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International Atomic Energy Agency

EMERGENCY PREPAREDNESS AND RESPONSE

EPR-NPP-CAP

2024

Classification, Assessment and Prognosis During Nuclear Power Plant Emergencies

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CLASSIFICATION, ASSESSMENT AND PROGNOSIS DURING NUCLEAR POWER PLANT EMERGENCIES

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EPR-NPP-CAP (2024)

CLASSIFICATION, ASSESSMENT AND PROGNOSIS DURING NUCLEAR POWER PLANT EMERGENCIES

GUIDELINES ON PROVIDING INFORMATION TO THE ON-SITE AND OFF-SITE PROTECTIVE ACTION DECISION MAKERS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2024

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FOREWORD

After the accident at the Fukushima Daiichi nuclear power plant in Japan in March 2011, the IAEA developed the IAEA Action Plan on Nuclear Safety, which was approved by the IAEA Board of Governors in September 2011 and unanimously endorsed by the IAEA General Conference during its 55th regular session that same month. As part of this plan, the IAEA was requested to provide Member States, international organizations and the general public with timely, clear, factually correct, objective and easily understandable information during a nuclear emergency on its potential consequences, including analysis of available information and prognosis of possible scenarios based on evidence, scientific knowledge and the capabilities of Member States.

In adopting the action plan, the IAEA's role in an emergency has been expanded to include a prognosis of the potential evolution of an accident and an assessment of its possible consequences. The IAEA shares the results of its assessment and prognosis with Member States and relevant international organizations to assist them in their own analysis of a nuclear or radiological emergency. This publication aims to provide on-site decision makers in Member States with practical guidance information on classifying a nuclear emergency at a nuclear power plant or a spent fuel pool as part of the assessment and prognosis process. A declared emergency class serves as the basis for the activation of off-site responders, as well as the preparation and implementation of public protective actions, as applicable, by the off-site decision makers.

The Operations Manual for IAEA Assessment and Prognosis During a Nuclear or Radiological Emergency, EPR-A&P (2019), describes the details of the IAEA assessment and prognosis process including its technical basis. It is supported by the IAEA Incident and Emergency Centre's assessment and prognosis tools web site, which contains specialized tools and describes the detailed information to be shared by Member States during a nuclear or radiological incident or emergency for assessment and prognosis.

Information on core damage assessment based on radiation levels in the containment and an overarching emergency classification flow chart are included as supplementary files and available on-line.

The IAEA officer responsible for this publication was F. Stephani of the Incident and Emergency Centre.

CONTENTS

1. INTRODUCTION

After the accident at the Fukushima Daiichi Nuclear Power Plant (NPP) in Japan in March 2011, the IAEA developed the IAEA Action Plan on Nuclear Safety [1]. The adoption of this Plan has expanded the IAEA's role in an emergency to include a prognosis of the potential evolution of an accident and an assessment of its possible consequences. The IAEA shares the results of its assessment and prognosis (A&P) with Member States and relevant international organizations to assist them in their own analysis of a nuclear or radiological emergency. An IAEA publication, The Operations Manual for IAEA Assessment and Prognosis during a Nuclear or Radiological Emergency (EPR-A&P 2019) [2] describes the A&P process, including its technical basis.

This publication provides practical guidance and tools on the symptom and event based emergency classification, as part of the A&P of a nuclear emergency at an NPP or spent fuel pool (SFP). The A&P process described here applies to all NPP and SFP emergencies, regardless of the cause. Two key needs are addressed. The first is for guidance to support the decision making process relating to the declaration of a nuclear emergency at an NPP or SFP. The second is for guidance to obtain a good understanding of the status of an NPP involved in a nuclear emergency.

Regarding the first need, the guidance developed in this publication elaborates on emergency action levels (EALs) derived from specific, predetermined and observable on-site criteria for on-site decision makers to declare an emergency class. On that basis, simple and unambiguously presented information is necessary for decision makers to decide on the implementation of protective actions and other off-site response actions.

Regarding the second need, this publication puts in perspective the emergency classification within the A&P process. This process allows assessment of a past or present release, and anticipation of a future release to the environment and its characterization.

1.1. BACKGROUND

In 1997, the IAEA published IAEA-TECDOC-955, titled Generic Assessment Procedures for Determining Protective Actions during a Reactor Accident, which provided practical guidance and tools for the assessment of an accident at a nuclear reactor. IAEA-TECDOC-955 also provided procedures for classifying an accident, projecting consequences, coordinating and interpreting environmental monitoring, determining public protective actions and controlling doses to emergency workers. It is intended for use by on-site and off-site authorities responsible for evaluating the consequences of a reactor accident and making recommendations for the protection of plant personnel, emergency workers and the public.

The following publications (given in chronological order) have either updated, further developed, or in some cases replaced, certain sections and elements of IAEA-TECDOC-955:

— IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [3], which provides recommendations on the use of a classification system that includes EALs spanning the full range of postulated emergencies, regardless of their probability of occurrence.
— IAEA Safety Standards Series No. GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [4], which provides recommendations on the generic and operational criteria needed to support decision making on protective actions and other response actions in an emergency. These criteria

also superseded the concept of intervention levels and generic action levels contained in GS-G-2.1 [3][1].
- IAEA EPR Series, EPR-NPP Public Protective Actions 2013, Actions to Protect the Public in an Emergency due to Severe Conditions at a Light Water Reactor [5] provides those responsible for making, and acting on, decisions with an understanding of the actions necessary to protect the public in the event of an emergency, including actual or projected severe damage to the fuel in the reactor core, or the SFP, at a light water reactor.
- IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [6], which establishes requirements necessary to ensure an adequate level of preparedness and response for a nuclear or radiological emergency, irrespective of its cause. In particular, Requirement 6 of GSR Part 7 details the requirement for assessing the information necessary to make decisions about the allocation of resources for all response organizations during a nuclear or radiological emergency. Requirement 9 of GSR Part 7 [6] details the assessment of emergency conditions to support effectively urgent protective and other response actions in a nuclear or radiological emergency.
- IAEA EPR Series, EPR-NPP-OILs 2017, Operational Intervention Levels for Reactor Emergencies [7], provides selected default values for operational intervention levels, together with a detailed description of the methodology for their derivation, as well as practical tools for their use.

This EPR Series publication replaces IAEA-TECDOC-955[2].

1.2. OBJECTIVE

The objective of this publication is to provide guidance and tools on the symptom and event based emergency classification, as part of the A&P of a nuclear emergency at an NPP or SFP. To support this overall objective, this publication describes:

- The necessary information to be provided to on-site decision makers to support the declaration of an emergency class;
- The types of EAL on which the methodology for assessing an emergency class is based;
- How to determine the appropriate emergency class for any given on-site condition, within a predefined emergency classification process and based on the use of EALs;
- How to anticipate the potential progression of the on-site conditions and, consequently, a change in the emergency class;
- The A&P methodology to assess the potential consequences and the possible progression of a nuclear emergency at an NPP.

1.3. SCOPE

The scope of this publication is restricted to facilities in Emergency Preparedness Category I (such as NPPs and SFPs: see Table 1 of GSR Part 7 [6]), for which on-site events are postulated that could give rise to severe deterministic effects off the site that would warrant precautionary

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[1] The concept of intervention levels and generic action levels was first published in GS-R-2, titled Preparedness and Response for a Nuclear or Radiological Emergency, which has been superseded by GSR Part 7 [6].
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Assessment Procedures for Determining Protective Actions During a Reactor Accident, IAEA-TECDOC-955, IAEA, Vienna (1997).

urgent protective actions[3], urgent protective actions[4], or early protective actions[5], and other response actions. This publication was primarily developed considering NPPs in an initial state (i.e. prior to the emergency triggering event) at full nominal thermal power, and SFPs fully loaded including spent fuel freshly removed from a reactor. Numerical values given in this publication refer to those initial states and might be too conservative for other initial states (e.g. shutdown or refuelling)[6]. However, the concepts, methodology for A&P, as well as structure and navigation in EALs may apply to all reactor states regardless of whether or not a reactor is initially at full nominal power or an SFP at maximum residual heat.

This publication provides guidance and tools for the A&P of a nuclear emergency based on on-site observables and predetermined criteria. It does not consider the emergency alert and notification, activation of emergency response organizations, or implementation of specific on-site and/or off-site protective actions.

This publication is aimed at emergency preparedness and response personnel at various levels (government, national and local response organizations, nuclear operators and regulatory bodies) responsible for providing and/or analysing information on the status of an unfolding emergency at an NPP or SFP. The goal is to support the decision making process regarding the protection of the public, plant personnel and emergency workers.

1.4. STRUCTURE

Section 1 introduces the publication, including its scope and objectives. Section 2 provides an overview of the use of symptom and event based criteria in the A&P process, including a discussion of the application of the A&P methodology and the use of the IAEA's Reactor Assessment Tool for this purpose. It also provides the definitions for the emergency classes used in the remainder of the publication. Section 3 defines the EALs for each emergency class across four categorizations, namely 'Radiation and Dose Levels', 'Fission Product Barriers', 'Conventional Emergencies, Natural Events, Security Events', and 'Safety Systems and Equipment'. Section 3 also provides details on the ways in which each EAL may be met during an emergency situation, and demonstrates the escalation criteria between each emergency class in the form of four operational flowcharts.

The Appendix provides guidance on the assessment of an occurrence of a loss of coolant accident or a steam generator tube rupture, which might result in a significant release of radioactive material to the environment.

Annex I presents the orders of magnitude of the extent of core damage, based on the length of time that spent fuel is uncovered, either partially or completely. Annex II presents ranges of containment radiation levels for core melt and gap release situations for a number of different

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[3] An urgent protective action taken before or shortly after a release of radioactive material, or an exposure, on the basis of the prevailing conditions to avoid or to minimize severe deterministic effects.

[4] A protective action in the event of a nuclear or radiological emergency which must be taken promptly (usually within hours to a day) in order to be effective, and the effectiveness of which will be markedly reduced if it is delayed.

[5] A protective action in the event of a nuclear or radiological emergency that can be implemented within days to weeks and can still be effective.

[6] In IAEA-TECDOC-955, numerical time criteria were doubled from status at full power to cold shutdown or refuelling states.

reactor types. Annex III references a supplementary file that presents an overarching flowchart for a better use of the four operational flowcharts.

2. EMERGENCY CLASSIFICATION AS PART OF THE ASSESSMENT AND PROGNOSIS PROCESS

This publication describes the use of specific, predetermined, observable on-site criteria (e.g. an instrument reading, equipment status, or other observable occurrences such as a fire or earthquake) to identify the appropriate emergency class. These criteria are called emergency action levels (EALs). For a given emergency class, each response organization has predetermined response arrangements. The highest classification is a general emergency, which warrants taking precautionary urgent protective actions, urgent protective actions, early protective actions and other response actions on the site and off the site. As such, the declaration of a general emergency constitutes a simple and unambiguous manner to present information for a decision maker on the need to promptly implement public protective actions.

The process for conducting the A&P of a nuclear emergency at an NPP is also described, based on relevant on-site information (and, to a limited extent, off-site information such as weather conditions or weather data). The A&P methodology for a nuclear emergency at an NPP relies on the analysis and anticipation of the status of the NPP fission product (FP) barriers and the systems associated with them. Such on-site information and related insights are crucial for estimating a source term or characterizing a radioactive release. After the implementation of precautionary urgent protective actions or urgent protective actions, as applicable, technical results from A&P may help refine or update decisions on taking public protective actions and other response actions. For instance, technical results from A&P may suggest expanding the implementation distances of protective actions.

2.1. ON-SITE AND OFF-SITE RESPONSE ORGANIZATION INTERFACE

The on-site and off-site decision makers need to coordinate their efforts in the development, implementation and maintenance of emergency preparedness and response arrangements with respect to the following:

- The emergency management system (see Requirement 1 of GSR Part 7 [6]), which needs to provide a systematic, proactive approach to guide departments and agencies at all levels of government, non-governmental organizations and the private sector to work together seamlessly and manage emergencies involving all threats and hazards, regardless of cause, magnitude, location, or complexity, in order to reduce loss of life, property and harm to the environment.
- The emergency roles and responsibilities, such that within the framework of the emergency management system and applicable regulations and guidance, the roles and responsibilities are identified and agreed upon between the on-site and off-site emergency response organizations under the national coordination mechanism (see para. 4.10 of GSR Part 7 [6]).
- The hazard assessment (see Requirement 4 of GSR Part 7 [6]), such that it identifies all sources of exposure to radiation, analyses the event sequences and radiation doses that could be received by workers and the public. Non-radiation-related hazards also need to be identified in the hazard assessment.
- The protection strategy (see Requirement 5 of GSR Part 7 [6]), developed and understood by all response organizations within the emergency management system.

The on-site and off-site decision makers need to coordinate the development of a ‘response interface’ to maintain a common understanding of the following:

- The predefined emergency notification process (see Requirement 7 of GSR Part 7 [6]), such that the on-site decision maker can initiate immediate communication with the off-site notification contact point, so that the decision maker in charge can decide on, and order the implementation of, public protective actions in a timely manner.
- The predefined approach for the declaration of a given emergency class, such that the on-site and off-site decision makers are knowledgeable about the process through which specific, predetermined, observable criteria are used to detect events, recognize symptoms and determine the emergency class.
- The predefined characterization process of an actual or anticipated radioactive release.

2.2. SYMPTOM BASED AND EVENT BASED CRITERIA

Symptom based parameters or conditions that are measurable over some range using plant instrumentation, such as core exit temperature, reactor coolant level or activity concentration at the stack, are used for the majority of EALs to identify the appropriate emergency class. Event based conditions that are easily identifiable and have potential or actual safety significance, such as station blackout, fire or earthquake, and security based events, are also used to identify the appropriate emergency class. The EALs are to include thresholds that will allow the operating organization to declare the appropriate emergency class promptly, and with a minimum of effort.

2.3. ASSESSMENT AND PROGNOSIS PROCESS FOR A NUCLEAR EMERGENCY AT A NUCLEAR POWER PLANT

2.3.1. Principles of the IAEA’s assessment and prognosis methodology

The A&P methodology for an NPP emergency relies on the analysis of the actual and future status of the NPP’s FP barriers. An FP barrier is defined as a physical barrier placed between a radiation source or radioactive material and workers, members of the public or the environment [8][7]. Most NPPs in operation have three FP barriers. To each FP barrier, one or more safety functions are associated for the purpose of A&P. For the purpose of emergency response, ‘critical safety functions’ are considered. Critical safety functions include, but are not limited to, the fundamental safety functions considered in the NPP design and safety demonstration[8].

For a three barrier reactor[9], the three FP barriers considered, and their related critical safety functions, are given in Table 1.

[7] See also ‘defence in depth’ in Ref. [8].

[8] For the purpose of emergency response, a ‘critical’ safety function is a function that ensures the integrity of its related FP barrier. However, an FP barrier can be damaged and a related critical safety function can still be satisfactory (e.g. a medium size break on the RCS means the damage of the second FP barrier on a pressurized Water Reactor (PWR); however, the critical safety function ‘RCS heat removal’ can still be satisfactory due to the size of the break).

[9] Such as PWR, boiling water reactor (BWR), water cooled, water, moderated power reactor (WWER).

TABLE 1. FP BARRIERS AND CRITICAL SAFETY FUNCTIONS OF A THREE FP BARRIER REACTOR

Fission product barriers	Critical safety functions
Fuel (and cladding)	Reactivity control RCS inventory control
Reactor coolant system (RCS)	RCS heat removal
Containment building (and its possible extensions)	Containment heat removal Isolation Hydrogen control

A fourth FP barrier can be considered for a pressurized heavy water reactor/Canada deuterium–uranium reactor (PHWR/CANDU) for the purpose of emergency response. Indeed, in addition to the fuel matrix and the fuel clads (first FP barrier), the pressurized heat transport system (PHTS) (second barrier) and the containment building (fourth FP barrier), the calandria vessel (third FP barrier) is considered a physical barrier between the fuel and the environment. Therefore, for a four barrier reactor, the four FP barriers considered, and their related critical safety functions, are listed in Table 2.

TABLE 2. FP BARRIERS AND CRITICAL SAFETY FUNCTIONS OF A FOUR-FP BARRIER REACTOR

Fission product barriers	Critical safety functions
Fuel (and cladding)	Reactivity control Core inventory control
PHTS	PHTS heat removal
Calandria vessel	Calandria vessel heat removal
Containment building (and its possible extensions)	Containment heat removal Isolation Hydrogen control

For all NPP technologies with at least three FP barriers, if two, or more, FP barriers are no longer intact or are no longer performing their function, there might be a release of radioactive material in the environment (i.e. there might be a pathway for FPs from the reactor core to be released to the environment). This principle constitutes the basis for the IAEA's A&P process for a nuclear emergency at an NPP.

2.3.2. Assessment of a nuclear power plant emergency

Assessing the status of each FP barrier (is the FP barrier still intact?) allows a conclusion on the likelihood of a past or a current release. Then, assessing the status of the critical safety functions related to the FP barriers facilitates a broader and deeper understanding of the technical situation. Based on this assessment and on the emergency class declared, the actual release condition can be characterized.

Based on the technical information shared by the NPP operating organization, an assessment of the status of the FP barriers can be performed. Then, the characterization of the critical safety functions related to each FP barrier provides a further understanding of the situation: what is/are the nature(s) of the identified failure(s) (is it regarding injecting coolant into the reactor vessel? Is it regarding removing the heat from the RCS? Is it regarding containing of the FPs?). Finally, assessing the equipment or systems that ensure each critical safety function (e.g. the emergency core cooling system (ECCS) for injecting coolant into the reactor vessel) helps in preparing an inventory of the key equipment and systems which are available and operational, and on which the on-site situation is reliant.

The emergency class declared is part of the initial information shared by the NPP operating organization with off-site decision maker(s). In fact, it is likely that the very first emergency communication to off-site decision maker(s) will only include the name and location of the affected NPP, the emergency class declared along with the date and time of the emergency declaration. Subsequent emergency communications to off-site decision makers will include a description of, or an assumption on, the triggering event and a summary description of the emergency progression. Therefore, the emergency class declared constitutes one of the first pieces of information shared with off-site decision maker(s) at the beginning of the urgent response phase, and on which the assessment of the affected NPP is reliant.

2.3.3. Prognosis of a nuclear power plant emergency

The prognosis process leverages step 3 of the assessment process (see Fig. 1). The three steps of the assessment and the three steps of prognosis follow a ‘mirror approach’, as illustrated in Fig. 2.

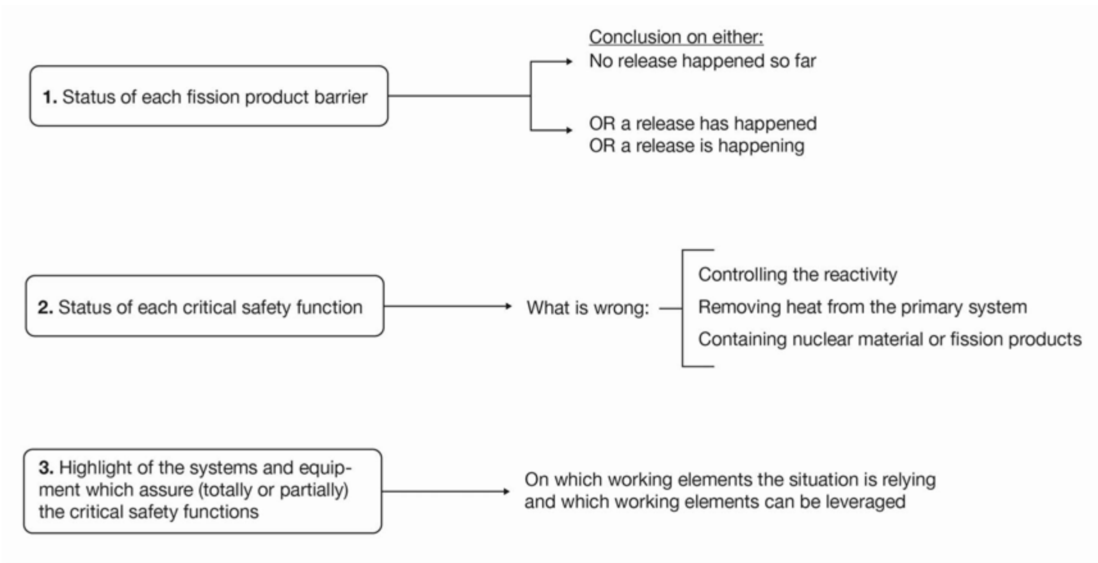

FIG. 1. Process to perform an assessment during a nuclear emergency at an NPP.

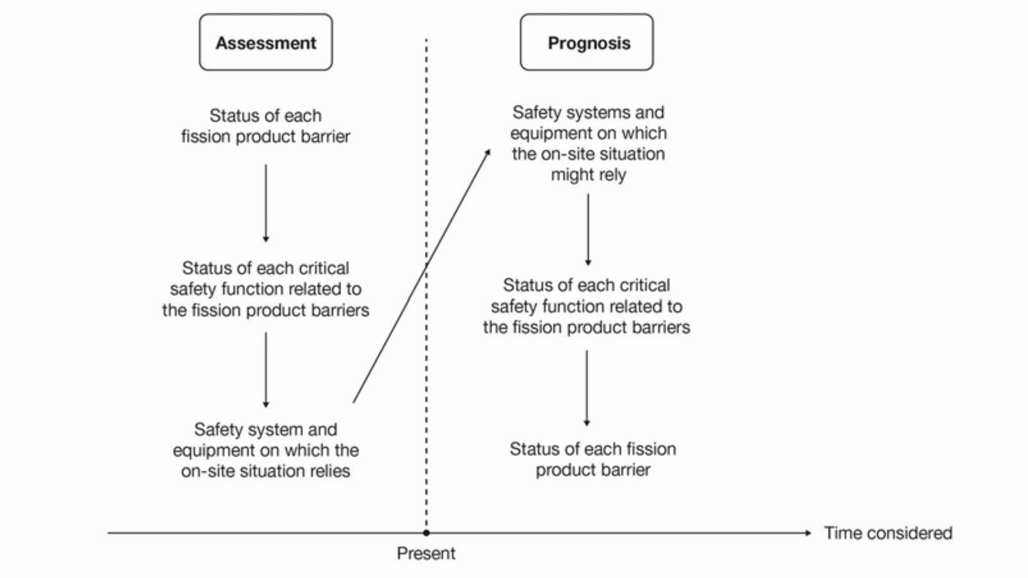

FIG. 2. Mirror approach between the assessment and the prognosis.

Based on the safety systems and equipment available at the reactor at a given time, possible developments can be identified following this general question:

> Will this equipment, which is currently operating/available, still be operating/available in the future? If not, will this loss have an impact (degradation or failure) on the related critical safety function? In this case, will this degradation or failure of a critical safety function have an impact on the integrity of the related FP barrier or on the severity of a possibly current release?

Therefore, one output of the prognosis is a possible anticipation of a change in the emergency class declared. Indeed, if the assessment of a given NPP emergency concludes on the absence of a past or current release, but the prognosis concludes on a future release, this means that the emergency class is likely to change. These outputs from the A&P need to be shared with off-site decision makers so that they can plan emergency response actions ahead of time. Restored or fixed safety equipment, as well as mobile means, can then be given credit in the prognosis once the confirmation has been received that the safety equipment is connected/installed and effectively working. The prognosis approach is summarized in Fig. 3.

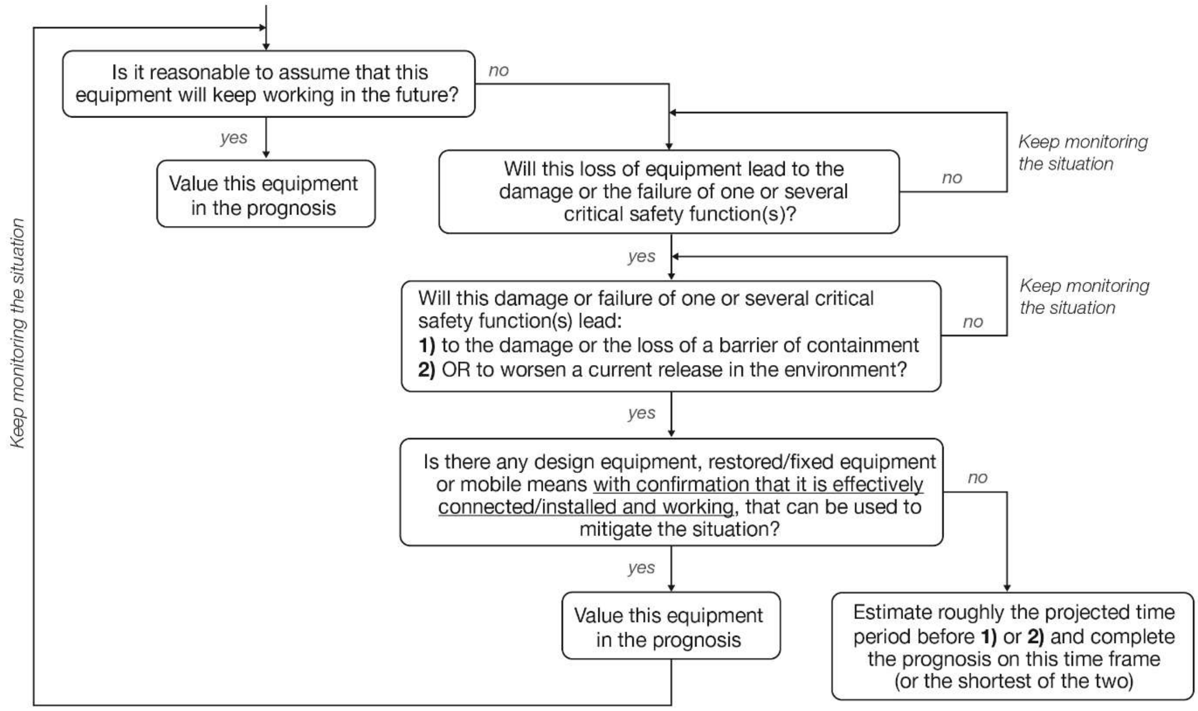

FIG. 3. Process to answer the first question of a prognosis — Is a release going to happen?

A classic question is: **What time frame needs to be covered by the prognosis?** For example, will the prognosis try to anticipate the next two hours or the next 48 hours? A specific time frame is not to be considered a priori when starting a prognosis; instead **the relevant time window will be determined by completing the prognosis process.** If the assumed loss of equipment leads to the damage or the failure of a critical safety function which results in a key event relevant for the protection of the public (i.e. the damage or the loss of an FP barrier), then this event needs to be a time marker and possibly be defined as a key milestone in the prognosis window. If several assumed losses of equipment lead to key events, the shortest prognosis window defined by one of these losses needs to be considered in order to ‘stress’ the timeline of the emergency response and to anticipate possible developments of the situation in a conservative manner.

Table 3 presents a typical list of equipment related considerations necessary to perform a prognosis.

TABLE 3. EXAMPLES OF QUESTIONS TO RAISE WHEN INITIATING A PROGNOSIS FOLLOWING A COMPLETED ASSESSMENT

Equipment available and working at the time of the assessment	Classic questions to raise when performing a prognosis
A single ECCS pump is working and is able to compensate for the leak from a primary break (ensuring the critical safety function 'RCS inventory control')	Will this final pump keep working? — What about the projected time period before the water tank is empty? — Is it technically possible to switch to recirculating mode? — Is the motor bearing temperature of the pump increasing?
After a station blackout, the power supply relies on emergency diesel generators and batteries	What is the projected time period before the diesel generators and batteries are no longer working?
After a total failure of feeding the steam generators (SGs) with water, the heat can still be removed from the primary circuit due to the large volume of liquid water still present in the SGs	How long before the SGs are ineffective or need to be isolated by the operators in the control room?

Assuming that a release to the environment is likely, this release needs to be characterized. Many technical questions can be raised about the future emergency developments. However, only a few need be considered for the purpose of characterizing the future release in the emergency response phase. There are three core questions that need to be considered (the characterization of an anticipated release in the emergency response phase would not be complete if not based on all three questions below):

(1) **When** is the anticipated release expected to occur?
(2) **How long** is the expected duration of the release?
(3) **How much radioactive material will be released and are public protective actions expected to be necessary?**

It might be difficult to provide accurate numerical answers to these questions, which involve fast running validated calculation tools. However, answers to them can be based on expert judgement using relevant criteria. **The criteria listed in Sections 2.3.3.1, 2.3.3.2 and 2.3.3.3 are not to be considered as strict numerical values.** Response to a nuclear or radiological emergency is a national responsibility. Depending on the national emergency framework and arrangements, the criteria listed in Sections 2.3.3.1, 2.3.3.2 and 2.3.3.3 may be chosen within a certain interval.

After the declaration of a General Emergency, **following the A&P process, obtaining replies to the prognosis questions needs to happen in parallel with taking precautionary urgent protective actions and urgent protective actions**. The overall conduct of A&P cannot delay such off-site public protective actions taken on the basis of prevailing conditions at the facility.

2.3.3.1. *When is the anticipated release expected to occur?*

The purpose of this question is to estimate the time period before a release starts. A useful approach is to characterize the pending release as either an immediate release or a deferred release, i.e. depending on whether the release is going to occur soon or later. 'Soon' is understood as being too early to be able to complete an evacuation of people within the precautionary action zone (PAZ)[10] (and possibly extended to a larger area within the urgent protective action planning zone (UPZ)[11]) before the start of the anticipated release. 'Later' is understood as sufficient time, estimated to be available to complete the evacuation of people within the PAZ, possibly extended to a larger area within the UPZ, before the start of the anticipated release.

Discussions with experts from several Member States involved in the A&P role at the national level led to the conclusion that, typically, the border between 'soon' and 'later' can be set around ten hours after the decision maker in the accident State has received the recommendation for protective actions (see Fig. 4) or, for the IAEA, after the information was received on the need to implement public protective actions.

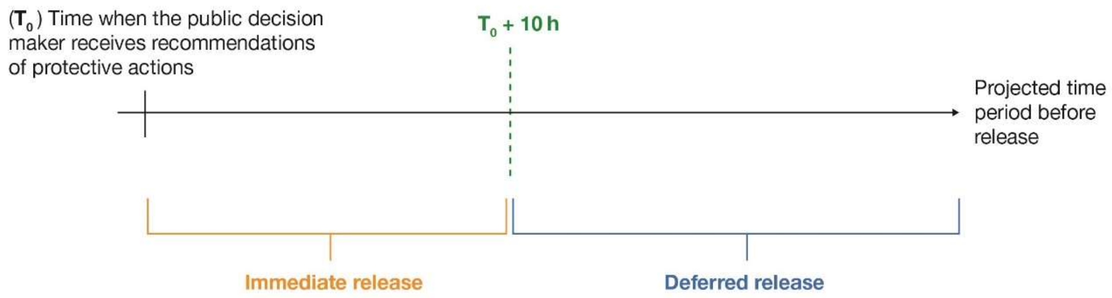

FIG. 4. Characterizing an anticipated release — When will it happen?

A ten hour marker is suggested as a reasonable amount of time needed by domestic authorities to successfully implement the first actions requested for an evacuation. According to the input received from Member States, there is a significant risk of failing to complete all the requested actions in less than ten hours before the release.

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[10] The PAZ is defined as an area around a facility for which emergency arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avoid or to minimize severe deterministic effects off the site. Protective actions within this area are to be taken before or shortly after a release of radioactive material or an exposure, on the basis of prevailing conditions at the facility [8].

[11] The UPZ is defined as an area around a facility for which arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avert doses off the site, in accordance with international safety standards. Protective actions within this area are to be taken on the basis of environmental monitoring or, as appropriate, prevailing conditions at the facility [8].

2.3.3.2. How long is the expected duration of the release?

The purpose of this question is to estimate the duration of a release based on a prognosis. A 'short term' release is a 'few hours' release, while a 'long term' release can continue for days.

Conducting an evacuation in response to an anticipated release that is projected to be 'immediate' (see 2.3.3.1) and 'short', especially under degraded off-site conditions (e.g. bad weather, degraded transport or communications following a natural event), might not be justified in the response phase. More particularly, in this situation, evacuation of special population groups (e.g. hospitals, nursing homes, schools) is not advised. On the contrary, being able to anticipate a 'long term' release, which is expected to last over several hours (or days), is very useful for the decision maker who needs to implement actions to protect the public with sufficient resources allocated accordingly.

An effective approach to answer this question is to characterize the upcoming release as either a short term release or a long term release.

The implementation of an evacuation might result in non-radiological consequences, such as over-stressed people, possible casualties among vulnerable people, or road accidents. Therefore, it may be useful to assess whether or not the release duration will be a severe health hazard to the general public if people do not evacuate and stay sheltered at home.

In addition, an anticipated long term release might occur with changing weather conditions, which notably increase the uncertainty of the projected atmospheric dispersion of the release. In this case, implementing the evacuation of people within the PAZ extended to a larger area within the UPZ may need to be considered as a precaution.

The case of a night evacuation can be used to develop a criterion to differentiate a short term and long term release. Considering the potential for non-radiological hazards due to the implementation of an evacuation (potentially at night and under bad weather conditions), a release starting at the beginning of the night but ending before daybreak might not present a risk high enough to require the public to leave (evacuate) their homes (shelters) or to implement the evacuation of special population groups. Therefore, assuming an average night of eight hours, an eight hour criterion between a short term release and a long term release can be defined (Fig. 5).

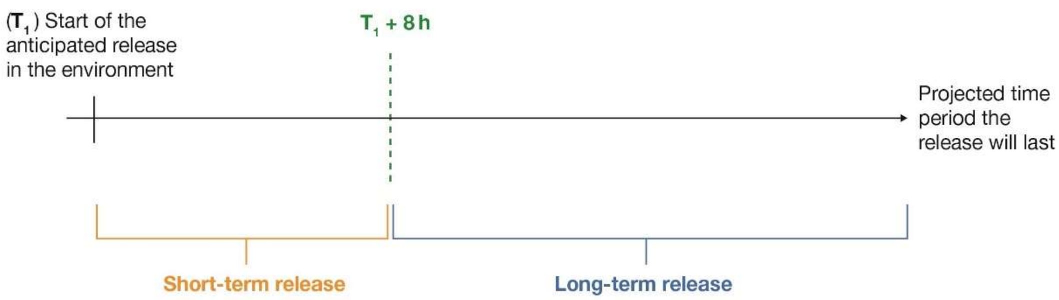

FIG. 5. Characterizing an anticipated release — How long will it last?

2.3.3.3. *How much radioactive material will be released?*

The purpose of this question is to estimate the order of magnitude of the projected source term[12] related to the release based on a prognosis. Indeed, the previous two questions ('When?' and 'How long?') characterize the release in terms of times (start and duration). This question is the first step to characterize the release in terms of impacted areas (considering the extent of the atmospheric dispersion of the release).

The options to characterize the upcoming release are as follows:

(1) The release is not expected to be of radiological safety significance[13];
(2) The radiological safety significance of the release is not expected to extend beyond the PAZ;
(3) The radiological safety significance of the release is expected to extend beyond the PAZ.

The PAZ needs to be defined and established in the preparedness phase. In the response phase, either pre-calculated results of the magnitude and environmental dispersion of the release, or live, fast-running calculations of the expected source term and its dispersion into the environment, need to be performed to assess whether the radiological safety significance of the release is likely to extend beyond the PAZ.

The off-site decision maker needs to be promptly informed once the radiological safety significance of an anticipated release is likely to extend beyond the PAZ. This will maximize the time available to implement public protective actions and other response actions, as needed, beyond this zone.

2.3.4. The IAEA Reactor Assessment Tool

The A&P methodology described in subsections 2.3.1–2.3.3 has been implemented in the IAEA Reactor Assessment Tool, a web-based tool hosted on the IAEA A&P tools website, which aims to support the response to a nuclear emergency at an NPP. Access to this tool can be granted to experts in Member States on request.

2.4. EMERGENCY CLASSES

Emergency classes are used for communicating the level of response needed for the response organizations. The situations that belong to a given emergency class are defined by the facility, source, or practice specific criteria, which, if exceeded, trigger classification at the prescribed level. For each emergency class, the initial actions of the response organizations are predefined.

There are four emergency classes, which in order of increasing severity are: alert, facility emergency, site area emergency and general emergency. The definitions of all four emergency classes from para. 5.14 of GSR Part 7 [6], are as follows:

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[12] A source term is a defined as "the amount and isotopic composition of radioactive material released (or postulated to be released) from a facility" [8].

[13] The term "radiological safety significance" is used in the Convention on Early Notification of a Nuclear Accident. For the purpose of this document, a release with radiological safety significance is understood as a release that leads to severe deterministic effects and/or an increase in the risk of stochastic effects off the site.

- **Alert**.“[A]n event that warrants taking actions to assess and to mitigate the potential consequences at the facility. Upon declaration of this emergency class, actions shall promptly be taken to assess and to mitigate the potential consequences of the event and to increase the readiness of the on-site response organizations.”
- **Facility emergency.** “[A]n emergency that warrants taking protective actions and other response actions at the facility and on the site but does not warrant taking protective actions off the site. Upon declaration of this emergency class, actions shall promptly be taken to mitigate the consequences of the emergency and to protect people at the facility and on the site. Emergencies in this class do not present an off-site hazard.”
- **Site area emergency.** “[A]n emergency that warrants taking protective actions and other response actions on the site and in the vicinity of the site. Upon declaration of this emergency class, actions shall promptly be taken: (i) to mitigate the consequences of the emergency on the site and to protect people on the site; (ii) to increase the readiness to take protective actions and other response actions off the site if this becomes necessary on the basis of observable conditions, reliable assessments and/or results of monitoring; and (iii) to conduct off-site monitoring, sampling and analysis.”
- **General emergency.** “[A]n emergency that warrants taking precautionary urgent protective actions, urgent protective actions, and early protective actions and other response actions on the site and off the site. Upon declaration of this emergency class, appropriate actions shall promptly be taken on the basis of the available information relating to the emergency, to mitigate the consequences of the emergency on the site and to protect people on the site and off the site.”

3. PROCESS FOR NUCLEAR POWER PLANT EMERGENCY CLASSIFICATION

3.1. EMERGENCY ACTION LEVEL CATEGORIES

EALs are specific and predetermined criteria for observable on-site conditions used to detect, recognize and determine the appropriate emergency class. They can be symptom based or event based. For NPPs, EALs can be divided into four categories:

- Radiation and dose levels (described in Section 3.2.1);
- Fission product barriers (described in Section 3.2.2);
- Conventional emergencies, natural events, security events (described in Section 3.2.3);
- Safety systems and equipment (described in Section 3.2.4).

The first step is to characterize the emergency situation by determining which of these four EAL categories apply. Note that more than one EAL category might apply to the emergency. The EALs in the applicable category(ies) are then reviewed to determine the emergency class. The highest (i.e. the most severe) emergency class reached from any of the four EAL categories is selected.

Once the regulatory body receives the emergency class declared by the on-site decision maker, it may make its own assessment of the declared emergency class. If there is a need for the regulatory body to receive additional information to fully understand the basis for the emergency class declaration, the on-site decision maker is to be contacted to provide the necessary clarification. The guidance in this publication aims to support such a discussion.

3.2. EMERGENCY ACTION LEVELS

Once an EAL category is selected, the EALs in this category need to be reviewed before reaching a conclusion on an emergency class. If more than one EAL category applies, EALs in each category that applies will need to be reviewed. For ease of use, the navigation through EALs in a given category is captured in the shape of flow charts using Yes/No questions (see Annex III).

The abbreviations for EALs follow the convention below:

- First letter: EAL category
  - ‘L’ refers to the EAL category ‘Radiation and dose levels’;
  - ‘B’ refers to the EAL category ‘Fission product barriers’;
  - ‘E’ refers to the EAL category ‘Conventional emergencies and/or security events’;
  - ‘S’ refers to the EAL category ‘Safety systems and equipment’.
- Second letter: Emergency class
  - ‘A’ refers to ‘Alert’;
  - ‘F’ refers to ‘Facility emergency’.
  - ‘S’ refers to ‘Site area emergency’;
  - ‘G’ refers to General emergency.
- Third letter: Sequential number in the scheme.

The four EAL categories and their respective EALs are shown in four flow charts in Sections 3.2.1–3.2.4.

The overarching emergency classification flow chart to facilitate the use of the four EAL flow charts is available on this publication's individual web page at www.iaea.org/publications.

Additionally, the overarching emergency classification flow chart is provided to facilitate the use of the four EAL flow charts (See Annex III).

3.2.1. Emergency Action Level category 'Radiation and dose levels'

3.2.1.1. Declaration of an 'Alert'

If either of the EALs outlined below (LA1, LA2) is met (assessment), or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Alert'.

LA1: Rise in radiation levels in the facility

This EAL addresses situations in which a rise in radiation levels in the facility has an impact on the following:

- Areas that require continuous occupancy to maintain safe operation. For instance, such a situation might occur in the event of a release of radioactive material (e.g. from a reactor unit or an effluent storage tank on site) that leads to exposure of control room operators.
- The ability to bring the reactor to a safe shutdown state. For instance, a situation might occur in the event of a release of radioactive material that leads to an increase in radiation levels in an auxiliary building or equipment room that do not require continuous occupancy, but in which a technician needs to manually open or close a valve as part of actions to shut down the reactor.

Paragraph 5.55 of GSR Part 7 [6] states:

> "The operating organization and response organizations shall ensure that no emergency worker is subject to an exposure in an emergency that could give rise to an effective dose in excess of 50 mSv other than:
>
> (1) For the purposes of saving human life or preventing serious injury;
> (2) When taking actions to prevent severe deterministic effects or actions to prevent the development of catastrophic conditions that could significantly affect people and the environment;
> (3) When taking actions to avert a large collective dose."

If it is anticipated that an effective dose of 50 mSv will be exceeded, para. 5.57 of GSR Part 7 [6] requires that emergency workers are to decide to perform their duties only after being fully and clearly informed, as well as comprehensively trained (to the extent possible), in advance of the associated health risks, and then choose to do so voluntarily.

LA2: Damage to irradiated fuel, or decreasing water level, in the SFP (fuel assemblies still under water in the SFP)

This EAL addresses situations involving irradiated fuel that still remains under water in the SFP (i.e. the level of coolant in the SFP is still higher than the top of the fuel). These situations include:

— *Suspected or confirmed damage to irradiated fuel.* Such a situation might be identified based on a visual inspection and/or rise in radiation monitoring in the SFP building. For instance, this event might be caused by faulty handling during unloading of an irradiated fuel assembly or by strong mechanical stress to irradiated fuel assemblies (e.g. due to an earthquake).
— *Decreasing level of coolant in the SFP.* Such a situation might be identified based on a visual inspection and/or decrease in the monitored level of coolant. For instance, it might be caused by a significant leak of coolant from a coolant injection pipe or pump, or a crack on the SFP structure.

While a radiation monitor could detect an increase in dose rate (e.g. due to a drop in the water level), it might not be a reliable indication of whether or not the fuel assemblies are still covered. For this reason, increased radiation monitor readings need to be combined with another indicator (or personnel report) of water loss.

3.2.1.2. Declaration of a 'Facility emergency'

If either of the EALs outlined below (LF1, LF2) is met (assessment) or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Facility emergency'.

LF1: On-site radiation levels requiring on-site protective actions

This EAL addresses on-site radiological conditions that necessitate on-site protective actions (either on the whole site or in specific on-site areas/buildings).

Such a situation can be identified based on fixed radiation monitoring in buildings and/or individual dosimeters carried by workers in various on-site areas. It might arise due to a release of radioactive material over the site, or affecting specific on-site areas, caused by a LOCA (loss of coolant accident) at a reactor unit, a LOCA at an SFP or a leak at an effluent storage tank.

LF2: The SFP coolant is saturated

This EAL covers situations where the coolant level in the SFP has started to decrease and the thermal-hydraulic status of the coolant changes from undersaturated to saturated, i.e.

$$T_{Saturation}\,(P_{SFP}) - T_{Fuel\ assemblies} < 0°C$$

where:

$T_{Saturation}\,(P_{SFP})$ is the saturation temperature at the pressure in the SFP building;
$T_{Fuel\ assemblies}$ is the temperature of the fuel assemblies in the SFP.

For an SFP located in a dedicated building (i.e. not located in the same building as a reactor), the value of P_{SFP} would be likely to stay equal to about the atmospheric pressure; therefore, $T_{Saturation}\,(P_{SFP})$ would be approximately 100˚C.

For an SFP located in the same building as a reactor, i.e. a reactor containment building, P_{SFP} can go higher than the atmospheric pressure, for instance in case of a primary break LOCA on the RCS. In such a case, $T_{Saturation}\,(P_{SFP})$ would be higher than 100˚C.

The Appendix provides guidance on how to assess the occurrence of a primary break LOCA at a reactor.

3.2.1.3. *Declaration of a 'Site area emergency'*

If either of the EALs outlined below (LS1, LS2) is met (assessment), or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Site area emergency'.

LS1: Radioactive release resulting in actual or projected off-site doses reaching (at least) 10% of a domestic generic criterion to take urgent protective actions

This EAL addresses situations in which a radioactive release to the environment results in actual or projected off-site doses that are equal to, or higher than, 10% of a generic criterion to take urgent protective actions.

On site, such a situation may be identified by monitoring the ambient dose rate in the SFP building. A typical order of magnitude of ambient dose rate for an operator to declare a 'Site area emergency' is a few tens of milligray per hour (mGy/h). Off-site, monitoring environmental dose rate values exceeding the order of magnitude of 1 microsievert per hour (µSv/h) might provide additional information to identify such a situation. Off-site decision makers need to consider using the corresponding values for generic criteria as established by the regulatory body.

As stated in Table II.2 of GSR Part 7 [6], generic criteria for taking protective actions and other response actions in an emergency to reduce the risk of stochastic effects include levels for the projected dose, or the dose that has been received, that exceed the following:

— 100 mSv effective dose in the first seven days;
— 50 mSv equivalent dose to the thyroid in the first seven days.

Corresponding values to avoid or to minimize severe deterministic effects are higher (see Table II.1 of GSR Part 7 [6]).

LS2: Fuel assemblies are still under water in the SFP (the SFP coolant is saturated, its level is below its nominal level and keeps decreasing)

This EAL addresses situations in which the tops of the fuel assemblies are still under water in the SFP; however, the thermohydraulic status of the SFP coolant is degraded (i.e. saturated), the SFP coolant level is below its nominal level and the decrease cannot be stopped.

Unless a sudden and prompt draining of an SFP is considered, an emergency at the SFP would lead to a decrease in the coolant level and the degradation of its thermohydraulic status, as follows:

— In the case of a station blackout or complete loss of cooling capabilities, provided the thermal power stored in the SFP is high enough, the saturation of the coolant would happen first, followed by a decrease in the coolant level by evaporation and boiling.
— In the case of a break leading to a LOCA at the SFP, the decrease in the coolant level would happen first, followed by coolant saturation (before reaching the top of the fuel assemblies).

Regardless of the order in which these two events take place, if the SFP coolant inventory cannot be compensated (i.e. stabilized or increased) through coolant injection after it has started decreasing and has reached saturation, escalation to 'General emergency' is very likely.

3.2.1.4. *Declaration of a 'General emergency'*

If either of the EALs outlined below (LG1, LG2) is met (assessment) or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'General emergency'.

LG1: Radioactive release resulting in actual or projected off-site doses above a domestic generic criterion to take urgent protective actions

This EAL addresses situations in which a radioactive release to the environment results in actual or projected off-site doses that exceed a generic criterion to take urgent protective actions.

On site, such a situation may be identified by monitoring the ambient dose rate in the SFP building. A typical order of magnitude of ambient dose rate for an operator to declare a 'Site area emergency' is a few Grays per hour (Gy/h). Off-site, monitoring environmental dose rate values exceeding the order of magnitude of 100 microsievert per hour (μSv/h) may provide additional information to identify such a situation. Off-site decision makers need to consider using the corresponding values for generic criteria as established by the regulatory body.

As stated in Table II.2 of GSR Part 7 [6], generic criteria for taking protective actions and other response actions in an emergency to reduce the risk of stochastic effects include levels for the projected dose, or the dose that has been received, that exceed the following:

— 100 mSv effective dose in the first seven days;
— 50 mSv equivalent dose to the thyroid in the first seven days

Corresponding values to avoid or to minimize severe deterministic effects are higher (see Table II.1 of GSR Part 7 [6]).

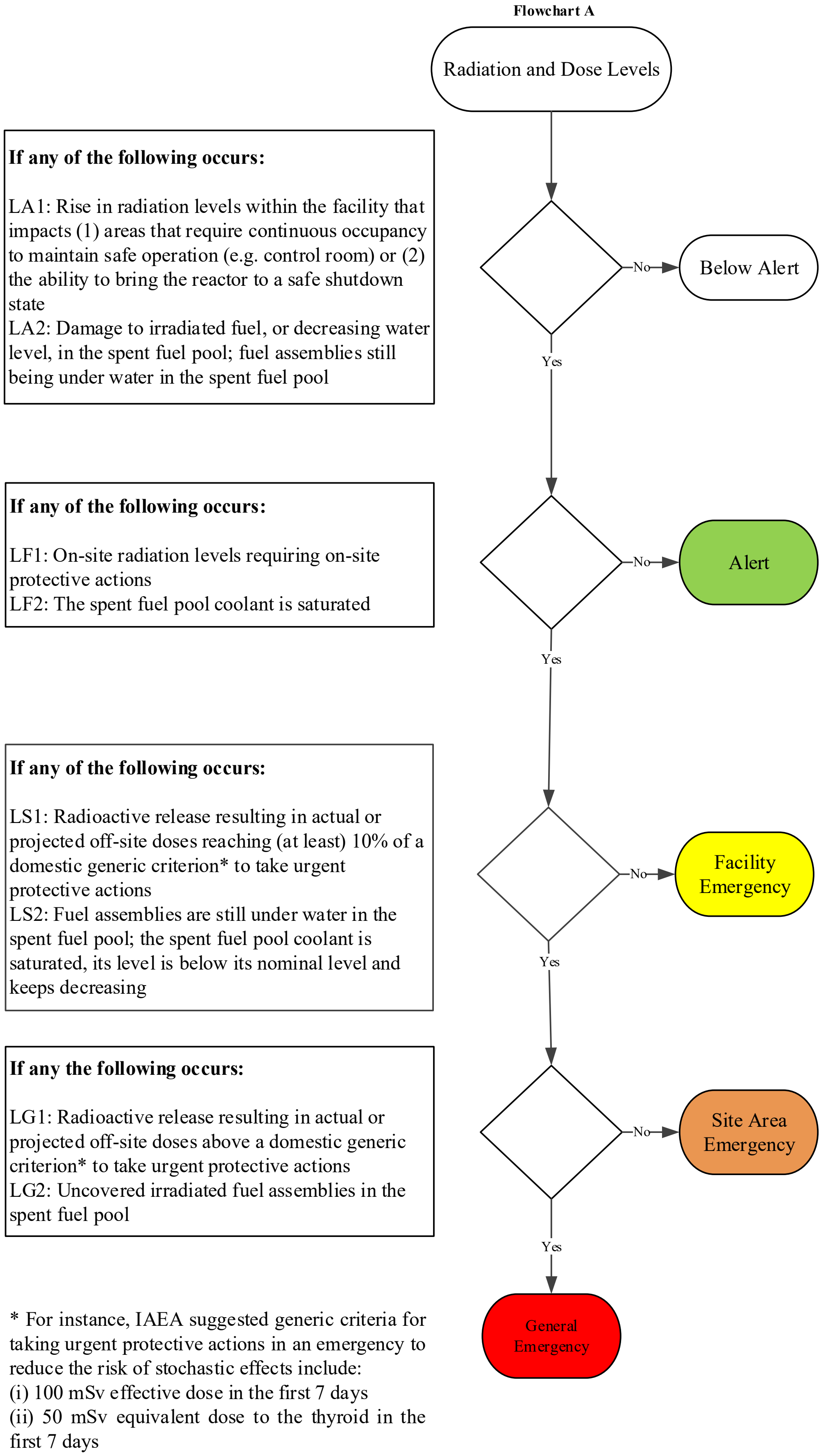
LG2: Uncovered irradiated fuel assemblies in the spent fuel pool

This EAL addresses situations in which irradiated fuel assemblies in the SFP have become uncovered (i.e. not covered by water). As soon as fuel assemblies are no longer fully covered by water, conservatively a gap release can be considered to start. The gap release is the release of FPs accumulated in the fuel pin gap. The gap release starts immediately after the failure of the fuel cladding.

The time at which the top of the fuel assemblies are expected to become uncovered can be assessed following the evolution of the gamma dose rate monitoring results in the building (a dramatic increase in the gamma dose rate would be likely to indicate the start of fuel degradation). However, inconsistent monitor readings might happen (e.g. due to a faulty monitor or by detection of radiation from a contaminated system nearby). For that reason, increased radiation monitor indications need to be combined with another indicator (e.g. personnel report) of water loss.

The extent of damage to spent fuel can be assessed based on the duration during which spent fuel is (partially or completely) uncovered. Annex I presents (in orders of magnitude) the extent of core damage depending on that duration.

3.2.1.5. 'Radiation and dose levels' flowchart



3.2.2. Emergency Action Level category 'Fission product barriers'

Section 2.3.1 elaborates the term 'FP barrier' and provides a list of these barriers considered for NPP designs.

3.2.2.1. Declaration of an 'Alert'

If the EAL outlined below (BA1) is met (assessment), or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Alert'.

BA1: Loss of integrity of at least one FP barrier

This EAL addresses situations in which at least one FP barrier is no longer intact. This might be due to:

— Mechanical failure or damage of an FP barrier (e.g. a steam generator tube rupture (SGTR)).
— Active (by the control room operators) or passive (due to overpressure) opening of a valve located on the envelope of a barrier.
— Failure of a valve (located on the envelope of a barrier) to close following a protection signal and/or an action by the control room operators.

The loss of integrity of the fuel barrier (alone) can be identified based on the following factors (they do not constitute an exhaustive list):

— Chemical analysis of the coolant sample showing that the coolant activity is higher than expected (the coolant activity expressed in 'equivalent I-131' is of particular interest).
— A sharply increasing, or already high, core exit temperature (e.g. for a 1000 MW(e) PWR and, for the purpose of emergency response, a core exit temperature reaching 700°C is considered as corresponding to the start of clad burst).
— The saturation margin monitored in the control room showing that the reactor coolant is overheated (i.e. $T_{Saturation}$ (P_{RCS}) – $T_{Core\ Exit}$ << 0 °C).
— The reactor coolant level reaching the top of the fuel assemblies (this applies to BWRs, PWRs and WWERs).

The loss of integrity of the RCS (alone) can be identified based on the following factors (they do not constitute an exhaustive list):

— Opening of the pressurizer relief valves due to overpressure in the RCS.
— An alarm indicating high activity in the steam phase of an SG (that is, the symptom of an SGTR with activity from the primary coolant being transferred to the damaged SG).
— Decreasing primary pressure in the RCS and/or decreasing coolant level in the pressurizer and/or decreasing coolant level in the reactor vessel and/or automatic start of the high head ECCS/safety injection pumps.
— Increasing coolant level and/or temperature in the reactor building sumps.
— Increasing temperature and/or pressure in the reactor building and/or automatic start of the containment spray system.

The loss of integrity of the containment building (alone) can be identified based on the following factors (they do not constitute an exhaustive list):

- Its isolation is not fully successful, i.e. at least one isolation valve outside (e.g. in a BWR, the main steam isolation valve) or inside containment failed to close, either automatically (e.g. on the signal of automatic ECCS start), or manually by action of the control room operators.
- There is a bypass of the containment building through which radioactive material can be released to the environment (e.g. in the event of an SGTR together with an outside of containment steam valve stuck open on the steam line of the affected SG).
- The mechanical integrity of the containment building is challenged (e.g. visible crack on the concrete following a severe earthquake).

3.2.2.2. Declaration of a 'Facility emergency'

If the EAL outlined below (BF1) is met (assessment) or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Facility emergency'.

BF1: Loss of integrity of the fuel barrier or the RCS barrier

This EAL addresses situations in which the integrity of the fuel or the RCS FP barrier is lost. In the event of a loss of integrity of the fuel or the RCS FP barrier, BA1 and BF1 are met. However, in the event of a loss of integrity of the containment building, BA1 is met but BF1 is not. This reflects the principle of the A&P methodology described in Section 2.3.1: if at least two FP barriers are no longer intact, a release to the environment might be possible. In this minimum of two FP barriers, the fuel or RCS barrier needs to be included for an emergency to involve FPs from the core or the primary coolant.

The loss of integrity of the fuel barrier (alone) can be identified based on the following factors (they do not constitute an exhaustive list):

- Chemical analysis of the coolant sample showing that the coolant activity is higher than expected (the coolant activity expressed in 'equivalent I-131' is of particular interest for that purpose).
- The sharply increasing, or already high, core exit temperature (e.g. for a 1000 MW(e) PWR and, for the purpose of emergency response, a core exit temperature reaching 700°C is considered as corresponding to the start of clad burst).
- The saturation margin monitored in the control room showing that the reactor coolant is overheated (i.e. $T_{Saturation}$ (P_{RCS}) – $T_{Core\ Exit}$ << 0 °C).
- The reactor coolant level reaching top of the fuel assemblies (this applies to BWRs, PWRs and WWERs).

The loss of integrity of the RCS (alone) can be identified based on the following factors (they do not constitute an exhaustive list):

- Opening of the pressurizer relief valves due to overpressure in the RCS;
- An alarm indicating high activity in the steam phase of an SG (that is, the symptom of an SGTR with activity from the primary coolant being transferred to the damaged SG);
- Decreasing primary pressure in the RCS and/or decreasing coolant level in the pressurizer and/or decreasing coolant level in the reactor vessel and/or automatic start of the high head ECCS/safety injection pumps;

- Increasing coolant level and/or temperature in the reactor building sumps;
- Increasing temperature and/or pressure in the reactor building and/or automatic start of the containment spray system.

3.2.2.3. Declaration of a 'Site area emergency'

If the EAL outlined below (BS1) is met (assessment), or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Site area emergency'.

BS1: Loss of integrity of at least two FP barriers

This EAL addresses situations involving the loss of integrity of at least two distinct FP barriers (regardless of which barriers). As explained in Section 2.3.1, in such a situation, a release of radioactive material to the environment might be possible.

As with BF1, BS1 is met in the event of the loss of:

- Both the fuel and the RCS FP barriers;
- The fuel or the RCS FP barrier and another FP barrier (i.e. the containment building for a PWR, BWR or WWER; or either the containment building or the calandria vessel for a PHWR/CANDU).

Regarding the loss of both the fuel and the RCS FP barriers, this can be assessed based on a dramatic increase in the gamma dose rate in the containment building. Annex II provides charts of standard ranges of release to containment either for a gap release or a core melt occurring 1h or 24 h after the reactor trip.

Regarding the containment building, this FP barrier can be seen to be lost (i.e. as no longer constituting a "physical barrier placed between a radiation source or radioactive material and workers, members of the public or the environment" [8]) when:

- Its isolation is not fully successful, i.e. at least one isolation valve outside or inside the containment failed to close, either automatically (e.g. on the signal of automatic ECCS start), or manually by action of the control room operators;
- There is a bypass of the containment building through which radioactive material can be released to the environment (e.g. in the event of an SGTR cumulated with an outside of containment steam valve stuck open on the steam line of the affected SG);
- The mechanical integrity of the containment building is challenged (e.g. visible crack on the concrete following a severe earthquake).

3.2.2.4. Declaration of a 'General emergency'

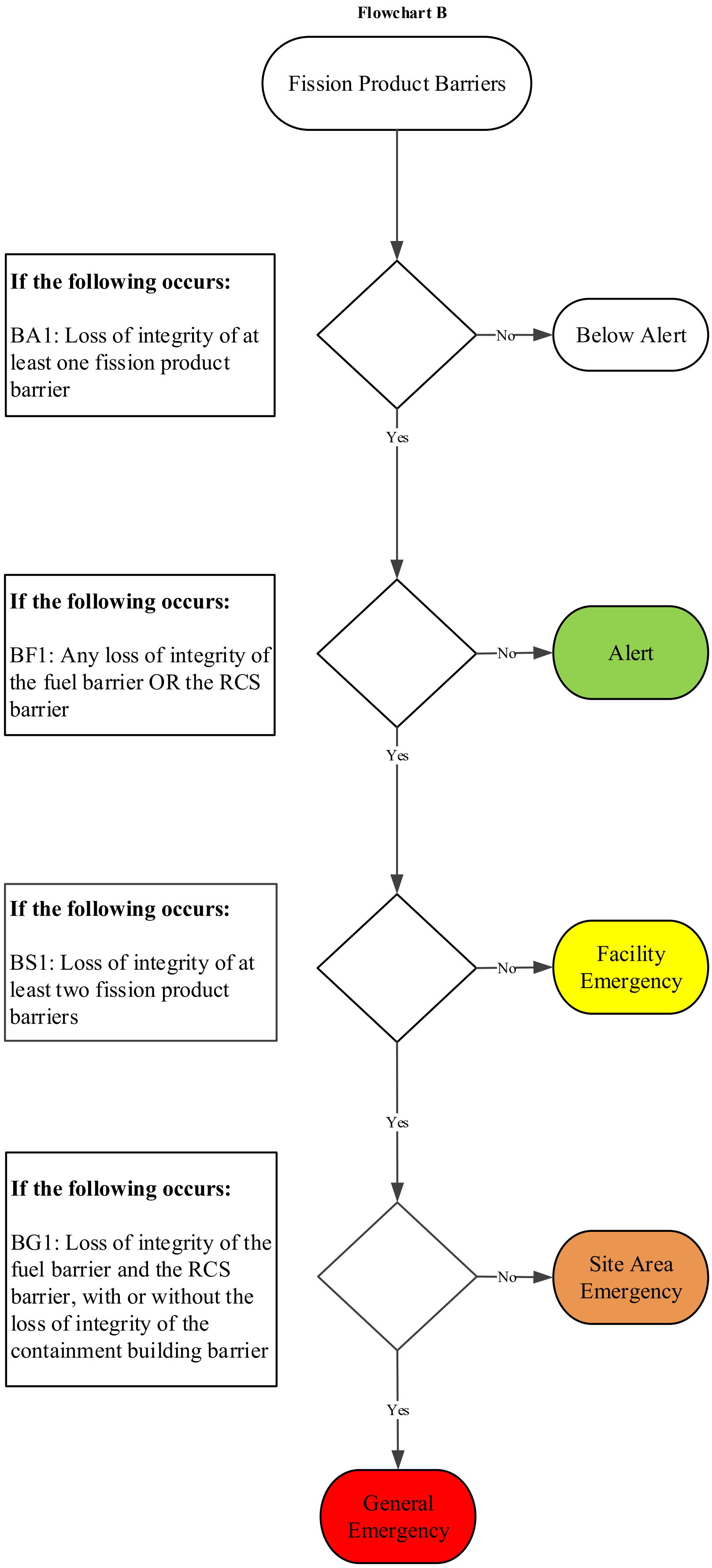
If the EAL outlined below (BG1) is met (assessment), or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'General emergency'.

BG1: Loss of integrity of the fuel barrier and the RCS barrier, with or without the loss of the containment building barrier

This EAL addresses situations involving the loss of integrity of both the fuel and the RCS FP barriers. This includes LOCA (primary break or SGTR) with fuel degradation. Overall, such situations lead to a release of radioactive material either through: (1) the stack chimney or containment venting systems after iodine filtration (iodine charcoal filters); or (2) the containment building itself in case the difference of pressure between the inside and outside of the containment building becomes too high.

The loss, or the anticipated loss, of integrity of the containment building, in addition to the loss of integrity of both the fuel and the RCS FP barriers, is not a necessary condition to meet BG1. Indeed, the additional loss of integrity of the containment building would accelerate the kinetics and the magnitude of the release to the environment. However, as stated in para. 5.14 of GSR Part 7 [6], “a general emergency…warrants taking precautionary urgent protective actions”. A precautionary urgent protective action is defined as an “action taken before or shortly after a release of radioactive material, or an exposure, on the basis of the prevailing conditions to avoid or to minimize severe deterministic effects” [8]. Therefore, EALs for the declaration of a general emergency need to reflect any situation in which protective actions need to be taken, even before the actual start of a significant release of radioactive material to the environment. In terms of FP barriers, this happens as soon as both the fuel and the RCS FP barriers lose their integrity.

3.2.2.5. *'Fission product' barriers flow chart*



3.2.3. Emergency Action Level category ‘Conventional emergencies, natural events, security events’

3.2.3.1. Declaration of an ‘Alert’

If any of the EALs outlined below (EA1–EA5) is met (assessment) or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, ‘Alert’.

EA1: Natural event affecting reactor safety equipment and/or the access to reactor safety equipment

Natural events include severe weather, earthquakes, volcanic eruptions, tsunamis, floods, and mudslides. This EAL addresses situations involving a natural event that leads to:

- A decrease in the availability or functioning of reactor safety equipment. For instance, this might happen in the following cases (they do not constitute an exhaustive list):
  - Very high flow rates in the cold source (river, sea) that degrades the ultimate heat sink (UHS) water intake;
  - Significant amount of green waste carried in the cold source stream that clogs the UHS water intake following a heavy rain;
  - Frozen and clogged UHS water intake during severe winter conditions;
  - Mandatory decrease in the UHS water intake flow rate during summer heat waves in accordance with national environmental regulations;
  - A sandstorm that clogs engine filters (such as for emergency diesel generators).
- Complete loss of safe access for verification, maintenance, manual start/stop, or fixing of reactor safety equipment. For instance, this might happen in the following cases (they do not constitute an exhaustive list):
  - On-site partial or total flooding (e.g. due to heavy rains) that prevents access to an auxiliary building or room where safety equipment is located;
  - Very high temperature in an auxiliary building or equipment room (e.g. due to heat wave) that prevents technicians from staying long enough to complete maintenance or reparation.

EA2: Conventional emergency affecting reactor safety equipment and/or access to reactor safety equipment

Conventional emergencies include fires, chemicals, explosions (non-security related), aircraft crash and any other events not related to radiation emergencies, which are included in the national all hazards emergency plans. This EAL addresses situations involving a conventional emergency that lead to:

- A decrease in the availability or functioning of reactor safety equipment.
- Complete loss of safe access for verification, maintenance, manual start/stop, or repair of reactor safety equipment. Access to reactor safety equipment might be prohibited due to toxic, corrosive, asphyxiant or flammable gases.

EA3: Control room evacuation needed

This EAL addresses situations in which evacuation of the control room is needed regardless of the cause of such an evacuation. Possible reasons for such an evacuation might be for safety (e.g. presence of pollutants in the building) or security (e.g. arrival of adversaries).

EA4: Security event affecting, or within, the limited access area

This EAL addresses situations triggered by a nuclear security event, such as:

- Hostile actions affecting, or within, the limited access area (a designated area containing a nuclear facility and nuclear material to which access is limited and controlled for nuclear security purposes [9]);
- An airborne attack threat or an actual airborne attack targeting the limited access area.

Such events might disrupt normal safe operation and/or diminish the level of nuclear security within the limited access area.

EA5: Other conditions that, in the judgement of the decision maker, warrant taking actions to assess and mitigate potential consequences at the facility

This EAL addresses any situation dealing with conventional emergencies, natural events, or security events that, in the judgement of the decision maker in charge of emergency declaration, warrant **taking actions to assess and mitigate the potential consequences at the facility**. These additional situations would not be covered by EALs EA1–EA4, but might still:

- Affect the availability or functioning of reactor safety equipment;
- Affect the safe access for verification, maintenance, manual start/stop, repair of reactor safety equipment;
- Affect, or take place within, the limited access area.

3.2.3.2. Declaration of a 'Facility emergency'

If any of the EALs outlined below (EF1–EF4) is met (assessment) or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Facility emergency'.

EF1: Conventional emergency or natural event that meets SF1–SF5, BF1, LF1, or LF2

This EAL addresses a conventional emergency or natural event that result in any of the following:

- Total loss of AC power (station blackout) with the reactor coolant temperature higher than the saturation temperature.
- Loss of all direct current (DC) power for at least 15 min.
- Failure of automatic and manual actions to shut down the reactor and maintain subcriticality. The reactor remains at power.
- Major loss of alarm(s) and/or indicator(s) in the control room, concurrently with indications that an emergency, or a potential emergency, is happening.
- Loss/unavailability of safety equipment, leading to a degradation of capability to remove heat from the RCS.
- Any loss of integrity of the fuel barrier OR the RCS barrier.
- On-site radiation levels requiring protective actions.
- The SFP coolant is saturated.

EF2: Control room evacuation needed and the supplementary control room cannot be reached or operated

This EAL addresses situations in which evacuation of the control room is needed, regardless of the cause, and it is not possible for control room operators (or other authorized personnel) to reach or operate the supplementary control room. The cause for failing to reach or operate the supplementary control room might be of a safety (e.g. presence of pollutants in the building) or security (e.g. arrival of adversaries) nature.

EF3: Security event affecting the protected area

This EAL addresses situations triggered by a nuclear security event, such as:

- Hostile actions affecting (but not necessarily taking place within) the protected area (the protected area is an area inside a limited access area containing Category I or II nuclear material and/or sabotage targets surrounded by a physical barrier with additional physical protection measures [9]);
- An airborne attack threat or an actual airborne attack affecting the protected area.

Such events might disrupt the normal safe operation and/or diminish the level of nuclear security within the protected area.

EF4: Other conditions that, in the judgement of the decision maker, warrant protective actions on the site

This EAL intends to address any other situations with regard to conventional emergencies, natural events, or security events that, in the judgement of the decision maker in charge of emergency declaration, warrant protective actions on the site.

3.2.3.3. Declaration of a 'Site area emergency'

If any of the EALs outlined below (ES1–ES3) is met (assessment) and/or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Site area emergency'.

ES1: Conventional emergency or natural event that meets SS1, S2, BS1, LS1 or LS2

This EAL addresses a conventional emergency or a natural event that results in:

- Total loss of AC power (station blackout) AND loss of integrity of the RCS FP barrier;
- Major loss of RCS cooling capability AND loss of integrity of the RCS FP barrier;
- Loss of integrity of at least two FP barriers;
- Radioactive release resulting in actual or projected off site doses reaching (at least) 10% of a domestic generic criterion to take urgent protective actions;
- Fuel assemblies are still under water in the SFP; the SFP coolant is saturated, its level is below its nominal level and keeps decreasing.

ES2: Security event within the protected area

This EAL addresses situations triggered by a nuclear security event, such as:

- Hostile actions within the protected area;
- An airborne attack threat or an actual airborne attack targeting the protected area.

ES3: Other conditions that, in the judgement of the decision maker, warrant protective actions on the site and the increase of readiness to take protective actions and other response actions off the site

This EAL intends to address any other situations with regard to conventional emergencies, natural events or security events that, in the judgement of the decision maker in charge of emergency declaration, warrant protective actions on the site and the increase of readiness to take protective actions and other response actions off the site.

3.2.3.4. Declaration of a 'General emergency'

If any of the EALs outlined below (EG1–EG3) is met (assessment) and/or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'General emergency'.

EG1: Conventional emergency or natural event that meets BG1, LG1, or LG2

This EAL addresses a conventional emergency or a natural event that results in either:

- Loss of integrity of the fuel barrier and the RCS barrier, with or without the loss of integrity of the containment building barrier;
- A radioactive release resulting in actual or projected off-site doses above a national generic criterion to take urgent protective actions;
- Uncovered irradiated fuel assemblies in the SFP.

EG2: Security event (e.g. hostile action) affecting, or within, the vital area

This EAL addresses situations triggered by a nuclear security event, such as:

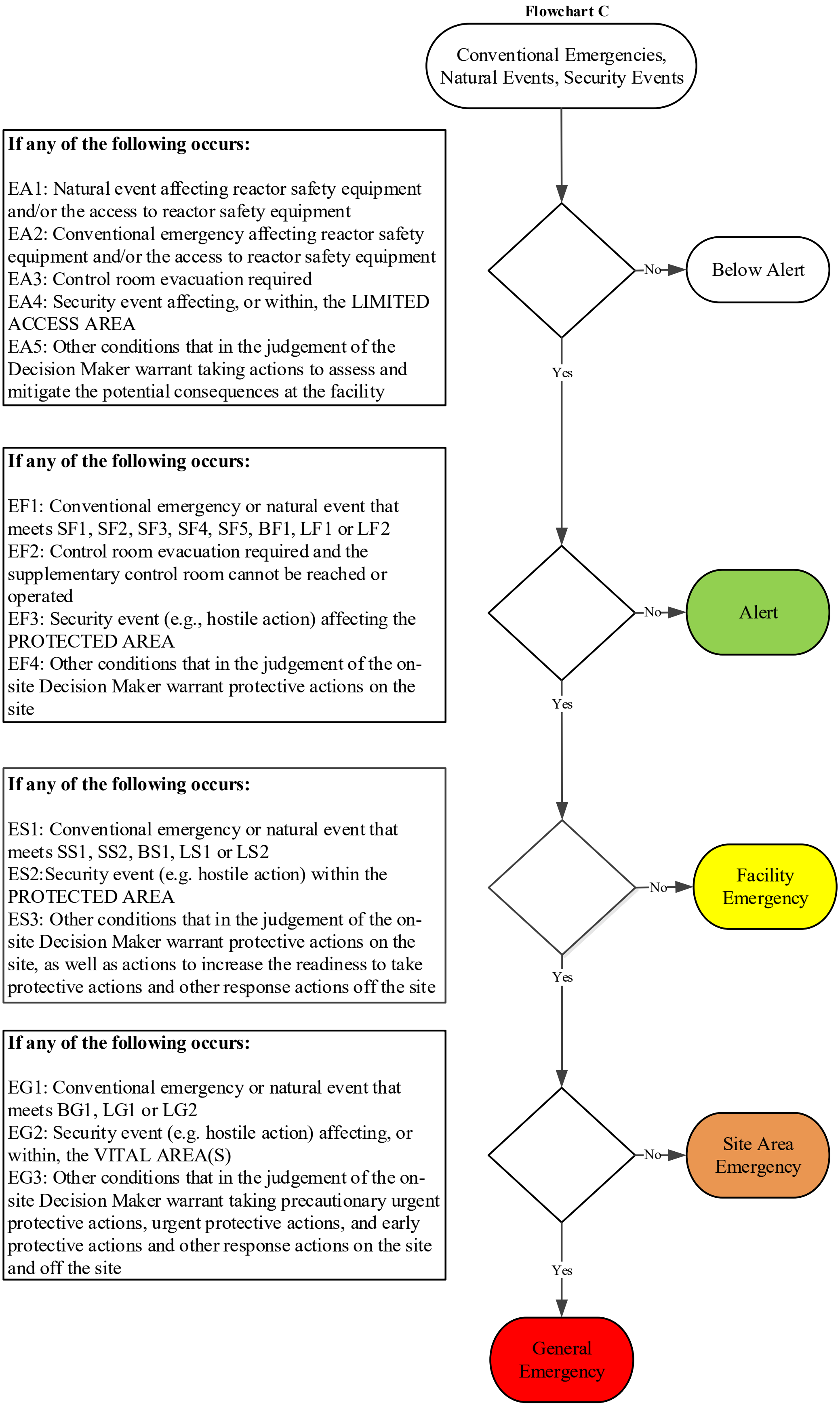
- Hostile actions affecting, or within, the vital area (the vital area is an area inside a protected area containing equipment, systems or devices, or nuclear material, the sabotage of which could directly or indirectly lead to high radiological consequences [9]);
- An airborne attack threat or an actual airborne attack targeting the vital area.

Such events might disrupt the normal safe operation and/or diminish the level of personnel security, equipment, system and device availability, or nuclear material security within the vital area.

EG3: Other conditions that, in the judgement of the decision maker, warrant taking precautionary urgent protective actions, urgent protective actions, and early protective actions and other response actions on the site and off the site

This EAL addresses any other situations with regard to conventional emergencies, natural events, or security events that, in the judgement of the decision maker in charge of emergency declaration, warrant taking precautionary urgent protective actions, urgent protective actions, and early protective actions and other response actions on the site and off the site.

3.2.3.5. Flow chart for conventional emergencies, natural events and security events



3.2.4. Emergency Action Level category ‘Safety systems and equipment’

3.2.4.1. Declaration of an ‘Alert’

Under this EAL category, it is considered that at least one key safety system or equipment is unavailable and/or malfunctioning (assessment), and/or is likely to become unavailable or to malfunction in the future evolution of the emergency. If any of the EALs outlined below (SA1–SA4) is met (assessment) and/or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, ‘Alert’.

SA1: Total loss of AC power (station blackout); or AC power to reactor safety system buses reduced to a single source for at least 15 min

The first part of the EAL SA1 addresses situations involving the complete loss of off-site and on-site AC electrical power to the essential and non-essential switchgear buses of the reactor unit. This first part of EAL SA1 can be met regardless of the time occurrence of the station blackout and regardless of the initial state (i.e. before the event that triggered the emergency) of the reactor unit (e.g. 100% nominal power, cold shutdown or refuelling).

The second part of the EAL SA1 covers the degradation of the off-site and on-site AC power systems, such that any additional single failure would result in a station blackout. This condition could occur due to a loss of off-site power, with a concurrent failure of all but one emergency diesel generator to supply power to its reactor safety system buses. Another related condition could be the loss of all emergency diesel generators, and a concurrent failure of all but one off-site AC power source. This second part of EAL SA1 can apply regardless of the initial state of the reactor unit. However, it is especially relevant with the reactor being initially in a ‘hot state’ and at power. A time interval of 15 min is suggested as a criterion to exclude transient or momentary power losses.

SA2: Failed automatic shutdown but successful manual shutdown with the reactor initially at-power

This EAL addresses situations in which an automatic shutdown signal failed to be triggered, or the automatic shutdown signal was triggered but the automatic shutdown failed. Such a failure results in the non-insertion or incomplete insertion of at least one control rod in the core of the reactor. In addition, this EAL is met when, following such a failure, a manual shutdown is successfully actioned by the operators.

SA3: Major loss of alarm(s), indicator(s) and/or instrumentation and control system(s) in the control room

This EAL addresses situations in which alarms, indicators and/or instrumentation and control systems that are important to safety in the control room are unavailable, unreliable or malfunctioning, which might substantially decrease the ability to detect and/or mitigate an emergency triggering event and/or to characterize its aftermaths.

The term ‘major loss’ can be understood as quantitative or qualitative. The loss of only a few alarms or indicators might prevent or delay control room operators from switching from a normal operation procedure to an emergency operation procedure. In addition, the cumulated losses of alarms or indicators, not necessarily used for the detection or characterization of an accident transient, might eventually impact/delay the ability of control room operators to take actions in response to such a transient. Additionally, the loss of a few but ‘critical’ (i.e. for the unfolding event) instrumentation and control systems might delay or impair the ability of control room operators to take mitigatory actions.

An alert is suggested to be declared if the major loss of control room alarms, indicators and/or instrumentation and control systems is longer than 15 min. This time interval is suggested as a criterion to exclude transient or momentary power losses.

SA4: Total loss of the connection with the UHS or major loss of safety equipment with cooling purpose (e.g. loss of the feedwater to the steam generators)

This EAL covers situations in which the overall cooling capacity is degraded and/or degrading. This includes situations in which the connection between the reactor unit and the UHS is lost. This can be caused by the loss or malfunction of pumps. Or it can be caused by a natural event such as strong winter conditions that lead to the freezing of the inlet channel, or large quantities of vegetal waste following heavy rains that clogs the inlet channel. In addition, this EAL addresses the loss or malfunction of various pumps intended to cool the primary circuit, the secondary circuit, or auxiliary systems. Such a loss or malfunction can be due to various technical (e.g. electrical, mechanical) reasons.

3.2.4.2. Declaration of a 'Facility emergency'

If any of the EALs outlined below (SF1–SF5) is met (assessment) and/or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Facility emergency'.

SF1: Total loss of AC power (station blackout) with the reactor coolant temperature higher than the saturation temperature

This EAL does not apply to BWRs. It addresses situations in which the reactor unit has suffered a station blackout for a period of time during which the reactor coolant temperature has increased above the saturation temperature (the saturation temperature is a function of the pressure and temperature in the reactor vessel). In such a case, heat exchange between the core and the coolant is degraded, which might lead to the start of fuel degradation.

SF2: Loss of all DC power for at least 15 min

This EAL intends to address situations involving the loss of all DC power, which compromises the ability to monitor and control plant safety functions, as well as associated systems and equipment. This EAL is met when the voltage on all DC buses is less than the minimum bus voltage necessary for the operation of safety related equipment. This criterion on the voltage value needs to incorporate a time consideration, which is suggested to be of at least 15 min of operation before declaring the inability to operate those loads. The voltage is usually near the minimum voltage selected when battery charging is performed. A time interval of 15 min is suggested as a criterion to exclude transient or momentary power losses.

SF3: Failure of both automatic and manual actions to shut down the reactor and maintain the subcriticality

This EAL addresses situations in which the reactor fails to be shut down or maintained subcritical (i.e. the chain reaction in the core failed to be completely stopped). Such a situation might arise when the automatic shutdown system and manual shutdown action failed to fully insert all the control rods in the core, also known as a 'reactor trip'. Nevertheless, a successful (automatic or manual) reactor trip is necessary, but not sufficient, to completely stop the chain reaction in the core of the reactor. In addition to the reactor trip, automatic or manual actions are expected to inject neutron poison that is to be diluted in the whole RCS. This is a necessary condition to make sure that the reactor is shut down and it will remain subcritical.

SF3 can be met regardless of the initial state (i.e. before the event that triggered the emergency) of the reactor unit (e.g. 100% nominal power, cold shutdown, or refuelling).

SF4: Major loss of alarm(s), indicator(s) and/or instrumentation and control system(s) in the control room despite indications that an emergency, or a potential emergency, is occurring

This EAL addresses situations in which alarms, indicators and/or instrumentation and control systems that are important to safety in the control room are unavailable, unreliable, or malfunctioning, i.e. in which control room operators have lost the ability to detect, characterize and/or mitigate an emergency triggering event and/or to characterize its consequences, together with the evidence that an emergency has been triggered. Such a major loss of alarms, indicators and/or instrumentation and control systems might be considered as 'confirmed' if it happens for more than 15 min. As for SA3, the term 'major loss' can be understood as quantitative or qualitative in terms of alarms, indicators and/or instrumentation and control systems lost in the control room. SF4 covers situations such as the loss of ability for control room operators to action systems or equipment (e.g. open/close a valve; start/stop a pump), but with confirmation from local monitoring that a system or equipment is in a different state than reflected in the control room. Two examples are provided below (they do not constitute an exhaustive list):

(1) The control room operators cannot confirm whether or not a steam valve outside of containment is opened or closed (which might constitute a bypass of containment in the case of an SGTR on the same SG of a PWR, CANDU or WWER), but a technician sent to the equipment room confirms that the valve is opened (based on the loud noise in the room).
(2) The control room operators reported the loss of the gamma radiation monitoring from the fuel building. In the meantime, a technician who was working in the fuel building had to suddenly leave the building due to the rapid increase of the value read on his/her individual dosimeter.

SF5: Loss or unavailability of safety equipment leading to a degradation in the heat removal capability from the RCS

This EAL addresses situations in which systems and equipment designed to remove heat from the RCS are lost or unavailable. Such a situation would lead to a decrease in the overall liquid coolant volume in the RCS. Later, the fuel would stop being covered by/surrounded with coolant, leading potentially to the start of fuel degradation. Classically, this EAL would be the escalation from the SA4 EAL.

3.2.4.3. Declaration of a 'Site area emergency'

If any of the EALs outlined below (SS1, SS2) is met (assessment) and/or anticipated to be met (prognosis), the emergency class associated with the event is, at a minimum, 'Site area emergency'.

SS1: Total loss of AC power (station blackout) AND loss of integrity of the RCS fission product barrier

This EAL addresses situations in which the reactor unit suffers a station blackout and a LOCA. The primary break causing the LOCA might occur independently of the station blackout or might be a consequence of the station blackout. Indeed, following a certain period of time after the station blackout, the status of the RCS/primary circuit envelope might become degraded. For instance, a station blackout might lead to:

- Loss of RCS cooling capability and, consequently, to an active (by the control room operators following emergency operating procedures) or passive (due to overpressure) opening of the pressurizer relief valves or other pressure exhaust valves on the RCS. This constitutes a LOCA which would lead to fuel degradation if the amount of coolant lost cannot be compensated for through coolant injection.
- Loss of coolant injection at the primary pump seals and, consequently, the occurrence of a primary break (i.e. a LOCA) at primary pump seal(s). This would lead to fuel degradation if the amount of coolant lost cannot be compensated for through coolant injection.

The loss of integrity of the RCS (alone) can be identified based on the following factors (they do not constitute an exhaustive list):

- Opening of the pressurizer relief valves due to overpressure in the RCS;
- An alarm indicating high activity in the steam phase of an SG (that is the symptom of an SGTR with activity from the primary coolant being transferred to the damaged SG);
- Decreasing primary pressure in the RCS and/or decreasing coolant level in the pressurizer and/or decreasing coolant level in the reactor vessel and/or automatic start of the high head ECCS/safety injection pumps;
- Increasing coolant level and/or temperature in the reactor building sumps;
- Increasing temperature and/or pressure in the reactor building and/or automatic start of the containment spray system.

SS2: Major loss of RCS cooling capability AND loss of integrity of the RCS fission product barrier

This EAL addresses situations in which the reactor unit suffers major equipment losses or malfunction in terms of RCS heat removal and a LOCA. The primary break causing the LOCA might occur independently of the degraded overall cooling capability or might be a consequence of it. Indeed, following a certain period of time after a major loss of cooling capability, the status of the RCS/primary circuit envelope might become degraded, for instance:

- The total loss of the UHS might lead to a loss of coolant injection at the primary pump seals and, consequently, the occurrence of a primary break (i.e., a LOCA) at the primary pump seal(s). This would lead to fuel degradation if the amount of coolant lost cannot be compensated through coolant injection; or
- The total loss of feedwater to the SGs might lead to an active (by the control room operators following emergency operating procedures) or passive (due to overpressure) opening of the pressurizer relief valves or other pressure exhaust valves on the RCS. This constitutes a LOCA which would lead to fuel degradation if the amount of coolant lost cannot be compensated through coolant injection.

The loss of integrity of the RCS (alone) can be identified based on the following factors (they do not constitute an exhaustive list):

- Opening of the pressurizer relief valves due to overpressure in the RCS;
- An alarm indicating high activity in the steam phase of an SG (that is, the symptom of an SGTR with activity from the primary coolant being transferred to the damaged SG);
- Decreasing primary pressure in the RCS and/or decreasing coolant level in the pressurizer and/or decreasing coolant level in the reactor vessel and/or automatic start of the high head ECCS/safety injection pumps;
- Increasing coolant level and/or temperature in the reactor building sumps;
- Increasing temperature and/or pressure in the reactor building and/or automatic start of the containment spray system.

3.2.4.4. Declaration of a 'General emergency'

If the EAL outlined below (SG1) is met (assessment) and/or anticipated to be met (prognosis), the emergency class associated with the event is 'General emergency'.

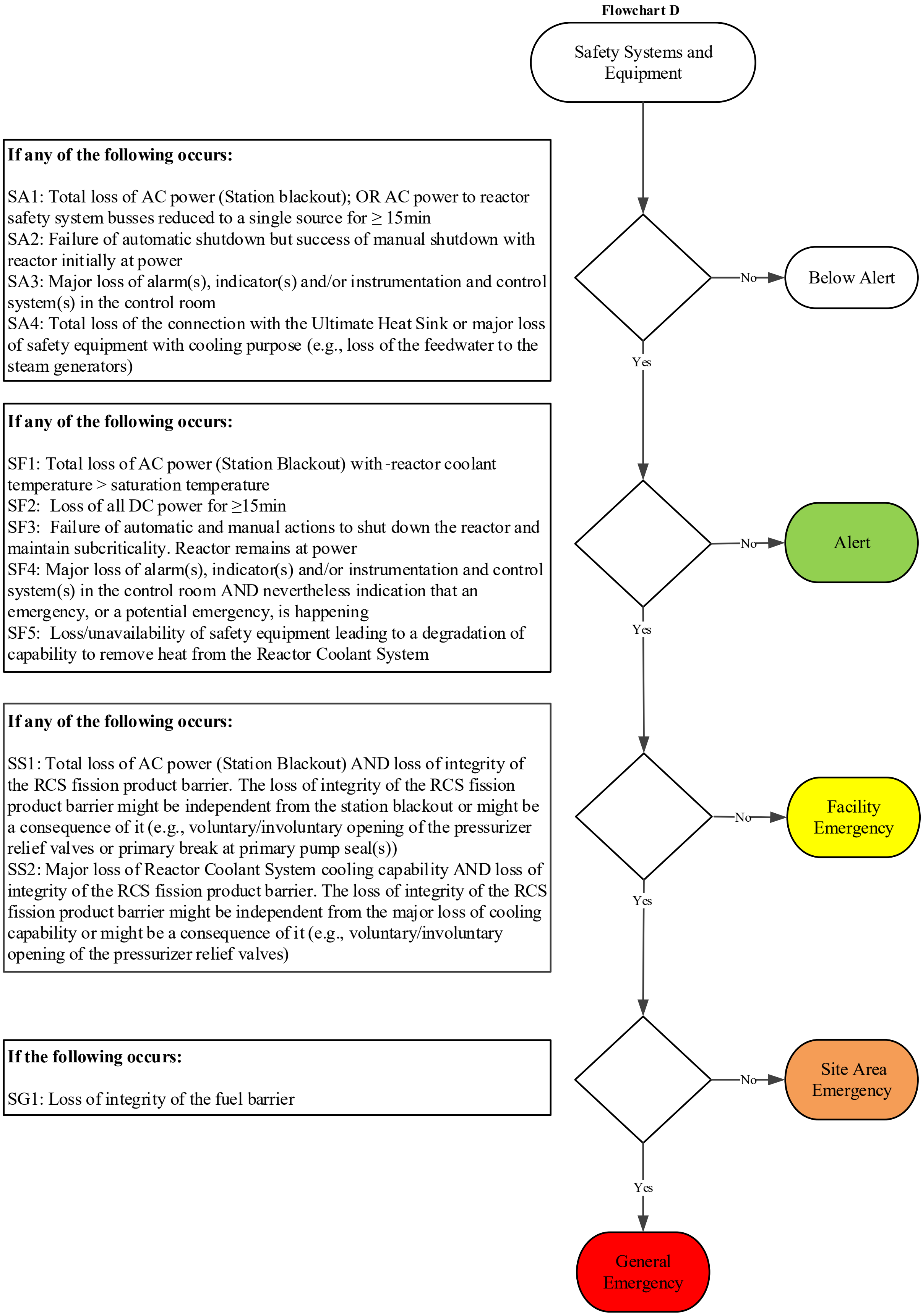
SG1: Loss of integrity of the fuel fission product barrier

This EAL constitutes the final escalation towards the emergency classification through the EAL category 'Safety systems and equipment'. As with SS1 and SS2, when considering SG1, a station blackout or a major loss of cooling capability, together with a LOCA (the primary break being independent from, or a consequence of, the station blackout or the major loss of cooling capability), have occurred at the reactor unit. SG1 addresses situations in which, coming from SS1 or SS2, the fuel degradation has already started (assessment), is imminent, or is anticipated (prognosis). Mostly, a situation cumulating in a station blackout and a primary break, or a major loss of cooling capability and a primary break, might progress with the decrease of the coolant level in the reactor vessel. The kinetics of this decrease is a function of several parameters, such as the primary pressure at the time of occurrence of the primary break, and the size and location of the primary break. As a consequence of such a decrease in the coolant level, the fuel might no longer be fully covered by, or surrounded with, coolant, leading to the rapid rise of the core exit temperature and start of fuel degradation.

The loss of integrity of the fuel barrier (alone) can be identified based on the following (they do not constitute an exhaustive list):

- The sharply increasing, or already high, core exit temperature (e.g. for a 1000 MW(e) PWR and for the purpose of emergency response, a core exit temperature reaching 700 C is considered as corresponding to the start of clad burst);
- The saturation margin monitored in the control room shows that the reactor coolant is overheated (i.e. $T_{Saturation} (P_{RCS}) - T_{Core\ Exit} << 0°C$);
- The reactor coolant level reaching the top of the fuel assemblies (this item applies to BWRs, PWRs and WWERs).

3.2.4.5. *Flow chart for 'Safety systems and equipment'*



APPENDIX: ASSESSING THE OCCURRENCE OF A LOSS OF COOLANT ACCIDENT OR A STEAM GENERATOR TUBE RUPTURE

For a significant release of radioactive material to the environment to happen, an SGTR or a LOCA needs to occur. However, while it is necessary it is not sufficient, as additional events need to take place, such as degradation or bypass of the containment building. Events such as a station blackout or loss of the UHS do not lead to a significant release of radioactive material to the environment unless they induce a primary break LOCA.

I.1. LOSS OF COOLANT ACCIDENT

A LOCA is a general term to designate primary breaks (i.e. breaks that occur on the RCS), which can range from a small break LOCA to a large break LOCA. Both the location and the size of the primary break are key factors for accident kinetics. Depending on the size of the break, the LOCA may be compensated for by coolant injection through charging pumps or the ECCS. In addition, the break location drives the water phase at the break, if the LOCA is not compensated for by injection of coolant: the higher the height of the primary break location on the RCS, the less coolant is lost in the liquid phase and the sooner coolant at the break is lost in the steam phase. For a given size but different locations of primary break LOCA, a steam phase LOCA leads to fuel degradation later than a liquid phase LOCA. Most LOCA scenarios start with a liquid phase LOCA, which turns to a steam phase LOCA when the primary coolant level becomes lower than the primary break height.

The occurrence of a primary break LOCA can be assessed based on the following events (the list is not exhaustive), which might occur in combination:

- Automatic reactor trip, decrease in the pressurizer level and primary pressure, automatic increase in the charging flow rate or automatic start of the ECCS;
- Increase in the pressure (and temperature) inside the containment building, which may lead to the automatic start of the containment spray system;
- Increase in the water level in the sumps;
- Increase in the gamma dose rate measured in the containment building.

I.2. STEAM GENERATOR TUBE RUPTURE

An SGTR, as a loss of integrity of the RCS, is a specific type of LOCA. When an SGTR occurs, primary coolant leaks into the damaged SG. Activity from the primary coolant can then be released in the environment in the steam or liquid phase. Control room operators have emergency operating procedures to respond to an SGTR and mitigate the release of radioactivity into the atmosphere. Without additional failures, an SGTR can be mitigated, the release stopped promptly, and the short steam release in the atmosphere would not constitute a hazard to the public or the environment.

However, if additional failures occur, such as an abnormal primary coolant activity prior to the SGTR and/or a bypass of containment through a steam valve in a stuck open position, the associated release in the environment may warrant taking protective actions.

The occurrence of an SGTR can be assessed based on the following events (the list is not exhaustive), which might occur in combination:

- An alarm in the control room indicates an abnormal level of radioactivity in the steam phase of the damaged SG;
- The water level increases in the damaged SG even after reducing or stopping the feedwater flow rate;
- An automatic reactor trip, decrease in the pressurizer level, automatic increase in the charging flow rate, and no increase in the pressure inside the containment building.

The increase in the water level in the damaged SG constitutes the main concern when taking actions to mitigate an SGTR, as a liquid water overflow in the damaged SG would lead to a liquid phase release. Indeed, for the same amount of water released in the environment from the damaged SG, a liquid phase release leads to a higher source term than a steam phase release, as only a small fraction of iodine and caesium (radionuclides that contribute most to the dose to people exposed to such a release) can migrate to steam.

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[8] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Nuclear Safety and Security Glossary: Terminology Used in Nuclear Safety, Nuclear Security, Radiation Protection and Emergency Preparedness and Response, Interim Edition, IAEA, Vienna (2022).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 27-G, IAEA, Vienna (2018).

ANNEX I:

PREDICTED CORE DAMAGE FOR DIFFERENT LENGTHS OF TIME THAT THE CORE IS UNCOVERED

TABLE I–1. CORE DAMAGE VS. TIME THAT THE CORE IS UNCOVERED (adapted from [I–1])

Length of time the core is uncovered	Estimated core damage	Insights
0	Normal coolant	Core remains covered and there is a slow reduction in power and pressure
0	Coolant with 10–100 times normal isotope concentrations (spike)	Core remains covered and there is a rapid shutdown, de-pressurization, or primary system
More than 15 mins	100% gap release	Exothermic Zr-H_2O reaction (self-sustaining) with rapid H_2 generation Fuel heat-up rate increases by a factor of 2 or 3 Rapid fuel cladding failure and local fuel melting
More than 30 mins	10–50% core melt	Rapid release of volatile fission products Possible relocation (slump) of molten core Possible uncoolable core even if recovered with water
More than 1 hour	100% core melt	Possible melt-through of vessel and containment failure even if core is re-covered with water.

REFERENCE TO ANNEX I

[I–1] NUCLEAR REGULATORY COMMISSION, Response Technical Manual, RTM93, NUREG/BR-0150, Vol 1, Rev. 4, USNRC, Office for Analysis and Evaluation of Operational Data, Washington, DC (1996).

ANNEX II: SUPPLEMENTARY FILE

CORE DAMAGE ASSESSMENT BASED ON RADIATION LEVELS IN THE CONTAINMENT

The supplementary file (figures that support core damage assessment based on radiation levels in the containment of a large PWR, BWR Mark I & II, BWR Mark III, WWER-230 and WWER-213, and formula and table for unknown containment types) for this publication can be found on the publication’s individual web page at www.iaea.org/publications

ANNEX III: SUPPLEMENTARY FILE

EMERGENCY CLASSIFICATION PROCESS OVERVIEW

The supplementary file (overarching emergency classification flow chart to facilitate the use of the four EAL flow charts) for this publication can be found on the publication's individual web page at www.iaea.org/publications

LIST OF ABBREVIATIONS

A&P	assessment and prognosis
BWR	boiling water reactor
CANDU	Canada deuterium uranium reactor
EAL	emergency action level
ECCS	emergency core cooling system
FP	fission product
LOCA	loss of coolant accident
PAZ	precautionary action zone
PHTS	pressurized heat transport system
PHWR	pressurized heavy water reactor
PWR	pressurized water reactor
RCS	reactor coolant system
SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
UHS	ultimate heat sink
UPZ	urgent protective action planning zone
WWER	water, water, energy, reactor

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