Safety Reports Series

No. 86

Safety Aspects of Nuclear Power Plants in Human Induced External Events: General Considerations

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SAFETY ASPECTS OF NUCLEAR POWER PLANTS IN HUMAN INDUCED EXTERNAL EVENTS: GENERAL CONSIDERATIONS

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SAFETY REPORTS SERIES No. 86

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FOREWORD

Many human actions pose challenges to the safe operation of a nuclear installation, such as a nuclear power plant. These challenges may arise from activities human beings undertake as a part of routine life. The challenges arising from intentional and accidental events need to be evaluated given the current design robustness of the installation and the vulnerability of the location of such events.

This publication is the first of three Safety Reports on the safety assessment of nuclear facilities subjected to extreme human induced external events. These publications address the assessment of nuclear installations subjected to accidental or unintentional human actions. They provide the general framework for approaches to obtaining the overall plant performance with regard to the fundamental safety functions from the performance of individual components. It includes safety assessments, the characterization and quantification of loadings, and appropriate analysis techniques and material properties for capacity assessments. This publication explores established methodologies in the light of recent advances in the understanding of material behaviour under such extreme loading conditions and computational techniques that can incorporate such behaviour in the analytical modelling.

These three Safety Reports were developed using funding from Member States voluntarily contributing to, and participating in, the extrabudgetary programme of the External Events Safety Section (EESS-EBP). Established in 2007, the EESS-EBP has developed technical documents considered a priority for Member States, given the current experience with severe external events globally. The aim of the programme is to provide technical inputs to current and future IAEA safety standards. The EESS-EBP implements these activities by assimilating the latest technical issues and practical methodologies in Member States, and disseminates the information through technical publications, sharing them in the working groups, and by participating in global conferences and forums.

The work of all the contributors to the drafting and review of this publication is greatly appreciated. In particular, the IAEA gratefully acknowledges the contributions of J.J. Johnson (United States of America) to the drafting of this publication, and of A. Blahoianu and N. Orbovic (Canada) to its review. The IAEA officers responsible for this publication were A. Altinyollar and F. Beltran of the Division of Nuclear Installation Safety.

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CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [1], establishes that the process of safety assessment of a nuclear installation needs to be repeated periodically — in whole or in part, as necessary — in order to take into account changed circumstances with respect to those considered for the design. Following this principle, the IAEA initiated a major effort in 2001 targeted at the development of guidelines for the assessment of vulnerability against accidental or postulated human induced external events not foreseen in the design basis. Examples of accidental events include explosions caused by pipeline failures, train crashes or hazardous material leaks from tanks. Examples of postulated external events include station blackout and the loss of ultimate heat sink due to unidentified causes.

The evaluation of the effects of accidental events on nuclear facilities can be based on all of the tools for assessment, including screening by probability of occurrence based on statistical evaluation of historical data. In the case of postulated external events, screening by probability of occurrence is not a tool available to the analyst, simply because the definition of the problem is such that the facility is assumed to be in a hypothesized state (operating or shutdown), without specification of how the facility reached this state. Consequently, the evaluation is focused on addressing the hypothesized state rather than the cause.

There is general agreement among experts that the current practice for nuclear facility design provides such a level of robustness that some external events not explicitly considered at the design stage may be accommodated by the nuclear facilities in their current configuration without significant radiological consequences. This is believed to be true for nuclear facilities in general and for nuclear power plants specifically. However, quantification is needed in order to understand with a high level of confidence which events can be screened out in a safety evaluation process and which events require a detailed assessment of the actual plant level performance.

In this context, a series of three Safety Reports has been produced that describe, generally and specifically, the approaches to addressing human induced external events with a focus on nuclear power plants. The three reports provide guidance to support the quantitative evaluation of the engineering safety

of facilities subjected to design basis external events and beyond design basis external events. In addition to this Safety Report, they include:

(a) Safety Aspects of Nuclear Power Plants in Human Induced External Events: Assessment of Structures, Safety Reports Series No. 87 [2];
(b) Safety Aspects of Nuclear Power Plants in Human Induced External Events: Margin Assessment, Safety Reports Series No. 88 [3].

This publication is the first in the series. It provides the general framework and includes a roadmap for performing the design and the evaluation of the protection against human induced external events. This Safety Report concentrates on an overall view of the methodology and on the important considerations for its application to existing and new nuclear power plants. Topics covered include elements of the design and evaluation approach, developed in five phases:

— Phase 1: Event identification;
— Phase 2: Hazard evaluation and load characterization;
— Phase 3: Design and evaluation approaches to structures, systems and components (SSCs);
— Phase 4: Plant performance assessment and acceptance criteria;
— Phase 5: Operator response.

The second report in the series addresses phases 2 and 3 of the general framework. It provides detailed guidelines for the safety assessment of nuclear power plant structures against mechanical impacts, explosions and fire hazards caused by human induced external events. The report covers the characterization of loading, the assessment of structural integrity using both simplified methods and more elaborated methodologies, and the assessment of induced vibration. Acceptance criteria are given in the report for different failure modes: overall stability, overall bending and shear, local failure modes and induced vibrations. In addition, since many of the human induced external events may result in a fire, the process of analysing the fire consequences is also given. Approaches to assessing the barrier fire performance and the fire performance of SSCs are also given.

The third report in the series addresses phases 1 and 4 of the general framework. The report describes the procedures for assessing the safety margins of nuclear power plants against human induced external events. Both postulated and accidental hazards are considered. The report focuses on plant and systems performance evaluations. A tiered approach to margin assessment is provided. The first tier consists of a deterministic procedure in which, for each scenario, the

existence of at least one undamaged success path[1] to comply with the fundamental safety function is investigated. This procedure can be extended to calculate probability measures such as the conditional core damage probability and the conditional probability of loss of spent fuel pool cooling and spent fuel damage, given the scenario. In the most elaborated stage, probabilistic safety assessment (PSA) techniques are introduced, giving consideration to the probabilistic aspects of hazards and of SSC capacity (fragility). Event tree and fault tree models are used to compute usual PSA metrics, such as core damage frequency, large early release frequency, and frequency of loss of spent fuel pool cooling and spent fuel damage.

In summary, these three publications in the Safety Reports Series provide methodologies that can be used in the evaluation of SSC capacity of nuclear power plants subjected to extreme human induced external events and in the assessment of the resulting safety margin of the facilities. The three publications may be useful to nuclear facility owners, operators and regulators who need an understanding of the safety issues in relation to human induced events. They contain descriptions of internationally accepted methods applied by the engineering community and some examples that may be useful in the evaluation of the need for plant upgrading. Many references are also provided for more detailed guidance, and the publications rely on many IAEA safety standards and relevant technical publications.

The three Safety Reports have a common thread and are closely related to each other. Together, they provide an approach to the assessment against extreme human induced external events fully consistent with the methods used for evaluation of nuclear facilities subjected to extreme natural events, such as earthquakes and floods.

1.2. OBJECTIVE

The objective of this Safety Report is to provide the framework of methods to be implemented for the design and evaluation of nuclear facilities subjected to extreme human induced external events. This framework addresses the performance of the overall facility, and individual SSCs, from the standpoint of complying with the fundamental safety functions.

[1] A success path is a set of systems and associated components that can be used to bring the plant to a stable hot or cold shutdown condition and to maintain this condition for a specified period of time.

1.3. SCOPE

This Safety Report concentrates on the development of a methodology and highlights considerations for its application to existing and new nuclear power plants. The methodology, although directly applicable to nuclear power plants, can easily be extended to other nuclear and non-nuclear facilities with complex processes (e.g. fuel processing and reprocessing facilities, and research reactors) with suitable grading based on the potential radioactive releases. The external events considered are limited to human induced events of accidental origin and exclude events that may be the result of malevolent action. Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.4. STRUCTURE

Section 2 provides an overview of the approach to engineering safety design and assessment against human induced external events. The section describes the key phases of the overall approach. It also includes general considerations about design and assessment principles and uncertainties. Section 3 is dedicated to the first phases of the methodology; namely, the external event identification and the definition of the loads to be considered in the assessment against each identified event. Section 4 is devoted to the plant specific evaluations. It describes in greater detail the approaches to addressing human induced external events taking into account the robustness of typical nuclear power plant designs, and it introduces the processes of PSA and safety margin assessment (SMA) to such assessments. It covers the selection of SSCs important for safety, the capacity evaluation procedures and the assessment of performance of fundamental safety functions. The section also includes examples of acceptance criteria. Section 5 presents considerations for the design of new nuclear power plants, including the selection of the design basis events and the basic design principles against extreme external events. Section 6 briefly explains the differences between evaluations for existing and for new installations. Finally, Section 7 addresses management of the efforts for the assessment.

As a supplement to Section 3, the Annex provides three examples to illustrate the preparation and use of the extreme loading definition matrices corresponding to human induced external events.

1.5. DEFINITIONS

Event

Any occurrence that is unintended by the operator of the nuclear facility, the consequences or potential consequences of which are not negligible from the point of view of safety.

Human induced external event

An event that is unconnected with the operation of the nuclear facility. In this publication, human induced external events are defined by loading conditions to be evaluated or to be taken into account in the design. Examples treated in Ref. [2] include aircraft crashes, explosions and large fires.

Design basis external event

An external event explicitly taken into account in the design of a facility, according to established criteria, such that the facility can withstand without exceeding authorized limits by the planned operation of safety systems or structures.

Beyond design basis external event

An external event that is more severe than a design basis external event (DBEE). The term refers to external events either not included as a design basis or included with a lower degree of severity.

Design extension external event

An external event that is not considered as a DBEE, but that is considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Hence, a design extension external event (DEE) refers to a rare and severe external event that is considered in the design process using realistic, rather than conservative, assumptions and acceptance criteria (see Fig. 1).

Loading conditions

Loading conditions for each external event are defined probabilistically or deterministically by the Member State. Tiered loading conditions are adopted

in Refs [2, 3]. These multiple loading conditions are denoted as follows: DBEE, DEE 1 and DEE 2. Increased magnitudes of loads are defined for DBEE, DEE 1 and DEE 2. Design extension conditions are events to be used in the design or evaluation process that correspond to rare and severe external events. Design processes of most existing facilities do not consider design extension conditions (see Fig. 1).

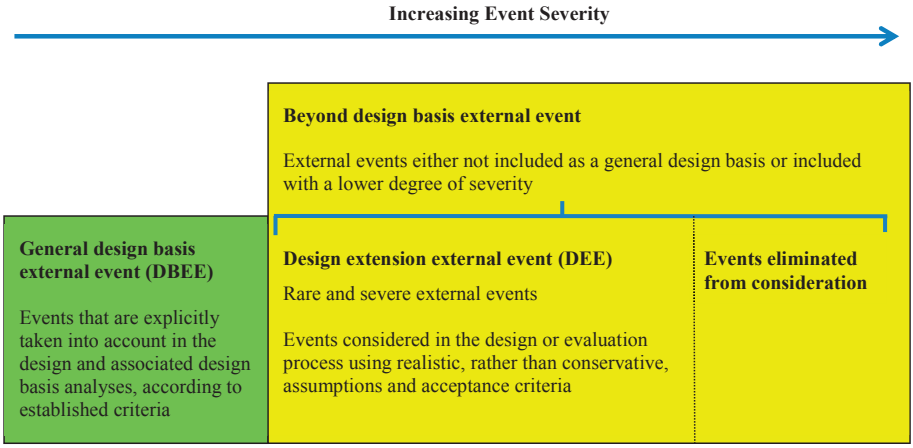

FIG. 1. Classification of external events from the point of view of design requirements.

Performance criteria

A defined function that the plant and SSCs is required to perform when subjected to the events (especially important for design extension events). Examples of performance criteria for DBEE are the design criteria (e.g. for structures — essentially elastic behaviour). Examples DEE 1 include system redundancy, reduced functionality (but adequate for cold shutdown), and structure integrity and leaktightness. Examples for DEE 2 include reduced functionality (but adequate for structure integrity) and cold shutdown. System redundancy is not required.

Acceptance criteria

Criteria that the plant and SSCs are required to satisfy when subjected to the events in order to show that the performance criteria are met. They may be design criteria or less conservative criteria. They may be success paths (multiple success paths or a single success path, depending on the loading condition). They may

be PSA metrics, such as conditional probability of failure of the nuclear power plant or of specific SSCs. They may be based on best estimate procedures and parameter values or on conservatively biased values. Tiered acceptance criteria corresponding to the tiered loading conditions are adopted in Refs [2, 3].

Structures, systems and components margin

A generic term defining the relative or absolute measure of a key performance parameter compared to acceptance criteria. For instance, 'design margin' is the relationship between SSC state and the design allowable state; 'high confidence of low probability of failure margin' is the 95% confidence of not exceeding a 5% probability of failure of the SSC state compared to a load descriptor; 'median failure margin' is the best estimate of margin (50% probability of failure compared to a load descriptor).

Plant margin

In the context of Ref. [3], the level of a hazard, generally beyond the design basis, that compromises the safety of the plant according to a specified metric, such as plant high confidence of low probability of failure margin and plant median margin. Here, the compromising of safety means that the plant is rendered incapable of achieving safety objectives under the impact of the hazard.

2. ELEMENTS OF ENGINEERING SAFETY EVALUATIONS

2.1. ASSUMPTIONS

For the design or evaluation of nuclear facilities subjected to extreme human induced external events, a series of assumptions needs to be made on the basis of which the evaluation or design is performed. The assumptions are to be agreed with the regulatory authority, and examples include:

(a) Loss of off-site power: This may occur owing to failure of on-site or off-site physical elements, such as switchyard (on-site) or grid infrastructure (most likely transmission towers, although substations and power generation plants could also be vulnerable). The hypothesized event (e.g. an aircraft crash)

may easily affect on-site elements, such as the switchyard, and still possess energy to impact buildings and structures.

(b) Plant state when event occurs: The external event can take place, for example, at full power and normal state, during hot shutdown due to advance warning of extreme human induced event occurring imminently, and at shutdown condition during refuelling outage.

(c) Time before aid from outside the plant boundary can arrive.

2.2. KEY ELEMENTS

As introduced in Section 1.1, the engineering safety evaluation process is composed of five phases:

— Phase 1: Event identification;
— Phase 2: Hazard evaluation and load characterization;
— Phase 3: Design and evaluation approaches to SSCs;
— Phase 4: Plant performance assessment and acceptance criteria;
— Phase 5: Operator response.

2.2.1. Phase 1: Event identification

Phase 1 is the first and highest level of screening implemented in the process. In this phase, human induced external events with the potential to cause unacceptable consequences are identified for further evaluation. IAEA Safety Standards Series No. NS-R-3 (Rev. 1), Site Evaluation for Nuclear Installations [4], identifies aircraft crashes and associated phenomena (impacts, fires and explosions), other explosions, fire and hazardous material releases as human induced events to consider.

IAEA Safety Standards Series No. NS-G-3.1, Human Induced External Events in Site Evaluation for Nuclear Power Plants [5], supports the requirements in NS-R-3 (Rev. 1) [4]. Tables 1–3 are reproduced from NS-G-3.1 [5], linking potential sources of human induced events to potential initiating events, to event development and to impact on the nuclear power plant (or other nuclear installation). Table 1 categorizes sources of events as stationary or mobile and identifies potential initiating events (also called hazards). Table 2 lists potential evolution of the events into specific loading conditions on the installation. Table 3 expands on the loading conditions on the installation and their consequences, indicating the parameters that can be used to assess the impact on the SSCs of the facility.

TABLE 1. IDENTIFICATION OF SOURCES AND ASSOCIATED INITIATING EVENTS

Facilities and transport systems to be investigated	Relevant features of the facilities and traffic	Initiating event
Stationary sources		
Oil refinery, chemical plant, storage depot, broadcasting network, mining or quarrying operations, forests, other nuclear facilities, high energy rotating equipment	Quantity and nature of substances Flow sheet of process involving hazardous materials Meteorological and topographical characteristics of the region Existing protective measures in the installation	Explosion Fire Release of flammable, explosive, asphyxiant, corrosive, toxic or radioactive substances Ground collapse, subsidence Projectiles Electromagnetic interference Eddy currents into the ground
Military facilities (permanent and temporary)	Types of activities Quantities of hazardous materials Features of hazardous activities	Projectile generation Explosion Fire Release of flammable, explosive, asphyxiant, corrosive, toxic or radioactive substances
Mobile sources		
Railway trains and wagons, road vehicles, ships, barges, pipelines	Passage routes and frequency of passage Type and quantity of hazardous material associated with each movement Layout of pipelines including pumping stations, isolation valves Characteristics of the vehicle (including protective measures) Meteorological and topographical characteristics of the region	Explosion Fire Release of flammable, explosive, asphyxiant, corrosive, toxic or radioactive substances Blockage, contamination (such as from an oil spill) or damage to cooling water intake structures Impacts of derailed vehicles

TABLE 1. IDENTIFICATION OF SOURCES AND ASSOCIATED INITIATING EVENTS (cont.)

Facilities and transport systems to be investigated	Relevant features of the facilities and traffic	Initiating event
Mobile sources		
Airport zone	Aircraft movements and flight frequencies Runway characteristics Types and characteristics of aircraft	Abnormal flights leading to crashes
Air traffic corridors and flight zones (military and civil)	Flight frequencies Types and characteristics of aircraft Characteristics of air traffic corridors	Abnormal flights leading to crashes

Source: Table I of NS-G-3.1 [5].

TABLE 2. EVOLUTION OF EVENTS AND IMPACT ON THE NUCLEAR POWER PLANT

Initiating event	Development of event	Possible impact of each event on the plant[a]
Explosion (deflagration, detonation)	Explosion pressure wave Projectiles Smoke, gas and dust produced in the explosion can drift towards the plant Associated flames and fires	(1) (2) (3) (4) (5) (6) (7)
Fire (external)	Sparks can ignite other fires Smoke and combustion gas of fire can drift towards the plant Heat (thermal flux)	(3) (4) (5) (6)

TABLE 2. EVOLUTION OF EVENTS AND IMPACT ON THE NUCLEAR POWER PLANT (cont.)

Initiating event	Development of event	Possible impact of each event on the plant[a]
Release of flammable, explosive, asphyxiant, corrosive, toxic or radioactive substances	Clouds or liquids can drift towards the plant and burn or explode before or after reaching it, outside or inside the plant Clouds or liquids can also migrate into areas where operators or safety related equipment can be prevented from functioning	(1) (2) (3) (4) (5) (6)
Aircraft crashes or abnormal flights leading to crashes, collision of planes, projectiles Vehicle impacts	Projectiles Fire Explosion of fuel tanks	(1) (2) (3) (4) (5) (6)
Ground collapse	Ground collapse Interference with cooling water systems	(7) (8) (9)
Blockage or damage to cooling water intake structures	Interference with cooling water systems	(12)
Electromagnetic interference	Electromagnetic fields around electrical equipment	(10)
Eddy currents into ground	Electric potential into ground	(11)

Source: Table II of NS-G-3.1 [5].

[a] See Table 3 for an explanation of the numerals.

TABLE 3. IMPACT ON THE NUCLEAR POWER PLANT AND CONSEQUENCES

Impact on the plant		Parameters	Consequences of impact
(1)	Pressure wave	Local overpressure at the plant as a function of time	Collapse of parts of structure or disruption of systems and components
(2)	Projectile	Mass Velocity Shape Size Type of material Structural features Impact angle	Penetration, perforation or spalling of structures or disruption of systems and components Collapse of parts of structure or disruption of systems and components Vibration induced false signals in equipment
(3)	Heat	Maximum heat flux and duration	Impaired habitability of control room Disruption of systems or components Ignition of combustibles
(4)	Smoke and dust	Composition Concentration and quantity as a function of time	Blockage of intake filters Impaired habitability of control room and other important plant rooms and affected areas
(5)	Asphyxiant and toxic substances	Concentration and quantity as a function of time Toxicity and asphyxiant limits	Threat to human life and health and impaired habitability of safety related areas Prevention of fulfilment of safety functions by operators
(6)	Corrosive and radioactive liquids, gases and aerosols	Concentration and quantity as a function of time Corrosive, radioactive limits Provenance (sea, land)	Threat to human life and health and impaired habitability of safety related areas Corrosion and disruption of systems or components Prevention of fulfilment of safety functions
(7)	Ground shaking	Response spectrum	Mechanical damage
(8)	Flooding (or drought)	Level of water with time Velocity of impacting water	Damage to structures, systems and components
(9)	Subsidence	Settlement, differential displacement, settlement rate	Collapse of structures or disruption of systems and components, including buried pipes, cables

TABLE 3. IMPACT ON THE NUCLEAR POWER PLANT AND CONSEQUENCES (cont.)

Impact on the plant	Parameters	Consequences of impact
(10) Electromagnetic interference	Frequency band and energy	False signals on electric equipment
(11) Eddy currents into ground	Intensity and duration	Corrosion of underground metal components Grounding problems
(12) Damage to water intake	Mass of the ship, impact velocity and area, degree of blockage	Unavailability of cooling water

Source: Table III of NS-G-3.1 [5].

Phase 1 comprises three steps: general assessment of potential events, consequence evaluation, and screening and categorization of events. These steps are described in the following subsections.

2.2.1.1. Step 1: Event assessment

Event assessment means a complete identification and evaluation of previously and newly defined events, categorizing them for inclusion in the evaluation. An event assessment comprises: (i) identification of the event; (ii) definition of the loading conditions associated with the event; and (iii) selection of the evaluation process (including methodologies of evaluation and parameters to be considered), consequence identification and acceptance criteria.

Comprehensive and well organized human induced event assessment programmes need to be conducted by the operator of the nuclear power plant and reviewed, as appropriate, by national, state and local government organizations, as these organizations are the source of much of the information on the hazards evaluated.

2.2.1.2. Step 2: Consequence evaluation

In this step, the Member State identifies the consequences of interest to determine the events to be taken into account. Typical considerations include:

(a) Safety of the public: To control the radiation exposure to people during operational and accidental states, which is the overarching metric, with

intermediate metrics such as:
— Core damage frequency;
— Containment and containment systems failure;
— Large early release frequency;
— Release of radioactive material to the environment (dispersion in air, water and ground);
— Collateral effects (e.g. explosions and release of hazardous materials).

(b) Environmental consequences: Short, medium and long term effects on the environment (air, water and ground).
(c) Safety of plant personnel: Short, medium and long term health and welfare of plant personnel.
(d) Energy security of the Member State: The need for the power generated by the nuclear power plant for Member State welfare.
(e) Economic considerations: Short, medium and long term effects on the Member State economy.

2.2.1.3. Step 3: Screening and categorization of events

This step is the decision process, in which the Member State screens and categorizes the identified events from Step 1, and applies the consequence criteria from Step 2 to define the events to be considered for the nuclear power plant of interest. Categories include:

(a) Not to be considered: This includes those events that are not applicable owing to reasons such as:
— Physical conditions of the nuclear power plant of interest (e.g. an event defined by a barge carrying large quantities of chlorine, which could be accidently released, but with no navigable water near the nuclear power plant);
— Events that remain the responsibility of the Member State to ensure no impact on the nuclear power plant (e.g. a state owned dam whose water should be systematically released to prevent overtopping or dam failure under extreme flooding conditions).
(b) DBEEs: Generally, these are events considered in the design.
(c) DEEs: These are rare and extreme events for which realistic, rather than conservative assumptions and acceptance criteria, can be used. They are the principal subject of the design and evaluation process and methodology of Refs [2, 3]. In existing plants, these events were generally not considered in the design.

2.2.1.4. Phase 1 end products

Phase 1 end products are:

(a) A comprehensive list of identified human induced events that may be applicable to the nuclear power plant (Step 1);
(b) Documentation of the Member State defined consequence criteria, which will be the basis for screening in, or out, human induced external events (Step 2);
(c) The disposition of the identified human induced events into three bins (Step 3).

DBEEs and DEEs are not to be considered.

2.2.2. Phase 2: Hazard evaluation and load characterization

The result of Phase 1 is the list of DBEEs and DEEs and their specification as input to Phase 2. In Phase 2, the list is refined by more detailed assessment of the range of potential events for their applicability to the specific nuclear power plant (or other nuclear installation) under design or assessment.

In Phase 2, a second level of screening based on-site and nuclear power plant specific characteristics is implemented. Typical screening parameters to be applied in this phase are probability, magnitude and distance of event specifics, and on-site characteristics (e.g. design conditions and zones of influence). These screening parameters are discussed in Section 3.1. An additional consideration is the type and number of co-located facilities on the site (see Section 3.1.5.1).

The screened-in human induced external events are further evaluated and loading functions are defined for the engineering evaluation. Additional screening may be performed at this stage. The result is a final list of events to be considered in the evaluation.

The load characterization is the link between the events and the definition of the loading environment for the plant engineering organization to evaluate. The resulting matrix of loading conditions produced by the events is to be applied to the entire facility or to portions of it (see Table 4).

Tables 5–7 expand on Scenario 1 in Table 4 (aircraft impact event) to identify the following parameters for engineering evaluation: impact, heat/fire and vibration. Reference [2] describes the engineering evaluation process in detail. These matrices, with their backup data, define the engineering loading environments.

TABLE 4. EXTREME ENVIRONMENT MATRIX

		PHYSICAL LOADING CONDITIONS						
Scenario No.	Scenario description	Impact	Blast	Heat/fire	Hazardous materials release	Smothering	Flooding	Other
1	Crash of a large passenger aircraft with a fully fuelled fuselage into a nuclear power plant	1, 2, 3 (Table 5)	No	1, 2 (Table 6)	No	No	No	No
2	Chlorine release	No	No	No	1 (Table A–6)	No	No	No
3	Blast pipeline	No	1 (Table A–10)	3, 4 (Table A–8)	2, 3 (Table A–9)	No	No	Debris

Note: Numbers under the physical loading conditions columns are explained in the specific table in parentheses. This example is explored in greater detail in the Annex.

TABLE 5. IMPACT PARAMETER DEFINITION MATRIX: SCENARIO 1 (AIRCRAFT IMPACT)

		MISSILE IMPACT					ANCILLARY EFFECTS			
Missile No.	Description	Mass (kg)	Shape/configuration	Impact angle	Impact velocity (m/s)	Relative hardness	Fire	Explosion	Vibration	Other
1	Crash of a large passenger aircraft with a fully fuelled fuselage into a nuclear power plant	396 900	Fuselage 40 m^2/52 m^2	3–10° to horizontal	110	Flexible	No	No	1, 2, 3 (Table 7)	No
2	Large passenger aircraft engines as projectiles	4 300	Circular body 2.7 m fan diameter	3–10° to horizontal	110	Semi-rigid	No	No	No	No
3	Debris	5 000	Rigid body	3–10° to horizontal	110	Rigid	No	No	No	No

TABLE 6. HEAT/FIRE PARAMETER DEFINITION MATRIX: SCENARIO 1 (AIRCRAFT IMPACT)

		FIRE SOURCE OUTSIDE FACILITY					COMBUSTIBLES INSIDE FACILITY				
Fire No.	Description	Combustible/ ignition	Quantity (L)	Spreading surface	Heat potential/ temp. (°C)	Burn duration (h)	Building/ yard	Quantity (L)	Type	Ignition likelihood	Burn duration (h)
1	Jet fuel fire from a large passenger aircraft	Yes	216 000	Half surface of outer shell	1 200	<1	Inside outer shell	1 500	Kerosene	Yes	1.5
2	Horizontal crash of a large passenger aircraft, with a fully fuelled fuselage, on the yard of a nuclear power plant	Yes	216 000	60 m × 500 m	1 200	1–8	No	No	Kerosene	No	No

TABLE 7. VIBRATION PARAMETER DEFINITION MATRIX: SCENARIO 1 (AIRCRAFT IMPACT)

	LOADING FUNCTIONS				
Vibration loading condition No.	Load application (building)	Location (elevation)	Loading function time histories		
			Rigid target	Flexible target	Flexible/non-linear target
1	Reactor building	All	No	No	Yes
2	Emergency cooling water building	All	No	No	Yes
3	Diesel generator building	All	No	Yes	No

2.2.2.1. Phase 2 end products

Phase 2 end products are:

(a) A potentially reduced set of screened-in human induced external events for detailed evaluation;
(b) The engineering loading conditions to be considered for each of the screened-in events to be evaluated.

2.2.3. Phase 3: Design and evaluation approaches to structures, systems and components

Phase 3 broadly covers many topics applicable to the evaluation of existing nuclear power plants and new designs. Phase 3 is where the majority of the engineering studies are performed.

Systems modelling is treated in detail in Ref. [3]. There are several possible approaches as a function of the characteristics of the screened-in events:

— Demonstrating that design loading conditions and/or the design robustness encompass the loading conditions of the screened-in event (discussed in Section 3);
— Probabilistic safety assessment (PSA), detailed or simplified, discussed in detail in Ref. [3];
— Safety margin assessment (SMA), discussed in detail in Ref. [3].

PSAs are composed of the following elements:

(a) Hazard analysis leading to initiating events: The human induced external events are the result of the hazards, which materialize in initiating events. Each of the events may directly, or indirectly, generate event sequences within the plant that have the potential to lead to core damage, containment failure and radioactive material release. For DEEs, in the order of two to ten initiating events may be identified for evaluation.
(b) Plant response: Two aspects of plant response are of interest — plant system behaviour and the behaviour of SSCs to the imposed loading conditions for the events. Plant systems are typically modelled by a combination of event trees and fault trees. The event (e.g. an aircraft crash) is the initiator of the accident sequences. Plant accident sequences are initiated by a faulted condition, such as a loss of coolant accident, and are modelled by event trees. The ability of the plant systems to mitigate the consequences of the

faulted condition depends on the degradation or failure of those safety systems. For example, assuming a coincidental loss of off-site power, an aircraft crash causes a loss of coolant accident, and the tertiary effect of the aircraft crash is a fire in the yard that damages the emergency power system.

(c) Fragility analysis: The analysis is performed on fragility functions of SSCs subjected to the DEE loadings of the event. Fragility is defined as the probability of failure as a function of the size of the input load. Generally, the fragility function is in terms of a single load parameter. This single parameter could integrate a number of factors into the single parameter; for example, for an aircraft crash, variability associated with impact parameters, such as velocity, angle and location of impact, and physical characteristics, such as the mass of the aircraft, could be integrated into a resulting single parameter. Alternatively, a multivariate fragility analysis could be performed, but this is not typically used in nuclear facility PSA methodology. For simplified evaluations, fragility functions for SSCs may be assumed to be binary (i.e. zero or one), leading to screening of SSCs based on the location relative to the damage footprint, or zone of influence.

(d) Accident sequence and systems analysis: In the majority of PSA applications for internal and external events, event trees are developed to model accident sequences and fault trees are developed to model failures of elements within the event trees, such as systems and structures. For the current applications of PSA modelling of human induced events, existing systems models for internal and external initiators provide a valuable starting point. Modifications to these systems models are required to include failures normally not considered credible for previously modelled hazards. One example is underground cable chases, which may be assumed to be robust against internal events and external events, such as earthquakes, wind storms and floods. However, a jet fuel fire in the yard may seep into the cable chase and cause failure. Hence, for specific events, previously screened out components need to be revisited for potential inclusion.

(e) Plant damage state: Accident sequences lead to core damage or core melt end points to which containment performance trees need to be added. The analyst needs to recognize that containment performance may be directly, or indirectly, affected by the loading conditions of the containment systems.

(f) Containment assessment: Containment performance criteria vary among Member States. In some Member States, containment damage and failure may be allowed if core damage failure does not occur. In addition, emergency management equipment as specified in emergency management equipment guidance (EMEG) and diverse and flexible coping strategies (FLEX) are important elements in containment performance achievement [6, 7].

(g) Off-site release: Off-site release may or may not be considered depending on the risk acceptance criteria.

In using the plant specific PSA approach, each event is in theory modelled by a set of event trees and fault trees similar in structure but with significant differences in loading conditions and consequently plant failure probabilities. To improve efficiencies of the analyses, the enveloping of loading conditions needs to be considered to the extent possible, assuming this does not cause excessively conservative results.

Many nuclear installations, in particular nuclear power plants, have developed a plant specific PSA. These studies typically model internal events and, in some cases, external events, such as earthquakes, fire, flooding and high wind loads, including tornadoes. The plant specific PSA can be adapted to perform an in depth safety analysis for extreme human induced external events. The adaptation of the PSA for this analysis has some advantages, including:

— The existing plant logic models are available, can be adapted and used, and these models are the most accurate description of plant behaviour.
— End metrics consistent with high level acceptance criteria can be modified and used (e.g. core damage frequency and large release frequency can be calculated).
— Relative risk ranking of events can be made.
— Risk ranking of overall effectiveness of existing and proposed SSCs is possible.
— Effects of human error and unavailability of systems can be included.

On the other hand, the PSA approach also has some disadvantages. Unless a very simplified PSA approach is used, such as a simplified event tree method, it is only cost effective to use the PSA approach if an internal events PSA has been performed. It is preferable that both internal events and external events, such as earthquakes, high winds, floods or fire, are modelled using PSA techniques. Then, it is only necessary to modify the systems models to include those basic events that were screened out for previous studies but are potentially relevant to the human induced external events. In addition, fragility functions are required for a large number of components, depending on the detail of the systems models and the number of events. The total number of SSCs may require significant effort even after grouping and screening components according to similar behaviour and capacities. Furthermore, specialized expertise is required of the engineering team that develops these fragility functions.

Another possibility is the SMA. In general, the SMA procedure comprises the following elements:

(1) Input given by the extreme environment definition matrices (see Tables 4–7).
(2) Definition of overall nuclear facility performance criteria when subjected to the extreme external events: For example, for a nuclear power plant subjected to a DEE, the overall performance criteria may be defined as hot or cold shutdown for 24 hours after the event occurs. A further assumption is that additional aid from outside the plant boundary can be effectively mobilized within the 24 hour period. The performance criteria, including the duration of plant shutdown before aid from outside the plant can be mobilized, need to be established.
(3) Assumptions for the engineering evaluations: For example, loss of off-site power conditions, operating state of facility (full or partial operation), system criteria (redundancy) or SSC capacity criteria (code based or less conservative).
(4) Definition of one or more ways to achieve safe shutdown or success paths.
(5) Identification of SSCs that comprise the safe shutdown paths and are required to function during and after the event, given the aforementioned assumptions: Definition of the specific functions these SSCs need to perform during and after the event. The SSCs are itemized on the selected equipment list (SEL).
(6) Evaluation of SSC capacity (items on the SEL) when subjected to the extreme loading conditions specified: For the SMA, the measure of capacity needs to be established when subjected to the specified loading conditions (e.g. the high confidence of low probability of failure, median centred capacity or other criteria). This step entails in-office and in-plant evaluations. The in-plant evaluations are the plant walkdowns.
(7) Definition of a measure of plant capacity, such as the size of the event, for which there is best estimate likelihood that the nuclear power plant will achieve hot or cold shutdown (or other end metrics): The plant end state is compared with the acceptance criteria.

2.2.3.1. Phase 3 end products

Phase 3 end products are:

(a) Execution of assessments according to the agreed upon methodologies for human induced external events (i.e. the demonstration of design robustness, PSA or SMA);

(b) Results of the assessments for comparison with acceptance criteria;
(c) Sensitivity study results, if performed.

2.2.4. Phase 4: Plant performance assessment and acceptance criteria

Acceptance criteria are in the form of end metrics that may be risk oriented, for example core damage frequency (conditioned on a human induced external event occurring), or may be in the form of capacity values, such as best estimate or high confidence that the nuclear power plant reaches cold shutdown when subjected to the event (e.g. the impact of a specified aircraft). Other important items are spent fuel pool structural integrity and cooling.

The concept of a tiered approach to defining acceptance criteria is introduced. For example, the consequences of less severe events on the nuclear power plant of interest need to meet more stringent acceptance criteria than those of the most severe events. For less severe events, requirements include redundant success paths to arrive at safe shutdown and conservatism in defining both the environmental loading functions and the performance criteria of SSCs — essentially elastic material behaviour for structures, components, equipment and distribution systems. For the most severe events, verifying a single success path to safe shutdown using realistic analyses may be acceptable. Table 8 presents an example following this concept. Reference [2] follows this approach when discussing structural capacity acceptance criteria.

The information in Table 8 highlights the importance of defining the required civil structure functional behaviour when subjected to human induced external events. Typically, civil structure functional requirements range from leaktightness (e.g. containment) to providing structural support of systems, equipment, components and distribution systems important to safety (e.g. anchorage), to providing barriers to protect SSCs important to safety (e.g. fire barriers and explosive protection walls). It is the civil structure function that requires design and evaluation. The performance assessment results are compared with acceptance criteria and on this basis the vulnerabilities are identified.

2.2.4.1. Phase 4 end products

Phase 4 end products are, as a function of the event, plant performance assessment and identified vulnerabilities for the nuclear power plant subjected to DBEEs and DEEs.

TABLE 8. PLANT ACCEPTANCE CRITERIA: EXAMPLE

Event level	Civil structure	Safety functions	No. of shutdown paths	No. of decay heat removal paths	Capacity assessment
DBEE	Essentially elastic	Safe shutdown Decay heat removal Containment	Multiple	Multiple	Conservative
DEE 1	Plastic	Safe shutdown Decay heat removal Containment	2	2	Best estimate (median)
DEE 2	Plastic	Safe shutdown Decay heat removal	1	1	Best estimate (median)

Note: DEE — design extension external event; DBEE — design basis external event.

2.2.5. Phase 5: Operator response

Given the results of Phase 4, the operator may take additional steps to address the identified vulnerabilities. These steps include prioritization and implementation of the compensatory and/or upgrading measures (e.g. design changes, operating procedure changes and administrative measures).

2.3. DESIGN AND EVALUATION PRINCIPLES

2.3.1. Event agnostic effects: Loading conditions

A philosophy of design and evaluation consisting of defining a set of event agnostic loading conditions and treating those as generally enveloping anticipated future defined human induced external events is prudent (see Section 5.2).

2.3.2. Defence in depth

The most general approach to the design and evaluation of nuclear power plants subjected to human induced external events is the utilization of the concept of defence in depth in the safety domain [1, 8]. The layers of defence in depth may be intrinsic or extrinsic, on-site or off-site. Furthermore, some layers of defence in depth will be related to prevention of the event and others to mitigation when, for example, core damage is considered the metric. Section 4 presents the basic concepts of defence in depth when applied to the assessment of safety against human induced external events.

2.4. INPUT TO NUCLEAR POWER PLANT ASSESSMENT

In accordance with the phases defined in Section 2.2, the following aspects are to be stipulated by the Member State as input to the safety evaluation against human induced external events.

2.4.1. Plant performance criteria

Plant performance criteria may be defined in one or more of the following:

(a) High level requirements, such as core damage frequency or large early release frequency, lower than threshold values. Methods of demonstration to be specified.
(b) Plant safe state achieved: hot or cold shutdown and cooling over a required time period (e.g. 24 or 72 hours).
(c) High confidence of survival of specified SSCs, such as containment structure and containment systems, spent fuel pool and spent fuel pool support functions.

2.4.2. Plant acceptance criteria

Acceptance criteria are innately tied to plant performance criteria, as follows (see also Table 8):

(a) Risk metrics are in terms of frequencies of occurrence per annum, and the acceptance criteria may be a fraction of the required core damage frequency or large early release frequency specified in the Member State.

(b) Plant safe states may be ensured through the identification of success paths to arrive at hot or cold shutdown. One can envision the Member State (or the operator) requiring:

(i) Conservatism in design or evaluation for less severe scenarios, such as requiring redundancy in success paths and conservatism in the evaluation processes, thereby ensuring with high confidence that hot or cold shutdown is achieved and future restart of the nuclear power plant is likely;

(ii) For extreme DEEs, liberalized acceptance criteria may be permitted with one shutdown path ensured, SSC acceptance criteria liberalized, best estimate or median centred evaluations permitted, among other things.

Plant acceptance criteria are then defined by the Member State as [9]:

— DBEE: All safety system functions and capabilities continue to be available for DBEEs. The design provides for the ongoing availability of fundamental safety functions during DEEs. These provisions will depend on the severity of the event.
— DEE 1: For more severe events, there is to be a safe shutdown path that comprises at least one means of reactor shutdown, fuel cooling and retention of radioactive material from the reactor. There should be sufficient structural integrity to protect important systems. Two such success paths are to be identified, where practicable.
— DEE 2: For extreme external events, there is to be at least one means of reactor shutdown and core cooling. Degradation of the containment barrier may allow the release of radioactive material. However, the degradation should be limited, with the goal that the dose acceptance criteria are not exceeded. In these cases, the response includes on-site and off-site emergency measures.

2.4.3. Operational status

Full power and shutdown operational modes for maintenance and refuelling are to be considered.

2.4.4. Consideration of multi-unit sites

Multi-unit sites and multi-use sites present special circumstances. Special attention needs to be given to common cause events, in particular considerations such as shared systems between units or assumptions about assistance of one unit to another. Such assistance may be unavailable owing to physical situations or the confusion that may accompany a DEE. Multi-use sites, such as chemical plants co-located at, or near to, a nuclear power plant may present additional loading environments if impacted by the DEE, for example breaching of chemical storage tanks due to an accidental explosion thereby releasing a hazardous material could be a threat to the adjacent nuclear power plant.

2.4.5. Severe accident prevention and management

In addition to existing measures, additional measures to prevent and mitigate severe accidents have been implemented in many Member States in the light of the Fukushima Daiichi accident, on 11 March 2011. For instance, Refs [6, 10] detail an approach broadly termed FLEX to supplement existing safety systems in nuclear power plants.

As stated in Ref. [6]:

> "The consequences of postulated beyond-design-basis external events that are most impactful to reactor safety are loss of power and loss of the ultimate heat sink. This document outlines an approach for adding diverse and flexible mitigation strategies — or FLEX — that will increase defense-in-depth for beyond-design-basis scenarios to address an ELAP [extended loss of AC power] and loss of normal access to the ultimate heat sink (LUHS) occurring simultaneously at all units on a site. ...
>
> "FLEX consists of the following elements:
> - **Portable equipment that provides means of obtaining power and water to maintain or restore key safety functions for all reactors at a site.** This could include equipment such as portable pumps, generators, batteries and battery chargers, compressors, hoses, couplings, tools, debris clearing equipment, temporary flood protection equipment and other supporting equipment or tools.
> - **Reasonable staging and protection of portable equipment from BDBEEs [beyond design basis external events] applicable to a site.** The equipment used for FLEX would be staged and reasonably protected from applicable site-specific severe external events to provide reasonable

assurance that *N* sets of FLEX equipment will remain deployable following such an event [where *N* = number of units on a site].

- **Procedures and guidance to implement FLEX strategies.** FLEX Support Guidelines (FSG), to the extent possible, will provide pre-planned FLEX strategies for accomplishing specific tasks in support of Emergency Operating Procedures (EOP) and Abnormal Operating Procedures (AOP) functions to improve the capability to cope with beyond-design-basis external events.
- **Programmatic controls that assure the continued viability and reliability of the FLEX strategies.** These controls would establish standards for quality, maintenance, testing of FLEX equipment, configuration management and periodic training of personnel.

"The FLEX strategies will consist of both an on-site component using equipment stored at the plant site and an off-site component for the provision of additional materials and equipment for longer-term response."

To incorporate FLEX into the evaluations being performed, several of the issues described as elements of the FLEX system need to be verified:

(a) If there is no redundancy, diversity and separation (i.e. if FLEX is a single train system), then verification is required of the following:
  (i) There will be no damage to FLEX equipment from the extreme human induced external event (no damage from impact, collateral damage from debris, damage from fire, or from smoke from aircraft crash; and no damage from an explosion);
  (ii) There will be no damage in the yard to impede the personnel from performing their required function to implement FLEX;
  (iii) There will be no missing equipment or inoperative equipment;
  (iv) There will be no human error, so the single train system is 100% operative (zero failure).

(b) If FLEX is a single train system, the likelihood, potentially a small likelihood based on evaluations as described above, that a single train FLEX system does not perform its function needs to be considered.

2.5. ASSESSMENTS OF EXTREME PLANT CONDITIONS

In the light of the Fukushima Daiichi accident, some Member States require operators to evaluate extreme plant conditions without consideration for the mechanism to reach such a state, for example:

— Station blackout;
— Loss of primary ultimate heat sink;
— Both occurring simultaneously.

These extreme plant conditions need to be evaluated without consideration of applying probability arguments as to why they are not credible. Within the DBEE and DEE environment, one can easily hypothesize scenarios that could produce these plant states. An example scenario is:

— A loss of off-site power;
— All emergency diesel generators in a single building — this building is disabled by an extreme flood;
— No possible recovery in the short term.

The evaluation needs to include mitigation strategies, implementation and training.

2.6. UNCERTAINTY

In treating extreme external events of all types, it needs to be recognized that all elements of the evaluation are subject to two types of variability or uncertainty: aleatoric and epistemic [11]:

> "Aleatoric uncertainty is the uncertainty inherent in a non-deterministic (stochastic, random) phenomenon. Aleatoric uncertainty is reflected by modelling the phenomenon in terms of a probabilistic model. In principle, aleatoric uncertainty cannot be reduced by the accumulation of more data or additional information (sometimes called 'randomness').

.....

"Epistemic uncertainty is the uncertainty attributable to incomplete knowledge about a phenomenon that affects the ability to model it. Epistemic uncertainty is reflected in ranges of values for parameters, a range of viable models, the level of model detail, multiple expert interpretations, and statistical confidence. In principle, epistemic uncertainty can be reduced by the accumulation of additional information (also called 'modelling uncertainty')."

The composite variability is the total uncertainty including the aleatoric and epistemic uncertainties. In many cases, aleatoric and epistemic uncertainty are modelled by log-normal distributions with the log-normal standard deviation for aleatoric uncertainty represented by β_R and the log-normal standard deviation for epistemic uncertainty represented by β_U. For this case, the logarithmic standard deviation of composite variability, β_c, is expressed as:

$$\beta_c = (\beta_R^2 + \beta_U^2)^{\frac{1}{2}} \quad (1)$$

These concepts are introduced here to emphasize the fact that representations of the extreme human induced external events are subject to uncertainty in the phenomena themselves and in the modelling of the phenomena. These concepts are discussed in significant detail in Ref. [3].

3. EVENT IDENTIFICATION AND LOAD CHARACTERIZATION

3.1. SCREENING OF EVENTS

The first level of screening was introduced in Phase 1 (event identification), as described in Section 2.2. A second level of screening based on-site and nuclear power plant specific characteristics is implemented in Phase 2 as a function of the specific site and nuclear power plant characteristics. Typical screening parameters to be applied in this phase are: probability, distance and magnitude of event specifics, and on-site characteristics (e.g. design conditions and zones of influence). Additional screening-out of events may be possible. For nuclear power plants, the number of co-located facilities on the site is another consideration. Each of these is discussed briefly in the following.

3.1.1. Screening by design robustness

In the design and evaluation process, the inherent strengths in facilities due to the design and construction conditions need to be recognized. SSCs designed to the wide range of conditions imposed by the design may possess significant margin for loading environments owing to defined human induced external events. For extreme external events, the focus is on the SSCs required to safely shut down the facility and to maintain it in a safe state through the time necessary for additional resources from outside the plant to assist, if necessary.

SSCs are designed and evaluated for a large number of conditions:

(a) Structures: Generally, structures provide one or more of the functions of pressure retention, shielding and confinement, and support to systems and components. Structures and structural elements are designed for operational and accident conditions throughout the facility's life. Operating loads include dead load, live load, atmospheric temperature, thermal loads, vibration, radiation effects, pressure retention and aging effects (radiation, corrosion and other material degrading effects). Structures are designed for accidental loads, such as missile impact (internally and externally generated), extreme wind, flood, earthquake, explosions and blasts (internally and externally generated), extreme heat loads, extreme radiation effects, impulse loads due to pipe whip and other phenomena, and heavy load drops. Some of these loading conditions are considered in design to act simultaneously. Design load combinations, along with conservative acceptance criteria, lead to robustness that needs to be taken into account in the evaluation of the consequences of the extreme human induced external events. Load combinations in DEEs often include only the normal operating loads and the load due to the extreme human induced event.

(b) Systems: Generally, systems are designed for a companion set of operating and accident conditions to structures. In addition, system design includes considerations of redundancy of function, and separation, segregation and diversity of trains and elements to provide high reliability for successful system performance for normal operating and accident conditions. This robustness in system behaviour is characterized in systems models, such as fault tree models.

(c) Components: Generally, components are designed for a companion set of operating and accident conditions to structures and systems. However, the environments for which components are designed, qualified and maintained are typically more extensive. Operating conditions mean component

function (e.g. pumps delivering fluid at a specified flow rate) under a wide range of specified conditions (e.g. temperature, humidity, radiation, cooling and vibration). Accident conditions mean components performing required functions during a specified period and under specified environmental conditions.

Nuclear facilities are designed for a wide range of extreme loading conditions. The design basis internal and external events, such as fire, pipe whip, loss of coolant accident, earthquake, extreme winds, explosions and aircraft crashes, provide an envelope of protection for a nuclear facility. It is important to take advantage of this designed protection when evaluating extreme human induced external events. In fact, some scenarios may be screened out due to effectively being bounded by design basis or beyond design basis conditions already considered. Bounding can be demonstrated on the basis of the event (for the whole facility), the extreme load (for each item), or the sizing requirement derived from the loads. The screening process becomes increasingly more difficult as one moves from the event to the load and to sizing.

Design robustness can be physical in the sense of an enveloping of loading conditions for design basis events compared to the imposed loading conditions due to the particular extreme human induced external event being analysed. For example, design for tornado loads may envelope loading conditions due to external explosions. Aircraft impact imposes structure loading conditions of global response, local response and vibration effects, for which either of these effects could be demonstrated to be enveloped by other DBEE loading conditions. Global response could be enveloped by earthquake loading, in-structure response spectra due to design basis earthquake may envelope the spectra due to vibration effects caused by aircraft impact, among other things.

Design robustness may be system wise due to defence in depth and redundancy. Many of the events may be screened out from detailed analysis, because their maximum impact is enveloped by design basis accidents, for example by analysis of transients with the assumption of the worst possible single failure which has been analysed, so their consequences are known.

In some cases, the critical scenarios developed for an extreme human induced external event cannot be enveloped by the analysis of design basis events, but they can be related to scenarios analysed for the assessment of other extreme external events and the plant specific PSA. The fault trees and event trees are helpful in understanding the systems logic (this is discussed in much more detail in Ref. [3]).

3.1.2. Screening by distance and magnitude and by probability

When the events cannot be screened out based on design robustness, two other screening methods are available: screening by distance and magnitude and screening by probability of occurrence.

3.1.2.1. Distance and magnitude screening

Following this method, the minimum distance and the maximum magnitude of the event are postulated with respect to the nuclear power plant site and the potential damaging effects on plant safety are assessed. If the effects are found to be insignificant, the event is screened out with respect to the assessed parameter.

An example where distance and magnitude screening may be effective is the screening of vehicles containing explosives. The plant boundary and Member State administrative procedures may be judged to be effective in keeping vehicles at safe distances from the nuclear power plant SSCs.

Another example is for extreme human induced external events of flood conditions (dam failure or overtopping) and consequent site inundation. It may be possible to assess the maximum flood height and take into account the site topography and possible drain paths to exclude effects on SSCs located at higher elevations. This can be applied to both off-site water sources and on-site tank or piping systems, reducing the possible impact of the flood on equipment important to safety and the number of events to be analysed in detail.

3.1.2.2. Probability screening

Screening by probability is generally more complex and uncertain, but it may be applied to events not screened out by distance or magnitude. The probability level used for screening is generally one or two orders of magnitude smaller than that used for design purposes, in order not to exclude any events due to the approximate nature of the probabilistic screening procedure. For those scenarios not screened out using the distance and magnitude approach, probabilistic screening criteria may be utilized. Generally, a threshold screening criteria is an annual probability of exceedance of 10^{-7}.

3.1.3. Screening by zone of influence

In addition to screening by the many factors discussed previously, this section discusses conceptual tools that may aid in excluding some events from further analysis, as the consequences are deemed to be sufficiently low or success paths for the safe shutdown can be shown to remain intact. On the other hand,

these concepts, when applied, could also serve to identify some safety concerns that are clearly vulnerabilities, without requiring extensive analysis. The concepts are illustrated using an aircraft crash as an example.

In the case of an aircraft crash, the first issue that national authorities need to address is which scenario is to be assessed. This requires definitions of, for example, the types of aircraft, velocities, altitudes and payloads (including the amount of fuel and the existence of passengers and cargo) which are to be used in the analysis. For example, with regard to accidental aircraft crash scenarios, Germany has required designs to deterministically protect against a Phantom or similar, fast military jet crashing into the nuclear power plant site. In contrast, France has ruled out accidental commercial and military crashes based on probabilistic considerations, but it requires consideration of general aviation crashes (Learjet 23 and Cessna 210).

These decisions require taking into account current and predicted air traffic for the country or region. It may involve setting two levels: one for best estimate survival and another (possibly more unreasonably burdensome) for best estimate consequences. Having developed national criteria which may be defined down to a list of site specific approach directions, a case can be made to perform a high level worst case analysis.

Methodologies for the analysis of impact have been developed over a number of years and have been updated, as needed, taking into account new information such as the events of 11 September 2001, testing performed subsequent to that event, and extensive analytical and numerical studies performed. These analyses make use of the results of the extensive work that has been performed on the few cases of aircraft impacts on engineered structures (i.e. buildings). Reference [2] discusses these approaches in detail.

The Pentagon Building Performance Report [12] and World Trade Center Building Performance Study [13] provide useful guidance that may be applied to the nuclear facility case. In Ref. [12], the results of the detailed study assessing the impact indicated that the damage was initially confined to a roughly triangular shape, extending along the direction of the approach. The damage swath was approximately 23–24 m at the point of entry into the building and extended to a depth of approximately 70 m. The damage caused by the landing gear was shown to extend beyond the initial zone of impact. Fire damage, due to burning of the jet fuel and to secondary fires caused by the ignition of on-site combustibles, extended into the areas unaffected by the impact, until contained by the building fire suppression systems.

Hence, the zone of influence concept can be applied for the purpose of preliminary screening. Aircraft crashes and explosions are two events where the zone of influence is a valuable tool. The concept is applied to aircraft crashes by imposing the damage and debris triangles on a scaled representation of a nuclear

plant, aligned along each or all determined approach paths. An approximation of the areas of damage likely to occur to the relevant building can be obtained. The footprint of the fire and smoke damage can be obtained by extending the zone of influence until it is met by a fire barrier that has not been damaged by the initial impact or subsequent debris.

The expectation is that this concept may provide reasonable initial estimates of the damage caused by an aircraft crash on a nuclear facility based on the evidence from past events. Clearly, this methodology could not be directly applied to certain structures within a nuclear facility. Hardened and robust structures, such as the containment building, would provide additional protection when compared to the structure of the Pentagon building. These key structures, whose failure could lead to significant, immediate consequences, would require additional evaluation to ensure that their integrity can be maintained. However, this concept could serve to focus the evaluation on those SSCs critical to the plant achieving safe shutdown, simultaneously eliminating those SSCs that are highly likely to fail in the crash.

Implementation of this concept could result in a visual representation similar to that in Fig. 2 for an aircraft crash in one direction (several diverse directions may be assessed as probable and each would need to be considered).

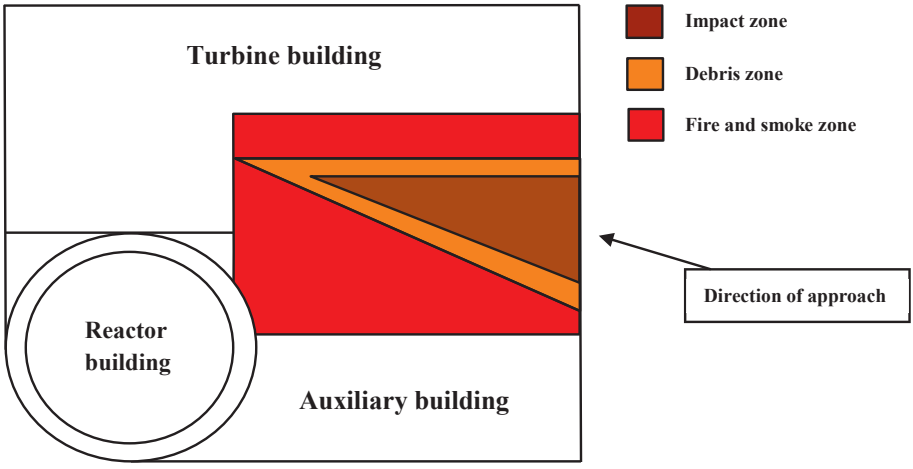

FIG. 2. Simplified schematic of a nuclear power plant indicating the three zones of influence following an aircraft crash.

Assuming a loss of all SSCs contained within the zone of influence, and using the defined success criteria (i.e. the redundancy and survivability requirements), the effect of the aircraft crash on the plant can be estimated. Systematic tools, such as PSAs, simplified event trees and SMAs, can be used along with the zone of influence to determine whether successful shutdown of the nuclear power plant remains feasible for those SSCs outside the zone of influence.

The list of SSCs needed for safe shutdown may populate an SEL plus structures. It should be noted, however, that the SEL will depend greatly on each scenario. In the following example, the emergency cooling system is assumed to be located primarily on the north side of the plant, whereas the shutdown maintenance cooling system is located on the south side, with the reactor building between them. Preliminary analysis has shown that either cooling system may be relied on to maintain the basic cooling requirements for removal of decay heat. The aircraft crash is to be considered a viable event (i.e. not screened out by previous methods). The aircraft is considered to be approaching from the north and the south directions for evaluation by the zone of influence approach. For the case of approaching from the north, and using the zone of influence concept, the emergency cooling system is assumed to be unavailable. Based on reasonable assurances, however, the maintenance cooling will survive the impact, debris and fire, thus ensuring that basic cooling functions are maintained. The opposite would be true for the approach from the south.

Caution is to be taken, however, if this methodology is to be used to exclude scenarios from further consideration. Owing to the significant uncertainty associated with this method, there needs to be a high degree of certainty that the essential safety functions are maintained. Furthermore, for open areas such as a turbine building, the zoning might be an underestimate, while for a cellular structure with many interior walls, such as a control building, the effects might be more restricted. Detailed consideration therefore needs to be given to postulated affected buildings and plants.

The zone of influence methodology may serve to identify clear vulnerabilities. For example, some nuclear power plants may locate the primary and secondary control rooms in close proximity to each other. When the damage footprint is imposed on the plant layout, assuming that it is feasible for the aircraft to approach from an alternative direction, there would be a good possibility that both control rooms may be lost simultaneously, or that the access to the secondary control room may be impeded owing to the severe fires expected.

3.1.4. Example of systematic approach to defining scope: Aircraft impact

Significant effort has been expended internationally to develop cost effective approaches to addressing the issues of extreme human induced external events. In the United States of America, significant effort was devoted to evaluating aircraft impact effects on existing and new nuclear power plants. The methodology, available in Ref. [14], is described here as an example. The focus is on two buildings — the reactor containment building and the spent fuel storage building. These methodologies were developed as a result of the events of 11 September 2001. They are equally applicable to extreme human induced external events of accidental aircraft crashes.

An approach similar to the zone of influence approach is used. The concept of defining areas of consequence for each of the hypothesized impact locations is employed. The areas of consequence are denoted damage footprints. Damage footprints are defined for impact, shock and fire loading conditions. The systematic approach to the evaluation looks first to the buildings containing nuclear fuel and, afterwards, to the buildings housing the equipment for heat removal.

3.1.4.1. Reactor containment building and spent fuel storage building

Two buildings — the reactor containment building and the spent fuel storage building — are to be evaluated for direct aircraft impact effects:

(a) Impact locations to be considered are defined, which are identified based on the aircraft parameters (such as angle of impact and manoeuvrability of aircraft), shielding by topography, nuclear power plant site buildings, transmission lines and other considerations.
(b) Conservative assumptions about the angle of aircraft impact, for example perpendicular to the centreline of the containment building and perpendicular to the spent fuel storage building are made.
(c) Local response, global response and vibration loading conditions are considered.

Damage footprints due to any consequences of the aircraft crash are developed, including structure failure modes, fire and vibration effects. The end product is aircraft impact locations and damage footprints. Studying the effects of an aircraft crash requires evaluations of global structural response, local response and vibration effects, as described in Ref. [2].

3.1.4.2. Heat removal capability

In addition to the evaluation of the reactor containment building and the spent fuel storage building, all structures containing equipment necessary to prevent damage of fuel in the reactor or the spent fuel pool will be identified for screening or evaluation. Front line and support systems needed for safe shutdown of the reactor or continued cooling of the spent fuel pool are identified.

In general, each building identified will be evaluated to define impact locations for structural response. Exterior faces of the buildings will be evaluated to screen out the need for further evaluation or to determine impact locations:

(a) The faces or partial faces of buildings could be screened out from further consideration due to shielding by adjacent structures, intervening structures, or other site features (see the rules in table 3-1 of Ref. [14]).
(b) Faces of buildings that are partially screened out are subdivided into portions for which aircraft impact is possible and not possible;
(c) The impact of multiple buildings during the event is considered, the result being the identification of multiple buildings vulnerable to a single aircraft crash;
(d) Candidates for aircraft impact assessment are the end products.

Damage footprints for each building, for each impact location of the building, and for each mode of failure or excitation (global or local structure response, fire and vibration effects) are developed for evaluation.

Screening rules were developed to aid in the evaluations:

— SSCs are assumed to be unable to perform their function or may perform their required function depending on parameters such as proximity to impact location, proximity to fire initiation, spalling, scabbing and perforation hazards.
— Screened out SSCs (those assumed to not be able to perform their functions) and screened in SSCs are identified for each impact scenario.

Excessive conservatism of screening rules may be reduced by performing more detailed evaluations, such as those defined in Refs [2, 3]. For each impact scenario, a success path to hot or cold shutdown is defined by sets of screened-in SSCs. Finally, the results of the evaluation are documented (safeguard information).

3.1.5. Special topics

3.1.5.1. Type and number of co-located facilities on the site

The type and number of co-located facilities on the site can have positive and negative effects on prevention, detection, control of consequences (normal and severe conditions) and emergency response.

(a) Positive effects: Multiple critical facilities located on a site (or in the vicinity) permit the pooling of resources to prevent or mitigate the consequences of a human induced external event. Various on-site and off-site measures can be deployed cost effectively.
(b) Negative effects: If there are shared systems between nuclear power plant units, or on the site in general, a human induced external event may neutralize the system performance.

The effects need to be carefully considered by key personnel. In addition, site-side planning, which is a plan for co-located facilities, needs to be implemented, with on-site and off-site considerations — including emergency response, such as evacuation of the site, if deemed necessary.

3.1.5.2. Considerations in siting

NS-R-3 (Rev. 1) [4] identifies aircraft crashes and associated phenomena (impacts, fires and explosions), other explosions, fire and hazardous material releases as important human induced events to consider at all stages of the nuclear installation siting, design, construction and evaluation (these events are also considered in Refs [2, 3]).

In addition, NS-R-3 (Rev. 1) [4] emphasizes the link between the site characteristics and the ultimate heat sink and the need to consider human induced events that could cause a loss of function of systems required for the long term removal of heat from the core, such as ship collisions, oil spills and fires.

It is important to note that consideration of human induced external events goes far beyond the site selection stage, since human activities around a selected site can change considerably during the life of the nuclear installation. Hence, the set of human induced external events considered applicable to the facility is not to be frozen at the site selection stage. The site needs to be periodically assessed for the potential new or increased hazards derived from the updated human activities in the site vicinity. Hence, new sites are assumed to follow NS-R-3 (Rev. 1) [4], but existing sites need to demonstrate, through the methodologies of this Safety

Report and Refs [2, 3], that the consequences of current potential human induced events are reduced to acceptable levels.

3.2. EVENT, HAZARD AND LOAD CHARACTERIZATION

3.2.1. Load evaluation for design extension external events

The general approach is to define loading matrices with extensive backup information, which are provided to the engineers for design and evaluation purposes. Examples of these matrices are provided in the following sections to describe the process. The Annex includes three instances of implementation. All examples in this publication are given for illustration purposes only; actual requirements are to be defined by Member States.

3.2.1.1. Design extension external events loading matrix

This matrix is a road map connecting extreme human induced external events to DEE loading. For each event identified in Phase 2, the potential extreme environment to be imposed on the facility is identified. An example is provided in Table 9. This example is purely hypothetical, and the extent of the phenomena and the parameters defined are not intended to be complete.

The columns of the matrix are defined as follows:

(a) Human induced event No.: An alphanumeric identifier, values ranging from 1 to *N*, where *N* is the total number of events.
(b) Human induced event description: A brief description of the event for identification purposes. (Example: Large passenger aircraft crash into a nuclear power plant site.)
(c) Physical loading conditions: Numerical identifiers on the type and specifics of loading conditions caused by the event. The numerical identifiers correlate directly with the other loading matrices: impact, explosion and blast, heat and fire, hazardous material release and other environmental consequences. The end result provides guidance to plant engineering on engineering disciplines required in the evaluation. It also provides background on the source of environmental load combinations required.
(d) Impact matrix: One or more impact loading conditions identified by number and reference to the impact matrix described in Section 3.2.1.2. (Example: DEE impact loading 1 and 2.)

TABLE 9. EXAMPLE OF DESIGN EXTENSION EXTERNAL EVENT LOADING MATRIX TABLE

		PHYSICAL LOADING CONDITIONS						
Human induced event No.	Human induced event description	Impact matrix	Explosion/blast matrix	Heat/fire matrix	Hazardous material release matrix	Smothering	Flooding	Other
1	Crash of a large passenger aircraft with a fully fuelled fuselage into a nuclear power plant	1, 2	None	1	None	None	None	None
2								
N								

(e) Explosion/blast matrix: One or more explosion or blast DEE loadings identified by number and reference to the explosion/blast matrix described in Section 3.2.1.3. (Example: 'None', i.e. no explosion or blast loads associated with Event 1 or as ancillary to the aircraft crash.)
(f) Heat/fire matrix: One or more heat or fire loading conditions identified by number and reference to the heat/fire matrix described in Section 3.2.1.4. (Example: Heat/fire environmental loading condition 1.)
(g) Hazardous material release matrix: One or more hazardous material release conditions identified by the number and reference to the hazardous material release matrix described in Section 3.2.1.5. (Example: 'None', i.e. no hazardous material release condition associated with Event 1.)
(h) Smothering, flooding and other phenomena are identified with examples for future consideration (see Section 3.2.1.6 for flooding):
— Smothering, choking or depriving SSCs of necessary air for operation is suggested as a potential concern (e.g. a lack of air to emergency diesel generators could prevent startup and operation). Smothering due to firefighting techniques (i.e. foam) may need to be evaluated.
— Flooding of the site due to internal or external sources may need to be evaluated (e.g. the failure of an up stream dam, which leads to the release of large quantities of water that floods the site).

3.2.1.2. Impact parameter definition matrix

This matrix identifies the impact parameters to be used by plant engineering for the evaluation of SSC capacity. Each Member State defines the characteristics of missiles as rigid, semi-rigid or flexible, and also defines the impact parameters, such as the angle and velocity of impact. An example is provided in Table 10.

The columns of the matrix are defined as follows:

(a) Missile type/No.: Missile load identifier. In general, values range from 1 to *M*, the total number of missile impact scenarios. (Example: Missile No. 1 is the fuselage from a large passenger aircraft, including fuel; Missile No. 2 is the engines.)
(b) Description: Brief description of source of loading condition. (Example: Missile No. 1 is a crash of a large passenger aircraft with a fully fuelled fuselage; Missile No. 2 is the engines.)
(c) Mass: Mass of the missile. (Example: Missile No. 1 is 157 000 kg, including fuel; Missile No. 2 is 4800 kg per engine.)

TABLE 10. IMPACT PARAMETER DEFINITION MATRIX

		MISSILE IMPACT					ANCILLARY EFFECTS			
Missile type/No.	Description	Mass (kg)	Shape/ configuration	Impact angle	Impact velocity	Relative hardness	Fire	Explosion	Vibration	Other
1	Crash of a large passenger aircraft with a fully fuelled fuselage into a nuclear power plant	157 000	Flexible	Defined by Member State	Defined by Member State	Flexible	1	No	Yes	No
2	Large passenger aircraft engines as projectiles	4 800	1.40 m diameter/ hollow cylinder	Defined by Member State	Defined by Member State	Semi-rigid	No	No	No	No
3										
M										

(d) Shape/configuration: General and specific description of missile. Dimensions specified, if available at this stage. (Example: Missile No. 1 is a flexible fuselage, dimensions to be determined; Missile No. 2 engines to be assumed rigid and dimensions as shown.)
(e) Impact angle: Angle or range of potential impact angles taking into account the physics and human capability necessary to achieve objective. (Example: Impact angle in the range of 0–30° from the horizontal.)
(f) Impact velocity: Velocity of missile taking into account the physics and human capability necessary to achieve objective. (Example: 180 m/s.)
(g) Relative hardness: Important parameter in assessing effect of missile on SSCs. Qualitative or quantitative measure. (Example: Missile No. 1 fuselage is considered flexible; Missile No. 2 is considered rigid.)
(h) Ancillary effects: These are effects that are consequential to the direct impact. They may be specified in other places in the specification such as fire in the example. They may be consequences to the impact, such as spalling or scabbing of concrete, which may be an ancillary effect on components in the vicinity of the impact.
(i) Fire: Missile impact causes a fire due to the missile impacting a combustible, such as a diesel oil tank. (Example: Missile No. 1 refers to heat and fire, '1' within the fire matrix, which is a jet fuel fire associated with the aircraft crash; Missile No. 2 has no related fire.)
(j) Explosion: Missile impact causes explosion due to impacting an explosive storage facility in the surrounding area of the plant. (Example: No explosions assumed.)
(k) Vibration: Missile impact causes overall vibration of the impacted building. Vibration can affect sensitive equipment. (Example: Missile No. 1 impact is considered able to produce significant vibration affecting all the building.)
(l) Other: Other hazards identified, such as intruders in coordination with missile attack. (Example: No other hazards identified.)

3.2.1.3. Explosion/blast parameter definition matrix

This matrix identifies a simplified set of parameters for explosion and blast loading conditions for the use of plant engineering in the evaluation of SSC capacity. An example is provided in Table 11.

TABLE 11. EXPLOSION/BLAST PARAMETER DEFINITION MATRIX

		EXPLOSION PARAMETERS		PRESSURE PULSE	
Explosion No.	Description	TNT equivalent mass	Reference distance	Incident (side-on)	Reflected
1					
2					

The columns of the matrix are defined as follows:

(a) Explosion No.: Explosion and blast condition identifier. Values range from 1 to the total number of blast conditions considered. (Example: No explosion or blast conditions were assumed.)
(b) Description: Description of the explosion scenario.
(c) Explosion parameters: Table 11 presents example descriptors of the characteristics of the explosion. For general descriptions, trinitrotoluene (TNT) equivalent mass and reference distance (measured from a facility reference point) is the most general information. Other descriptors can be given if the multi-energy method is used [2].
(d) Pressure pulse: Table 11 presents example descriptors of the pressure pulse created by the explosion. Specific information about the incident and reflected waves would be developed for the nuclear power plant under evaluation. Typically, side-on and reflected peak overpressures are used. The details are a function of numerous site specific characteristics.

3.2.1.4. Heat/fire parameter definition matrix

This matrix identifies the heat and fire characteristics to be used by plant engineering for the evaluation of SSC capacity. An example is provided in Table 12.

The columns of the matrix are defined as follows:

(a) Fire No.: Heat and fire condition identifier. Values range from one to the total number of fire conditions.
(b) Description: Brief description of the source of the fire. (Example: Jet fuel fire from a large passenger aircraft.)

TABLE 12. HEAT/FIRE PARAMETER DEFINITION MATRIX

		FIRE SOURCE OUTSIDE FACILITY					FIRE SOURCE OR COMBUSTIBLES INSIDE FACILITY				
Fire No.	Description	Combustible/ ignition	Quantity (kg)	Heat potential/temp. (°C)	Burn duration (h)	Other	Building/ yard	Quantity (kg)	Type	Ignition likelihood	Burn duration (h)
1	Jet fuel fire from a large passenger aircraft	Yes	50 000	1 000	1–8						
2											
3											
N											

(c) Fire source outside facility: These entries define the fire hazard on the basis of its source being outside the facility. For an aircraft crash or other similar event, the distribution of the combustibles within and outside the facility boundary is important. Two obvious distributions are on plant yard and penetration into buildings; another is outside the facility boundaries, which could inhibit access by emergency responders and others. Examples of important parameters include type and quantity of combustible, estimates of heat potential and temperature, and duration of burn. (Example: Jet fuel from a large passenger aircraft spilled and ignited. No penetration of building. Quantity is 50 000 kg. Burn duration at high temperature, 1000°C, is 1 hour maximum and 5–7 hours of residual fire at 300°C.)

(d) Fire source or combustibles inside facility: These entries define the fire hazard on the basis of the source being inside the facility or ignited inside as a consequence of an outside source. Examples of important parameters include type and quantity of combustibles, location and estimated duration of burn. (Example: None.)

3.2.1.5. Hazardous material release definition matrix

This matrix identifies important parameters for hazardous material release conditions at the nuclear power plant. An example is provided in Table 13.

The columns of the matrix are defined as follows:

(a) Case No.: Hazardous material release number. Values range from 1 to the total number of hazardous material release conditions. (Example: No hazardous material release was assumed.)
(b) Material description: Brief description of the hazardous material.
(c) Quantity: Quantity of the material released and over what time frame.
(d) Smothering effect (personnel): Physical effects on personnel (e.g. plant operating staff) need to be itemized. Indicate whether personnel protective gear is required and the time frame for implementation.
(e) Smothering effect (components): Smothering or choking of components as a possible effect is to be identified. For example, if emergency diesel generators could be adversely affected by the atmospheric dispersion of a particular chemical, it needs to be identified here.
(f) Lethal or disabling effects (personnel): Potential effect on plant personnel.
(g) Duration: Time frame in which hazardous material is present. Occurrence of dispersion.
(h) Penetration extent: Hazardous material migrates into buildings through flow paths, including heating, ventilation and air-conditioning systems, or remains in the plant yard.

TABLE 13. HAZARDOUS MATERIAL DEFINITION MATRIX

		HAZARDOUS MATERIAL LOADING CONDITIONS						
Case No.	Material description	Quantity	Smothering effect (personnel)	Smothering effect (components)	Lethal or disabling effect (personnel)	Duration	Penetration extent	Other
1								
2								
N								

3.2.1.6. Flooding

Generally, flooding scenarios can be initiated from inside the plant boundary or outside the plant boundary. Inside the plant boundary, there may be a combination of causes of flood scenarios. Generally, flood would be a secondary effect to another event, for example aircraft crashes or explosions could damage multiple yard tanks simultaneously. Well designed nuclear power plants will have considered common cause events that could cause yard tanks to fail simultaneously (e.g. earthquakes). However, the design conditions may have introduced capacity that would be effective when considering extreme DEEs. For example, well anchored flat bottom tanks may easily resist the effects of explosions, but not aircraft or rigid missile impact. Consequently, the effect on safety systems may have already been evaluated. However, designers of new plants and evaluators of existing plants need to revisit the issue to verify that it has been treated properly for the extreme human induced external event.

Outside the plant boundary, a nuclear power plant sited on a river or other body of water constrained up stream by one or a series of dams could be vulnerable to externally induced failures. Multiple dam failure could be caused by a variety of phenomena progressively failing dams up stream of the plant. These scenarios are best treated by a combination of Member State measures, such as monitoring fill levels of the dams to be certain overtopping or failure does not occur. The nuclear power plant may participate in the minimization of the consequences by having in place an emergency procedure for shutdown if advance warning of a potential flood is given. To be confident of the effectiveness of such an emergency procedure, it needs to be incorporated into normal operator training.

3.2.1.7. Summary table

This table provides a summary of loading conditions for the evaluation of the facility. The table and its backup information is the pinch point between the human induced external event definition and the evaluation requirements for the engineering safety experts. The table contains the loading identifiers and the load combinations to be considered. An example is provided in Table 14, which has been simplified for illustrative purposes. For all items on the SEL, there is a set of loading conditions and load combinations to be considered.

TABLE 14. EXTREME LOADING MATRIX SUMMARY TABLE

			PHYSICAL LOADING CONDITIONS						
Plant area	Vital area	Description	Impact	Blast	Heat/fire	Hazardous material release	Smothering	Flooding	Other
Building 1									
Building 2									
Building 3									
Zone 1									
Zone 2									
Zone 3									
Zone 4									
Yard 1									
Yard 2									
SEL item 1									
SEL item 2									
N									

4. PLANT SPECIFIC EVALUATION

4.1. DEFENCE IN DEPTH

The basic approach to the overall plant specific evaluations is utilizing the concept of defence in depth [1, 8]. In a full scope safety evaluation against human induced external events, all layers of defence in depth are assessed. The layers of defence in depth may be intrinsic or extrinsic, on-site or off-site, and related to safety, security or a combination. Some layers of defence in depth are related to prevention (prevention of the human induced event from adversely impacting the nuclear power plant) and others to mitigation when, for example, 'core damage' is considered the consequence of the human induced event.

Five levels of defence in depth are defined as follows [8]:

— Level 1: Prevention of abnormal operation and failure, and prevention of the consequences of human induced event scenarios from reaching the nuclear power plant;
— Level 2: Control of abnormal operation and detection of failures, and control of damage caused by human induced events through design principles, including design robustness, redundancy, physical separation and diversity;
— Level 3: Control of accidents within the design basis;
— Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of consequences of severe accidents;
— Level 5: Mitigation of the radiological consequences of significant external releases of radioactive materials.

Owing to the interdependencies of their roles and responsibilities, the Member State and the operator share the responsibility of the defence in depth as a whole:

(a) At level 1, human induced external events are initiated outside the plant boundary. Consequently, there is a shared responsibility between the Member State organizations to prevent human induced events from occurring.
(b) At levels 2–4, if human induced events are initiated, the majority of the initial burden for level 2–4 activities is the responsibility of the operator. However, there is an understanding of approach between Member State organizations and the responsible plant personnel.

(c) At level 5, emergency response activities clearly involve the Member State, not only the operator. Hence, the related activities are also interdependent.

A coordination plan is the key element to ensure coordination of off-site and on-site activities. A coordination plan is to be established for the integration of on-site and off-site action plans. The coordination plan needs to define roles and responsibilities, communication (including command and control, alarms, responsible government organizations, media, public) and emergency response coordination with existing procedures.

4.2. SUCCESS PATH AND FAILURE PATH

Two modelling and evaluation approaches are tools in the plant specific evaluation process: probabilistic safety assessment (PSA) and safety margin assessment (SMA). These methods are discussed in detail in Ref. [3].

The SMA approach relies on defining success paths. A success path is a set of systems and associated components that can be used to bring the plant to a stable hot or cold shutdown condition and to maintain this condition for a specified period of time. A complementary definition is that a success path is defined by SSCs whose successful performance will put the nuclear power plant in a safe state (i.e. hot or cold shutdown).

Once the front line and support systems are identified, the success paths consisting of combinations of safety systems, equipment and structures will be developed. These required SSCs will be listed on an SEL augmented by required structures. All screening tools, including those identified in Section 3 will be implemented.

In its entirety, the PSA approach models the process from initiating event to end metrics of interest, including mitigation systems, containment SSCs (level 2), and on-site and off-site consequences (level 3). This constitutes defining the failure paths and their quantification in probabilistic terms. Similar to the success paths, the potential failure paths comprise SSCs that define the SEL for further evaluation. When SEL items cannot be screened out based on conservative performance criteria, a detailed computation of capacities is necessary (see Section 4.5).

4.3. SELECTED EQUIPMENT LIST

SSCs that require evaluation for the SMA or the PSA are identified depending on the methodology to be implemented. Along with identification of the SSCs, their required performance needs to be identified.

For the SMA approach, multiple equipment lists could be defined as a function of the event under evaluation. Different sets of lists will be required for different human induced events based on the location and extent of effects of the event. The important element here is to systematically assess the human induced events and the accompanying equipment lists. It is expected that the number of items on the equipment lists will be in the hundreds.

4.4. AREA DEPENDENT EVENT EVALUATION

There are many events for which area dependent evaluations are performed for design basis external events (DBEEs) and for design extension external events (DEEs). Three examples are security events (design basis threats and beyond design basis threats), internal and external fires, and internal and external floods.

In the case of physical protection systems, security evaluations are based on identifying vital areas and protecting those vital areas. A vital area is an area within the protected area containing equipment, systems, devices or nuclear material, the damage of which could directly or indirectly lead to unacceptable radiological consequences. Depending on the safety philosophy of the nuclear power plant (and the Member State), the set of vital areas could include all designated safety systems or a subset of safety systems and equipment. The number of vital areas and their extent depend on the physical protection philosophy of the Member State. In some Member States, all safety related items are to be protected. This translates into a small number of vital areas, but with very large areal extent (e.g. an entire building might be defined as one vital area). Alternatively, a minimum set of equipment may be a subset. This latter philosophy would be parallel to the SMA approach to human induced external events.

In the case of fire and flood, location dependent evaluations are also implemented and can be used to assess the survivability of a minimum subset of SSCs. Location dependency is especially true when one assumes items in a compartment fail when fire or flood engulf the compartment. In that case, survivability is 100% dependent on location.

It is important to note that existing models and results can be used in the development of the elements of the evaluation methodology. These models could be probabilistic or deterministic (i.e. event or fault tree or success path based).

4.5. PERFORMANCE OF STRUCTURES, SYSTEMS AND COMPONENTS

4.5.1. Civil engineering structures

Civil engineering structures subjected to various levels of DBEEs and DEEs are evaluated by techniques described in Ref. [2] (see Table 8, in Section 2.2.4, for acceptance criteria for structures at the three levels).

4.5.2. Mechanical and electrical equipment

Important considerations regarding the evaluation of performance of mechanical and electrical equipment in extreme loading environment are:

(a) To define the functions to be performed by the item of interest, in what time frame and under what environmental conditions;
(b) To define the extreme loading environment to be imposed on the item of interest (including amplitude and duration), for example direct heat/flame, compartment temperature as a function of time, and mitigating factors that are evaluated to be effective (e.g. the fire suppression system);
(c) To determine the support systems required for the item of interest to operate as required (e.g. emergency power and room cooling);
(d) To determine the likelihood of the item of interest to fulfil the required function by using engineering evaluations (e.g. test data, analyses and computer simulations);
(e) To transmit this information to the systems modelling discipline [3].

4.5.3. Piping, cabling, instrumentation and control, and service lines

Distribution systems are the life blood of the front line and supporting systems. Distribution systems are best evaluated by a combination of analytical tools and in-plant evaluations. The most significant vulnerabilities to piping, cabling, instrumentation and control, and heating, ventilation and air-conditioning systems are due to direct effects, such as explosions, mechanical impacts (aircraft crashes), fire and flood. In addition, indirect failures, such as structure elements failing, falling or otherwise damaging the distribution system, are important.

It is extremely important to identify the required functions to be performed by the distribution system of interest when subjected to an event and the subsequent hazard and effect. This is especially true for instrumentation and control systems. Generally, instrumentation and control systems are more vulnerable than other distribution systems, which are typically rugged when

subjected to extreme loading conditions. Consequently, required instrumentation and control systems need to be carefully considered in all areas of the evaluation.

It is difficult to evaluate these potential failure modes only from drawings without an in-plant evaluation. For existing nuclear power plants, in-plant evaluations are to be performed. For new nuclear power plants, in-plant evaluations can be performed at system turnover.

4.6. PLANT PERFORMANCE EVALUATION

The three fundamental safety functions to be maintained for human induced external events imposed on the nuclear reactor are control of reactivity, fuel cooling and confinement of radioactive material. To achieve these safety functions, front line and support systems are required to perform their functions. As part of the front line and support systems, essential monitoring and control capabilities may be required.

In the light of the Fukushima Daiichi accident, some Member States require additional conditions to be met for the integrity of the spent fuel pool and for spent fuel pool cooling. One Member State requires the following plant states to be demonstrated to be achievable through design or verification for loading conditions of extreme human induced external events:

— Safe shutdown (DBEE, DEE 1 and DEE 2);
— Containment function (DBEE and DEE 1);
— Spent fuel pool integrity (DBEE and DEE 1);
— Spent fuel pool cooling (DBEE, DEE 1 and DEE 2);
— Ultimate heat sink (DBEE, DEE 1 and DEE 2).

In some Member States, successful performance is defined as [14]:

— The reactor core remains cooled, or the containment remains intact.
— Spent fuel cooling or spent fuel pool integrity is maintained.

Reference [3] examines these criteria and alternatives in detail. The following sections explore the main aspects to be considered in the assessment of performance of the safety functions.

4.6.1. Safe shutdown

Safe shutdown refers specifically to the nuclear reactor system and includes hot or cold shutdown, prevention of recriticality and decay heat removal.

Verification of the ability of the nuclear power plant SSCs to shut down the plant and to maintain it in a hot or cold shutdown state is the objective of the safety assessment. Verification methods and acceptance criteria may be tiered depending on the severity of the human induced external event.

Three operating states need to be considered: full power, low power and outage. The most vulnerable state of operation is not known a priori, consequently all states require evaluation or, as a minimum, consideration.

In addition to the evaluation of the reactor containment building, all structures containing equipment necessary to prevent damage to fuel in the reactor or the spent fuel pool will be identified for screening or evaluation. Front line and support systems needed for safe (cold) shutdown of the reactor or continued cooling of the spent fuel pool are identified.

4.6.2. Containment function

The containment function relies on maintaining the containment structure's integrity when subjected to the human induced events. This is essential if the event has induced core damage (Level 4 layer of defence in depth).

The first level of containment integrity is to ensure that direct damage to the containment structure induced by the event is highly unlikely to occur. Assurance is gained through evaluations of the structure capacity of the containment when subjected to mechanical and other loading conditions owing to the event. The containment structure is considered to maintain its structural integrity when it is satisfactorily evaluated for the event under consideration by the methodologies of Ref. [2] and the accompanying acceptance criteria.

In addition, the containment's ultimate pressure capability, given a core damage event, should not be exceeded before effective mitigation strategies can be implemented. Effective mitigation strategies are those that, for a period of time, provide sufficient cooling to the damaged core or containment to limit temperature and pressure challenges below the ultimate pressure capability of the containment. Generally, this means that front line and support systems needed to ensure that these preventative actions are successfully achieved will be available when under demand after the initial actions of the human induced external event.

The next level of containment evaluation is to ensure that, given a level of fuel damage, the following occurs:

— Prevention of hydrogen deflagration or hydrogen detonation (inerting, recombiners or igniters), taking into account venting processes;
— Prevention of overpressurization of the containment;
— Prevention of recriticality;
— Prevention of basemat melt-through.

In the PSA language, one can envision a high level OR gate: (reactivity control and core cooling are maintained) OR (containment integrity is uncompromised due to direct effects of the human induced external event AND front line and support systems needed to maintain containment integrity operate successfully).

4.6.3. Spent fuel pool integrity and cooling

As in the evaluation of the safe shutdown and the containment, two levels of evaluation are required for the spent fuel: direct damage to the spent fuel pool due to the event and indirect damage, as described in the following.

Direct damage is a direct consequence of the scenario, for example aircraft impact on the fuel storage building, wall failure or a projectile penetrates the outer wall and ruptures the spent fuel pool wall, thereby releasing water and leading to the uncovering of the spent fuel elements.

Indirect damage is an indirect consequence of the scenario, for example the spent fuel pool stays intact but the spent fuel pool cooling relies on a series of front line and support systems (some identical to those required for the reactor cooling) and these systems fail. The cooling function cannot be achieved, the water boils off, the spent fuel rods eventually become uncovered and radioactive material is released.

The evaluation process remains the same as for ensuring reactivity control, cooling of the core, and reactivity confinement. The engineering safety evaluation methodologies of Refs [2, 3] apply directly.

4.6.4. Ultimate heat sink availability

The ultimate heat sink (UHS) is a medium to which the residual heat from the reactor is transferred. In some cases, the nuclear power plant has a primary UHS, such as the sea or a river, and a secondary UHS, such as another water source or the atmosphere. In the engineering safety evaluation, it is important to recognize multiple UHSs and their characteristics (i.e. location, form, reliance on front line and support systems). The redundancy offered by a secondary UHS is important, since in many cases the primary UHS and its systems may not be well protected from an extreme human induced external event. Emergency measures, such as FLEX, may provide the means for an alternative UHS.

4.7. ASSESSMENT OF EXTERNAL PLANT CONDITIONS

External plant conditions are conditions which exist in the surrounding areas of the nuclear power plant that may be a help in, or a hindrance to, preventing the consequences of the human induced external event. Typical conditions that would be helpful in preventing a human induced external event from having a significant effect on the nuclear power plant and the surrounding area are administrative controls on potential sources of these events implemented by Member States:

— Buffer zones in the air, land and water, for example maintaining hazardous material boundaries outside the plant boundary and so preventing land or water vehicles from entering areas where explosions or hazardous material releases could affect the nuclear power plant;
— Maintaining no-fly zones around the nuclear power plant;
— Maintaining a buffer zone of no combustibles on the land around the plant boundary.

A typical condition that would be a hindrance to preventing a human induced external event from having a significant effect on the nuclear power plant and the surrounding area is high population density in the vicinity of the nuclear power plant.

4.8. ACCEPTANCE CRITERIA

As introduced in Section 2.2.4, Phase 4 of the overall methodology deals with acceptance criteria, which may be in the form of end metrics:

(a) Risk oriented, for example core damage frequency, large early release frequency, total effective dose equivalent to personnel and total effective dose equivalent to the public.
(b) Capacity oriented, for example conservative containment capacity when subjected to specified aircraft impact loads, best estimate capacity (only slightly conservatively biased). Table 8, in Section 2.2.4, presents an example of these acceptance criteria for SSCs.

Acceptance criteria may expand on to the SSC level:

— Systems requirements: The number of safety trains to be protected or demonstrated to be available, including capacities of SEL items in the safety trains. Section 2.3 provides commentary on this subject.
— SSC capacity values: Treating the loading environment as best estimate and the resistance of SSC in the same manner, or conservatively defining the loading environment and conservatively defining the systems acceptance criteria, for example demonstrate one or two trains of safety systems to be verified to perform (redundancy).

IAEA publications introduce the tiered approach to defining safety assessment methodologies and acceptance criteria [2, 3]. For example, the consequences of less severe events (i.e. more frequent events) on the nuclear power plant of interest need to meet more stringent acceptance criteria than those of the most severe events (i.e. rare or less frequent events). For less severe events, requirements include redundant success paths to arrive at safe shutdown and conservatism in defining both the environmental loading functions and the performance criteria of SSCs — essentially elastic material behaviour for structures, components and equipment, and distribution systems. For the most severe hazards, verifying a single success path to safe shutdown using realistic analyses may be acceptable. Table 8, in Section 2.2.4, suggests one approach where systems requirements tier down depending on the event severity level. References [2, 3] follow this approach when discussing system and structure capacity acceptance criteria.

4.9. EVALUATION PROCEDURE

In summary, the general procedure for plant specific evaluation consists of:

(a) Event and load characterization: This is the input to the plant specific evaluation, originating from Phases 1 and 2 (see Section 3).
(b) Systems analysis: Depending on the selected approach (SMA or PSA), success paths or failure paths are identified for each event given as input (see Section 4.2).
(c) SEL: Using the results of the systems analysis and the area review, a list of SSCs required to perform the selected safety functions under the plant conditions generated by the considered events is compiled for capacity assessment (see Section 4.3).

(d) Area dependent event evaluation: Areas of influence corresponding to the events are assessed in order to identify the portions of the plant at which SSCs will likely not be available to perform their intended safety functions (see Section 4.4).
(e) Assessment of SSCs: Performance of selected SSCs for the loading conditions given as input is assessed (see Section 4.5). As a result of the assessment, SSCs not able to perform their intended safety functions are identified.
(f) Assessment of plant performance: Using the results of the systems analysis and the assessment of SSCs carried out in the previous step, the overall performance of the plant to keep the fundamental safety functions under the external events is assessed (see Section 4.6).
(g) Acceptability of plant performance: Plant performance is assessed against the acceptance criteria in the Member State (see Section 4.8). This is already Phase 4 of the overall evaluation (see Section 2.2).

A very important element to emphasize is that the evaluation of safety against human induced external events is very much a multidisciplinary activity. It involves specialists with expertise in safety engineering, operations, engineering (e.g. civil, structural, mechanical, electrical, instrumentation and control, and geotechnical) and emergency response. The interaction of these disciplines is essential to obtaining a holistic approach to dealing with design extension human induced external events.

5. DESIGN OF NEW PLANTS

5.1. DESIGN PROCESS

For new nuclear power plants, it is essential to consider human induced external events during the design process, taking into account lessons learned from evaluations of existing nuclear power plant SSCs. Future nuclear power plants are being designed for operating lives up to 60 years. It is prudent to verify that new designs have the capacity to resist various human induced external events and, if necessary, provide additional design features to ensure that the capacity exists.

The general design process is shown in Fig. 3 and includes important elements such as the following:

(a) Safety goals in terms of metrics such as core damage frequency, large release frequency and dose limits to the public are applicable in numerous Member States. Member States set the specific target values for these safety goals through law. In general, they are applicable to accidental human induced external events. For non-accidental human induced events, some Member States have specific acceptance criteria.
(b) Three fundamental safety functions are defined as: (i) the control of reactivity; (ii) the removal of the heat from the core; and (iii) the containment of radioactive material. In some Member States, these three safety functions are further broken down, in particular item (ii).
(c) Member States define and specify deterministic and probabilistic success criteria. Both deterministic and probabilistic success criteria are utilized in the implementation of the defence in depth principles as shown.
(d) The defence in depth concept belongs to the nuclear safety fundamentals. The layers of defence in depth may be intrinsic or extrinsic, on-site or off-site. Some layers of defence in depth are related to prevention and others to mitigation.

The general aspects of siting and design make a very important contribution to the protection of nuclear power plants against human induced external events. For new nuclear power plants, generally two stages of design development exist: standard or reference design (e.g. in the United States of America, Certified Designs are licensed); and site specific issues once a site is selected.

DEEs may not be available for a new design. It is then necessary to perform the analysis using a generic DEE (site and state independent). It is also possible to use events that are beyond a design level in order to attain graded protection. This will serve the purpose of designing the facility to some level of human induced external event. After the site is selected and the DEEs are known, it is necessary to complement the design with extrinsic measures so that all the human induced events scenarios that can be generated are addressed.

5.2. HUMAN INDUCED EVENT: AGNOSTIC DESIGN

The details of human induced events to be considered in current designs are specified and available. However, considering the expected operating life of new nuclear power plants to be 60 years, it is prudent to think beyond the current environment of possible events. Larger and larger aircraft are being

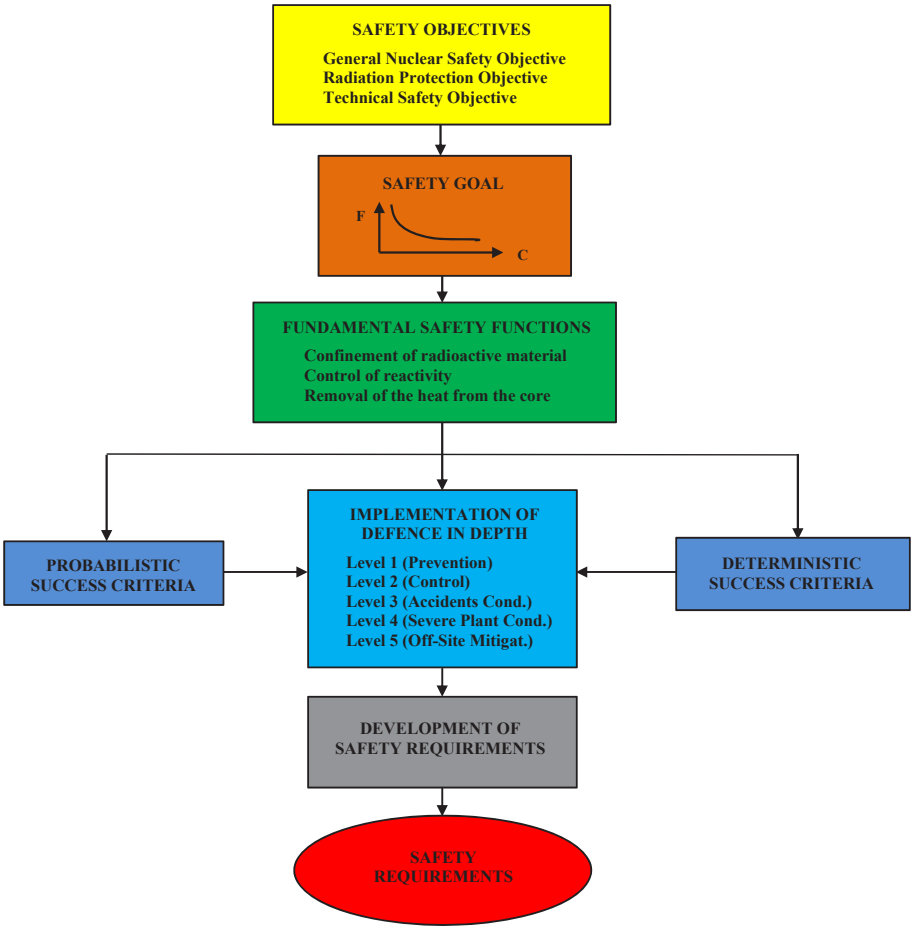

FIG. 3. Flow chart for design requirements.

designed, built and put into operation. Transportation routes, pipelines and other infrastructure are anticipated to change during the life of the nuclear power plant. Relying strictly on administrative control to prevent such hazards from impacting new nuclear power plants may be unreasonable. A philosophy of design and evaluation of defining a set of human induced event agnostic scenarios and treating those as generally enveloping anticipated future defined human induced events is prudent.

One such enveloping case is aircraft crashes, which includes impact loading conditions of global response, local response, and vibration effects and fire effects. Hence, larger aircraft than the existing fleet could be specified by the Member State for one DEE. These specific phenomena are treated in

Ref. [2]. In addition, there may be other human induced events that are defined (or need to be defined) that encompass currently identified human induced events or hypothesized future events. It is recognized that some Member States are implementing this process or philosophy. Implementing this process will provide decision makers information on the capacity of newly defined environmental loading conditions in the decades to come while existing and new nuclear power plants are in operation.

An additional consideration that falls into the category of human induced event agnostic design is the evaluation of the nuclear power plant for extreme external events without consideration for how the following states could arise:

— Station blackout (i.e. the loss of on-site normal power, off-site power or emergency power);
— Loss of UHS (e.g. the loss of primary UHS);
— Simultaneous station blackout and loss of UHS.

5.3. REDUNDANCY, DIVERSITY AND SEPARATION

Redundancy, diversity and separation have been introduced into the design concept for most new nuclear power plant designs. New nuclear power plant designs are required to have redundant systems to provide front line or supporting functions, and the reactor vendors take the opportunity of a new design to introduce diversity and separation into system design. Diversity in systems and components is introduced for a number of reasons, one of which is to avoid correlation of failure of identical systems and components located in near identical locations. Correlation of failure of redundant systems due to their identical nature may defeat one of the important attributes of redundancy; that is, no safety improvement gained for particular initiating events, for example common cause events, such as earthquakes. Identical components residing close to each other so that they receive the same seismic input are identically aligned, with the same supports and anchorage that will likely have correlated failure modes. If there were four emergency diesel generators, one required for each train, and one train required for system success, correlated failures would assume that if one generator fails, they all fail.

The concept of separation is also very important for design implementation, especially for dealing with human induced events that have a large area of influence. An example of the consequences of lack of separation is four pumps providing water to the component cooling water system. All four pumps are located in the intake structure. In a fire initiated following an aircraft crash and fuel fire: if one pump fails, they all fail.

5.4. LAYOUT

Layout is particularly effective in minimizing the consequences of human induced external events. Sections 2 and 3 discuss zones of influence and damage footprints, which take into account the advantages of layout in preventing a single human induced event from disabling multiple SSCs.

5.5. DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

Perhaps the most important issue for new nuclear power plant designs is the implementation of shielding structures that shield the reactor containment building (and in some cases other structures important to safety) from damage due to extreme external events, such as an aircraft crash. The majority of these shielding structures are designed to preclude phenomena associated with aircraft crashes (i.e. overall and local structure failure), which then precludes the possibility of a jet fuel fire being initiated inside the safety related structures. Often, a gap exists between the shielding structure and the safety related structure to provide additional defence in depth for impact, vibration and fire loading conditions.

Although these shielding structures are designed specifically for aircraft crashes, they are also effective against other loading conditions, such as large explosions.

5.6. DESIGN CAPACITY OF PLANT

Throughout the world, DEEs of human induced origin, such as the impact of a large commercial aircraft on a nuclear power plant, are being taken into account in the design process. In the light of the Fukushima Daiichi accident, human induced event agnostic design scenarios are also being evaluated for all new designs. Thus, there is high confidence that many extreme human induced external event scenarios are being taken into account in the design of new nuclear power plants.

6. SAFETY EVALUATION OF EXISTING PLANTS

The methodologies for existing and for new nuclear power plants are basically the same. The differences are that new plants have a focus on explicitly considering some human induced events in the design process — events that may not have been recognized at the time of the design of existing plants. In other words, DEEs are considered in the design process for new plants, whereas beyond design external events were not taken into account in the design process of most existing plants (see Fig. 1, in Section 1.5).

There are distinct differences between the evaluation of existing plants for beyond design external events and the design of new plants when subjected to DEEs. The obvious difference is that the physical existence of the existing plant limits the physical modifications that can be implemented easily or at all. Hence, management of human induced event scenarios for existing plants will rely on the robustness of the existing designs, relatively small physical modifications (if deemed necessary and cost effective), operational or procedural changes, and emergency management (e.g. FLEX and EMEG). Detailed evaluation procedures are contained in Refs [2, 3].

7. MANAGEMENT OF THE ASSESSMENT

Member State regulatory authorities need to provide operators at the outset with the criteria to be used in the assessments. At the same time, they need to agree to a timescale for the conduct of the assessment and for reporting back the conclusions and recommended actions. The regulatory authorities may undertake their own prioritization of the programme for the total number of nuclear power plants and operators under their responsibility. That prioritization may be influenced by factors such as national information, source terms and release fraction estimates, and potential effects on the public (dispersion and concentration of radioactive materials). Similarly, the effects of reported incidents, in particular major events (e.g. the chemical plant explosion in Toulouse, France, the events of the 11 September 2001 and the Fukushima Daiichi accident), can influence scheduling and prioritization.

Peer review is highly desirable. As discussed throughout this publication and in Refs [2, 3], assessment of the effects of human induced external events is a multidisciplinary activity with highly specialized elements, perhaps requiring experts in the field to perform the work. Consequently, peer review by other experts in these areas is required. Records retention needs to be robust to enable future retrieval when new human induced events may be postulated.

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Annex

DEVELOPMENT OF THE EXTREME LOADING DEFINITION MATRICES: THREE EXAMPLES

A–1. INTRODUCTION

In this annex, representative load cases are defined following the format of Section 3. The objective is to acquaint the analyst on the type of information to be provided, recognizing that it may need to be expanded when the actual analysis is performed. Three examples pertaining to extreme human induced external events are presented: aircraft crash, hazardous chemical release and blast.

Section 3 introduced the load definition matrices as examples of the types of input and response expected to be produced by the engineering staff. These extreme loading definition matrices can be thought of in three categories:

(a) Level 1: Extreme Environment[1] Matrix provides the correlation between the human induced external event and the loading conditions.
(b) Level 2: Parameter Definition Matrices:
    — Impact Parameter Definition Matrix;
    — Explosion/Blast Parameter Definition Matrix;
    — Heat/Fire Parameter Definition Matrix;
    — Hazardous Material Release Definition Matrix.
(c) Level 3: Extreme Loading Matrix.

Examples are presented here to further clarify these concepts. Reference [A–2] presents very detailed assessments. The examples presented here are only for familiarization with the issues. The example Extreme Environment Matrix for a nuclear power plant subjected to three human induced external event scenarios is given in Table A–1.

[1] For consistency with other IAEA publications (see Ref. [A–1]), the term ‘environment’ is used in this publication in reference to the set of concurrent loads associated with a single event. Hence, reference is made to ‘extreme environments’ such as ‘aircraft crash environment’ or ‘pipeline blast environment’.

TABLE A–1. EXTREME ENVIRONMENT MATRIX

		PHYSICAL LOADING CONDITIONS						
Scenario No.	Scenario description	Impact	Blast	Heat/fire	Hazardous materials release	Smothering	Flooding	Other
1	Crash of a large passenger aircraft with a fully fuelled fuselage into a nuclear power plant	1, 2, 3 (Table A–2)	No	1, 2 (Table A–3)	No	No	No	No
2	Chlorine release	No	No	No	1 (Table A–6)	No	No	No
3	Blast pipeline	No	1 (Table A–10)	3, 4 (Table A–8)	2, 3 (Table A–9)	No	No	Debris

Note: Numbers under the physical loading conditions columns are explained in the specific table in parentheses.

A–2. PARAMETER DEFINITION AND LOADING CONDITIONS FOR AIRCRAFT CRASH (SCENARIO 1)

Human induced external event scenario 1 is defined as the accidental crash of a large passenger aircraft into a nuclear power plant site. The loading conditions are: aircraft impact on nuclear power plant structures, jet fuel fire in buildings if perforation occurs, and aircraft crash induced vibration on all buildings potentially impacted by the aircraft.

This is intended to be a realistic example, but, of course, the specifics for a site depend greatly on the local conditions. In this hypothesized case, a large passenger aircraft is considered and the site is assumed to be such that the impact velocity of 110 m/s is realistic, given the topography of the site and other considerations. Tables A–2 and A–3 provide the impact parameters and the heat/fire parameters, respectively, corresponding to this scenario.

Vibration loading conditions are described in Table A–4. The analysis to be performed is assumed to account for the flexible and non-linear behaviour of the buildings of interest. It should be noted that this vibration loading condition was only associated with the fuselage impact. A separate vibration loading condition could have been specified for the engine or debris loads, if deemed appropriate.

Finally, Table A–5 itemizes the loading conditions to be considered for the buildings on the nuclear power plant site being evaluated. Load combinations are as shown unless it can be demonstrated that it is unrealistic to consider the indicated loads simultaneously.

A–3. PARAMETER DEFINITION AND LOADING CONDITIONS FOR CHLORINE RELEASE (SCENARIO 2)

Human induced external event scenario 2 is that of the accidental release of 10 tonnes of chlorine gas from a truck. The truck is able to reach within 200 m of the plant boundary and release the entire inventory of 10 tonnes. Detailed dispersion modelling could be used for this and other more severe cases of chlorine release. For example, a rail car of 90 tonnes of chlorine could be accidentally derailed, releasing the inventory of one or more rail cars transporting chlorine. Modelling of the dispersion is a key component of the assessment. Table A–6 gives general information on the release. Table A–7 lists structures to be evaluated. Of course, this information needs to be supplemented with the details.

TABLE A–2. IMPACT PARAMETER DEFINITION MATRIX: SCENARIO 1

		MISSILE IMPACT					ANCILLARY EFFECTS			
Missile type/No.	Description	Mass (kg)	Shape/ configuration	Impact angle	Impact velocity (m/s)	Relative hardness	Fire	Explosion	Vibration	Other
1	Crash of a large passenger aircraft with a fully fuelled fuselage into a nuclear power plant	396 900	Fuselage 40 m^2/52 m^2	3–10° to horizontal	110	Flexible	No	No	1 (Table A–4)	No
2	Large passenger aircraft engines as projectiles	4 300	Circular body 2.7 m fan diameter	3–10° to horizontal	110	Semi-rigid	No	No	No	No
3	Debris	5 000	Rigid body	3–10° to horizontal	110	Rigid	No	No	No	No

TABLE A–3. HEAT/FIRE PARAMETER DEFINITION MATRIX: SCENARIO 1

		FIRE SOURCE OUTSIDE FACILITY						COMBUSTIBLES INSIDE FACILITY				
Fire No.	Description	Combustible/ ignition	Quantity[a] (L)	Spreading surface	Heat potential/ temp. (°C)	Burn duration (h)	Other	Building/ yard	Quantity (L)	Type	Ignition likelihood	Burn duration (h)
1	Jet fuel fire from a large passenger aircraft	Yes	216 000	Half surface of outer shell	1 200	<1	No	Inside outer shell	1 500	Kerosene	Yes	1.5
2	Horizontal crash of a large passenger aircraft, with a fully fuelled fuselage, on the yard of a nuclear power plant	Yes	216 000	60 m × 500 m	1 200	1–8	No	No	No	Kerosene	No	No

[a] 50% is considered to be consumed in a fire ball, 50% as pool fire.

TABLE A–4. VIBRATION PARAMETER DEFINITION SCENARIO 1

	LOADING FUNCTIONS				
Vibration loading condition No.	Load application (buildings)	Location: elevation	Loading function time histories		
			Rigid target	Flexible target	Flexible/non-linear target
1	Reactor building	All	—	—	Yes
2	Emergency cooling water building	All	—	—	Yes
3	Diesel generator building	All	—	Yes	—

TABLE A–5. EXTREME LOADING MATRIX: SCENARIO 1

		PHYSICAL LOADING CONDITIONS						
Plant area	Engineering load description	Impact	Blast	Heat/fire	Hazardous materials release	Smothering	Flooding	Other
Reactor building	Aircraft crash	1, 2, 3 (Table A–2)	No	1 (Table A–3)	1 (Table A–3)	Yes	Yes	Vibration
Emergency cooling water building	Aircraft crash	1, 2, 3 (Table A–2)	No	1 (Table A–3)	1 (Table A–3)	Yes	Yes	Vibration
Pump house	Aircraft crash	1, 2, 3 (Table A–2)	No	1 (Table A–3)	1 (Table A–3)	No	Yes	Vibration
Diesel generator building	Aircraft crash	1, 2, 3 (Table A–2)	No	1 (Table A–3)	1 (Table A–3)	Yes	Yes	Vibration
Cooling water pipelines	Aircraft crash	No	No	2 (Table A–3)	2 (Table A–3)	No	No	No

TABLE A–6. HAZARDOUS MATERIAL RELEASE DEFINITION MATRIX: SCENARIO 2

		HAZARDOUS MATERIAL LOADING CONDITIONS						
Case No.	Material description	Quantity (t)	Smothering effect (personnel)	Smothering effect (components)	Lethal or disabling effect (personnel)	Duration	Penetration extent	Other
1	Chlorine	10	Yes	Yes	High	Multiple hours	High through HVAC systems	No

Note: HVAC — heating, ventilation and air-conditioning.

TABLE A–7. EXTREME LOADING MATRIX: SCENARIO 2

		PHYSICAL LOADING CONDITIONS						
Plant area	Engineering environmental load description	Impact	Blast	Heat/fire	Hazardous materials release	Smothering	Flooding	Other
Reactor building	Hazardous material	—	—	—	1 (Table A–6)	—	—	—
Emergency cooling water building	Hazardous material	—	—	—	1 (Table A–6)	—	—	—
Pump house	Hazardous material	—	—	—	1 (Table A–6)	—	—	—
Diesel generator building	Hazardous material	—	—	—	1 (Table A–6)	—	—	—
Cooling water pipeline	Hazardous material	—	—	—	1 (Table A–6)	—	—	—

A–4. PARAMETER DEFINITION AND LOADING CONDITIONS FOR GAS PIPELINE EXPLOSION (SCENARIO 3)

Human induced external event scenario 3 is defined in Tables A–8 to A–11. The event is characterized by an accidental gas pipeline explosion. The gas pipeline is located a distance of 280 m from the closest safety related structure of the nuclear power plant. The gas in the pipeline is methane (CH_4) and liquefied under pressure. The total amount of gas potentially ejected before isolation of the pipeline up stream of the break location is 19 900 kg. The volume of the gas cloud (air–gas mixture volume) that could lead to the explosion is estimated at 338 000 m^3. The assumed relative flame front propagation α is 0.3. Characteristics of the blast are calculated based on the multi-energy method described in Ref. [A–2].

REFERENCES TO THE ANNEX

[A–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Engineering Safety Aspects of the Protection of Nuclear Power Plants against Sabotage, IAEA Nuclear Security Series No. 4, IAEA, Vienna (2007).

[A–2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Aspects of Nuclear Power Plants in Human Induced External Events: Assessment of Structures, Safety Reports Series No. 87, IAEA, Vienna (in preparation).

TABLE A–8. HEAT/FIRE PARAMETER DEFINITION MATRIX: SCENARIO 3

		FIRE SOURCE OUTSIDE FACILITY					COMBUSTIBLES INSIDE FACILITY				
Fire No.	Description	Combustible/ ignition	Quantity	Heat potential/temp.	Burn duration (min)	Other	Building/ yard	Quantity	Type	Ignition likelihood	Burn duration
3	Fire ball	CH_4 and other hydrocarbons	80–100% of ejected gas before break isolation	800–1000°C inside fire ball	<5	—	—	—	—	—	—
4	Laminar burning after suction into building via ventilation system	CH_4 and other hydrocarbons	≤25% of ejected gas volume	800–1000°C inside fire plume	—	—	—	—	—	Transportation and ignition likelihood low	Long due to potential internal fire loads

TABLE A–9. HAZARDOUS MATERIAL RELEASE DEFINITION MATRIX: SCENARIO 3

		HAZARDOUS MATERIAL LOADING CONDITIONS						
Case No.	Material description	Quantity	Smothering effect (personnel)	Smothering effect (components)	Lethal or disabling effect (personnel)	Duration	Penetration extent	Other
2	CO, CO_2	ca. 50% of stechiometric equilibrium	No	No	Yes	Short term depending on weather conditions	Via ventilation system into main control room	No
3	CH_4, air mixture (late ignition case)	ca. 80% of released	No	No	No	No	No	Burnable inside building

TABLE A–10. EXPLOSION/BLAST PARAMETER DEFINITION MATRIX: SCENARIO 3

					PRESSURE PULSE		
Explosion No.	Flammable mask (kg)	Explosion energy (MJ)	Blast strength (multi-energy method)	Reference distance (m)	Peak side on pressure (kPa)	Blast wave shape	Positive phase duration (ms)
1	19 900	1 183 000	7	280	31	B to A	212

Note: The multi-energy method described in Ref. [A–2] is used.

TABLE A–11. EXTREME LOADING MATRIX: SCENARIO 3

		PHYSICAL LOADING CONDITIONS						
Plant area	Engineering environmental load description	Impact	Blast	Heat/fire	Hazardous materials release	Smothering	Flooding	Other
Reactor building	Gas pipeline	—	—	3, 4 (Table A–8)	2, 3 (Table A–9)	—	—	—
Emergency cooling water building	Gas pipeline	—	—	3, 4 (Table A–8)	2, 3 (Table A–9)	—	—	—
Pump house	Gas pipeline	—	—	3, 4 (Table A–8)	2, 3 (Table A–9)	—	—	—
Diesel generator building	Gas pipeline	—	—	3, 4 (Table A–8)	2, 3 (Table A–9)	—	—	—
Cooling water pipelines	Gas pipeline	—	—	3, 4 (Table A–8)	—	—	—	—

ABBREVIATIONS

DBEE	design basis external event
DEE	design extension external event
EMEG	emergency management equipment guidance
FLEX	diverse and flexible coping strategies
PSA	probabilistic safety assessment
SEL	selected equipment list
SMA	safety margin assessment
SSCs	structures, systems and components
UHS	ultimate heat sink

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