section 5).

The safety reassessment method described above also has to be applied for facilities associated with the research reactor, such as experimental loops and radioisotope production installations.

4. REASSESSMENT OF THE SITE

4.1. REVIEW OF THE SITE CHARACTERISTICS

Requirements, guidance and methodologies for assessing external hazards for research reactors are provided in IAEA safety standards and other publications [1, 13–18][10]. As well as being useful for the initial siting, design and safety evaluation of a new research reactor, these publications can also be used for reassessment of the ability of an existing research reactor to cope with extreme external events.

[10] Reference [14], on the Safety of New and Existing Research Reactor Facilities in Relation to External Events (Safety Reports Series No. 41), is currently under revision. The revised version is to include examples of acceptable safety margins of SSCs important to safety.

A site-wide review of safety may need to refer to the security of the site affected by a beyond design basis event. Methodologies for reviewing the security of the site are beyond the scope of this publication; further information is provided in IAEA Nuclear Security Series publications [19–23].

Research reactor site characteristics are reviewed to ensure the continued acceptability of the site. This review focuses on determining whether any changes have occurred in the characteristics of the site area[11] and site vicinity that may result in an increase in the magnitude or the frequency of occurrence of potential external hazards or in changes to the associated radiological consequences. Information on reassessment of the external hazards for verification of compliance with the reactor design basis is provided in Section 3.

Changes within the site area and in its characteristics to be reviewed to verify the site area's continued acceptability may include:

— *Changes in the distribution of workers on the site.* In many States, workers designated as non-radiation workers within the site area are treated as the general public with respect to radiation dose limits. Therefore, the effects of changes in the distribution of workers on the site (including changes in the proportion of workers designated as radiation workers versus those designated as non-radiation workers) need to be assessed to verify whether the potential consequences of an event have changed.

— *Changes at the other facilities within the site area, including changes in their use.* Such changes can have a positive or negative impact on the safety of the research reactor and its associated facilities through increasing or decreasing the potential hazards. For example, decommissioning of another research reactor located on the site will significantly reduce the potential hazard.

— *Changes in the site infrastructure and support services.* Such changes can also have a positive or negative impact on accident prevention measures and/or planned mitigatory actions. For example, improvements to the site control and alarm systems can be expected to have benefits in relation to severe accident management.

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[11] The site area is a geographical area that contains an authorized facility (e.g. research reactor) and within which the management of the facility may directly initiate emergency actions. This is typically the area within the security perimeter fence or other designated property marker. The site area is often identical to the operations area, except in situations where the authorized facility is on a site where other activities are being carried out beyond the operation area but where the management of the authorized facility can be given some degree of authority over the whole site area [7].

Changes in the characteristics of the site's surroundings need to be reviewed to verify the site's continued acceptability. They may include:

— *Changes in the surrounding population.* Analysis needs to be performed to verify the effect of any changes in the surrounding population or its distribution on the potential consequences of an event. For example, the consequences of an accident at a research reactor and its associated facilities originally located far from any urban area will be different if they are now located within an urban area.
— *Changes in local land use.* Analysis needs to be performed to verify the effect of any changes in land use that could represent a new hazard to the reactor facility and installations on the site. For example, new industrial facilities located nearby could represent a new external hazard if they involve the use of large volumes of toxic or explosive gases. Conversely, the removal of a nearby industrial facility could reduce the potential hazard to the reactor facility and the installations on the site.
— *Changes in local transportation routes.* New roads, railways or airports constructed near the reactor facility could increase or decrease the potential hazards to the reactor facility and the on-site installations.
— *Changes in the hydrology and topography of the site vicinity.* Changes in hydrology and topography could present a new hazard or remove an existing hazard to the reactor facility and the installations on the site. For example, if the site is located near a river, construction of a dam upstream or downstream from the reactor facility may increase the potential flooding risk.

4.2. SITE-WIDE EVENTS

In order to assess the impact of extreme external events on the site area and the site vicinity, the entire site has to be considered so that the potential for simultaneous accidents at different facilities can be assessed, including the ability of on-site response personnel and off-site response organizations to respond effectively.

A process for reassessing the potential impact of extreme external events across the whole site could be as follows:

— Identification of possible new hazards to the research reactor and its associated facilities that may arise from other facilities on the site;

— Identification of other on-site facilities for which the external event may result in more serious consequences than those that would result for the research reactor and its associated facilities;
— Determination of the impact of the external event on the site infrastructure and supporting services with respect to both its impact on the on-site facilities (the research reactor and its associated facilities) due to consequential and common cause failure, and its impact on the accident prevention measures to be applied or mitigatory actions to be taken within the research reactor and its associated facilities;
— Reassessment of the ability of the on-site response personnel and/or the off-site response organizations to effectively manage events occurring simultaneously within multiple facilities on the site, including the research reactor and its associated facilities (see also Section 5), possibly initiated by the same external event, taking into consideration the impact of this initiating event on the site infrastructure.

Reassessment of the potential impact of extreme external events on the site vicinity may include:

— *The impact on access to the reactor site.* For example, access of reactor operating personnel to the site to replace those on duty needs to be evaluated. Access of the off-site response organizations to the reactor site has to be properly coordinated with the local competent authorities and site authorities. Alternative access paths to the site need to be investigated in the case of blockage of the normal access path (e.g. due to an external event).
— *The availability of on-site response personnel.* Staffing different positions relevant to the performance of emergency response functions with on-site response personnel for an emergency initiated by an extreme external event could be challenging, particularly due to the possibility of: (i) several facilities being affected simultaneously; (ii) the need for prolonged response; and (iii) loss of the access path to the site (see Section 5). In addition, consideration needs to be given to the possibility that, in the case of an extreme external event, there may be a natural tendency of on-site response personnel to absent themselves from their workplace in order to assist their families, which may result in deficiencies in the staff and pose additional difficulties for accident management.
— *The availability of the off-site response organizations.* Assuming that access to the reactor site is ensured, it also needs to be verified that the off-site response organizations are appropriately trained and capable of fulfilling their intended role (see also Section 5), particularly in emergency response to a beyond design basis accident initiated by an extreme external

event affecting several facilities simultaneously. This needs to include performance of appropriate drills and exercises (see Section 5). It has to be guaranteed (possibly through formal agreements or memoranda of understanding) that the operating organization will receive the services of the off-site response organizations.

— *The use of the site as an emergency centre.* Locating the emergency centre for directing the on-site emergency response on the site itself may affect its operability and habitability under severe conditions combined with an extreme external event and in accessibility issues.

5. REASSESSMENT OF EMERGENCY PREPAREDNESS AND RESPONSE

This section provides information on performance of a review of the emergency arrangements in place in order to verify their adequacy in addressing the consequences of beyond design basis accidents.

As a first step, the existing emergency plan and associated emergency procedures have to be reviewed to ensure that the existing arrangements are adequate and that their implementation is possible (particularly for an accident initiated by an extreme external event affecting several facilities simultaneously). Safety requirements and the associated guidelines on emergency preparedness and response in relation to research reactors are established in IAEA Safety Standards and other publications [24–28]. These publications form the basis for this review.

In the second step, the following topics have to be reassessed and verified in the light of feedback from the Fukushima Daiichi accident:

— *Chain of command.* The objective of the reassessment is to verify that a clearly defined command and control system is in place and is well understood by all in order to: (i) effectively manage the response to an emergency; and (ii) allow prompt decisions on protective actions and other response actions to be taken. The emergency plan and associated procedures have to provide for the implementation of the chain of command, including decision making in all phases of the emergency response. Training, drill and exercise programmes need to provide for training and realistically testing the chain of command and decision making in an emergency response, particularly for a beyond design basis accident initiated by an extreme external event affecting several facilities simultaneously.

— *Communication.* The objective of the reassessment is to verify the existence of adequate procedures and means for effective communication during an emergency. Pre-established communication procedures need to be in place to ensure effective communication between on-site response personnel at different positions and off-site response organizations, and effective communication to the public. The means of communication used are to be redundant and diverse in nature, in order to ensure their operability (e.g. considering possible damage due to the initiating extreme external event) and their availability for an extended period of time, which may be necessitated by the emergency. Communication to the public is not to interfere with the prompt implementation of the protective actions and other response actions in an emergency and needs to be carried out in a coordinated manner so that consistent messages are given.
— *Readiness of the on-site response personnel and off-site response organizations.* The objective of the reassessment is to ensure that arrangements are in place and are implemented to ensure that on-site response personnel and off-site response organizations have the capability to respond to a beyond design basis accident initiated by an extreme external event affecting several facilities simultaneously. Emergency drill and exercise programmes need to be revised to cover response to such events.
— *Emergency equipment.* The objective of the reassessment is to ensure that equipment necessary for use in response to an emergency (e.g. radiation survey meters, weather related instruments) is continuously available for its intended use under severe conditions and is subject to periodic verification. The availability and suitability of the analysis tools to be used in an emergency response to a beyond design basis accident initiated by an extreme external event affecting several facilities simultaneously also need to be covered by the reassessment. Moreover, the reassessment has to verify that any additional equipment intended to maintain the basic safety functions is suitable and available for use under severe conditions, and is working properly. This review has to cover the availability of power supplies such as mobile diesel generators and batteries, as well as water supplies for core cooling and spent fuel cooling and for maintaining the ultimate heat sink, where applicable. Furthermore, interfaces for use of off-site equipment have to be prepared and available. The operating organization needs to have agreements in place to ensure that the emergency equipment is available from off-site sources.

— *Accessibility and logistical support.* The objective of the reassessment is to ensure that alternative means of accessibility to the site for off-site response organizations are in place and that logistical support is available when needed, with due consideration of the possible impact of an extreme external event. It is also important to ensure that those personnel tasked with providing analytical support for decision making can properly validate and verify the results in order to minimize any potential confusion that may arise from the receipt of conflicting information.
— *Role of the regulatory body*. See Section 2.

The training and qualification programme for the reactor operating personnel has to be revised to cover the operator’s response to beyond design basis events as well as to ensure that operators are adequately trained to fully recognize the potential for an event to be beyond the design basis and to respond effectively.

6. APPLICATION OF A GRADED APPROACH

Given the different types and sizes of research reactors and the associated utilization programmes, a graded approach should be applied to the safety reassessment commensurate with the potential hazard of the reactor facility [2]. Aspects of the reassessment that may be subjected to grading include the scope, extent and details of the analysis, and the required human and financial resources, which may be significantly less for low power research reactors than for high power research reactors.

Factors affecting the application of a graded approach are those related to the risk and the potential hazard, including, for example:

— The reactor power;
— The fission product inventory and the radiological source term;
— The amount and enrichment of fissile material;
— Fuel design;
— Inherent safety features of the design;
— The presence of high pressure or high energy piping (experimental loops);
— The quality of the means of confinement (containment and ventilation systems);
— The presence of experimental facilities and experimental devices, and the reactor utilization programme;

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

Grading may be applied to the scope and level of detail of review of design basis events and the assessment of the beyond design basis events of the reactor facility. Certain accident scenarios may not apply or may need only limited analysis in low power research reactors compared with high power research reactors. For example, the analysis and management of a loss of coolant accident may vary significantly depending on the power and design of the reactor.

The graded approach may be also applicable to the selection of site related design basis events (and beyond design basis events) to the extent that the examination of events may show that some of them pose a minimal hazard to the reactor facility on a particular site.

A graded approach may also be used in the application of the safety requirements related to the levels of the defence in depth, in the sense that level 5, and sometimes level 4, may be met by the inherent safety characteristics of the reactor instead of through engineered safety features of the design. If the research reactor is designed without confinement or containment, for example, this needs to be justified on the basis that, under accident conditions, there is no potential for release of radioactive material from the facility that may result in unacceptable off-site consequences.

Grading may be applicable to the emergency arrangements to be established based on the potential hazard associated with the research reactor facility in line with the requirements established in Ref. [24]. Grading may also be applied to the number and types of escape routes, based on the layout and size of the reactor facility. It may also be applied to the necessary emergency equipment, and to the scope and frequency of the emergency drills and exercises.

A graded approach can be also applied to the organizational aspects, including human and financial resources, of performing the safety reassessment and to the management of implementation of the findings of the reassessment. Application of the graded approach should be based on the potential hazard of the research reactor facility, and should take into account the existence of other nuclear installations on the site, including those facilities associated with the research reactor [2].

Nevertheless, certain organizational factors, such as safety culture and human performance, are required to be maintained by the managers at the highest level [29], since weakness in these areas can have an impact on the effectiveness of an emergency response.

7. USE OF THE REASSESSMENT FINDINGS

The objective of this section is to provide information on establishing a process for using the findings of the safety reassessment described in Sections 3–5.

The regulatory body may analyse the extent to which the lessons learned from the Fukushima Daiichi accident have been understood and have been applied to the site and facilities (see also Section 2). To ensure effective implementation of the findings of the reassessment, the operating organization may establish an action plan to be submitted to the regulatory body for review and subjected to peer review, as required.

Such an action plan will define those actions for which a temporary shutdown of the reactor is required. The operating organization is responsible for ensuring the availability of the human and financial resources necessary for implementation of the action plan. The action plan can include short term and long term actions, depending on the impact of each action on the safety of the reactor. For example, it is expected that actions that will result in direct safety improvements will have priority over other actions. A cost–benefit analysis may be applied in the area of operational improvements, including improvements to the reactor availability and utilization programme. Opportunities for improvement that are identified may be addressed over a longer period of time within the normal allocation of resources and normal planning for the facility. The operating organization also has to review the collective findings of the safety reassessment to determine if the aggregate of potential impacts makes the risk associated with continued operation of the facility greater than the benefit that would be derived from its continued operation.

Upon completion of the safety reassessment, the operating organization may have identified findings or opportunities for improvement that require follow-up actions. These may include:

- Findings, such as a non-compliance with a regulatory requirement, which need corrective action to ensure that an adequate margin of safety is maintained;
- Opportunities for improvement, such as deviations from best management practices or design requirements that do not currently have a significant impact on safety.

Corrective actions may take one or more of the following forms:

— Improving safety culture, leadership and organizational aspects, including definition of the roles and functions of groups and individuals within the reactor operating organization, including the lines of communication, and incorporation of the emergency management system into the integrated management system. Requirements and guidance on application of the management system for facilities and activities are provided in Refs [29, 30].
— Enhancing training and qualification programmes for the reactor operating personnel and ensuring adequate human resources for safe operation of the reactor.
— Establishing management system processes, as needed.
— Modifying (or upgrading) those SSCs that are not adequately designed or that would not perform as expected during a beyond design basis event.
— Strengthening emergency arrangements, including training and qualification programmes, drills and exercise programmes, the emergency plan and procedures, and the availability and operability of the necessary equipment and facilities, etc., for dealing with a beyond design basis accident initiated by an extreme external event affecting several facilities simultaneously.

In addition, the results of the evaluation of beyond design basis events need to be shared with the reactor operating personnel in order to promote a better understanding of the prevention of beyond design basis accidents and mitigation of their consequences, and to improve safety culture.

Despite the fact that research reactors are of different types and sizes and have different utilization programmes, the results of the safety reassessment, in particular the evaluation of beyond design basis events, is likely to reveal common issues and generic lessons to be learned by the whole research reactor community. Therefore, the results of the safety reassessment need to be shared, particularly those related to generic lessons learned[12] [31–33]. It is not expected that facilities report information that could identify security vulnerabilities or commercial information associated with their specific reactor design.

[12] Some research reactor organizations have already published the results of the safety reassessment (‘stress tests’ or ‘complementary safety assessment’) of their research reactor facilities [31–33].

Operating organizations could consider independent review of the safety reassessment through, for example, peer review processes, taking into consideration the relevant national regulatory requirements. Peer reviews can also be conducted as a partnered review with the facility personnel or as an independent review of the process and the conclusions reached by the facility review. Peer reviews may also be conducted under the auspices of the IAEA.

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Annex

SELECTED POSTULATED INITIATING EVENTS FOR RESEARCH REACTORS

The list is reproduced from the appendix to Ref. [A–1].

(1) Loss of electrical power supplies:
- — Loss of normal electrical power.

(2) Insertion of excess reactivity:
- — Criticality during fuel handling (due to an error in fuel insertion);
- — Startup accident;
- — Control rod failure or control rod follower failure;
- — Control drive failure or system failure;
- — Failure of other reactivity control devices (such as a moderator or reflector);
- — Unbalanced rod positions;
- — Failure or collapse of structural components;
- — Insertion of cold water;
- — Changes in the moderator (e.g. voids or leakage of D_2O into H_2O systems);
- — Influence by experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material);
- — Insufficient shutdown reactivity;
- — Inadvertent ejections of control rods;
- — Maintenance errors with reactivity devices;
- — Spurious control system signals.

(3) Loss of flow:
- — Primary pump failure;
- — Reduction in flow of primary coolant (e.g. due to valve failure or a blockage in piping or a heat exchanger);
- — Influence of the failure or mishandling of an experiment;
- — Rupture of the primary coolant boundary leading to a loss of flow;
- — Fuel channel blockage;
- — Improper power distribution due, for example, to unbalanced rod positions in core experiments or fuel loading (power–flow mismatch);
- — Reduction in coolant flow due to bypassing of the core;
- — Deviation of system pressure from the specified limits;

— Loss of heat sink (e.g. due to the failure of a valve or pump or a system rupture).

(4) Loss of coolant:
- — Rupture of the primary coolant boundary;
- — Damaged pool;
- — Pump-down of the pool;
- — Failure of beam tubes or other penetrations.

(5) Erroneous handling or failure of equipment or components:
- — Failure of the cladding of a fuel element;
- — Mechanical damage to core or fuel (e.g. mishandling of fuel, dropping of a transfer flask onto the fuel);
- — Failure of an emergency cooling system;
- — Malfunction of the reactor power control;
- — Criticality in fuel in storage;
- — Failure of means of confinement, including the ventilation system;
- — Loss of coolant to fuel during transfer or storage;
- — Loss or reduction of proper shielding;
- — Failure of experimental apparatus or material (e.g. loop rupture);
- — Exceeding of fuel ratings.

(6) Special internal events:
- — Internal fires or explosions;
- — Internal flooding;
- — Loss of support systems;
- — Security related incidents;
- — Malfunctions in reactor experiments;
- — Improper access by persons to restricted areas;
- — Fluid jets and pipe whip;
- — Exothermic chemical reactions.

(7) External events:
- — Earthquakes (including seismically induced faulting and landslides);
- — Flooding (including failure of an upstream/downstream dam and blockage of a river and damage due to tsunami or high waves);
- — Volcano eruption (including lava flow, ash deposition, toxic gas emission, etc.);
- — Tornadoes and tornado missiles;
- — Sandstorms;
- — Hurricanes, storms and lightning;
- — Tropical cyclones;
- — Explosions;
- — Aircraft crashes;
- — Fires;

— Toxic spills;
— Accidents on transport routes (including collisions into the research reactor's building);
— Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities and waste management facilities);
— Biological hazards, such as microbial corrosion, structural damage or damage to equipment by rodents or insects;
— Extreme meteorological phenomena;
— Lightning strikes;
— Power or voltage surges on the external electrical supply line.

(8) Human errors.

REFERENCE TO THE ANNEX

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CONTRIBUTORS TO DRAFTING AND REVIEW

Abou Yehia, H.	International Atomic Energy Agency
Adelfang, P.	International Atomic Energy Agency
Alamsyah, R.	Nuclear Energy Regulatory Agency (BAPETEN), Indonesia
Barnea, Y.	International Atomic Energy Agency
Ciocanescu, M.	Institute for Nuclear Research — Piteşti, Romania
Crone, R.	Oak Ridge National Laboratory (ORNL), United States of America
Kastenmüller, A.	Technische Universität München, Germany
Krause, M.	Atomic Energy of Canada Limited (AECL), Canada
Oh, S.-Y.	Korea Atomic Energy Research Institute (KAERI), Republic of Korea
Ohba, T.	Japan Atomic Energy Agency (JAEA), Japan
Shokr, A.M.	International Atomic Energy Agency
Summerfield, M.W.	Australian Nuclear Science and Technology Organisation (ANSTO), Australia
van der Schaaf, B.	Consultant, Netherlands

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GERMANY

Goethe Buchhandlung Teubig GmbH
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