
	<p>Advanced Safety Assessment Methodologies: extended PSA</p>	
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"NUCLEAR FISSION "
Safety of Existing Nuclear Installations

Contract 605001

The PSA assessment of Defense in Depth Memorandum and proposals



This report has been proposed by NIER to support the development of the deliverable D30.7 vol 4 (PSA and DiD) of the ASAMPSA_E project. It has not been reviewed by the ASAMPSA_E partners and some issues may need to be discussed further. It has nevertheless been discussed during the final ASAMPSA_E workshop (Vienna, sept. 2017)

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<p>Summary : This report concerns the peculiar roles of the Defence-in-Depth (DiD) concept and the Probabilistic Safety Assessment (PSA) approach for the optimization of the safety performances of the nuclear installation. It proposes a conceptual framework and related process for the assessment of the “safety architecture” implementing DiD, which is articulated in four main steps devoted to (1) the formulation of the safety objectives, (2) the identification of loads and environmental conditions, (3) the representation of the safety architecture and (4) the evaluation of the physical performance and reliability of the levels of DiD. A final additional step achieves the practical assessment of the safety architecture and the corresponding DiD with the support of the PSA. The comprehensive safety assessment of the implemented architecture needs its multi-dimensional representation, i.e. for given initiating event, sequence of possible failures, affected safety function and level of DiD. The risk space (frequency/probability of occurrence, versus consequences) is the framework for the integration between the DiD concept and the PSA approach. Additional qualitative key-notions are introduced in order to address the compliance of the safety architecture with a number of international safety requirements. In this context, the role of the PSA is no longer limited to the verification of the fulfilment of probabilistic targets but includes different contributions to the assessment of the DiD identified in this report.</p>	

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The PSA assessment of Defense in Depth

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Executive Summary

The report discusses the possible role of the PSA in the assessment of the Defence-in-Depth (DiD) which should be the foundation for the definition and the implementation of the plant “safety architecture”.

The objective of the safety architecture representation is to identify, for each plausible plant condition, i.e. for each initiating event and for each sequence generated by any plausible failures, the provisions that embody the different levels of DiD. Consequently, within the framework of this document, the safety architecture of the installation is not seen as an univocal static set of provisions implemented to respond to abnormal situations (control and, as needed, mitigation of accident consequences); in practice, this set is organized, as appropriate, specifically for each initiating event and as a function of the possible sequence of the plausible provisions’ failures, in order to face the challenges raised vis à vis the various fundamental safety functions. As such, the representation of the safety architecture, which is the input for the safety assessment (deterministic or probabilistic), must be able to integrate this variability while guaranteeing, for each configuration, the compliance with the principles of DiD and, in particular, to guarantee the functional redundancy and the independence between the different levels. This is why the notion of “multidimensional” is introduced for the representation of the safety architecture.

As defined, the safety architecture shall meet the applicable safety objectives while guaranteeing the compliance with the (IAEA) Safety Fundamentals [2] and IAEA Safety Requirements [3]. Deterministic and probabilistic approaches are complementary tools to check the meeting of these objectives in the wider context of the safety assessment, as defined by the IAEA GSR Part 4 Rev1 [4].

Consistently with the IAEA Safety Fundamentals [2], the DiD concept, and all principles for its implementation, represents the foundation of the deterministic approach to build the safety architecture; in this context, there is the fundamental need to address the compliance with its principles i.e.: the appropriateness of the approach for the construction of the safety architecture, the adequacy of the implemented “layers of overlapped provisions” (INSAG 10, [11]), and the availability of adequate margins to correctly address the uncertainties. Unquestionably, the Probabilistic Safety Assessment (PSA) could support the demonstration of this compliance, even if - up to now - no formal and one-at-one links are established between DiD concept and PSA approach and no specific requirements are formulated for the assessment of DiD using PSA.

In order to contribute to fill-in this gap, the peculiar role of the Defence-in-Depth concept and the Probabilistic Safety Assessment approach for the optimization of the safety performances of the nuclear installation have been preliminarily investigated [30]: general indications have been provided about a global process for the assessment of the DiD, i.e. for the verification, through PSA, that the implemented safety architecture complies with the principles of the DiD.

The content of this report goes further making explicit the possible relationship between DiD and PSA. The proposed process is fully consistent with the indications provided by the IAEA GSR Part 4 Rev1 [4] and is based on some concepts introduced by the Generation IV Risk and Safety Working Group ([15] and [16]). It is articulated in four main steps devoted to 1) the formulation of the safety objectives, 2) the identification of loads and environmental conditions, 3) the representation of the safety architecture and 4) the evaluation of the physical performance and reliability of the levels of DiD. A final step achieves the practical assessment of the safety architecture and the corresponding DiD with the support of the PSA.

Concerning the safety objectives (cf. Section 2), the reference to 1) the risk space (frequency/probability of occurrence, versus consequences) is considered essential to assess the whole safety architecture with respect to the achievement of probabilistic targets, 2) the performance allocated to the safety functions to reduce the consequences of plausible events and, finally, 3) the reliability which has to be allocated to the provisions which achieve these functions. Additional qualitative key-notions are introduced, providing general indications about the criteria and metrics which should have to be defined in details and adopted. They refer to basic design goals (e.g. need for protective measures limited in times and areas in case of severe accidents) and to DiD principles (e.g. independence of DiD levels, practical elimination of events and sequences leading to early or large releases, demonstration of the availability of “adequate margins” against possible cliff edge effects).

The development of some Safety Fundamentals and Requirements leads to the definition of additional qualitative objectives; they address the search for exhaustiveness for the design basis events and the design extension conditions considered for the safety design and assessment, the need for progressiveness in the system’s response to abnormal events, the need for a forgiving and tolerant character of system safety response, and the suitable balanced contributions of the different events / sequences to the whole risk.

The identification and recognition of all plausible normal and off-normal loads and environmental conditions (cf. Section 3), that can affect the behavior of the installation, is the result of a detailed analysis of the system complemented, as needed, by the consideration of the experience feedback. Since the years 2000 the basis for the design evolved and, today, all the plausible conditions generated by internal and external hazards (Anticipated Operational Occurrences, Design Basis Events and Design Extension Conditions), have to be considered within the Design Basis and, more generically, for the definition of the Safety Case.

Moreover, an explicit one-at-one correspondence is suggested, for example, by WENRA [13] & NUREG 2150 [10] between, on one side, these conditions, the levels of DiD and, on the other side, their positioning within the risk space. This correspondence is essential for the designer who can so superpose the levels of DiD within the area of allowable risk and, simultaneously, gives explicit targets (success criteria, both in terms of performances and reliability) for these levels. It is worth nothing that these targets are essential to classify the Systems, Structures and Components, complementing the process defined by the SSG-30 Safety Guide [6], and to size the provisions associated with each level of the DiD.

The **Objective Provisions Tree (OPT)** methodology and the complementary notion of Line of Protection/Layers of Provisions (LOP), developed within the context of the IAEA activities and endorsed, among others, by the Generation IV International Forum / Risk & Safety Working Group (GIF/RSWG), are proposed for the representation of the safety architecture implemented by the nuclear installation (cf. Section 4); they are fully consistent with the safety assessment process as presented by the IAEA GSR Part 4 Rev1 [4]. If correctly implemented, these tools can support the identification of possible lacks or the weaknesses of DiD level(s), e.g. lack of independence between the DiD levels, inadequacy of the layers of provisions allocated to a given DiD level, etc. The OPT and LOP also provide the essential information for the subsequent development of probabilistic studies, by representing the whole safety architecture that is successively analytically described by the PSA, with all its internal interactions.

The availability of an exhaustive - as practicable - representation of the safety architecture allows the development of a PSA model with a structure that better complies with the DiD principles and that, in turn, allows the evaluation of the physical performance and reliability of the levels of DiD (cf. Section 5). This structure is based on Event Trees built to reflect the crossing of different levels of DiD and on Fault Trees which, at each crossing, allow assessing the reliability of the implemented layers of provisions.

This PSA re-structuring is recommended, but not an unquestionable need (i.e. the whole process for DiD assessment is not invalidated). Theoretically, different PSA models can embed the same information through different event tree-fault tree structures, and provide all the information required for the DiD assessment.

The structure proposed for the PSA/Event Tree integrates some qualitative notions about the practical elimination of both “short” sequences (e.g. in case of non-allowed failure of the first levels of DiD, for instance the rupture of the PWR vessel during normal operation or transients without core melt controlled by the safety systems) and of sequences which lead to unacceptable consequences, i.e. early or large releases (in case of failure of the 4th level of DiD). A partial practical example developed starting from the OPT of the IAEA TECDOC 1366 [8] is also presented.

In summary, the acceptability of a safety architecture shall be based on the degree of meeting the DiD principles while fulfilling the applicable Safety Fundamentals and Requirements. Deterministic and probabilistic considerations shall be integrated into a comprehensive implementation of Defence-in-Depth. The risk space (frequency/probability of occurrence, versus consequences) looks as being the appropriate framework for this integration. In this context, the role of the PSA is no longer limited to the verification of the fulfilment of probabilistic targets but includes essential contributions to the assessment of the DiD:

- PSA can provide additional evidences of the independence among DiD levels and specific insights about plausible dependent failures, also accounting for external (natural or man-made) hazards;
- PSA can support the deterministic design and sizing of provisions, by addressing the effects of their reliability and contributing to the definition of boundary conditions which are acceptable for their correct operation;
- PSA can support the demonstration of the “practical elimination” of plausible events and sequences which could lead to early or large releases;
- PSA can support the demonstration of the gradual degradation of the safety architecture in case of loss of safety functions, before that harmful effects could be caused to people or to the environment (Progressive character of the safety architecture);
- PSA can provide specific insights about the effectiveness of redundancies among implemented provisions, about the modelling of human factor (for immaterial provisions) and about the uncertainties on input data and their propagation through the model (tolerant character of the safety architecture);
- PSA can contribute to the demonstration of the proper priority in the operation of different means required to achieve safe conditions, through inherent characteristics of the plant, passive systems or systems operating continuously in the necessary state, systems that need to be brought into operation, procedures (forgiving character of the safety architecture);
- PSA can provide specific insights addressing the balanced/unbalanced contributions of the different events / sequences to the whole risk identifying the presence (to be avoided) of excessive or significantly uncertain contributors to risk (balanced character of the safety architecture).

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ABBREVIATIONS

AOO	Anticipated Operational Occurrence
BDBEE	Beyond Design Basis External Event
CCDP	Conditional Core Damage Probability
CERP	Conditional Early Release Probability.
CFDP	Conditional Fuel Damage Probability
CLRP	Conditional Large Release Probability
CDF	Core Damage Frequency
DBA	Design Basis Accident
DBC	Design Basis Conditions
DEC	Design Extension Condition
DiD	Defence-in-Depth
DPA	Deterministic and Phenomenological Analyses
DSA	Deterministic Safety Assessment
ERF	Early Release Frequency
ET	Event Tree
FDF	Fuel Damage Frequency
FT	Fault Tree
GIF/RSWG	Generation IV Risk and Safety Working Group
IE	Initiating Event
ISAM	Integrated Safety Assessment Methodology
LOCA	Loss of Coolant Accident
LOD	Line Of Defense
LOP	Line Of Protection
LWR	Light Water Reactor
MCCI	Molten Core Concrete Interaction
NPP	Nuclear Power Plant
OPT	Objective Provision Tree
PIE	Postulated Initiating Event
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Analysis
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
QSR	Qualitative Safety features Review
SA	Safety Architecture
SSC	Systems, Structures, and Components

1. INTRODUCTION

1.1. PREAMBLE

The safety architecture

The safety architecture (SA) of a plant is the set of technical, human and organizational provisions that are set up by the plant designer and the plant operator in order to:

- ensure the achievement of tasks allocated to the process¹ in acceptable conditions of safety, i.e. maintaining significant parameters (e.g. the fuel temperature) within allowable operational limits;
- prevent the degradation of the facility, i.e. exceeding operational limits;
- restore and keep the facility in a safe shutdown condition for the short and long term, in case of failures.

The system's response to each abnormal event is specific, i.e. it is achieved by organizing, manually or automatically, a subset of safety related provisions - material (e.g. SSC) or immaterial (e.g. procedures) - to manage the conditions (i.e. mechanisms challenging the safety function(s)) and to meet the safety objectives.

The safety architecture is not a static univocal representation of the related provisions but rather a multi-dimensional representation of the mode of operation of the installation and of its response to abnormal conditions. Looking beyond the factual identification of all the available provisions, the SA considers the given levels of Defense in Depth (DiD) in which provisions are required to intervene/operate, with specific reference to the initiating event and the safety function for which they are requested. The initiating event, the sequence of possible failures, the affected safety function and the level of DiD in which the provision is asked to achieve its mission (layer of provision) are the dimensions of the SA representation. A provision can be allocated to different "layers of provision" for given initiator, sequence, safety function and level of DiD. As detailed in the following sections, a necessary condition for the acceptability of the positioning of that provision in the architecture is the functional independence between the DiD levels.

Appendix 1 provides more details concerning the definition of the safety architecture.

The assessment of the safety architecture

The safety architecture of a nuclear installation shall meet the safety objectives while complying with the principles defined within the IAEA Safety Fundamentals (SF1, [2]) and the IAEA Safety Requirements [3]. Specifically, the optimization - through an Integrated Decision Making Approach - of plant's safety performances, both in terms of physical performances and reliability in achieving the requested safety functions, is a specific objective which summarizes the compliance with the applicable safety principles (5 to 10 of [2])²³; this search for optimization shall support the plant's design and its safety assessment (cf. §1.2.1) to check the compliance with the quantitative and qualitative safety objectives⁴.

¹ i.e. the energy production.

² Principle 5: Optimization of protection; Principle 6: Limitation of risks to individuals; Principle 7: Protection of present and future generations; Principle 8: Prevention of accidents; Principle 9: Emergency preparedness and response; Principle 10: Protective actions to reduce existing or unregulated radiation risks.

³ Concerning the term "optimization of the safety performances", this notion is largely evoked, explicitly or implicitly, by the IAEA Safety Fundamentals [2] and specifically by principles 5 and 10.

To support the optimization, the quantitative safety objectives are complemented by qualitative notions (see Section 2) related to design goals and including the implementation of principles of DiD and the selection of characteristics which will enable to meet safety requirements. Still within the logic of optimization, the assessment of the safety architecture against both quantitative and qualitative objectives, shall take full advantage of the possible complementarity between the deterministic and probabilistic approaches.

The DiD concept and all principles for its implementation represent the foundation of the deterministic approach to build the safety architecture. If correctly interpreted / implemented, DiD helps guaranteeing - as far as practically feasible - “exhaustiveness” in terms of coverage of plausible abnormal conditions and “progressiveness” in terms of plant response to these conditions. The correct design of the “layers of provisions” which characterize - and materialize - the different DiD levels help guaranteeing the “tolerant” and the “forgiving” character of the plant’s safety, i.e. its response versus the abnormal conditions.

On its side, the Probabilistic Safety Assessment (PSA) provides a comprehensive, structured approach for identifying failure scenarios and the corresponding damages to the facility and, as a last step, allows deriving numerical estimates of the risk to the workers, the public and the environment. If appropriately developed, and if the results are adequately interpreted, the PSA can provide a methodical support and an essential contribution for determining whether the safety objectives are met, the DiD requirements are correctly taken into account and the radioactive releases related to the operation of the installation are kept below the dose limits and are kept As Low As Reasonably Achievable (ALARA). The potential role of PSA in the assessment of DiD can be summarized in saying that it contributes to achieve the overall assessment of the implemented safety architecture; PSA allows quantifying the probability of detrimental or unacceptable events (e.g. severe fuel damage conditions - PSA Level 1) through the systematic evaluation of all relevant and plausible incidental and accidental scenarios. If appropriately developed, PSA can provide additional insights on the achievement of qualitative objectives as for example about the progressiveness of the plant response to abnormal conditions or, more specifically, about the degree of “balance” for the prevention, management, and limitation of detrimental or unacceptable consequences for the whole set of the considered design basis and design extension conditions.

It is worth noting that, despite the potential of the PSA approach and that of the deterministic DiD approach, and despite the recognition of their complementarity, no specific requirements are formulated about the use of PSA for the assessment of DiD. For instance, it is recognized that PSA results shall in particular be taken into consideration in the design of provisions associated to the last levels of DiD, but the criteria to be used to obtain an optimized design is an open question. In this context, the purpose of this report is the definition of a process to explore the relationship between DiD and PSA, with the objective to optimize their complementarity and to help improving the quality of the safety assessment of the installations.

-
- *Protection must be optimized to provide the highest level of safety that can reasonably be achieved.*
 - *Protective actions to reduce existing or unregulated radiation risks must be justified and optimized.*
- The definition that can be retained is also provided by the IAEA SF1 [2]: “*protective actions must be optimized to produce the greatest benefit that is reasonably achievable in relation to the costs*”. The protective actions are characterized in terms of physical performances and reliability in achieving the requested safety functions.
- ⁴ It is essential to acknowledge that the verification of a quantitative target, even if it is extremely ambitious, cannot be sufficient and that it is essential to check the compliance with the « how the goal is achieved » and this is defined through qualitative objectives that reflect the principles of defense in depth.

1.2. SAFETY ASSESSMENT: DID CONCEPT AND LINK WITH THE PSA

1.2.1. Safety Assessment Process

In order to define the link between the DiD concept and the PSA approach it is important to identify the specific role of the two components within the context of the safety approach for the design and the assessment of a nuclear installation. The organization of the safety assessment, as defined by the IAEA GSR Part 4 Rev1 [4], can allow achieving this task. Figure 1-1 shows the main elements of the process.

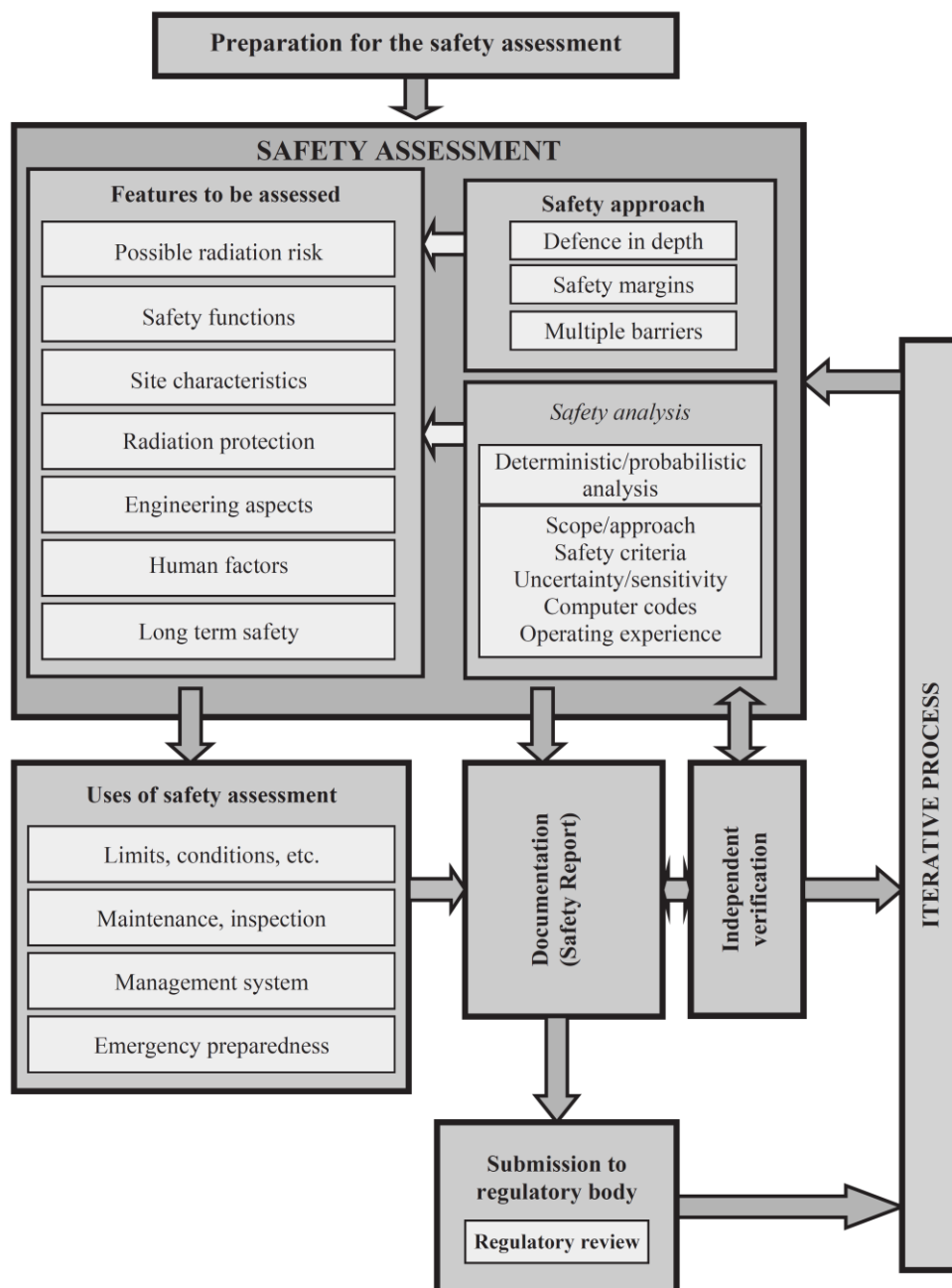


Figure 1-1 Overview of the safety assessment process [4], p. 13

According to the IAEA GSR Part 4: *Safety assessments⁵ are to be undertaken as a means of evaluating compliance with safety requirements (and thereby the application of the fundamental safety principles).... Safety assessment includes, but is not limited to, the formal safety analysis⁶*. Scope, responsibility, and purpose of the safety assessment are detailed in the GSR Part 4 Rev1 (Requirements 2, 3, 4).

Besides the “Features to be assessed” (Possible radiation risks, Safety Functions, Site characteristics, Radiation protection, Engineering aspects, Human factors, Long term safety), the two key components of the safety assessment are identified: Safety approach and Safety analysis.

The features to be assessed covers the plant’s safety architecture, its performances for the short, medium and long term versus the risk that is potentially generated⁷, and finally the “boundary conditions” which shall be considered to make the assessment.

Defence-in-Depth, Safety margins and Multiple Barriers are the basic elements of the safety approach⁸. They should be implemented in compliance with the fundamental safety principles as well as the full set of safety requirements.

Deterministic and probabilistic analyses (through the implementation of deterministic and probabilistic tools) are the key components of the safety analysis which - as a first goal - should confirm the compliance with the quantitative safety objectives.

As a complement to this goal, and coherently with the requirements of the GSR Part 4 Rev.1 (cf. below), the safety assessment should focus on the compliance of the safety architecture and its performance with the fundamental safety principle as well as with the full set of safety requirements discussed above, considering insights coming from deterministic and probabilistic evaluations.

The notion of Defence-in-Depth, if correctly interpreted, merges all the principles and requirements to be considered. These are the main topics of the approach proposed in this document.

1.2.2. Assessment of the DiD objectives and rationale

Two basic questions arise about the need of the assessment of the Defence-in-Depth:

- What does it mean “to assess defense in depth”?
- How PSA can help this assessment?

⁵ From IAEA glossary [1]: Safety assessment is the assessment of all aspects of a practice that are relevant to protection and safety; for an authorized facility, this includes siting, design and operation of the facility.

⁶ From IAEA glossary [1]: Safety analysis is the evaluation of the potential hazards associated with the conduct of an activity. Safety analysis is often used interchangeably with safety assessment. However, when the distinction is important, safety analysis should be used for the study of safety, and safety assessment for the evaluation of safety – for example, evaluation of the magnitude of hazards, evaluation of the performance of safety measures and judgment of their adequacy is safety analysis, or quantification of the overall radiological impact or safety of a facility or activity is safety assessment.

⁷ The consideration of the notion of risk, introduces implicitly the need to consider both the physical performances of the plant safety architecture, i.e. the capability to control and mitigate the possible consequences of an abnormal plant condition, and the frequency of occurrence of these consequences, i.e. the reliability of the safety architecture in achieving the requested mission

⁸ The authors believe that multiple barriers and safety margin should be considered as integral parts of DiD.

What does it mean "to assess defense in depth"?

As noted above, objectives, criteria and metrics for the DiD assessment must be clearly defined.

The assessment of the DiD shall verify the compliance with all the set of available requirements including the achievement of safety objectives (i.e. the safety analysis).

Criteria for the assessment should reflect:

- the quantitative deterministic (e.g. allowable doses) and probabilistic (e.g. frequency of occurrence of a dreaded event) objectives,
- and the qualitative objectives (i.e. the way followed to comply with the above criteria): independence between the levels of the DiD⁹, exhaustive, tolerant, forgiving, progressive and balanced plant response.

Metrics should translate the criteria above into physical parameters in order, for the designer, to be able to check their achievement.

How PSA can help this assessment?

Conventionally, the objectives of a Probabilistic Safety Analysis are to determine all the significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria (if defined).

Within the context of the present thought, the Probabilistic Safety Analysis should go beyond and support the verification that the DiD concept is adequately implemented, within the larger context of the safety assessment.

The availability of an adequate representation of the safety architecture is a prerequisite and the foundation for the analysis/assessment.

From this point of view it is interesting to recall ref. [33] which recognizes that "*current PSA studies lack a clear evaluation of all Defence-in-Depth levels*", defines an objective "*to investigate to what extent measures and parameters of PSA can be used in order to give estimates of the five levels of Defence-in-Depth*" and provides proposals for presenting the safety architecture in a way that should be consistent with the principles of the DiD.

The need for assessing the Defence-in-Depth is explicitly recognized by the GSR Part 4 Rev1 (Requirement 13) which details the objective to be pursued: "*It shall be determined in the assessment of Defence-in-Depth whether adequate provisions have been made at each of the levels of Defence-in-Depth.*"

The achievement of this objective is supported by further requirements (4.45-4.58) which can be merged and articulated into four complementary objectives defined as follows:

⁹ The requirements for the independence between the different DiD levels are sufficiently explicit and motivated to justify the fact that the "independence as far as feasible" shall be searched in designing or improving a safety architecture.
Some partial evaluation of NPP versus the principle of the DiD are available (IAEA SR 46 [9]; JANSI Activity post Fukushima on the Japanese NPP [24]) and they systematically show the benefits of the exercise.

1) Adequacy of the implemented provisions

“4.45. It has to be determined in the assessment of Defence-in-Depth whether adequate provisions have been made at each of the levels of Defence-in-Depth to ensure that the legal person responsible for the facility can:

- a) Address deviations from normal operation or, in the case of a repository, from its expected evolution in the long term;*
- b) Detect and terminate safety related deviations from normal operation or from its expected evolution in the long term, should deviations occur;*
- c) Control accidents within the limits established for the design;*
- d) Specify measures to mitigate the consequences of accidents that exceed design limits;*
- e) Mitigate radiation risks associated with possible releases of radioactive material.”*

2) Adequacy of the approach for the construction of the safety architecture

“4.46. The necessary layers of protection, including physical barriers to confine radioactive material at specific locations, and the necessary supporting administrative controls for achieving Defence-in-Depth have to be identified in the safety assessment. This includes identification of:

- a. Safety functions that must be fulfilled;*
- b. Potential challenges to these safety functions;*
- c. Mechanisms that give rise to these challenges, and the necessary responses to them;*
- d. Provisions made to prevent these mechanisms from occurring;*
- e. Provisions made to identify or monitor deterioration caused by these mechanisms, if practicable;*
- f. Provisions for mitigating the consequences if the safety functions fail.”*

3) Compliance with the principles of the Defence-in-Depth

“4.47. To determine whether Defence-in-Depth has been adequately implemented, it has to be determined in the safety assessment whether:

- a. Priority has been given to: reducing the number of challenges to the integrity of layers of protection and physical barriers; preventing the failure or bypass of a barrier when challenged; preventing the failure of one barrier leading to the failure of another barrier; and preventing significant releases of radioactive material if failure of a barrier does occur;*
- b. The layers of protection and physical barriers are independent of each other as far as practicable;*
- c. Special attention has been paid to internal and external events that have the potential to adversely affect more than one barrier at once or to cause simultaneous failures of safety systems;*
- d. Specific measures have been implemented to ensure reliability and effectiveness of the required levels of defence.”*

- 4) Availability of adequate margins to correctly address the uncertainties and avoidance of cliff edge effects.

“4.48. It has to be determined in the safety assessment whether there are adequate safety margins in the design and operation of the facility, or in the conduct of the activity in normal operation and in anticipated operational occurrences or accident conditions, such that there is a wide margin to failure of any structures, systems and components for any of the anticipated operational occurrences or any possible accident conditions. Safety margins are typically specified in codes and standards as well as by the regulatory body.

It has to be determined in the safety assessment whether acceptance criteria for each aspect of the safety analysis are such that an adequate safety margin is ensured.” Moreover, “where practicable, the safety assessment shall confirm that there are adequate margins to avoid cliff edge effects that would have unacceptable consequences.”

1.2.3. Deterministic and probabilistic approaches for the assessment

Deterministic and Probabilistic analysis are recognized as the two complementary elements for the safety analysis. This complementarity is formulated by the GSR Part 4 Rev1 [4]: “15. *Both deterministic and probabilistic approaches shall be included in the safety analysis.*” The reference recognizes that “*deterministic and probabilistic approaches have been shown to complement one another and can be used together to provide input into an integrated decision making process.*” Complementary insights, specific for their singular contributions are provided by the GSR Part 4 Rev. 1:

- *“4.54. The aim of the deterministic approach is to specify and apply a set of conservative deterministic rules and requirements for the design and operation of facilities or for the planning and conduct of activities.”*
- *“4.55. The objectives of a probabilistic safety analysis are to determine all significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined.”*

Coherently with the above indications, once principles, requirements, guidelines, objectives and safety options have been defined/selected, the full process (iterative as needed) for the design and the assessment of the installation (i.e. its retained¹⁰ safety architecture), including the safety analysis¹¹, can be summarized as in Table 1-1. The table resumes - roughly - the crosscutting relationships between, on one side, the steps of the process for the design and assessment of the safety architecture and, on the other side, the expected contribution of deterministic and probabilistic approaches. Its content demonstrates the complementary role of the two approaches. Appendix 2 details the analysis with the support of the Gen IV- RSWG ISAM methodology [15].

¹⁰ Typically, the designer selects among several options the design solution considered as “more pertinent” (e.g. “active” versus “passive” solution, “known” versus “innovative”). The selection is based on several criteria (not all related to safety, as result of an iterative process that ends on a “retained” architecture.

¹¹ As already indicated, according to GSR Part 4, “Safety analysis” is only a part of the “Safety assessment”.

Table 1-1 Relationships between the steps for the design and assessment of the safety architecture and the role of the deterministic and probabilistic approaches

Steps for the design and the assessment of the retained safety architecture ➡	Deterministic	Probabilistic
<i>Regulatory Framework (Goals, objectives, principles, requirements, guidelines)</i>	✓	✓
<i>Selection of Safety Options and provisional Provisions</i>	✓	✓
1. <i>Compliance / consistency of the design options with the principles, requirements and guidelines</i>	✓	
2. <i>Identification, prioritization and correction (if feasible) of discrepancies between design options with the principles, requirements and guidelines,</i>	✓	
3. <i>Identification of challenges to the safety functions,</i>	✓	
4. <i>Identification of mechanisms (initiating events) and selection of significant (envelope) plants conditions to be considered for the design basis events (DBE),</i>	✓	✓
5. <i>Selection and categorization of representative design extension conditions (without and with core melting; DEC A & DEC B with the WENRA terminology) to be considered for the design basis¹²</i>	✓	(✓) ¹³
6. <i>Selection of external events that exceed the design basis and for which safety systems are designed to remain functional both during and after the external event</i>	✓	(✓) ^{13?}
7. <i>Identification of plant event or sequences that could result in large or early radioactive releases that must be practically eliminated</i>	✓	✓
8. <i>Identification and selection of needed provisions, implementation within the corresponding “layers of provisions” for the different levels of the DiD</i>	✓	
9. <i>Design and sizing of the provisions,</i>	✓	✓
10. <i>Response to DBE and DEC events (safety analysis),</i>	✓	✓
11. <i>Final assessment for a safety architecture that shall meet the safety objectives and should be as far as reasonably possible¹⁴:</i>		
o <i>Exhaustive,</i>	✓	
o <i>Progressive,</i>	✓	✓
o <i>Tolerant,</i>	✓	✓
o <i>Forgiving,</i>	✓	✓
o <i>Balanced.</i>		✓

Table 1-1 shows the complementary role of the deterministic and probabilistic approaches, needed for:

- the identification of mechanisms (initiating events) and selection of significant (enveloping) plants conditions to be considered for the design basis events (DBE)⁹;
- the selection and categorization of representative design extension conditions (without and with severe fuel damage conditions; DEC A & DEC B) to be considered for the design basis;
- the selection of external events that exceed the design basis and for which safety systems are designed to remain functional both during and after these events;

¹² The integration of Design Extension Conditions (DEC) within the design basis is consistent with the current requirements expressed by the IAEA NSSR 2/1 Rev.1 [3] - Requirement 20.

¹³ The contribution to this step is essentially deterministic even if it is recognized that probabilistic assessment can help, for example, for the identification of complex events / sequences which probability of occurrence justify their consideration for the design and / or for the categorization of the selected initiating events.

¹⁴ See Section 2.3 for details about these objectives and their rationale / links with the IAEA Safety Standards.

- the identification of plant events or sequences that could result in high radiation doses or large or early radioactive releases that must be practically eliminated;
- the design and sizing of the provisions;
- the analysis of the response to DBE and DEC events (safety analysis);
- the verification of the progressive, tolerant and forgiving, character of the safety architecture.

The peculiar key role of the deterministic approach includes the verification of the compliance / consistency of the design options with the principles, requirements and guidelines for the identification, prioritization and correction (if feasible) of discrepancies (if any), the identification of challenges to the safety functions, the identification and selection of needed provisions and their implementation within the corresponding layers of provisions of the DiD, and the support to the exhaustiveness in the coverage of “unexpected” plant conditions. The assessment of the balanced character of the safety architecture is specific to the probabilistic assessment, which allows identifying the presence (to be avoided) of excessive or significant uncertain contributors to risk. Additional contributions that the PSA - if appropriately developed - can provide to the assessment of the implemented safety architecture, are introduced and discussed in the following sections.

1.3. THE ASSESSMENT OF DEFENSE IN DEPTH WITH THE PSA

Section 1.2 underlines that deterministic and probabilistic approaches, as they are implemented today, are complementary within the whole context of the design and the assessment of the safety architecture.

The same section points out that for the DiD assessment there is the fundamental need to determine “*whether adequate provisions have been made at each of the levels of Defence-in-Depth*” (cf. GSR Part-4.45). As indicated within the §1.2.2, the achievement of this objective is articulated into four complementary objectives relevant to (i) the Adequacy of the implemented provisions, (ii) the Adequacy of the approach for the construction of the safety architecture, (iii) the Compliance with the principles of DiD and (iv) the Availability of adequate margins to correctly address the uncertainties and avoidance of cliff edge effects.

The need for checking the “adequacy”, the “compliance”, and the “existence of adequate margins” generates the necessity for intermediate steps which should create the conditions for the analysis.

Consistently with this objective, the Figure 1-2 presents the whole process for the assessment of the DiD (i.e. for the assessment of the Safety architecture implementing the DiD).

The DiD assessment process is articulated in four main steps relevant to:

- the definition of safety objectives, both quantitative and qualitative;
- the identification and recognition of all loads and environmental conditions that may affect the operation;
- the representation, as comprehensive as practicable, of the safety architecture in a manner that shall be consistent with the principles of Defense in Depth¹⁵ and useful (and complemented) for the PSA¹⁶;
- the evaluation of the effectiveness of DiD levels, i.e. their physical efficiency and reliability.

¹⁵ Different independent levels, functionally redundant and with a clear identification of relevant provisions.

¹⁶ The basic idea is to achieve a representation of the safety architecture that allows answering the questions: what is doing what during the management of an abnormal / accidental condition? Are DiD levels (i.e. the layers of provisions for each initiating event) correctly identified, designed/sized and implemented? Are DiD principles, e.g. the independence between the layers of provisions, adequately guaranteed?

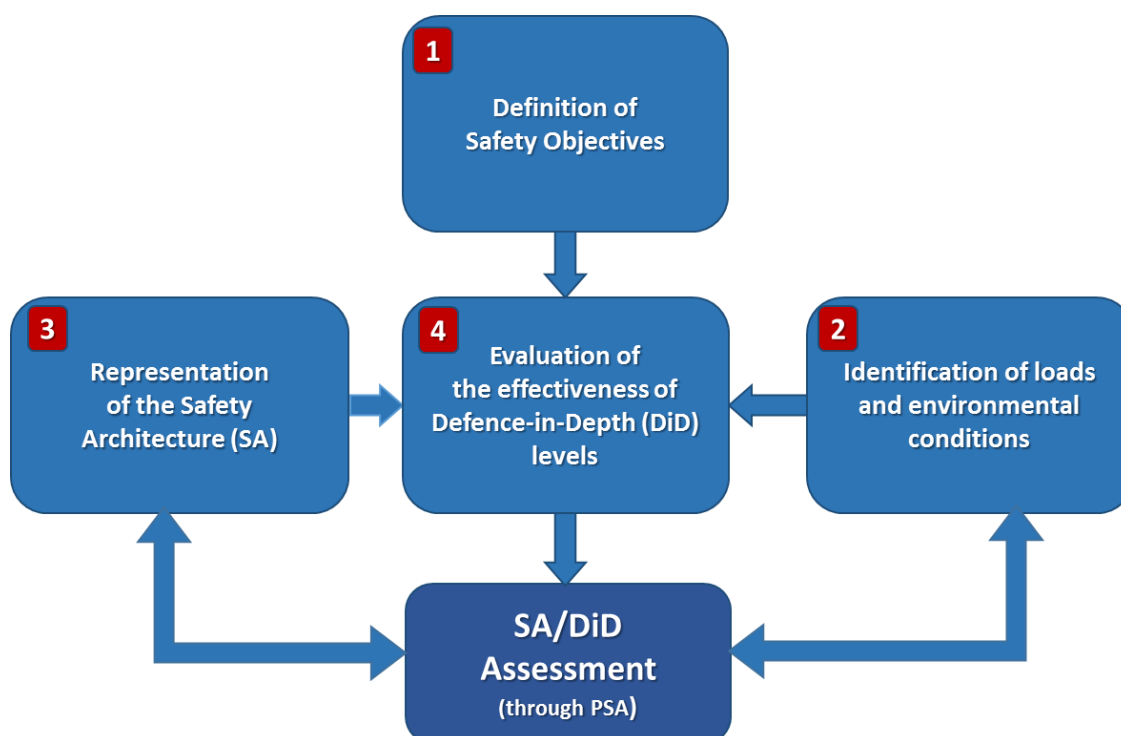


Figure 1-2 Process for the PSA assessment of Defense in Depth

The following sections - from 2 to 5 - provide details on each of the four steps of the process. Section 6 presents and discusses the final step with the practical assessment of the safety architecture, and the corresponding DiD, with the support of the PSA.

Some specific remarks on existing reactors and PSA are provided in the Section 7.

Conclusions are summarized in Section 8.

2. DEFINITION OF SAFETY OBJECTIVES

2.1. DETERMINISTIC APPROACH AND RELEVANT OBJECTIVES

For nuclear installations in general and for reactors in particular, to fit with the principle of the DiD, the implemented “layers of overlapped provisions” (INSAG 10, [11]), should ensure, both for normal operation of the system, as well as for postulated incidents and accidents, the achievement of the three basic safety functions: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel store; and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases (NSSR 2/1 Rev.1 [3]).

The nuclear safety analysis aims at showing that the implemented provisions are sufficient to ensure compliance with Safety Objectives proposed by the designer and endorsed by the safety authority. This analysis, conventionally “deterministic and conservative, is based on a limited set of bounding initiating events (plant conditions) that are categorized considering their estimated frequency of occurrence (e.g. cat. I to IV) and taken as references for the design basis, for the different possible states of the system (normal operation, shutdown, maintenance, etc.).

In addition to these “design basis” studies, the compliance with the DiD principles requires on one hand to take into account the possible lack of completeness in the deterministic analysis and, on the other hand, to demonstrate the potential for the prevention, control and mitigation of degraded conditions of the installation. To do this, the plant conditions of the facility, as defined above, are conventionally complemented with the consideration of 1) accident situations generated by multiple failures or total loss of redundant provisions without significant fuel degradation, 2) situations of severe accidents with significant fuel degradation (all these being considered as Design Extension Conditions - DEC) or initiating events induced by natural or human-made external hazards exceeding the design basis. Their analysis (using less conservative assumptions and rules) may leads to the design and the implementation of specific additional provisions and / or the adaptation of existing provisions to ensure that the corresponding safety objectives are met.

Finally, initiators, situations or sequences involving very energetic phenomena, whose consequences could not be mitigated by reasonable technical means and that could lead to large or early releases into the environment, should be identified and “practically eliminated” (see Section 2.4 and Fig. 3.2).

The deterministic approach, as described above, is consistent with the recommendations of WENRA for the new reactors¹⁷ and, in particular, with the revised structure of the levels of DiD as shown within the Table 2-1 [13].

For each plant condition category (Associated Plant Conditions), WENRA defines the allowable “radiological consequences” and suggests an unambiguous correspondence between these categories (last column) and the levels of DiD (first column).

¹⁷ These recommendations are as far as reasonably feasible applicable to plants currently operating or under construction [13]: “The safety objectives address new civil nuclear power plant projects. However, these objectives should also be used as a reference to help identify reasonably practicable safety improvements for “deferred plants” and existing plants during Periodic Safety Reviews”.

Table 2-1 WENRA RHWG Proposed revision of the level of DiD^{18,19}

Levels of defence in depth	Objective	Essential means	Radiological consequences	Associated plant condition categories
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)	Normal operation
Level 2	Control of abnormal operation and failures	Control and limiting systems and other surveillance features		Anticipated operational occurrences
Level 3 ⁽¹⁾	3.a Control of accident to limit radiological releases and prevent escalation to core melt conditions ⁽²⁾	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact ⁽⁴⁾	Postulated single initiating events
	3.b	Additional safety features ⁽³⁾ , accident procedures		Postulated multiple failure events
Level 4	Control of accidents with core melt to limit off-site releases	Complementary safety features ⁽³⁾ to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time	Postulated core melt accidents (short and long term)
Level 5	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off site radiological impact necessitating protective measures ⁽⁵⁾	-

¹⁸ It may be noted that, within Table 2-1, the level 3a (DiD level 3a - Postulated Single Initiating Events) covers the categories III and IV of the "design basis" (i.e.: the conventional design).

¹⁹ See [13] for detailed comments on the table.

2.2. QUANTITATIVE SAFETY OBJECTIVES

This section introduces some basic concepts through which the assessment of the safety architecture can be engaged with quantitative safety objectives.

2.2.1. The Risk Space

The allowable radiological consequences as defined within the Table 2-1 are - generally speaking - represented by an acceptable domain within the “risk space”. Figure 2-1 provides the “Generic F-C Curve, with ALARA region” as reported in the NUREG 2150 [10]²⁰. The boundaries of this domain are defined by pairs “frequency of occurrence - consequences” (Farmer curve) which allow defining the risk profile²¹. The ALARA concept is added as a systematic and essential complement (cf. Figure 2-1).

NB: The box “Situations practically eliminated” has been added by the authors to the Figure of [10]²².

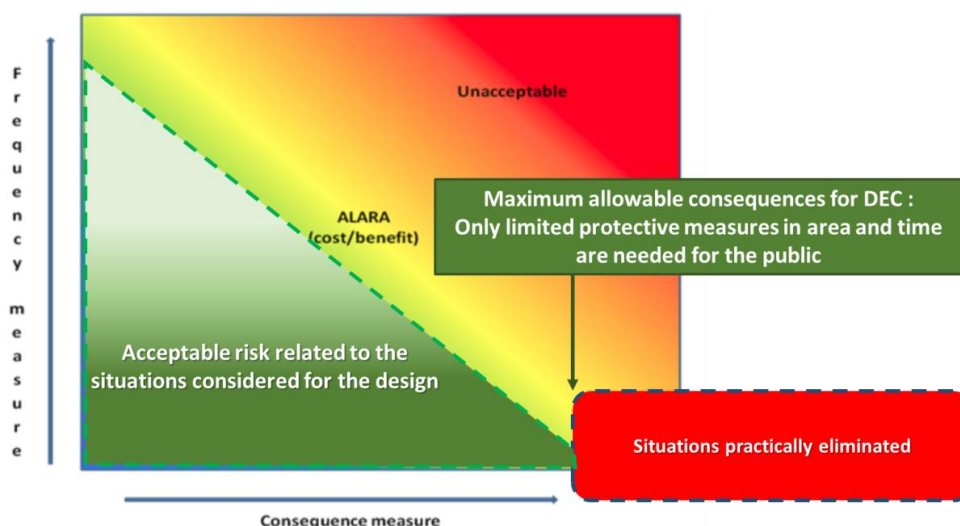


Figure 2-1 Risk space, NUREG 2150 [10]

²⁰ Cf. NUREG 2150 [10] : “The development and use of the F-C curve similar is perceived by some as a dramatic departure from past practices, but many programs already incorporate aspects of the approach by differentiating between high frequency-low consequence activities and the potential for low frequency-high consequence accidents. In the analyses to support licensing of nuclear power reactors, events have traditionally been defined within the categories of (1) normal operation, (2) anticipated operational occurrences, (3) design-basis accidents, and (4) beyond-design-basis accidents. The primary criteria for placing scenarios within the above categories are related to event frequencies. The allowable consequences (defined in terms of degree of fuel damage) are defined for the categories, and generally more damage is acceptable for scenarios with lower frequencies.” [10].

²¹ The figure has a purely conceptual interest; e.g. for the decision maker (designer - regulator) the curve may not necessarily be iso-risk.

²² The term “Residual risk” is intentionally avoided because no unequivocal definition exists. The term is not defined within the IAEA glossary [1]. Following the IRSN [27] two qualitative “Residual risks” should be considered: *Environmental residual risk*: Risk remaining after the reduction in exposure provided by the collective protection equipment; *Individual residual risk*: Risk remaining after the reduction in exposure provided by the individual protection provisions. Consistently, following the European Nuclear Society [27], the Residual risk is defined as the “Remaining risk which cannot be defined in more detail after elimination or inclusion of all conceivable quantified risks in a risk consideration”; according to this definition, the practical elimination coincides with the rejection into the residual risk. From a quite different perspective, IAEA [29] defines the Residual risk as *the risk which remains despite provisions made to prevent accidents and, if an accident occurs, to minimize the consequences*; according to this definition, roughly speaking, the Residual risk is conceptually the same as the risk accepted for the facility.

Conventionally, several metrics are associated with the risk space typically in terms of frequency related to a defined scenario, e.g.: Frequency of occurrence for the postulated initiating event (PIE); Core damage and fuel damage frequency; Large release and/or early release frequency(ies) (and other release frequency measures). For each initiating event and corresponding sequence, whose consequences are potentially unallowable and are positioned on the risk space, some “conditional metrics” can be defined: Conditional Core Damage Probability (CCDP); Conditional Fuel Damage Probability (CFDP), Conditional Large Release Probability (CLRP), and Conditional Early Release Probability (CERP)²³.

Some probabilistic targets are introduced in Section 2.2.2, in order to provide quantitative orders of magnitudes which are needed for the assessment. Following the authors, these targets have to be complemented (without modifying the principle for an acceptable region) with criteria and metrics that translate the requirements and recommendations stated by IAEA and WENRA:

- *“Defence-in-Depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. The independent effectiveness of the different levels of defence is a necessary element of Defence-in-Depth.” [2];*
- in case of severe accidents the consequences shall be consistent with the objective that *“only protective measures that are limited in terms of times and areas of application would be necessary and that off-site contamination would be avoided or minimized”²⁴;*
- sequences that could result in unacceptable radioactive consequences releases shall be practically eliminated;
- a complementary key notion, which is also critical vis-à-vis the safety objectives and must be taken into account, is that of *“cliff edge effects”²⁵* with the *“sufficient margin”* (or *adequate margins*) which must be guaranteed for the sequences which have the potential to trigger these unacceptable effects.

Complementary qualitative safety objectives, related to the notion of robustness, exhaustiveness, progressiveness, as well as tolerant, forgiving and balanced characters, are introduced in §2.3.

Indications on how to manage all the above criteria are provided in §2.4.

²³ The availability of these data can, for example, provide insights to the designer to decide if it is better to work on the “upstream” frequency of occurrence of the initiating event or, if it is more interesting to strengthen the reliability of the architecture (i.e. the different layers of provisions” which are implemented to manage the sequence).

²⁴ WENRA Objective O3 [13]:”*for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures”.*

²⁵ *“A cliff edge effect, in a nuclear power plant, is an instance of severely abnormal plant behavior caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input.” [3]*

2.2.2. Probabilistic targets

The prevention of severe accidents

To compensate for the possible lack of completeness in identifying situations considered for the design, in line with the principles of DiD, the designer is requested to conventionally consider plant degradations which mobilize, inside the containment, source terms for which a release outside of the facility would be unacceptable. These situations (generally termed “severe accident”) correspond to that identified with the term “postulated core melt accident” in the Table 2-1.

From a probabilistic point of view, discussing about orders of magnitude, as indicated in the INSAG-12 [11], the objective is a frequency of severe damage to the plant (e.g. core melting) lower than $\sim 10^{-5}$ /reactor year (CDF, equivalent PSA level 1) all initiators considered and combined²⁶. This objective shall be correlated with a further reduction of a factor 10 - ($10^{-5} > 10^{-6}$ /reactor year)²⁷ usually endorsed by regulators -, for the unacceptable offsite consequences²⁸, all events considered and combined (equivalent PSA level 2). On a conceptual level, the containment, acting as a final barrier, provides the necessary order of magnitude to ensure compliance with 10^{-6} /reactor year for unacceptable consequences (10^{-5} /reactor year + containment failure or bypass $\Rightarrow 10^{-6}$ / reactor year). These global objectives, even if simplified, are not directly usable for the design and need to be translated into practical intermediate goals that can guide the designer for the selection of adequate provisions and their implementation within the architecture of the entire plant and, at the same time, for the definition of the performance of these provisions, i.e. their sizing, as required for the achievement of the safety functions. These intermediate objectives must also provide margins to cover the uncertainties correlated with the probabilistic approach.

For instance, the first of these objectives addresses the internal events which, in practice, are the basis for the design of all the provisions of the safety architecture. It is therefore proposed to translate the need for improved margins retaining, for the prevention of severe accidents due to the internal initiating events (excluding hazards), an objective of about 10^{-6} per reactor per year. Assuming about ten to twelve independent families of initiating events, considered separately, this figure is reduced to about “ 10^{-7} per reactor per year per family of initiators”, all safety functions combined. This is the reliability which is requested for the whole set of actions implemented to manage a given initiating event (representing of a family of events), i.e. by the levels 1 to 3 of the DiD, to prevent severe accidents conditions.

²⁶ Cf. IAEA No. SSG-3 - Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants: *The objectives for core damage frequency suggested by INSAG12 are (a) 1×10^{-4} per reactor-year for existing plants and (b) 1×10^{-5} per reactor-year for future plants. It was not explicitly specified in INSAG for which scope of PSA the numerical values are applicable. It is assumed that a full scope PSA is meant.*

²⁷ Cf. INSAG 12: *Severe accident management and mitigation measures could reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response.*

²⁸ Note that, while the figure “ 10^{-6} /reactor year” does not change, it is the content of the notion of “unacceptable consequences offsite” that has been significantly reduced over the years. With the current recommendations the notion of unacceptable consequences offsite does correspond to the objective for having nothing more than (cf. WENRA) “... limited protective measures in area and time are needed for the public and that sufficient time is available to implement these measures”.

These actions for the control of safety functions, whose failure will lead to the severe accident configuration, are closely interconnected. In these conditions it is not feasible to deduce a specific probabilistic target for a specific safety function. However, the orders of magnitude are the same and the objective for the prevention of this failure can be considered as “fraction of 10^{-7} per reactor year, per family of initiators and per function”.

The management of Severe Accident with core degradation

In accordance with the indications set out in Table 2-1 (i.e. *Control of accidents with core melt to limit off-site releases*) and to guarantee the *Practical elimination of situation that could lead to early or large releases of radioactive materials*, it is necessary to consider the establishment of specific “layers of provisions” for the management of Severe Accident (more generically “conditions with plant degradation”). These layers materialize, for a given sequence, the 4th level of the DiD. The probabilistic targets for the whole sequence are those that are associated with unacceptable consequences, i.e. an order of magnitude over the prevention level: 10^{-6} /reactor year. This additional decade could be tentatively allocated to the reliability of the 4th level of the defense but in practice, given the indications post Fukushima, especially with the requirement for the practical elimination of sequences leading to large or early release, it is a higher reliability that should be guaranteed.

Natural hazards exceeding design basis conditions

The consideration of “*natural hazards exceeding those to be considered for design*” complements what is already done for severe accidents and, in these conditions, there is no reason to modify the probabilistic targets that are associated with unacceptable consequences. Similarly to what is done for Severe Accidents, and in accordance with the indications coming from the post Fukushima “Stress tests”, the designer must identify and implement specific provisions for the management of these situations (“Hardened Safety Core” (HSC) [18])²⁹. These provisions complement those provided for managing situations with core degradation / fuel damage; they are implemented to ensure proper operation in the extremely degraded conditions and the designer must guarantee the level of their physical efficiency and reliability (i.e. the capability to achieve the mission as requested).

Events, conditions or sequences practically eliminated

Finally, initiators, situations or sequences that lead to intolerable large or early releases in the environment, involving phenomena (e.g. very energetic) whose consequences could not be mitigated by reasonable technical means, should be identified and “practically eliminated”. To achieve this objective, the loss of provisions performing safety functions whose failure can cause these intolerable effects, should be significantly lower than 10^{-7} /reactor year³⁰, even if this “cut off value” cannot be used alone to justify the practical elimination (see Section 2.4).

²⁹ HSC indicates a limited number of material / organisational / human systems providing essential safety functions even in extreme circumstances, i.e. circumstances exceeding those adopted for the general design of the facility. This term has been used according to the indications provided by the European Nuclear Safety Regulators Group, among the measures imposed after the accident at Fukushima Daiichi to reinforce the safety requirements for the prevention of natural risks, the management of loss of electrical power and cooling systems situations and for management of severe accidents.

³⁰ It is the overall probability of the sequence which, after the appearance of the initiator continues with the loss - in cascade - of provisions (i.e.: the different LOP implemented at different levels of DiD for the different

2.3. QUALITATIVE SAFETY OBJECTIVES

This section introduces additional objectives, defined qualitatively, to be considered in the assessment of the safety architecture in order to verify the fulfillment of safety requirements stated in the NSSR 2/1 Rev.1 [3]. Even if no new notions are introduced (all the objectives being recognizable in the NSSR 2/1 requirements), formal definition are proposed as further step toward the definition of metrics (out of scope of this document).

Robustness

Among the qualitative objectives, the notion of “robustness” is systematically evoked both for the design and for the assessment of the safety architecture³¹. This notion cannot be reduced, but envelops, the request for “simplicity” of the safety architecture³² and to the meeting of values / figures consistent with the quantitative safety objectives, even if these figures are extremely low.

Obviously, the adequate consideration of uncertainty (either aleatory or epistemic) is essential to improve robustness but the way to achieve safety, which is strongly connected with the implemented Defense in Depth, is also a key contributor.

This “way to achieve safety” can be defined qualitatively through corollary notions concerning the essential characteristics required to the safety architecture, namely: Exhaustiveness, Progressiveness, Tolerance, Forgiving and Balanced character. These notions are detailed in the following starting from the proposals formulated by the GIF/RSWG (underlined) [16] and integrating them with complements and indications aimed at reducing potential ambiguities. The reference to the relevant requirement specified in the NSSR 2/1 Rev.1 [3] is provided for each one of these characteristics.

Exhaustiveness

The exhaustiveness character of the safety architecture represents the capacity to manage a comprehensive set of postulated initiating events, being considered in the design and even those unexpected or unidentified.

safety functions) implemented to prevent, control and minimize the consequences of the initiating event.

³¹ Within WENRA [13], the notion of robustness is evoked to meet several qualitative objectives:

- “For the DiD approach which is intended to provide robust means to ensure the fulfilment of each of the fundamental safety functions.
- To satisfy the basic safety expectations on the independence between different levels of DiD for which a more robust demonstration of the independence between levels of DiD is requested.
- For the analysis methodology, for which adequate methods have to be utilized in order to show the robustness and reliability of the approach.
- The robustness of a plant’s safety case which is requested to support the practical elimination.
- A robust design based on DiD with sizeable safety margins and diverse means for delivering fundamental safety functions as well as comprehensive operator response plans is required to fully integrate the lessons Learnt from the Fukushima Dai-ichi accident.
- Finally robust complementary safety features (DiD level 4) shall be specifically designed for fulfilling safety functions required in postulated core melt accidents.”

³² Considering, for example, a given probabilistic objective for a given sequence, conventionally represented through an event tree - e.g. $10^{-9}/ry$ -, one can easily understand that the same figure can be obtained in different manners, i.e. nine independent steps/failures each characterized by an unreliability of $10^{-1}/demand$ or three independent steps/failures each with a reliability of $10^{-3}/demand$. One can reasonably suppose that the demonstration will be more easy and robust for the second - more simple - safety case.

This qualitative objective is consistent with the NSSR 2/1 Rev.1 [3] Requirement 16: *“The design for the nuclear power plant shall apply a systematic approach to identifying a comprehensive set of postulated initiating events such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design.”*

The identification of risks, based on challenges to the fundamental safety functions, should look for exhaustiveness. In parallel, the identification of the corresponding scenarios (i.e. the mechanisms and the corresponding provision’s failures that materialize the challenges) to be retained to design and size the safety architecture provisions, must be as exhaustive as possible³³.

Among the strong motivations of the DiD, there is the objective to cover the potential lack of comprehensiveness in the identification of events. DiD, with all its principles, aims at supporting a robust demonstration about the acceptability of all “known” risk contributors (including “known” uncertainty), and at providing confidence that the adopted conservatisms allow enveloping the “unknown” ones.

One example is the selection and sizing of provisions for the 4th level of the DiD (i.e. for the management and the mitigation of severe accident), which can be the result of a mechanistic approach (i.e. sequential or “bottom-up”) or, rather, “top-down” selecting arbitrarily plant damaged states representing all plausible plant degraded conditions (i.e. symptom based).

Progressiveness character

The Progressiveness character of the safety architecture represents the capacity “to degrade gradually” in case of hazardous event and loss of safety functions, the objective is to avoid that the failure of a given provision (or layer of provisions) entails a major increase of consequences, without any possibility of restoring safe conditions at an intermediate stage.

This qualitative objective is consistent with the IAEA Safety Fundamentals [2]: *“3.31 The primary means of preventing and mitigating the consequences of accidents is ‘Defence-in-Depth’. Defence-in-Depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available.”*

A progressive degradation of the safety architecture requires, firstly and consistently with the principles of DiD, the implementation of subsequent functionally redundant levels of protection, each with appropriate performances and reliability, which have to fail before that harmful effects could be caused to people or to the environment and, secondly, to have consequences that evolve, as much as possible, in a linear manner crossing these levels without risk for cliff edge effects (cf. Fig. 2.1).

³³ The requirement for exhaustiveness is not formulated in the same way for proven technology of reactors or for new concepts: for the first, it is essentially the feedback experience that ensures completeness; for the latter it is the requirement of making a Phenomena Identification and Ranking Table (PIRT study, see appendix 2, that will confirm the list of events and phenomena to be considered.

Tolerant character

The Tolerant character of the safety architecture represents the capacity to manage intrinsically variations in the operating conditions of the plant, i.e. avoiding that small deviations of the physical parameters outside the expected ranges lead to significant consequences.

This qualitative objective is consistent with the requirement 5.8 in the IAEA NSSR 2/1 Rev.1 [3]: *“The expected behavior of the plant in any postulated initiating event shall be such that the following conditions can be achieved, in order of priority: (1) A postulated initiating event would produce no safety significant effects ...”*.

The Tolerant response of the system is guaranteed by appropriate design requirements aimed at ensuring the required efficiency, reliability and margins (i.e. conservative design) of material and immaterial provisions which achieve the safety functions (e.g. engineered safety features, inherent characteristics). The request for a *tolerant defense* includes the rejection of any risk for “cliff edge effects”. The corresponding criteria are established in terms of allowable ranges around the normal operating conditions.

Forgiving character

The Forgiving character of the safety architecture shall provide guarantee of the availability of a sufficient grace period and make if possible repairs (restorations) during accidental situations, but strongly underlined without compromises with the safety requirements; it is representative of the capacity to achieve safe conditions through - in priority order - inherent characteristics of the plant, passive systems or systems operating continuously in the necessary state, systems that need to be brought into operation, procedures.

This qualitative objective is consistent with the requirement 5.8 (1) stated in the IAEA NSSR 2/1 Rev.1 [3]: *“The expected behavior of the plant in any postulated initiating event shall be such that the following conditions can be achieved, in order of priority: (1) A postulated initiating event would produce only a change towards safe plant conditions by means of inherent characteristics of the plant; (2) Following a postulated initiating event, the plant would be rendered safe by means of passive safety features or by the action of systems that are operating continuously in the state necessary to control the postulated initiating event; (3) Following a postulated initiating event, the plant would be rendered safe by the actuation of safety systems that need to be brought into operation in response to the postulated initiating event; (4) Following a postulated initiating event, the plant would be rendered safe by following specified procedures.”*

Corresponding adequate grace delays have to be fixed by the designer and endorsed by the regulator.

Balanced character

The Balanced character of the safety architecture represents the evenness of contributions of different events / sequences to the whole risk, i.e.: no sequence participates in an excessive and unbalanced manner to the global frequency of radioactive releases.

The requirement and the corresponding criteria shall be consistent with the requirement 5.76 stated in the IAEA NSSR 2/1 Rev.1 [3]: “The design shall take due account of the probabilistic safety analysis(a) Establishing that a balanced design has been achieved such that no particular feature or postulated initiating event makes a disproportionately large or significantly uncertain contribution to the overall risks, and that, to the extent practicable, the levels of Defence-in-Depth are independent; ...”.

Remark: if excessive unbalance is detected following a PSA study it could be a sign of unsuitable safety provisions; for instance Loss Of Offsite Power (LOOP) is often an important contribution to risk and can be a cause of unbalanced results; in that case, additional provisions are needed to reduce core melt frequency of that family event to the same level as other accident families. However, the notion of “balanced design” should not be considered as excessively mandatory and should be associated with that of “reasonably feasible”.³⁴

2.4. CRITERIA & METRICS

All the key-notions mentioned above should be reformulated by criteria and metrics that contribute to the assessment of the safety architecture through the evaluation of the results coming from the safety analyses. Some indications are provided in the following in order to support the definition of these criteria and metrics (which is an on-going activity, not fully addressed by this document), and to clarify the role of PSA in their assessment.

Off-site measures limited in times and areas

The “limited protective measures in area and time ...needed for the public and (the) sufficient time ... available to implement ... measures” will be deterministically defined fixing acceptable amounts for the released source term and corresponding kinetics of release. WENRA [13] provides some quantitative data about this goal (cf. §3.4 - Position 4 - Provisions to mitigate core melt and radiological consequences).

The table in Figure 2-2 provides the interpretation by WENRA (Position 4, [13]) of “limited protective measures”. The tables define the acceptable conditions for Permanent relocation, Evacuation, Sheltering and Iodine prophylaxis applicable, as goals, in the design phase of new reactors.

Measure	Evacuation zone	Sheltering zone	Beyond sheltering zone
Permanent relocation	No	No	No
Evacuation	May be needed	No	No
Sheltering	May be needed	May be needed	No
Iodine Prophylaxis	May be needed	May be needed	No

Figure 2-2 Design goals for areas where limited protective measures may be needed [13]

The corresponding safety objectives, in terms of allowable amount and kinetic for the releases, shall be defined as a function of the site. PSA Level 1 and Level 2 play an essential role for the assessment.

³⁴ E.g., if one considers the objective of 10^{-6} /yr for the prevention of Severe Accident as a result of all the families of internal initiators, and if the majority of these families are already largely beyond this value (e.g. 10^{-7} /yr), for those who remain in the range of the target, is not necessary to seek the same type of performance unless it is “reasonably practicable” i.e. without unreasonable efforts.

Independence of DiD levels

The independence between the DiD levels is one of the key verification to be performed. This requires a comprehensive and appropriate representation of the safety architecture, in terms of provisions implemented for each initiating event, and for each safety function at each level of DiD.

The objective is to ensure that the failure of a DiD level does not affect the efficiency and the performance of the next one(s) (i.e.: functional redundancy).

The Objective Provision Tree (OPT) methodology and the complementary notion of Line of Protection (LOP), discussed in the Section 4, identify for each initiating event, for each safety function and for each level of the defence the provisions implemented and allow identifying possible lack of independence between the DiD levels (e.g. overlapping of provisions on different levels)³⁵. In this context, if correctly structured, the PSA can provide additional evidences of this independence by representing the concatenation between the failures of the different layers of provisions, and providing specific insights about plausible dependent failures, also accounting for external (natural or manmade) hazards.

Practical elimination of events and sequences

Following the IAEA NSSR 2/1 Rev.1 [3] and WENRA [13]: *The possibility of certain conditions arising, whose consequences would be large or early release, may be considered to have been “practically eliminated” if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise*³⁶. For these conditions the safety demonstration is focused on the prevention.

For those which need a demonstration *with a high level of confidence*, the strategy can be based on the analysis of the safety architecture.

³⁵ One can raise the question of the acceptability of an architecture in which a given provision would be used for different initiating events and / or at different levels of the DiD, i.e. the provision is part of LOPs allocated to different levels of the DiD, depending on the requesting initiating event. This should be possible and allowed if the events which require the provision under consideration are completely independent. Similarly, for a given initiating event, the presence of a provision within two different levels of the DiD (n and n+1) could be acceptable if one can demonstrate that the failure of the first of the two levels is not related to the failure of the provision and if this failure do not affect the performances of the latter within the context of the mission allocated to the next level of the DiD. Having said that, it is obvious that all these verifications are possible if a clear and univocal representation of the safety architecture is available.

³⁶ Following WENRA [13], the “*high degree of confidence*” is translated into the following indications: “*The degree of substantiation provided for a practical elimination demonstration should take account of the assessed frequency of the situation to be eliminated and of the degree of confidence in the assessed. Appropriate sensitivity studies should be included to confirm that sufficient margin to cliff edge effects exist. ...*”. The role of probabilistic criteria is established by a specific recommendation: “*Practical elimination of an accident sequence cannot be claimed solely based on compliance with a general cut-off probabilistic value. ...*”. Conversely, the limitation of the role of probabilistic analysis is clearly stressed by this recommendation: “*The most stringent requirements regarding the demonstration of practical elimination should apply in the case of an event/phenomenon which has the potential to lead directly to a severe accident, i.e. to pass from DiD level 1 to level 4. For example demonstration of practical elimination of a heterogeneous boron dilution fault would require a detailed substantiation....*”.

Contributions of both deterministic (physical efficiency over the time) and probabilistic (reliability performances over the time) studies will support the fulfillment of this recommendation: “*It must be ensured that the practical elimination provisions remain in place and valid throughout the plant lifetime.*”

Some conditions can correspond to the failure of the 1st level of the defence (e.g. PWR vessel rupture; SFR core support collapse) and the demonstration will be essentially based on Quality Assurance for the design, the fabrication, the implementation and the operation (including the maintenance).

Other conditions are the results of uncontrolled sequences, with the successive failure of the DiD levels and eventually that of the 4th level (e.g. PWR pressurized core melting; long term loss of the decay heat removal). For these conditions, the corresponding layers of provisions implemented in order to prevent, manage and mitigate the sequence's consequences (including those of the Hardened Safety Core, shall be sized in order to provide the demonstration, *with a high level of confidence*, that their number and quality will be sufficient to avoid the loss of the whole set of DiD levels and, in particular, that of the 4th level of DiD³⁷.

With the same logic, specific provisions are supposed to ensure adequate safety against extreme natural events, (i.e. whose magnitude exceed those considered for design), including adequate margins against cliff edge effects. These provisions are conventionally associated with the DiD level 3b for those that address multiple failures and contribute to the prevention of severe accident situations, and to the DiD level 4 for those devoted to the management of severe accident conditions (cf. Table 2.1).

For the assessment of the adequacy of all these provisions (physical performances and reliability), deterministic criteria and metrics have to be defined specifically for:

- events that, following the failure of the 1st level of the DiD (e.g. PWR pressure vessel rupture) could lead to prompt reactor core / fuel damage and consequent early containment failure, mainly translating Quality Assurance (QA) objectives, to guarantee the highly hypothetical character of the event; probabilistic studies will be implemented to check the effectiveness of the QA implementation and to bring the proof of this very low frequency of occurrence;
- the triggering of unallowable phenomena (e.g. the large gas bubble through an SFR core with positive void coefficient), mainly through the knowledge and the mastering of uncertainty (e.g. with a PIRT analysis, see Appendix 2);
- the elimination of by-pass sequences, by the systematic review of all the containment penetrations and the definition of criteria/constraints on design, operation, maintenance and accident intervention procedures;
- the physical performances and the reliability of the different layers of provisions in order to guarantee the highly hypothetical character of the failure of all the LOP.

A systematic effort should be done in order to minimize the number of situations and sequences to be practically eliminated; in other words, the design basis, including the design extension conditions should be as comprehensive as reasonable feasible.

³⁷ "(4) The purpose of the fourth level of defence is to mitigate the consequences of accidents that result from failure of the third level of Defence-in-Depth. This is achieved by preventing the progression of such accident and mitigating the consequences of a severe accident. The safety objective in the case of a severe accident is that only protective actions that are limited in terms of lengths of times and areas of application would be necessary and that offsite contamination would be avoided or minimized. Event sequences that would lead to an early radioactive release or a large radioactive release are required to be practically eliminated." [3]

In compliance with IAEA NSSR 2/1 Rev.1 [3], the acceptance criteria associated to the success of the 4th level shall be defined to comply with the objective of generating the need for *limited protective measures in area and time*.

Coherently with the logic described above, and for all the considered cases, the probability to lose the provisions which perform safety functions, and whose failure can cause to intolerable large or early releases in the environment, should be significantly lower than 10^{-7} /reactor year³⁸. So far, this "cut off value" cannot be used alone to justify the practical elimination. §bis

Demonstration of design robustness against cliff edge effects

Again, following WENRA [13], *"The degree of substantiation provided for a practical elimination demonstration should take account of the assessed frequency of the situation to be eliminated and of the degree of confidence in the assessed. Appropriate sensitivity studies should be included to confirm that sufficient margin to cliff edge effects exist. ..."*. This recommendation introduces another key notion which must be taken into account: *"cliff edge effects"* with the *"sufficient margin"* (adequate margins with the IAEA NSSR 2/1 Rev.1 [3] terminology)³⁹.

The design against cliff edge effects and the need of adequate margins generate deterministic and probabilistic criteria.

For DEC conditions, deterministic criteria about the performance of provisions should allow facing, without abrupt transition, possible small variations of the plant parameters. Probabilistic criteria are defined, in terms of reliability targets, for the physical performances required for the provisions which are an integral component of the 4th level of the DiD, performances which allow guaranteeing "adequate margins".

Specifically about *natural hazards*, the requirement 5.21a in the NSSR 2/1 Rev.1 [3] states that *"The design of the plant shall provide for an adequate margin to protect items ultimately necessary to prevent large or early radioactive releases in the event of levels of natural hazards exceeding those to be considered for design taking into account the site hazard evaluation"*⁴⁰.

This requirement affects the design and the margins that have to be guaranteed by the features implemented to mitigate severe accidents. It has the purpose to ensure that, should a severe accident occurs due to an external hazard, there would be appropriate assurances that sufficient mitigation means would remain available. The

³⁸ It is the overall probability of the sequence which, after the appearance of the initiator, continues with the loss - in cascade - of provisions (i.e. materializing the different LOPs at different levels of DiD, for the different safety functions) implemented to prevent, to control and manage and, to minimize the consequences of the initiating event.

³⁹ As stressed by the IAEA TECDOC [31], adopting margins in the design of a NPP is a common practice to improve the robustness of the design and providing an effective mean to deal with uncertainties even if, on one side, the extension of the design basis with the introduction of DEC has introduced new elements that need to be addressed and, on the other side, the Fukushima Daiichi accident has reinforced the importance of the effects of external events. Generic insights on *Safety margins for design basis accidents* and *Safety margins for design extension conditions* are provided by the IAEA TECDOC [31]. The reference concludes in particular that there could be a substantial difference between the safety margins for design extension conditions without significant fuel degradation and those for design extension condition with core melt, essentially due to the larger uncertainties which are associated with these conditions.

⁴⁰ Following WENRA [13]: *"Rare and severe external hazards are additional to the general design basis, and represent more challenging or less frequent events. This is a similar situation to that between Design Basis Conditions (DBC) and Design Extension Conditions (DEC); they need to be considered in the design but the analysis could be realistic rather than conservative."*

design of the corresponding provisions, which realize the 4th level of the DiD⁴¹, should be particularly robust and to include margins to withstand loads and conditions generated by the events exceeding those derived from the site evaluation and considered in the design basis. This implies that cliff edge effects should not occur not only for small variations in a plant parameter but also for significant variations of the loads and environmental conditions.

Characteristics required to the Safety architecture

Qualitative safety objectives have been introduced, discussed and motivated within the section 2.3. It is appropriate to provide indications about criteria that can be associated to these objectives and to clarify the role of PSA in the assessment of what is advocated in terms of exhaustiveness, progressiveness of the plant response, tolerance to possible alterations of plant conditions, forgiving reaction to abnormal conditions, and balanced contribution versus the whole risk. Insights are provided hereafter, complementing the information provided in Table 1-1 and within the section 2.3.

Exhaustiveness character

The Exhaustiveness character of the safety architecture is addressed deterministically. PSA, if appropriately developed, takes care that the comprehensive set of postulated initiating events is introduced into the model as risk contributor.

Progressiveness character

Among the strong motivations of the DiD concept there is the objective to have a progressive degradation of the safety architecture before that harmful effects could be caused to people and/or to the environment. Progressiveness of the safety architecture for a specific initiating event is first addressed deterministically by checking the presence of all the needed DiD layers of provisions.

PSA can complement the demonstration of the progressive character of the safety architecture, identifying the potential for “short” sequences and guaranteeing their practical elimination. Moreover, PSA - if appropriately developed - contributes to provide evidence of the implemented progressive defence, with the probabilistic assessment of the different intermediate states of the plants which correspond to the failures of the subsequent DiD levels and the corresponding conditional frequency of occurrence (see Section 6).

Tolerant character

The criteria to assess the Tolerant character of the safety architecture could be established mainly in terms of allowable ranges of operation around the normal operating conditions. Probabilistic studies can contribute to the verification of the Tolerant character of the safety architecture mainly by questioning the margins allowable for the correct behaviour of material (e.g. engineered systems) and immaterial (e.g. inherent characteristics, procedures) provisions, and by addressing uncertainty on input data and its propagation through the model.

⁴¹ A specificity of external hazards is the possibility that subsequent level of Defence-in-Depth (e.g. 4th level) may be impaired before the previous one (e.g. 3rd level); this motivates the requirement and allows covering the possibility that external hazards may challenge levels of DiD without regard to their order.

Forgiving character

PSA, if appropriately developed, can contribute to the demonstration of the Forgiving character of the safety architecture through the probabilistic representation of the chronology and the kinetic of the plausible degradations of the safety architecture. Specific chronology criteria (i.e. requested grace delay) refer to the priority in the operation of different means required to achieve safe conditions (inherent characteristics of the plant, passive systems or systems operating continuously in the necessary state, systems that need to be brought into operation, procedures). Kinetics allows assessing the availability of sufficient grace period for their implementation.

Balanced character

PSA is essential for the assessment of the balanced character of the safety architecture since it allows evaluating the contribution to the whole risk of each specific events and sequences.

Classification of provisions: material and immaterial

One of the principal activities within a risk-informed regulatory process is the ranking of structures, systems and components (SSCs; material provisions) with respect to their contributions to the (primary) risk measure. This can be done by secondary risk measures, computed through Importance and Sensitivity analysis⁴².

Importance measures allow identifying the more important risk contributors, while Sensitivity indices allow identifying the more significant uncertainties which affect the risk. Different importance measures are traditionally used to rank SSCs⁴³. They can be classified in risk-significance (if related to the role that the SSC plays in the measures of risk) and safety-significance (if related to the role that the SSC plays in the prevention of the occurrence of an undesired end state) importance measures⁴⁴. Analogous approach should be developed for non-material provisions (inherent characteristics, procedures).

⁴² Importance and Sensitivity analyses aim at quantifying the contribution of the input variables to the model output (Importance analysis) and to the related uncertainty (Sensitivity analysis).

⁴³ The aforementioned “traditional” importance measures are “local” ones. (i.e. they deal with a point value of the model output and input variables (basic events or parameters) and cannot be used for finite changes of the input variables or, in this case, they do not include the contributions of non-linear terms. Moreover, they are not “additive”. Further approaches are recently proposed for “Global” Importance and sensitivity analysis (i.e. focused on uncertainty on the model output with reference to the entire range of values of the input variables). See for details the Deliverable D30.5 “Risk Metrics and Measures for an Extended PSA” [25].

⁴⁴ They include, for instance, the (safety-significant) “Risk Achievement Worth” (RAW) and the (risk-significant) “Fussell-Vesely” measure (FV). RAW measures the “worth” of the component in achieving the risk level, by considering the maximum increase achievable when the component is always failed. FV is the probability (at a given time) that at least one “minimal cut set” that contains the component is failed (i.e. all components in the minimal cut set are failed), given that the system is failed (at that time).

3. THE IDENTIFICATION OF PLAUSIBLE LOADS AND ENVIRONMENTAL CONDITIONS

The identification and recognition of all plausible⁴⁵ normal and off-normal loads and environmental conditions that can affect the behavior of the installation is correlated with the identification of all plausible events which can strike the plant; this identification is the result of a detailed analysis of the system complemented, as needed, by the consideration of the experience feedback and the site characteristics.

Once the identification is complete, a step of grouping the events in a limited number of families, characterized by similar causes and responses, is performed. For each family, event(s) which may reasonably be considered - in terms of consequences - as envelope of others, are retained as Postulated Initiating Event (PIE) and are used to select and design the facility's provisions, i.e. the safety architecture.

The process is obviously iterative since the implementation of provisions for preventing, managing and / or mitigating abnormal situations can, itself, generate potential accident situations and / or introduce additional hazards whose consequences are not necessarily mitigated by measures already considered⁴⁶.

The list of PIE and the corresponding set of loads and environmental conditions shall be complemented by the consideration of internal and external hazards as well as the consideration of Design Extension Conditions (DEC, former Beyond Design basis Events).

Within the context of the holistic approach described in this document, the Internal or External hazards and DEC's should be considered as "specific environmental boundary conditions" for the multidimensional safety architecture implemented by the plant (see §1.1 and Appendix 1). All these conditions are affected by uncertainties on the definition of the corresponding "loadings" (in a general sense). In a context where the "proportionate approach" (or graduated approach) is preconized by the regulators, the amplitude of these uncertainties justifies different approaches (conservative, best estimate, bottom up (i.e. mechanistic analysis of the sequences), or top - down (symptom based or the analogous "approche par états"). The possibility to have dependent failures (common cause or propagating failures) shall also be considered. All these conditions are defined as Design Basis.

Finally, events or sequences that can lead to large or early releases shall be practically eliminated⁴⁷.

In this complex context, the list of available provisions (material and immaterial) can be defined unambiguously; their performances (physical and reliability) will, in turn, be function of "loadings". Having this list, the first key objective should be an adequate representation (multidimensional as needed) of the safety architecture, addressing all the plausible conditions / loadings generated by the Design Basis conditions and events and support - as required - the demonstration for the practical elimination.

⁴⁵ "Plausible" is all what it is not "physically impossible".

⁴⁶ For example, in standard PWRs the presence of boron in the primary circuit is a provision to help the "Control of chain reactions". It must appear from the provisions listed in the OPT PWR, for example within the first level of DiD. But the implementation of this provision introduces the risk of boron dilution and an initiator raise: the "plug of clear water" directly generated by the provision. Additional provisions are expected to address this specific risk (at the second and third level DiD).

⁴⁷ Cf. WENRA [13]: "... the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA SSR 2.1)."

The methodology for selecting initiating events and hazards for consideration in an Extended PSA, and specifically the screening approach used to select the External Hazards to be analyzed is addressed in a dedicated ASAMPSA_E deliverable (D30.3, [25]).

Figure 3-1 and Figure 3-2 [17] [31] resumes the evolution of the key elements for the design basis since the requirements of the years 2000 until those, updated, as introduced by IAEA [3] or WENRA [13]. Both references clearly place the conditions generated by internal and external hazards among the design basis⁴⁸.

NS-R-1, 2000

Operational states		Accident conditions	
NO	AOO	(a) DBAs	Beyond design basis accidents
		(b) Severe Accidents	
Included in the design basis		Beyond design basis	

SSR-2/1, 2012

Operational states		Accident conditions		Cond. practically eliminated
NO	AOO	DBAs	Beyond design basis accidents	
		Design Extension Conditions		
		Without CD	Severe Accidents	
Included in the design basis		Beyond design basis		

Design Basis ≠ Design Basis Accidents

Beyond Design Basis ≠ Beyond Design Basis Accidents

Figure 3-1 Plants states categorization following the IAEA requirements in 2000 and 2012

Design basis				Beyond design basis
Operational states		Accident conditions		Conditions practically eliminated
NO	AOO	DBAs	Design Extension Conditions	
		No core melt	Severe Accidents (core melt)	No cliff-edge effects
Conditions generated by External & Internal Hazards				
Criteria for the necessary capability, reliability and availability (for each plant state)				
Design basis of equipment for Operational states	Design Basis of Safety Systems including those SSCs necessary to control DBAs and some AOOs	Design Basis of safety features for DEC's including those SSCs necessary to control DEC's		No plant equipment is designed for these conditions
		Design Basis of the containment systems		

Figure 3-2 Plants states categorization following the updated IAEA requirements in 2016

⁴⁸ E.g. this is consistent with the WENRA indication: "For new reactors external hazards should be considered as an integral part of the design and the level of detail and analysis provided should be proportionate to the contribution to the overall risk."

The initiating events, once identified, are categorized following their estimated frequency of occurrence. This categorization allows assigning the PIE, in a conventional manner, to the various categories: Anticipated Operational Occurrences, Design Basis Events and Design Extension Conditions. For each category, quantitative safety objectives are usually suggested by the designer and endorsed by the regulators; this allows defining the space of acceptable risk (see Section 2.2.1). According to Figure 3-3, which is composed by contributions from WENRA [13] & NUREG 2150 [10], this categorization can be related with correspondent levels of DiD and their positioning within the risk space (see Section 2.2.1).

All PIEs are characterized by thermal, hydraulic & mechanical loads and specific environmental conditions. As indicated above these loads and the environmental conditions shall be taken into account to select and size the provisions to be implemented within the safety architecture.

It is worth noting that the considerations developed in the present document for the assessment of the whole “safety architecture”, i.e. the assessment of successive layers of provisions with their physical efficiency and reliability, should be considered to complement the process described by the SSG-30 Safety Guide [6] for the classification of System, Structures and Components⁴⁹.

⁴⁹ The objective of the SSG-30 [6] is “to provide recommendations and guidance on how to meet the requirements established in the NSSR 2/1 Rev.1 [3] and IAEA GSR Part 4 [4] for the identification of SSC important to safety and for their classification on the basis of their function and safety significance”. The classification process recommended by the SSG-30 is “consistent with the concept of Defence-in-Depth set out in the IAEA NSSR 2/1 Rev.1. *“All items important to safety shall be identified and shall be classified on the basis of their function and their safety significance.”* The functions to be addressed are “primarily those that are credited in the safety analysis and should include functions performed at all five levels of DiD”.

According to the SSG-30, the method for classifying the safety significance of items important to safety shall be based primarily on deterministic methods complemented, where appropriate, by probabilistic methods, with due account taken for: (i) the safety function(s) to be performed by the item; (ii) the consequences of failure to perform a safety function; (iii) the frequency with which the item will be called upon to perform a safety function; (iv) the time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function. The reliability required to the SSCs in order to meet the applicable safety objectives by the implemented safety architecture is a further key issue to be considered.

Levels of defence in depth	Objective	Essential means	Radiological consequences	Associated plant condition categories
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)	Normal operation
Level 2	Control of abnormal operation and failures	Control and limiting systems and other surveillance features		Anticipated operational occurrences
Level 3 ⁽¹⁾	3.a Control of accident to limit radiological releases and prevent escalation to core melt conditions ⁽²⁾	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact ⁽⁴⁾	Postulated single initiating events
	3.b	Additional safety features ⁽³⁾ , accident procedures		Postulated multiple failure events
Level 4	Control of accidents with core melt to limit off-site releases	Complementary safety features ⁽³⁾ to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time	Postulated core melt accidents (short and long term)
Level 5	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off site radiological impact necessitating protective measures ⁽⁵⁾	-

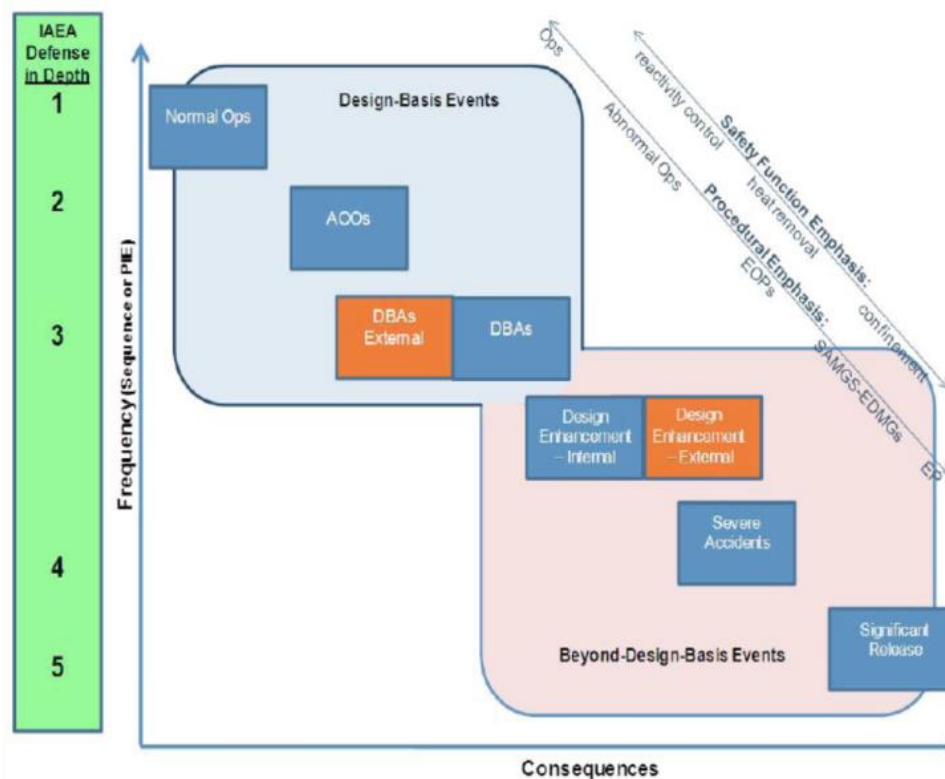


Figure 3-3 Categorization of the PIE and correspondence with the level of the DiD

4. SAFETY ARCHITECTURE REPRESENTATION

Around the “reactor process”, whose design and performances are defined to fulfil the basic requirements (power level, ranges of operating temperatures, efficiency, potential for fissile creation, potential for waste management, etc.), the safety related architecture is build up to insure the operability, the availability, and the safety of the system. As already indicated, the safety architecture is a complex multidimensional set of material (active and / or passive systems and components) and immaterial provisions (intrinsic characteristics, procedures) (cf. §1.1 and Appendix 1).

The system's response to each abnormal event is specific in the sense that it is achieved by organizing, manually or automatically, a subset of provisions to manage the corresponding mechanisms which challenge the safety function(s) and, finally, to meet the safety objectives. For a given abnormal event, in the logic of Defense in Depth, this subset corresponds to the notion of “layer of provisions” (LOP); if the implementation of the defense is correctly done, the safety architecture should address any possible deficiencies/failures, partial or total, of an LOP through the intervention of another “layer of provisions”, functionally redundant, allowing to ensure the achievement of the required mission.

Defense in Depth is so organized into successive levels whose role is defined conventionally (see Figure 3-3): Prevention of any abnormal plant condition (level 1), Detection of all abnormal situations and Control of Anticipated Operational Occurrences (AOO) (level 2), Protection against accidental situations, limitation of their consequences and, more generally, prevention of severe accidents situations (level 3a and 3b), Management of severe accident situations and mitigation of their consequences (level 4).

For each plausible condition (i.e. the Design Basis, cf. Fig. 3.2) the representation of the safety architecture shall allow identifying the content of each of these levels; their assessment shall be specific to the condition under examination. The overall evaluation of the system is the integral of these singular evaluations; this approach allows identifying any weaknesses among the range of the safety architecture responses.

The Objective Provision Tree (OPT) methodology and the concept of Line Of Protection (LOP), both described hereafter (Sections 4.1 & 4.2) implement this logic for the representation of the safety architecture while remaining fully consistent with the safety assessment process presented by the IAEA GSR Part 4 Rev1 [4].

4.1. THE OBJECTIVE PROVISIONS TREE

The Objective-Provisions Tree (OPT) methodology is suggested by the Generation IV Risk and Safety Working Group (GIF/RSWG), as part of the Integrated Safety Assessment Methodology (see Appendix 2).

OPT - see IAEA TECDOC 1366 [8] and IAEA Safety report 46 [9] - allow a standardized representation of the safety architecture by identifying, for each initiating event and for each safety function the different levels of Defense in Depth and the corresponding “layers of provision”. The references [22], [23], and [24]⁵⁰ show the results of recent activities on the OPT methodology and its possible use.

⁵⁰ OPT implemented by the Japan Nuclear Safety Institute (JANSI) to survey and evaluate the severe accident measures after the Fukushima accident.

The logic of the OPT lies on the systematic identification, for a given level of DiD and for given “safety functions” (SF), of the plausible “challenges” to this SF; for each of these challenges, the methodology identifies the corresponding relevant “mechanisms and phenomena” to be prevented or controlled by a set of “provisions” which are designed and implemented to meet specific acceptance criteria⁵¹ and to maintain or to bring the plant to controlled or safe states, meeting the safety objectives.

Conventionally, for a given DiD level, the objectives corresponding to a given safety function are translated into physical parameters or “decoupling criteria”⁵² that reflect the allowable consequences associated with the DiD level under consideration.

So, for each safety function, representative parameters can be identified with associated values/figures that reflect compliance with safety objectives.

For the safety function under consideration, the partial or total failure of an LOP means the failure of the DiD level. This failure leads to complementary conditions in terms, for example, of specific boundary conditions (e.g. temperature, pressure, humidity) that have, in turn, to be considered for the design of the successive LOPs.

⁵¹ The approach adopted by OPT is fully consistent with the IAEA GSR Part 4: “4.46. *The necessary layers of protection, including physical barriers to confine radioactive material at specific locations, and the necessary supporting administrative controls for achieving Defence-in-Depth have to be identified in the safety assessment. This includes identification of: (a) Safety functions that must be fulfilled; (b) Potential challenges to these safety functions; (c) Mechanisms that give rise to these challenges, and the necessary responses to them; (d) Provisions made to prevent these mechanisms from occurring; (e) Provisions made to identify or monitor deterioration caused by these mechanisms, if practicable; (f) Provisions for mitigating the consequences if the safety functions fail.*” [4]

⁵² *Decoupling Criteria*

The decoupling criteria are deterministic and are used to assess the physical performances of the architecture. Decoupling criteria and the corresponding metrics are physical parameters (e.g. number of clad failures) which make the link between the safety objectives, which are formulated in quite generic manner (e.g. health consequences ⇒ corresponding releases), and quantitative and measurable objectives or acceptance criteria (e.g. maximum clad temperature) which are usable by the designer to check the acceptability of the design. Moreover, through the assessment process, they allow defining measurable safety margins.

As a matter of example, the following decoupling criteria are conventionally used in the safety analysis of LWR: The Departure of Nucleate Boiling ratio (DNBR; >1) to guarantee the avoidance of the fuel clad failure; the fraction of fuel rods experiencing DNB during accident conditions (e.g. < 10%); Specific decoupling criteria are defined for the LOCA conditions, they address : the peak cladding temperature (e.g. 1204°C), the maximum percentage of oxidized cladding thickness (e.g. <17%); the maximum hydrogen generation amount (e.g. <1%); the core geometry that shall remain coolable; the fact that long term core cooling shall be ensured, etc.

Analogous definitions can be found within the IAEA terminology. The terms decoupling criteria is consistent with the notion of “acceptance criteria” defined, as follow, within the IAEA glossary [1]: “*Specified bounds on the value of a “functional indicator” or “condition indicator” used to assess the ability of a structure, system or component to perform its design function.*” The term is nevertheless more generic, including also the notion of “performance indicator” where, according to the IAEA glossary [1]:

- “*a Condition indicator is a characteristic of a structure, system or component that can be observed, measured or trended to infer or directly indicate the current and future ability of the structure, system or component to function within acceptance criteria.*”
- “*a Functional indicator is a condition indicator that is a direct indication of the current ability of a structure, system or component to function within acceptance criteria.*”;
- “*a Performance indicator is a characteristic of a process that can be observed, measured or trended to infer or directly indicate the current and future performance of the process, with particular emphasis on satisfactory performance for safety.*”

Figure 4-1 ([15] and [16]) shows the iterative process for the implementation of the safety architecture by the identification of the contents of all levels of DiD (left side of the figure) and the standard structure of the OPT (right side of the figure).

The OPT can be used to check that:

- all the initiators are adequately addressed;
- all levels of DiD are properly structured and organized (i.e. the necessary provisions are in place and are sufficient) to achieve the required missions;
- the mutual independence of the levels of DiD is guaranteed.

According to this last point, coherently with the principles of the DiD, the provisions associated with each level must be independent and, if possible, diversified from those allocated to the other levels of DiD⁵³.

The benefits from the implementation of OPT are even stronger when it is considered within the whole context of interaction with the other ISAM tools and, specifically, with the PSA for which the OPT represents an essential input in terms of presentation of the whole safety architecture which, once available, is analytically described and assessed by the PSA.

⁵³As already indicated, one can raise the question of the acceptability of an architecture in which a given provision would be used for different initiating events and / or at different levels of the DiD, i.e. the provision is part of LOPs allocated to different levels of the DiD, depending on the requesting initiating event. This should be possible and allowed if the events which require the provision under consideration are completely independent (cf. also foot note N° 35).

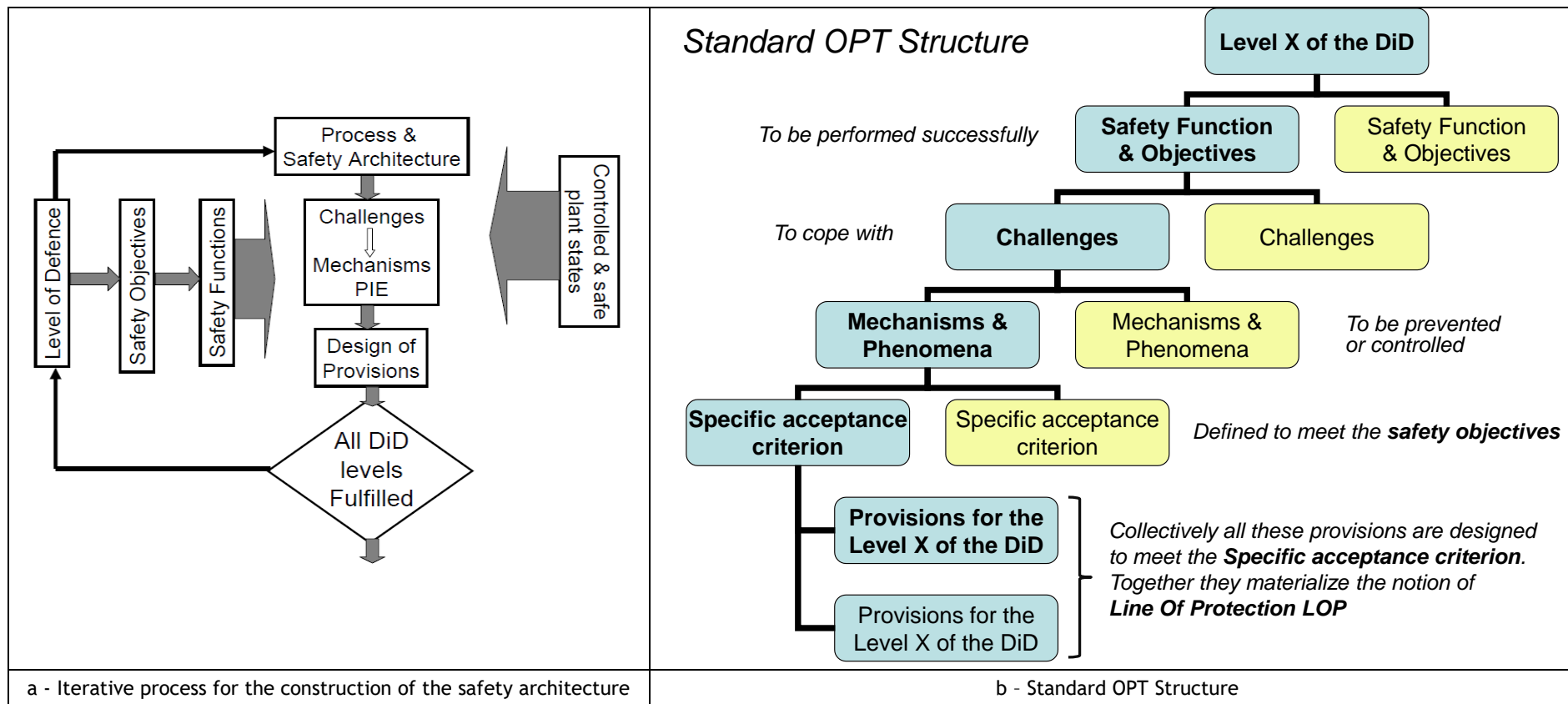


Figure 4-1 Process for the implementation of the whole safety architecture and Standard OPT structure

4.2. THE LINE OF PROTECTION METHODOLOGY

Once the probabilistic targets are defined, before the formal assessment through the PSA, it is possible to roughly sketch the safety architecture's characteristics with the Line of Defence (LOD) or Line of Protection (LOP) methodology.

The methodology, which was originally proposed as LOD ([19] and [20]) with, in particular, the notions of strong (noted « a ») and medium (noted « b ») lines of defense⁵⁴, evolved within the context of IAEA activities and GIF/ RSWG ([15] and [16]), to better consider the nature of the safety related provisions implemented within the innovative concepts (e.g. the passive systems, inherent characteristics, procedures, etc.). While the LOD method focused on engineered safety systems with a rough correspondence with the events categories⁵⁵ and without specific relationship with the Defence-in-Depth, the concept of LOP embraces the notion of "layers of provisions" as defined by the IAEA Safety Fundamentals, integrating all the possible contributions to achieve safety, i.e. material and immaterial provisions, with a direct correspondence with the DiD level, allowing a more comprehensive representation of the safety architecture⁵⁶.

The notion of LOP is an integral component of the Objective provisions tree (see Fig. 4.1 right side).

The LOP methodology can be used as a guide for the rough verification of compliance with the quantitative probabilistic objectives. The goal is to verify the implementation, for each plausible situation and for all possible pathways toward the severe accident, of a succession of LOPs with adequate reliability. The counting of these lines ensures that the probabilistic targets are met, and ensures that the overall risk associated with the sequence "initial plant condition + initiating event + possible failure of LOP" is acceptable.

The LOP methodology is based on the adoption of the following rule: ... Given a plant condition resulting from an initiating event applied to a given initial state of the installation. ... Given a safety function, whose control is requested by the initiating event under examination, the failure⁵⁷ of which will lead to potential consequences larger than that allowed in the category of the plant condition... In this situation there is a potential for a release higher than that allowable and therefore for an intolerable risk. For each plant condition taken into account for the design, and for each safety function that meets the above criteria, the designer shall identify the number and quality of LOP to be implemented in order to meet the objectives of the function, i.e. to ensure that the overall risk associated with the sequence "initial plant condition + possible failure of LOP" remain acceptable.

⁵⁴ The strong line of defence (a) corresponds to a LOD designed to meet high reliability performances (e.g. safety classified system, designed considering the single failure criterion). Its probability of failure can vary within a range of 10^{-3} to 10^{-4} per year or per demand.

The medium line of defence (b), does not meet the same design or implementation requirements (e.g.: less safety margins than the LOD "a") and it shows a lower reliability (e.g.: operator actions, etc.). Its probability of failure can vary within a range of 10^{-1} to 10^{-2} per year or per demand.

⁵⁵ It is worth noting that the revised structure for the DiD, as suggested by WENRA (cf. Table 2.1), creates this correspondence between the categories and the levels of the DiD.

⁵⁶ A recent application of LOD/LOP notion is done for the ASTRID project [21].

⁵⁷ Failure at the solicitation or at short or long term

The Appendix 3 and §5 provide insights about the performances required to the DiD levels, discussing the DiD principle, the notion of Line of Protection as well as the deterministic and probabilistic success criteria.

Fig. 4.2 shows schematically the logic of the methodology, as defined in the '90ies, including the line requested for the management of the severe accident conditions and the rejection of the failure of the 4th level of the DiD into the Residual Risk (left side). The figure also provides the representation of the different area covered by the different levels of DiD (right side).

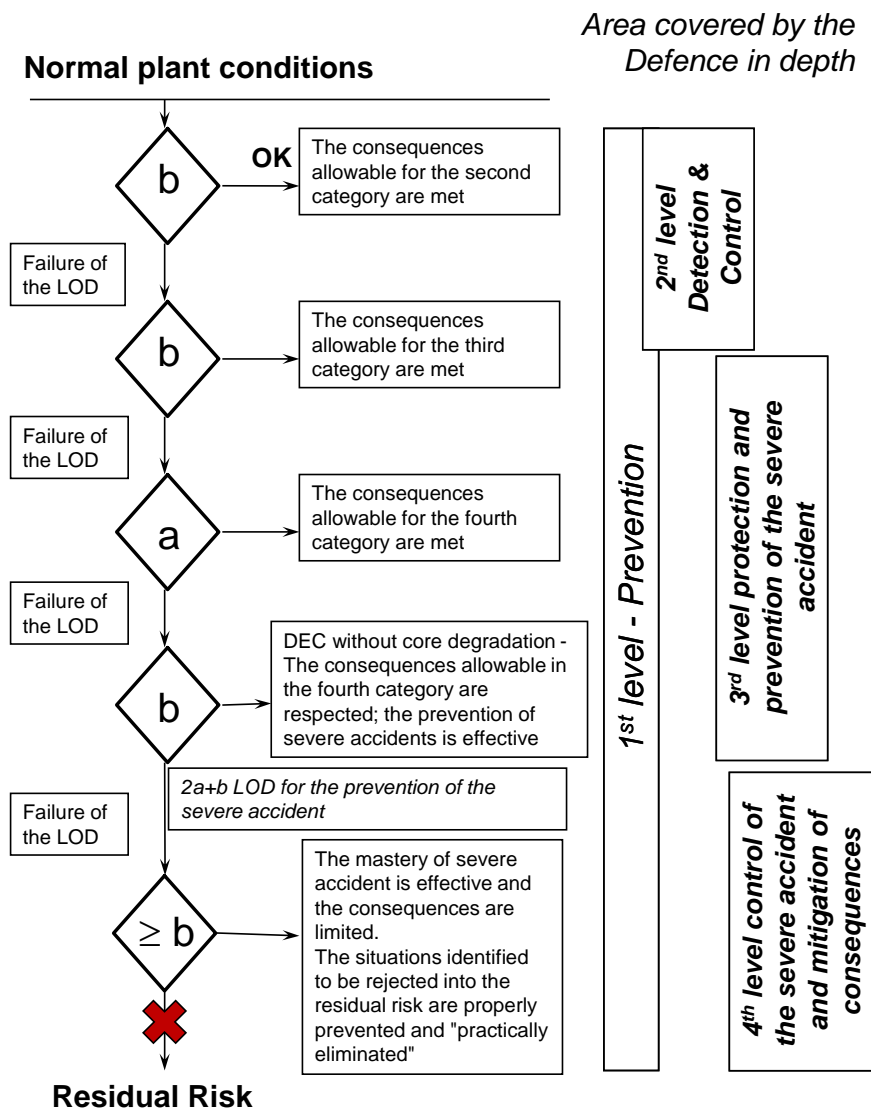


Figure 4-2 Principles for the Lines of Defense methodology

Note that the one strong Lines of Defense (a) plus three medium lines (3b) is assumed to be the equivalent to two strong lines (2a) and one medium line (b) (i.e. $a=2b$).

Fig. 4.3 shows the same logic for the concatenation of the different levels of the DiD, as suggested recently by WENRA [14] including the very last indications for the integration of the external hazards (i.e. the integration of the post Fukushima studies). The WENRA scheme is completed with indication of the different areas covered by the different levels of DiD, according to Fig. 3.3 (right side).

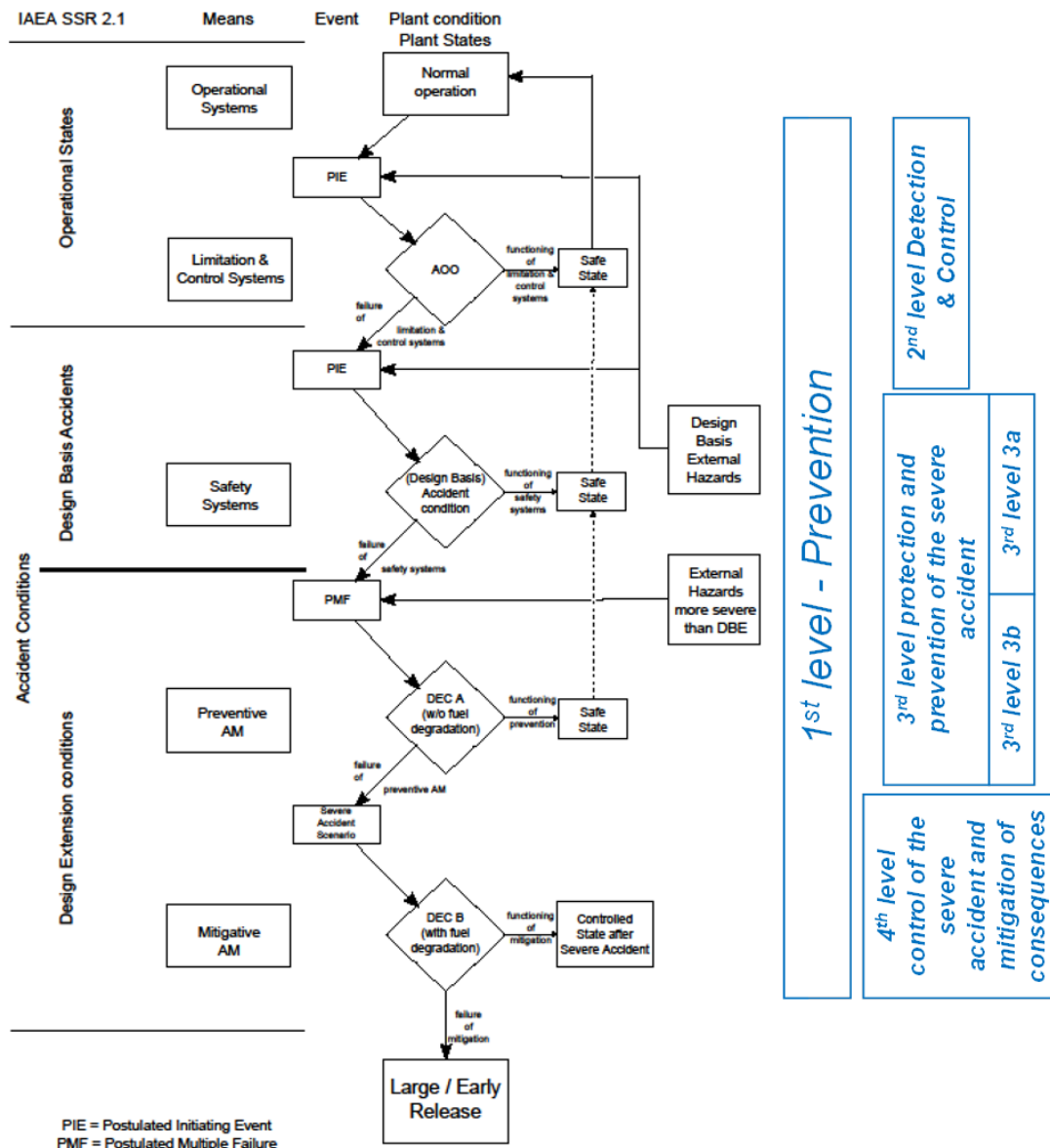


Figure 4-3 WENRA: Scheme of means, events and plant conditions⁵⁸

(The right side of the figure is added by the authors)

The content of each LOP figured by the boxes “means” within the figure, i.e. the provisions which compose the LOP, must be able to ensure the achievement of the requested mission in terms of needed physical efficiency and reliability.

As for the LODs, the LOPs can be classified in “Strong” $\left(\frac{(10^{-3} \div 10^{-4})}{\text{year}} \text{ or } \frac{(10^{-3} \div 10^{-4})}{\text{demand}}\right)$ and “Medium” $\left(\frac{(10^{-1} \div 10^{-2})}{\text{year}} \text{ or } \frac{(10^{-1} \div 10^{-2})}{\text{demand}}\right)$, with reference to different reliability targets.

⁵⁸ The depiction proposed by the WENRA scheme complies with the principles mentioned but, following the authors, the simplified representation of the chronology between the levels 3a and 3b (in series) seems not sufficient in so far it should be considered the possibility of having the two lines 3a and 3b in parallel, being designed to operate against different (single and multiple) postulated events.

With regard to probabilistic targets, the value "fraction of 10^{-7} / reactor year per family of initiators and per function" for the prevention of severe accidents (cf. Section 2.2.2) corresponds to the implementation, for each sequences starting with a PIE classified as AOO and which, potentially, can lead to a severe accident configuration, to the equivalent of two strong lines plus a medium line (i.e. $2a+b$ LOPs with, if necessary, $b+a$).

In case of failure of the prevention (level 1) and of the control of incidental / accidental conditions (levels 2 & 3), the severe accident management and the protection against intolerable releases to the environment, require the implementation of ad-hoc provisions for the management of the degraded situations and the mitigation of corresponding consequences (i.e. the layer of provisions which materialize the 4th level of the DiD). Before the Fukushima accident, the reliability allocated to this complementary line was in the range of 10^{-1} - 10^{-2} per demand (equivalent to a medium line, i.e. the order of magnitude reasonably allocated to the containment, cf. Section 2.2.2). After Fukushima, consistently with the new requirements, there is the need to practically eliminate large or early releases and, in this context, to implement specific provisions to achieve the requested missions both for the prevention and for the management of severe accident conditions even in case of *natural hazards exceeding those to be considered for design*. One can reasonably consider that the requirements in terms of reliability for the 4th level of the DiD is more stringent and ambitious than before and that the needed reliability of this additional line (Hardened Safety Core), while not being not defined precisely, should be rather in between "b" and "a" (b/a) or even equivalent to that of a strong line (i.e. "b" \Rightarrow "a").

In these conditions the practical elimination of a given sequence whose potential consequences are the large or early releases should corresponds to the failure of more than 3 strong LOPs ($>3a$)

Example of qualitative characteristics to distinguish Medium and Strong lines is provided in Table 4-1.

Table 4-1 Qualitative characteristics of Medium and Strong LOPs

Qualitative characteristics	Medium lines	Strong lines
Simplicity	desirable	desirable/recommended
Diversity	desirable	recommended/required
Independence	required	required
Redundancy	desirable/required	required
Fail-safe/Fail-tolerant	required	required
Single Failure Criterion	recommended	required
Testability	recommended	required
Om-service-inspectability	recommended/required	required
Human corrective actions	permitted	not permitted

Table 4-2 provides a proposal for the positioning of Lines Protection (LOP) as indicated by the modified WENRA/RHWG table ([15], [16]). It presents the architecture of LOPs to be interposed between the plant condition whose potential consequences are greater than those allowed for its category, and these consequences, depending on their level of gravity.

Table 4-2 Proposal for the positioning of LOPs, modified WENRA/RHWG table [15], [16]

Level of DiD		Associated plant condition categories			
Level 1		Normal operation			
Level 2		Anticipated operational occurrences		b	
Level 3	3a	DiD Level 3.a Postulated single initiating events 3 rd Category		b	
		DiD Level 3.a Postulated single initiating events 4 th Category		a	
	3b	DiD Level 3.b Selected multiples failures events including possible failure or inefficiency of safety systems involved in DiD level 3.a Consideration of "natural hazards exceeding those to be considered for design"		b	b/a
Total LODs to be implemented for the prevention of severe accidents				≥ 2a+b	
Level 4	4	DiD Level 4. Postulated core melt accidents (short and long term) Consideration of "natural hazards exceeding those to be considered for design"		b/a	b/a
Minimum number of LODs to be implemented to reject into the residual risk or to practically eliminate				> 3a	

In a similar way Table 4-3 provides an example of the complete representation of the safety architecture for the initiating events of 3rd and 4th categories and for events with multiple failures, in terms of positioning of LOPs between the plant condition under investigation and the resulting situation whose limits shall be met.

Table 4-3 Proposal for the positioning of LOPs, complete safety architecture

Category of the initial plant condition \Rightarrow « Level of the DiD » -Category of the resulting plant condition \Rightarrow	2	3	4	Selection of multiple failures events	Hazards exceeding those of design
« 2 » - Cat II	b				
« 3a » - cat III	b	b			
« 3a » - cat IV	a	a	a		
« 3b » - Multiple failure events « 3b » - Hazards exceeding those to be considered for design	b	b	b	b	
	b/a	b/a	b/a		b/a
Total number of LOD to be implemented for the prevention of severe accidents	$\geq 2a+b$	$\geq 2a$	$\geq a+b$	b	$\geq b$
Level 4 - Provisions for the management of degraded situations and mitigation of consequences: Severe Accidents - Provisions for the management of natural hazards exceeding those to be considered for design	b/a	b/a	b/a	b/a	
	b/a	b/a	b/a		b/a
Total LOD effort to practically eliminate situations whose consequences are unacceptable	$> 3a$	$> 2a+b$	$> a+2b$	$> 2b$	$> 2b$

4.3. OPT, LOP AND FMEA

The section discusses specifically the relationship between the Objective Provisions Tree (OPT), Line of Protection (LOP) and Failure Mode and effects analysis (FMEA).

In analyzing the potential consequences of a LOP / provision failure, all plausible failure modes are considered. It is at this stage that an FMEA like analysis is possible and - at the discretion of the designer - will complete the OPT approach in order to:

- Identify weaknesses in the system and make remedies;
- Identify ways to prevent certain failures;
- Study in detail the consequences of failures of the various provisions;
- Classify failures with selected criteria;
- Provide an optimization of the maintenance plan and insights to support the development of test plans;
- Optimize tests to check the proper operation of the installation;
- Motivate decisions for design's revisions;
- Etc.

The FMEA can provide an important validation step in the identification of the initiators but it applies to an already globally defined architecture. From this point of view, it is complementary to the OPT. In particular the FMEA can check - through the analysis of "Criticality" ($\text{Criticality} = \text{Severity of failure} \times \text{Frequency of occurrence} \times \text{Detectability of the failure}$) - that the possible interactions between provisions are properly taken into account in terms of their real importance, and the possible consequences of failures are optimized in terms of risk / criticality.

As illustrated above, the essential difference between the OPT and the FMEA is that the former helps to build an architecture according to the principles of defense in depth, while FMEA can provide, in addition, an interesting indication for the optimization of the architecture in terms of "criticality", as defined above.

This optimization phase is a step that can be considered as part of the iterative process in the implementation of the OPT and, as such, FMEA, or any other method of risk analysis (e.g. HAZOP - Hazard and Operability analysis), can certainly help during the phase of detailed engineering for new installations.

5. THE EVALUATION OF PERFORMANCE OF DID LEVELS

As already indicated, the acceptability of a safety architecture remains based on the degree of meeting the DiD principles. The deterministic and probabilistic considerations, including success criteria, shall therefore be integrated into a comprehensive implementation of Defence-in-Depth. Such success criteria are essential to design adequately the provisions implementing the levels of the DiD and refer to the required physical efficiency and reliability. The final goal of this process is the optimization of the whole safety related architecture in terms of performances and reliability.

Therefore, the objective pursued with the evaluation of the effectiveness of each level of Defense in Depth is twofold: firstly checking, for a given initiating event, that physical efficiency of the material and immaterial provisions, that are located on that level, allows achieving the task as required and, secondly, that this can be done with a reliability that is consistent with the expectations/needs.

The deterministic assessment is done with conventional rules that are specific to each category of PIE. So for Anticipated Operational Occurrences (AOO) and Design Basis Accidents (DBAs) analysis of physical performance is done with a conservative approach while for Design Extension Conditions (DEC) it is a “best estimate” approach that is adopted.

The equipment reliability is a key issue which integrates insights from probabilistic studies within the deterministic approach for safety assessment. The latter remains the basis for the construction and analysis of the safety architecture, in particular with the background of the Defense in Depth (DiD), but the contribution of probabilistic assessment to accompany the construction or to endorse the final structure is essential.

There are a number of deterministic design requirements and practices aimed at ensuring the required reliability of material and immaterial provisions, including physical separation, independence, fail safe design, redundancy, diversity, safety margins, conservative design, and single failure criterion (NSSR 2/1 Rev.1 [3]). Probabilistic studies contribute to the verification of the fulfilment of some of the above requirements, e.g. by questioning the effectiveness of redundancy or diversity among material provisions or by modelling human factor for immaterial provisions. Even more, PSA - if appropriately developed - contributes by modelling probabilistically the plausible degradations of the implemented safety architecture, due to the failure of one or more safety functions; PSA supports the selection of adequate design options, or the verification of the adequateness of implemented solutions, against two key objectives: efficiency (i.e. the capability to correctly achieve the requested mission) and reliability (i.e. to achieve the mission with the due reliability⁵⁹). The final goal of this process is the optimization of the whole safety related architecture in terms of performances, reliability and costs. Figure 5-1 summarizes the logic [16].

⁵⁹ The reliability of inherent or passive systems / provisions is a matter of extensive work for its assessment. The distinction between active and passive could be no longer justified. What seems essential to select and implement a given technical option, are the physical efficiency (i.e. the capability to achieve the requested mission) and the reliability that can be guaranteed to correctly achieve this mission. It is perfectly true that passive systems could have a higher reliability but it is also true that for several of them large uncertainties characterize their physical efficiency. Finally what seems essential, to motivate the selection of a given option, is the capability to provide an adequate demonstration of both the physical efficiency and the reliability.

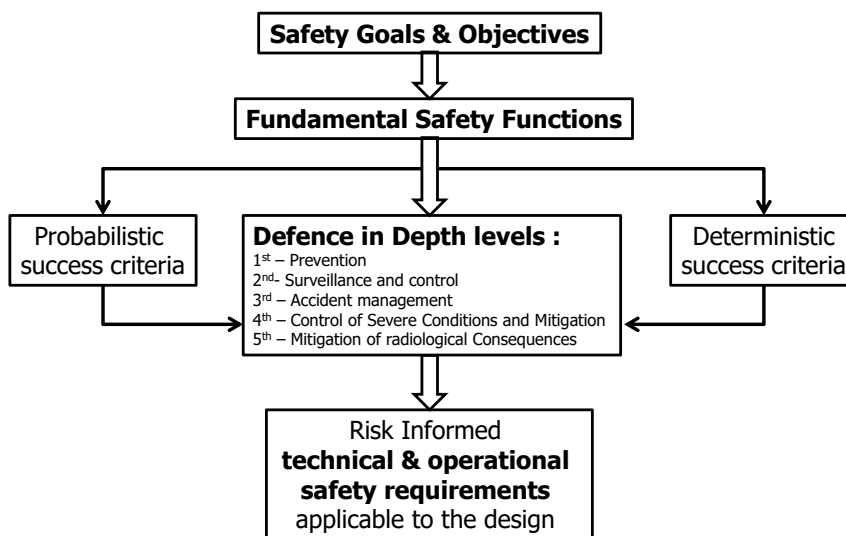


Figure 5-1 Defence-in-Depth and Risk-Informed Safety Philosophy [7]

The concept of risk space introduced in the Section 2 can be used to examine or define the deterministic or probabilistic success criteria, in terms of required performances and reliability of safety functions.

About this topic, it is interesting to mention Ref. [31] that indicates some overall perspectives about how Defence-in-Depth can be characterized, performing a historical review of Defence-in-Depth. Among the findings one can outline the one that indicates: *“There is almost no guidance on criteria for determining adequacy of Defence-in-Depth . The literature does suggest that the elements (e.g., layer of defense) should be quantified, that risk can be used to assess each defense system (e.g., safety measure), that compensatory measures can be graded in order to reduce risk, that any sequence (given all defense layers have failed) remain under a frequency consequence curve, that redundancy and diversity is sufficient to ensure risk guidelines are met, and that the adequacy of Defence-in-Depth can be assessed via a process that uses measures of risk.”*

The idea of using risk to assess the DiD is directly related to the indications provided for example by the Fig. 3.3 above; the objective to have all the consequences of the plausible sequences below the F-C curve defines the allowable risk to be guaranteed by the performances of the safety architecture.

The overall intent is consistent with that illustrated schematically in Figure 5-2. It shows that, for a given initiating event whose consequences are potentially unacceptable, design provisions are implemented⁶⁰ (cf. also Appendix 3):

- to keep or make the consequences acceptable with regard to the likelihood of the initiating event they are requested to control; this allows defining the success criteria in terms of requested physical efficiency that allow maintaining or bringing back the installation into the acceptable area (Control - Mitigation: deterministic success criteria);

⁶⁰ For initiating events whose consequences are very low there is no need for mitigation measures; the implementation of provisions to limit the consequences is not necessary.

- to decrease the likelihood of the accidental sequence; this allows defining the success criteria in term of reliability of the layer of provisions required to ensure that, in case of failure, the sequence “PIE + layers of provisions’ failure”, is within the acceptable area (Prevention: probabilistic success criteria).

N.B. The figure presents only two extreme configurations: layer of provisions’ success / failure. Obviously intermediate cases - partial success / failure - have to be considered in an analogous manner.

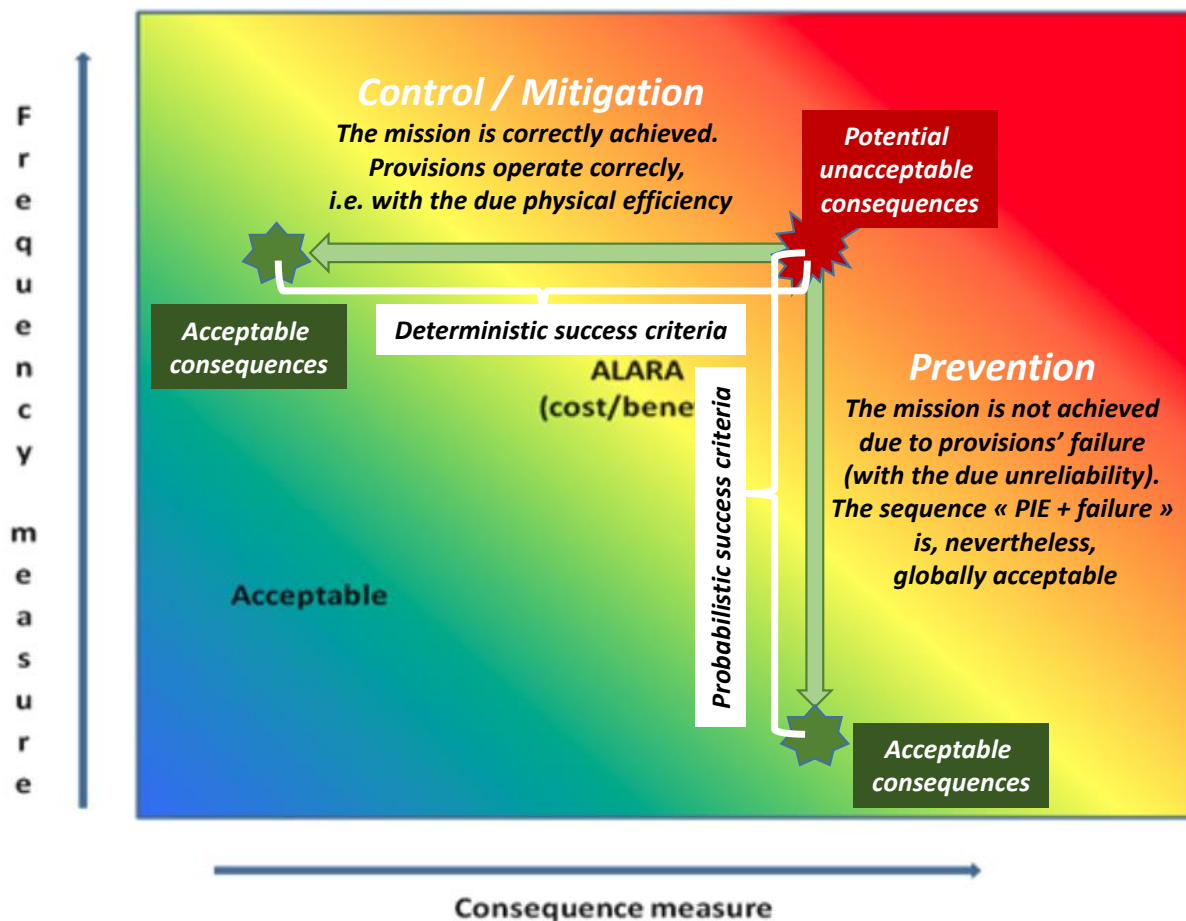


Figure 5-2 Risk space and deterministic / probabilistic success criteria

As a matter of example, guidelines 5.47 - 5.52 stated in the IAEA SSG-3 [5] allow defining the physical performances - in terms of success criteria to be defined deterministically within the risk space - for the singular provisions; they are fully applicable to the suggested approach:

“5.47. The success criterion for each safety system should be defined as the minimum level of performance required to achieve the safety function, taking into account the specific features of each sequence ...;

5.48. The safety systems that would fail as a result of the initiating event should be identified and taken into account in specifying the success criteria;

5.49. *The success criteria should specify the mission times for the safety systems, that is, the time that the safety systems will need to operate so that the reactor reaches a safe, stable shutdown state and that will allow for long term measures to be put in place to maintain this state;*

5.50. *The success criteria should also specify requirements for support systems, based on the success criteria for the front line systems, which are performing safety functions directly;*

5.51. *The success criteria should define the operator actions required to bring the plant to a safe, stable shutdown state as defined by the plant procedures ...;*

5.52. *The Level 1 PSA documentation should include a list of the safety functions, safety systems, support systems and operator actions that are required for each initiating event to bring the reactor to a safe, stable shutdown state.”*

6. THE PROBABILISTIC ASSESSMENT OF THE SAFETY ARCHITECTURE AND DID

The Level 1 PSA relies on event trees drawn to determine how, following a given initiating event, the accident sequences progress until the severe accident condition (i.e. a fuel damage state, cf. ASAMPSA_E D30.5 [25]). In order to enable a comprehensive evaluation of the safety architecture, the PSA has to consider all the initiating events, all the safety functions, and all the levels of the DiD.

The availability of an exhaustive - as practicable - representation of the safety architecture (see Section 4) allows the development of a PSA model with a structure that better complies with the DiD principles, based on Event Trees (ET) built to reflect the crossing of different levels of DiD and, for each level of DiD, on Fault Trees (FT) built to assess the reliability of the implemented layers of provisions⁶¹.

This PSA re-structuring is recommended, but not an unquestionable need (i.e. the whole process for DiD assessment is not invalidated). Theoretically, different PSA models can embed the same information through different event tree-fault tree structures, and provide all the information required for the DiD assessment. (being possible to recognize for each given initiator the subsequent layers of provisions that can fail, leading to the loss or degradation of safety function(s)).

The PSA's event trees can be built / re-structured directly starting from a representation of the safety architecture through the Objective Provision Trees. Each OPT is specific of a given level of the DiD, of a given safety function and of a given initiating event. For a given PIE, the PSA's event tree allows modelling the failure of LOPs addressing their concatenation, interactions (e.g. the amplitude and the kinetics of the reactivity control will affect the amount of heat to be removed) and plausible dependent failures (including common cause failures and propagating failures)⁶².

Figure 6-1 provides the standard structure of the Event Tree for a given PIE which demands for (all) DiD levels intervention. The sequence "hazard + failure of the DiD level 1" materializes the initiating event.

⁶¹ Each node of the ET represents the failure/success of the whole set of provisions (i.e. the layers of provisions, i.e. the Line of protection) which materialize the corresponding DiD level, with the respective conditional failure probability. The latter is assessed by a FT which includes all the provisions required to be operational in order to achieve successfully the requested mission: engineered safety systems and all support system components, passive systems and components (e.g. undetected filter blockages, pipe leaks, etc.) as well as procedures and operator interventions.

⁶² On its side, the specificity of the OPT approach is to identify, for a given initiating event and a given safety function, and for each level of DiD, the corresponding LOP with all its provisions. Obviously for different initiating events, but for the same safety function and / or the same DiD level, LOPs are built specifically and not necessarily with exactly the same provisions. Moreover, the provisions which appear at a given level of DiD for an initiating event and a safety function may intervene at another DiD level for another initiating event. Under these conditions (i.e. the multidimensional character of the safety architecture, cf. Appendix 1), concerning the degree of detail for the PSA input data, it is not interesting to introduce directly the failure of single provisions within the ET (this would certainly be very tedious due to the enormous quantity of possible combinations) but to model the failure of the whole LOP within the ET and the failures of its provisions through a dedicated FT.

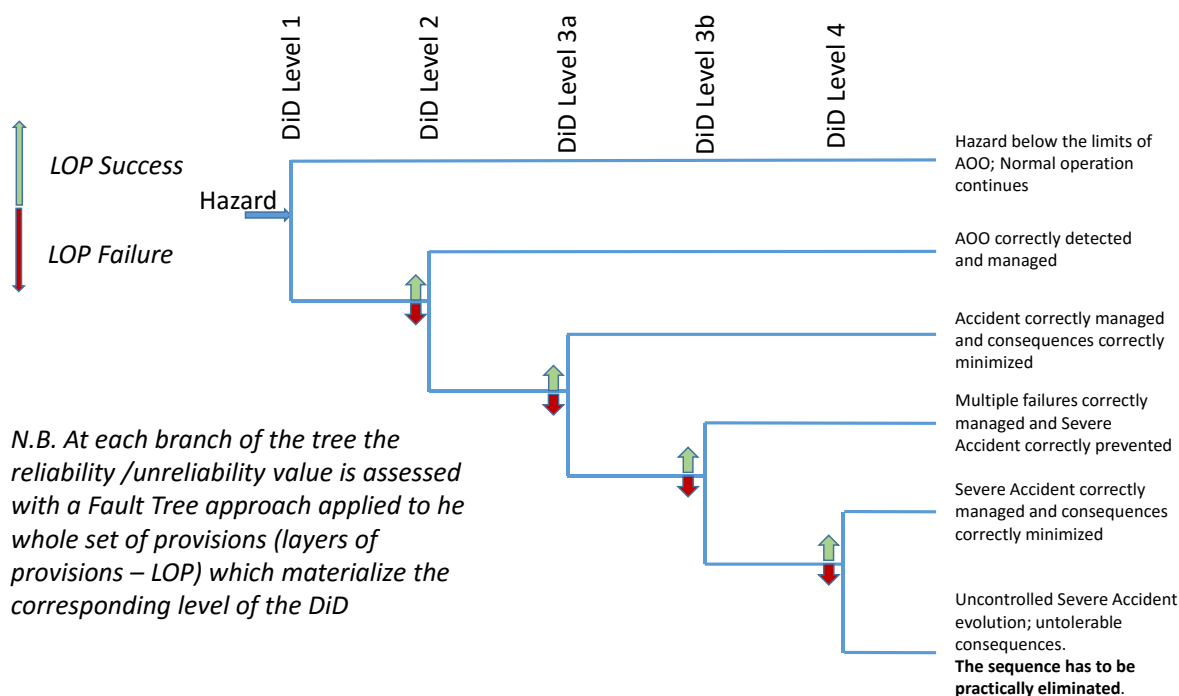


Figure 6-1 Example of Event Tree organized following the structure of the DiD

Figure 6-2 integrates into the standard Event Tree structure the indications about the practical elimination of (“short”) sequences by-passing the intermediate levels (2nd and/or 3rd) or leading to unacceptable consequences (in case of failure of the 4th level of DiD). It also shows the possible by-pass of the 3a level of DiD (following the WENRA definition) in case of multiple failures events.

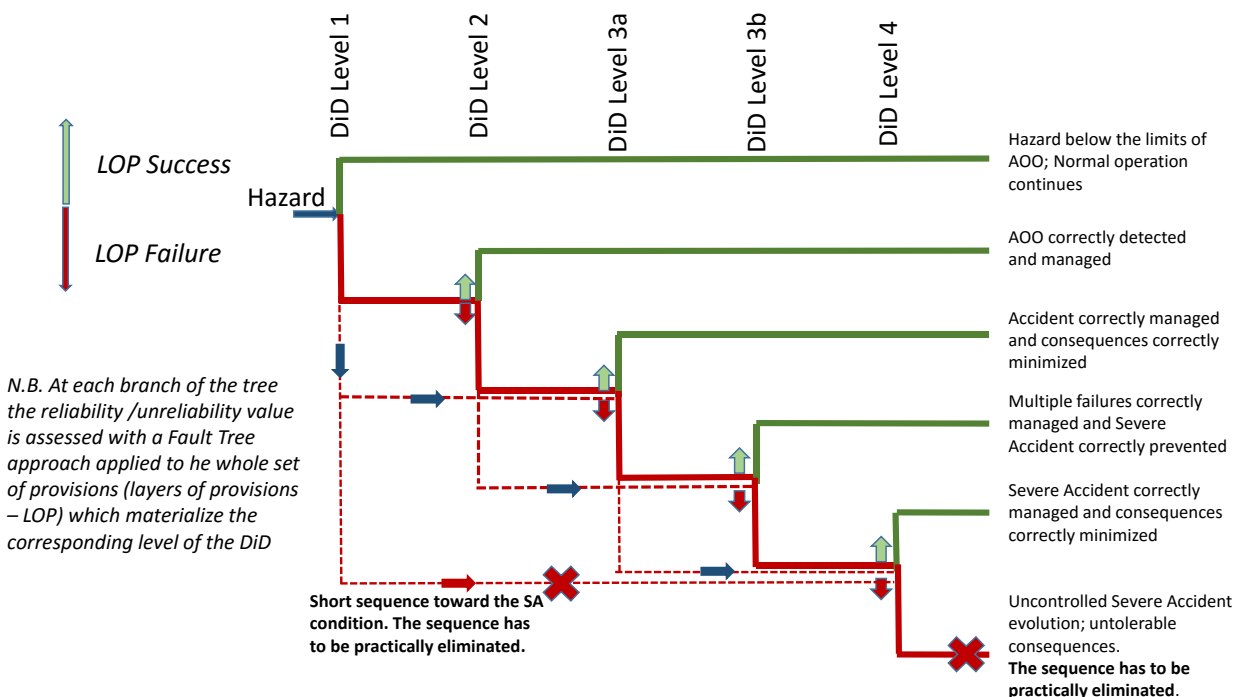


Figure 6-2 Updated example of Event Tree organized following the structure of the DiD

N.B.: Discussing the practical elimination it is worth noting that WENRA [13] explicitly indicates that *“In each level of DiD, some situations need to be practically eliminated as it cannot be demonstrated that, should they occur, their radiological consequences would be tolerable. Situations that could lead to early or large releases of radioactive materials have to be practically eliminated.”*

The guidelines specified in the SSG-3 [5] remains applicable to the provisions when aggregated within the LOP: *“5.83. Functional descriptions should be produced for each of the safety systems modelled in the Level 1 PSA to ensure that there is a valid and auditable basis for the logic model being developed. Functional descriptions typically include the following: a) The function of the system; b) The system failure modes; c) The system boundaries; d) The interfaces with other systems; e) The mode of operation being modelled (for systems with more than one mode); f) The components that need to operate or change their state and their normal configuration; g) Whether the component operations are manual or automatic; h) The conditions that must exist for automatic signals to be received by the components.”*

A partial practical example developed starting from the OPT of the IAEA TECDOC 1366 [8] is presented within the Appendix 4.

N.B. The trees as presented within the TECDOC 1366 are quite old (2001-2002) and should be updated, in particular to take into account the new requirements for the assessment. This step has not been done for the current exercise.

Because previous extensive applications of the proposed approach are not available, it is recommend a specific application exercise. From the perspective of the authors, this aims opening lines of thought to motivate further development.

Figure 6-3 summarizes the whole process for the assessment of the DiD with the support of the PSA, providing details with respect to Figure 1-2 about the significant issues for each one of the four main steps.

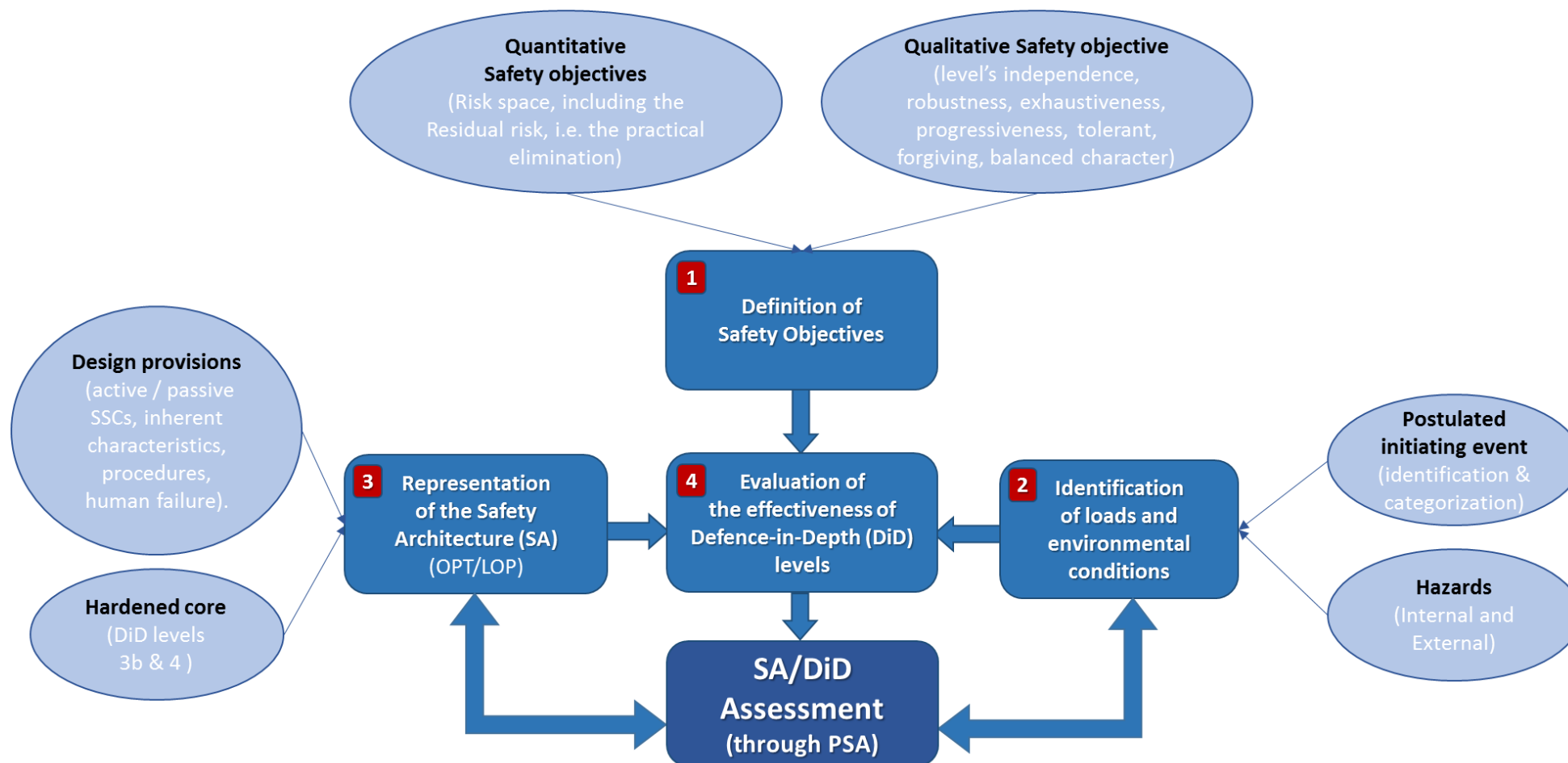


Figure 6-3 *Steps for the PSA assessment of DiD and details*

7. CONSIDERATIONS ABOUT EXISTING REACTORS AND PSA

For existing plants, the improvement of the safety architecture is a “design activity” and, as such, the indications of the NSSR 2/1 Rev.1 [3] shall be implemented as far as reasonably practicable. Specifically, NSSR 2/1 Rev.1 (1.3), coherently with WENRA [13], states: *“For the safety analysis of such designs, it is expected that a comparison will be made with the current standards, for example as part of the periodic safety review for the plant, to determine whether the safe operation of the plant could be further enhanced by means of reasonably practicable safety improvements”*. This also applies to the improvement of a methodology which should integrate all new knowledge and experience available. The evolution of the PSA approach, as described within the previous sections, should consider this objective while exploiting the available achievements.

The implementation of an updated methodology on an “old safety architecture” can obviously identify inconsistencies and determine whether the safe operation of the plant could be further enhanced by means of reasonably practicable safety improvements. The objective of the designer / analyst should be to try to correct - as far as feasible - these inconsistencies and propose / implement the improvements⁶³.

One key question can be raised to address the possible role and the needs of extension of the existing PSAs: what can be learnt from an appropriate / innovative representation of the safety architecture (e.g. through the OPT) about the adequacy of an existing PSA (not structured as described in section 6).

Two main answers/challenges concern the implemented safety architecture: Identification of missed DiD level(s) for particular initiators (short sequences), even if probabilistically non-significant (e.g. Fukushima); general insights for the improvement of the safety architecture.

The current PSA approach provides integral results, e.g. versus a given detrimental event (e.g. severe core damage - PSA level 1) and, as such, even if these results comply with the probabilistic success criteria (e.g. the CDF), the first of the above challenges cannot be fully addressed; in other terms: the achievement of a given figure for the CDF does not necessarily mean that all the DiD levels are correctly designed and implemented. One can imagine that the levels effectively implemented, even if insufficient in number, are sufficiently strong to guarantee the CDF objective.

Another plausible explanation could be the unjustified accentuation of a very low frequency of occurrence to the initiating event (cf. Fukushima). It is a possible deficiency of current PSA which should also be found in the probabilistic assessment of the safety architecture and DiD. But in the case of Fukushima, if any PSA have been carried out properly, the internal flooding initiator event analysis would give some important knowledge about electrical generator deficiencies in that situation.

⁶³ This sort of exercise has been done by JANSI with the support of the OPT to identify and correct weaknesses within the safety architecture of current Japanese plants. Only the synthesis of this exercise is available [24] and no details were found within the open literature.

In these conditions, the appropriate representation of the safety architecture (i.e. consistent with the DiD principles) allows identifying the possible inconsistency between the quantitative figures/results (e.g. the CDF) which can be acceptable (from numerical point of view) and the way followed to implement safety; the latter can be questionable from the viewpoint of the DiD principles (e.g. lack of DiD levels, insufficient independence between the DiD levels, etc.) which would not be, or not sufficiently, fulfilled.

The Objective Provisions Tree (OPT) and the related concept of Line Protection (LOP), already considered as well suited tool to support the design and assessment of future Gen IV systems, seems appropriate also for the re- assessment and extension of PSA of existing NPP. Specifically, OPT could support the identification of the initiators to be considered for PSA extension and the systematic representation of the implemented safety architecture, typically not easy to be understood from the PSA, even if conceptually modelled. The OPT and LOP allow identifying possible deficiencies in the DiD implementation, i.e. the lack of specific “layers of provisions” clearly allocated/correlated to specific DiD levels, and/or inconsistencies for these levels, e.g. lack of independence between the DiD levels. Deficiencies or inconsistencies identified, despite the acceptable results of PSA, rise the questions “how the safety is implemented?” and more generally “is the safety demonstration robust enough?”, and prove the weakness of the PSA approach applied alone.

The key objectives of the DiD logic is the coverage of “unforeseen” plant conditions, i.e. the possible lack of exhaustiveness in the identification of events or sequences. The PSA alone is not sufficient to address this eventuality, because it can only address sequences “expected” (although extremely unlikely). This is why, even if the PSA results are acceptable from “success criteria” point of view, it is also essential to comply with the DiD principles. Moreover, DiD principles can support the identification of missed events in the PSA (failure of provisions in the safety architecture) and the improvement of probabilistic model(s).

Concerning the identification of the initiating events, the contribution of the PIRT/OPT/LOP (see Appendix 2) can likely help to guarantee the exhaustiveness but nothing proves that this approach is more effective than the conventional methods as suggested, for example by the IAEA SSG-3⁶⁴.

⁶⁴ According to IAEA SSG-3 [5]: “5.13. A systematic process should be used to identify the set of initiating events to be addressed in the Level 1 PSA. This should involve a number of different approaches including:

- Analytical methods such as hazard and operability studies or failure mode and effects analysis or other relevant methods for all safety systems to determine whether their failures, either partial or complete, could lead to an initiating event;
- Deductive analyses such as master logic diagrams to determine the elementary failures or combinations of elementary failures that would challenge normal operation and lead to an initiating event;
- Comparison with the lists of initiating events developed for the Level 1 PSAs for similar plants and with existing safety standards and guidelines;
- Identification of initiating events on the basis of the analysis of operating experience from the plant under investigation and from similar plants;
- Review of the deterministic design basis accident analysis and beyond Design basis accident analysis and the safety analysis report”

Concerning the improvement of the probabilistic models, the insights produced by the proposed process for the assessment of the safety architecture can support:

- the systematic verification of the completeness and correctness of the representation made by the PSA model of the degraded states of the safety architecture, due to the failure of the layers of provisions materializing the levels of DiD; this can be done by checking that the information enclosed in the OPTs (PIE and failure conditions, with respect to the mechanisms challenging the safety function at the different DiD levels - and to the implemented line of protection) are integrated within the existing PSA;
- the re-structuring of the PSA model based on Event Trees (ET) reflecting the different levels of DiD and Fault Trees (FT) assessing the reliability of the implemented layers of provisions.

8. CONCLUSIONS

The definition and the implementation of the safety architecture of a nuclear installation must be coherent with the principles of the Defense in Depth (DiD). Moreover, for the upgrading for modernization of the NPP facilities / safety system in operation, as well as for the design of innovative plants, it is essential to remain consistent with the applicable Safety Fundamental [2] and Safety Requirements [3] and to fully integrate the recent recommendations related to the lessons learnt by the Fukushima Daiichi event.

Deterministic and probabilistic approaches are recognized to be complementary elements for the safety assessment of nuclear installations, including both the verification of the compliance with the applicable Safety Fundamentals and Requirements as well as the safety analysis, i.e. the meeting of safety objectives. Unquestionably, the assessment of the DiD, i.e. the verification of the compliance of the implemented safety architecture with the DiD principles can be supported by PSA. For this objective, a certain number of practices should evolve. This document presents and motivates the ways to be followed in order to achieve these evolutions.

Concerning the definition of the objectives to be considered for the safety assessment, the reference to the risk space is essential to integrate the insights and results coming from the deterministic and probabilistic studies and to evaluate the effectiveness of the levels of DiD in terms of 1) physical efficiency to keep the consequences of the event under examination allowable, and 2) reliability of the layers of provisions which perform the requested mission.

Moreover, the introduction of qualitative objectives allows complementing the probabilistic targets and expanding the application field of the PSA approaches, including the support to the verification of:

- 1) the achievement of basic design goals (protective measures limited in times and areas, exhaustiveness of the safety assessment),
- 2) the DiD principles (independence of DiD levels, practical elimination of events and sequences, demonstration of design against cliff edge effects), and
- 3) the additional characteristics required for the safety architecture (progressiveness, tolerant, forgiving and balanced characters).

The classification of the Postulated Initiating Event according to their frequency of occurrence, with reference to the plant operational states and their relationships with the DiD levels and with the allowable risk space, are essential for the identification of loads and environmental conditions to be considered in the design and sizing of provisions.

The prerequisite for optimizing the synergies between the deterministic and probabilistic approaches is the representation of the safety architecture that should, as far as possible, reflect the principles of implementations of DiD concept while being assessable by the PSA approach. Specifically, it is the reliability of the layers of provisions (or Lines of protection) that should be assessed by probabilistic studies.

The Objective Provisions Tree (OPT) and the Line Of Protection (LOP) are tools proposed to support the identification of possible deficiencies in terms of DiD level and to provide the essential information for the subsequent development of the PSA.

The definition of safety objectives, both in quantitative and qualitative aspect, the identification and taking into account of all loads and environmental conditions that may affect the operation of the installation and the representation, as comprehensive as practicable, of the safety architecture, allow the evaluation of the performance of each level of Defense in Depth. This is a twofold task: firstly it is verified that, for a given initiating event, the functional efficiency of the technological and other related provisions are capable to realize the mission required and, secondly, if needed, this is done with a reliability that meets the safety criteria.

The availability of an exhaustive - as practicable - representation of the safety architecture allows the development of a PSA model with a structure that better complies with the DiD principles. The standard updated structure of the PSA Event trees, whose nodes shall represent the failure/success of the DiD levels and the corresponding Lines of Protection, and a partial practical example of application are presented.

All the proposals of this document are based on consolidated terminology [1] and shared concepts ([5], [10], [13] and [15]), and consistent with the (IAEA) Safety Fundamentals [2], Safety Requirements [3] and process for the Safety assessment [4]. They contribute to clarify the possible peculiar role of the PSA approach for the assessment of DiD and to support the on-going evolution of the PSA approach through indications on the a conceptual framework and related process for the assessment of the “safety architecture” implementing DiD and on metrics to be adopted. Further activities are requested to finalize the proposed approach; they mainly concern the detailed definition of the above criteria and metrics, coherently with the indications provided within the document.

Remark

The last exchange of questions and remarks between the approver (IRSN) and the authors about the approach proposed for the PSA assessment of Defense in Depth is provided in the following.

IRSN - A PSA study assesses the risk (core melting, fission product release, dose in the plant vicinity...); the “new tool” proposed in the report wants to assess DiD level efficiency.

Authors - The “new tool”, as the previous, will allow assessing the risk while verifying, simultaneously, the compliance of the implemented safety architecture with the DiD principles and, more generally with the whole set of safety requirements. From this point of view, one can consider that the new tool is really an “Extended PSA”.

IRSN - Initiating events are chosen independently of the DiD levels in PSAs (several hundred initiating events); in the new tool, the initiating events are, (we suppose), those of the design accidental scenarios;

Authors - The initiating events are at least those selected for the design of the installation (DBE and DEC). Additional initiating events can be accounted in the PSA depending on the measures implemented in the different levels of Defense in Depth which may, in turn, generate specific incidental situations.

IRSN - For an initiating event, the accidental scenario in usual PSA is built taking into account the available means needed (human and systems) to perform fundamental safety function (reactivity control, core cooling and fission product containment); in the new tool, the available means are limited to the layer of provisions of a DiD level

Authors - The reference “to the layer of provisions of a DiD level” is inclusive rather than restrictive. In fact, the notion of LOP integrates all the safety related features which contribute - as requested - to the achievement of the safety function. By addressing step-by-step all the levels of the DiD allows the whole set of provisions *implemented* for the prevention, control, management and consequences mitigation to be considered for each initiating event.

IRSN - So, these differences could be emphasized and the report could explain that the new tool is a mean to verify the quality of the deterministic analysis.

Authors - The key objective of the tool is, as indicated above, assessing the risk while verifying, simultaneously, the compliance of the implemented safety architecture with the DiD principles.

IRSN - Is it useful?

As recommended, for example by IAEA GSR Part 4, the tool can allow making the whole “safety assessment”, i.e. the compliance with the safety requirements as well as the safety analysis.

IRSN - Is there limitation in the NPP SSCs to be taken into account (then the method may be easier to implement) or shall all NPP SCCs (like in PSA) be taken into account?

Authors - The objective is to achieve the whole comprehensive representation of the safety architecture. The selection of the appropriate “granulometry” is left to the designer / analyst. The notion of provisions includes material and immaterial (e.g. passive systems, inherent characteristics, procedures, ...) means.

IRSN - It seems important to us to keep a complete independence between deterministic analysis (useful for design) and PSA study (useful to assess the risk).

Authors - The above comment is exactly the vision, often supported by both the practitioners of DiD deterministic assessment and PSA, that the authors propose to overcome. This allows extending the use of the Probabilistic safety analysis toward the ultimate goal of the Safety assessment of the nuclear installation. Indeed, to assess the risk without assessing the compliance with the requirements is a partial approach that does not fully answer the need for the “safety assessment” as defined by the IAEA GSR Part 4 Rev1 (“Safety assessments are to be undertaken as a means of evaluating compliance with safety requirements and thereby the application of the fundamental safety principles Safety assessment includes, but is not limited to, the formal safety analysis” [4]).

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Appendix 1 - Insights concerning the concepts of “Defence-in-Depth” and “Safety architecture”

Several definitions of the concept of Defense in Depth (DiD), perfectly consistent among them, are available within a number of reference documents such as the European Council Directive (Ref. A1.2), the fundamentals and requirements of the IAEA (Ref. A1.3; Ref. A1.4), the WENRA's recommendations for current and future reactors (Ref. A1.5; Ref. A1.6) (see below) .

Following the GIF/RSWG (Ref. A1.1) the “Safety Architecture” is defined as being “*the full set of provisions - inherent characteristics, technical options and organizational measures - selected for the design, the construction, the operation including the shut down and the dismantling, which are taken to prevent the accidents or limit their effects.*” The notion shall be considered within the framework of the implementation of the defense in depth.

The notion of “Safety architecture” is consistent with these definitions, and in particular with that of **successive layers of provisions**, functionally redundant *so that in the event that a failure were to occur, it would be detected, compensated or corrected by appropriate measures* (Ref. A1.2).

The objective of the of safety architecture representation is to identify, for each plausible plant condition, i.e. for each initiating event and for each sequence generated by any plausible failures, the provisions that embody the different levels of defense in depth.

The safety architecture is therefore a multi-dimensional representation of the mode of operation of the installation and its response to abnormal situations. Besides the factual identification of all the available provisions, what is sought is their belonging, vis-à-vis the initiating event and the safety function for which they are requested, to the given level of Defense in Depth in which they are required to intervene/operate.

In other words, for a given provision, the dimensions of this “space” are: the initiating event, the sequence of possible failures, the affected safety function and the level of Defense in Depth (DiD) in which the provision is asked to achieve its mission (layer of provision). A comprehensive safety assessment, i.e. both the compliance with the safety requirements and the “safety analysis” (cf. IAEA GSR Part 4) need the modeling of the architecture in this space.

Once the safety architecture representation is available, the objective is the demonstration for each initiating event, of the performance of layers of provisions for the different levels of defense in depth, as a whole, both as regards their physical performance, i.e. the ability to achieve the mission required in case of failure of the previous level, but also the ability to achieve this mission with the required reliability.

This demonstration integrates the analysis of independence between levels, i.e. the ability to realize the mission despite the partial or total failure of the previous level.

Defence-in-Depth: Excerpts from reference documents

- Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations (Ref. A1.2)

*The concept of defence-in-depth is fundamental to the safety of nuclear installations and is the basis for implementing high level nuclear safety objectives. Application of the defence-in-depth principles, as recognised in international standards and guidance and by WENRA, ensures that safety activities are subject to, as far as reasonably practicable, **independent layers of provisions**, so that in the event that a failure were to occur, it would be detected, compensated or corrected by appropriate measures. The effectiveness of each of the different layers is an essential element of defence-in-depth to prevent accidents and mitigate the consequences should they occur. Defence-in-depth is generally structured in five levels. Should one level fail, the subsequent level comes into play. The objective of the first level of protection is the prevention of abnormal operation and system failures. If the first level fails, abnormal operation is controlled or failures are detected by the second level of protection. Should the second level fail, the third level ensures that safety functions are further performed by activating specific safety systems and other safety features. Should the third level fail, the fourth level limits accident progression through accident management, so as to prevent or mitigate severe accident conditions with external releases of radioactive materials. The last objective (the fifth level of protection) is the mitigation of the radiological consequences of significant external releases through the off-site emergency response.*

- IAEA - Safety Fundamentals (Ref. A1.3)

*3.31. The primary means of preventing and mitigating the consequences of accidents is 'Defence-in-Depth'. Defence-in-Depth is implemented primarily through the combination of a number of consecutive and **independent levels of protection** that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. When properly implemented, Defence-in-Depth ensures that no single technical, human or organizational failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability. **The independent effectiveness of the different levels of defence is a necessary element of Defence-in-Depth.***

- IAEA Safety Requirements - NSSR-2/1 (Rev. A1.1) (Ref. A1.4)

*2.12. The primary means of preventing accidents in a nuclear power plant and mitigating the consequences of accidents if they do occur is the application of the concept of Defence-in-Depth. This concept is applied to all safety related activities, whether organizational, behavioural or design related, and whether in full power, low power or various shutdown states. This is to ensure that all safety related activities are subject to **independent layers of provisions** so that if a failure were to occur, it would be detected and compensated for or corrected by appropriate measures. Application of the concept of Defence-in-Depth throughout design and operation provides protection against anticipated operational occurrences and accidents, including those resulting from equipment failure or human induced events within the plant, and against consequences of events that originate outside the plant.*

- WENRA - WENRA Safety Reference Levels for Existing Reactors (2014) (Ref. A1.5)

E2. Safety strategy

E2.1 Defence-in-depth shall be applied in order to prevent, or if prevention fails, to mitigate harmful radioactive releases.

*E2.2 The defence-in-depth concept shall be applied to provide **several levels of defence** including a design that provides a series of physical barriers to prevent uncontrolled releases of radioactive material to the environment, as well as a combination of safety features that contribute to the effectiveness of the barriers. The design shall prevent as far as practicable: challenges to the integrity of the barriers; failure of a barrier when challenged; failure of a barrier as consequence of failure of another barrier.*

- WENRA/RHWG - Safety of new NPP designs (Ref. A1.6)

*The primary means of preventing accidents in a nuclear power plant and mitigating the consequences of accidents is the application of the concept of Defence-in-Depth (DiD). This concept should be applied to all safety related activities, whether organizational, behavioural or design related, and whether in full power, low power or various shutdown states. This is to ensure that all safety related activities are subject to **independent layers of provisions**, so that if a failure were to occur, it would be compensated for or corrected by appropriate measures. Application of the concept of Defence-in-Depth throughout design and operation provides protection against anticipated operational occurrences and accidents, including those resulting from equipment failure or human induced events within the plant, and against consequences of events that originate outside the plant.*

*Therefore, Defence-in-Depth is a key concept of the safety objectives established by WENRA for new nuclear power plants. In particular, these safety objectives call for an extension of the safety demonstration for new plants, in consistence with the reinforcement of the Defence in-Depth approach. Thus **the DiD concept should be strengthened in all its relevant principles**. In addition to the reinforcement of each level of the DiD concept and the improvement of the independence between the levels of DiD (as stated in the WENRA safety objectives), this also means that the principle of multiple and independent barriers should be applied for each significant source of radioactive material. It shall also be ensured that the DiD capabilities intended in the design are reflected in the as-built and as-operated plant and are maintained throughout the plant life time.*

References A1

1. Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems - Revision 1; Report GIF/RSWG/2007/002; November 24, 2008
2. Council Directive 2014/87/EURATOM of 8 July 2014 amending the Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations
3. IAEA Safety Standards Series N° SF-1: Fundamental Safety principles; 2006
4. IAEA Safety Standards Series No. SSR-2/1 (Rev. 1): Safety of Nuclear Power Plants: Design; 2016
5. WENRA Report: Updating WENRA Reference Levels for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons learned; September 2014
6. WENRA RHWG Report: Safety of new NPP designs - Study by Reactor Harmonization Working Group RHWG; March 2013

Appendix 2 - The ISAM methodology

The process for the design and evaluation of a safety architecture

The process for the design and evaluation of a safety architecture follows a logic which begins with the identification of fundamental principles and requirements that are mandatory and imposed by the regulator. Among these principles the compliance with the logic of the Defense in Depth is certainly one of the most important. The strategy of DiD (i.e. the adoption of adequate safety architectures) ensures that the fundamental safety functions are reliably achieved, with sufficient margins, to compensate for equipment failure, human errors and hazards.

In parallel safety goals are established to ensure that public and environmental protection is properly secured. Principles and requirements are accompanied, if appropriate, by guidelines which, while not mandatory, can help selecting the options that will satisfy the principles, the requirements and objectives. Once principles, requirements, guidelines, objectives and safety options have been selected, the full process (iterative as needed) for the design and the assessment of the retained safety architecture (including the safety analysis⁶⁵) can be summarized as in Table 1-1.

The ISAM methodology

The full set of above steps can be achieved with the help of the Integrated Safety Assessment Methodology (ISAM) methodology, developed within the context of the Generation IV Risk and safety Working Group (GIF/RSWG) [15].

The methodology consists of five distinct analytical tools, each of which can be used to answer specific kinds of safety-related questions with different degrees of detail, and at different stages of design maturity.

Although individual analytical tools can be selected for individual and exclusive use, the full value of the integrated methodology is derived from using each tool, in an iterative manner and in combination with the others, throughout the development cycle.

Figure A2-1 details the overall task flow of the ISAM and indicates which tools are intended for use in each phase of Generation IV system technology development.

Each analysis tool is briefly described in the following [15].

⁶⁵ Following the indications of IAEA [4], the safety assessment is “the systematic process that is carried out throughout the design process to ensure that all the relevant safety requirements are met by the proposed (or actual) design of the plant. This would include also the requirements set by the operating organization and the regulators. Safety assessment includes, but is not limited to, the formal safety analysis”.

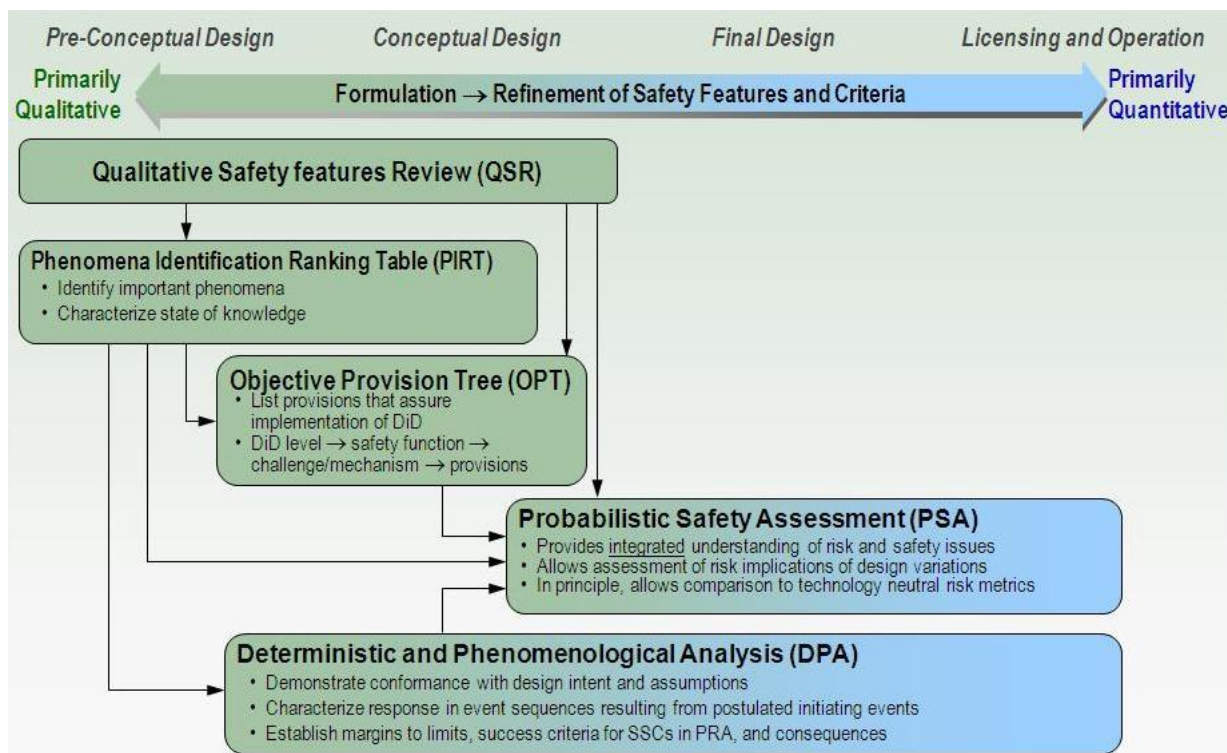


Figure A2-1 Proposed GIF Integrated Safety assessment Methodology (ISAM) Task Flow [15]

- *Qualitative Safety features Review (QSR)*

The QSR provides a systematic means of ensuring and documenting that the evolving Gen IV system concept of design incorporates the desirable safety-related attributes and characteristics that are identified and discussed within the significant references for principles, requirements and guidelines (IAEA, GIF, INPRO, etc.). The QSR provides a useful means of shaping designers' approaches to their work to help ensure that safety truly is "built-in, not added-onto" since the early phases of the design of Gen IV systems. The QSR serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the developing design in terms of safety issues or vulnerabilities.

- *Phenomena Identification and Ranking Table (PIRT)*

The PIRT technique has been widely applied in both nuclear and non-nuclear applications. As applied to Gen IV nuclear systems, the PIRT is used to identify a spectrum of safety-related phenomena or scenarios that could affect those systems, and to rank order those phenomena or scenarios on the basis of their importance (often related to their potential consequences), and the state of knowledge related to associated phenomena (i.e., sources and magnitudes of phenomenological uncertainties). The method relies heavily on expert elicitation, but provides a discipline for identifying those issues that will undergo more rigorous analysis using the other tools of ISAM. As such, the PIRT forms an input to both the **Objective Provisions Tree** (OPT cf. below) analyses, and the Probabilistic Safety Analysis (PSA). The PIRT is particularly helpful in defining the course of accident sequences, and in defining safety system success criteria. The PIRT is essential in helping to identify areas in which additional research may be helpful to reduce uncertainties.

- *Objective Provisions Tree (OPT)*

Following the logic illustrated by the Figure 4-1, the purpose of OPT is to ensure and document the provision of essential “lines of protection” to ensure successful prevention, control or mitigation of phenomena that could potentially damage the nuclear system. As such it can be considered as an innovative mean to represent the whole safety architecture.

There is a natural interface between the OPT and the PIRT in that the PIRT identifies phenomena and issues that could potentially be important to safety, and the OPT focuses on identifying design provisions intended to prevent, control, or mitigate the consequences of those phenomena.

The OPT can be extremely useful in helping to focus and structure the analyst’s identification and understanding of possible initiators and mechanisms of abnormal conditions, accident phenomenology, success criteria, and related issues.

- *Deterministic and Phenomenological Analyses (DPA)*

Conventional deterministic and phenomenological analyses, including the due consideration for the uncertainties, will be used to perform the quantitative analysis which supports the development and the sizing of the safety architecture. They will feed the PSA as an essential input to quantify the results. DPA is used from the late portion of the pre-conceptual design phase through ultimate licensing and regulation of the Generation IV system.

- *Probabilistic Safety Analysis (PSA)*

The PSA is a widely accepted, integrative method that is rigorous, disciplined, and systematic, and therefore it forms the principal basis of the ISAM. PSA can only be meaningfully applied to a design that has reached a sufficient level of maturity and detail. Thus, PSA is performed and iterated beginning in the late pre-conceptual design phase, and continuing until the final design stages.

In fact, as the concept of the “living PSA” is becoming increasingly accepted, the RSWG advocates the idea of applying PSA at the earliest practical point in the design process, and continuing to use it as a key decision tool throughout the life of the plant or system.

Although the other elements of the ISAM have significant value as stand-alone analysis methods, their value is enhanced by the fact that they serve as useful tools in helping to prepare for and to shape the PSA once the design has matured to a point where the PSA can be successfully applied.

Safety assessment and verification: the role of the ISAM tools

Table A-1 resumes - roughly - the crosscutting relationships between, on one side, the steps of the design / assessment process as described above and, on the other side, the different tools of ISAM.

The table content demonstrates the integrated character of the ISAM tools versus the safety assessment objective but, in particular, it shows the specific and complementary role of the deterministic tools (PIRT, OPT and DPA) and that of the probabilistic assessment (PSA).

Table A-1 *Relationships between the steps for the design and assessment of the SA and the role of the different tools of ISAM*

	QSR	PIRT	OPT	DPA	PSA
<i>Regulatory Framework (Goals, objectives, principles, requirements, guidelines)</i>	✓				
<i>Selection of Safety Options and provisional Provisions</i>		✓	✓	✓	✓
1. <i>Compliance / consistency of the design options with the principles, requirements and guidelines</i>	✓				
2. <i>Identification, prioritization and correction (if feasible) of discrepancies between design options with the principles, requirements and guidelines,</i>	✓	✓	(✓) ⁶⁶	(✓) ³⁵	(✓) ³⁵
3. <i>Identification of challenges to the safety functions,</i>		✓	✓		
4. <i>Identification of mechanisms (initiating events) and selection of significant (envelope) plants conditions to be considered for the design basis,</i>		✓	✓	✓	(✓) ⁶⁷
5. <i>Selection and categorization of representative design extension conditions (without and with core melting; DEC A & DEC B with the WENRA terminology) to be considered for the design basis</i>		✓	✓	✓	(✓) ³⁶
6. <i>Selection of external events that exceed the design basis and for which safety systems are designed to remain functional both during and after the external event</i>		✓		✓	(✓) ³⁶
7. <i>Identification of plant event sequences that could result in high radiation doses or radioactive releases must be practically eliminated</i>		✓		✓	✓ ⁶⁸
8. <i>Identification and selection of needed provisions, implementation within the corresponding "layers of provisions" for the different levels of the DiD</i>	✓	✓	✓		
9. <i>Design and sizing of the provisions,</i>			✓	✓	✓ ⁶⁹
10. <i>Response to transients (safety analysis),</i>				✓	✓
11. <i>Final assessment for a safety architecture that should be:</i>					
o <i>Exhaustive,</i>		✓	✓		
o <i>Progressive,</i>			✓	✓	✓
o <i>Tolerant,</i>				✓	✓
12. <i>Forgiving,</i>				✓	✓
o <i>Balanced.</i>					✓

⁶⁶ While QSR and PIRT are identified as the main ISAM tools for this step, the outcomes of other ISAM tools can be used in successive iterations.

⁶⁷ The contribution to this step is essentially deterministic even if it is recognized that probabilistic assessment can help, for example, for the identification of complex events / sequences which probability of occurrence justify their consideration for the design and / or for the categorization of the selected initiating events..

⁶⁸ The role of probabilistic studies is important even if not sufficient for the demonstration of the practical elimination

⁶⁹ While the design and sizing of the provisions will be essentially deterministic, the probabilistic studies will help guaranteeing the requested reliability

Appendix 3 - The performances of DiD levels

The principle of the DiD

The following definition is applicable to illustrate the DiD principle (cf. The Council Directive (Ref. A3.1)):

The concept of defence-in-depth is fundamental to the safety of nuclear installations and is the basis for implementing high level nuclear safety objectives.

Application of the defence-in-depth principles, ..., ensures that safety activities are subject to, as far as reasonably practicable, independent layers of provisions, so that in the event that a failure were to occur, it would be detected, compensated or corrected by appropriate measures.

The effectiveness of each of the different layers is an essential element of defence-in-depth to prevent accidents and mitigate the consequences should they occur.

Defence-in-depth is generally structured in five levels. Should one level fail, the subsequent level comes into play.

The notion of Line of Protection

Following the GIF/RSWG (Ref. A3.2): *"The Line of Protection (LOP) integrates all sort of provisions and characterizes them, in a homogeneous way, through their performances, their reliability and the conditions of their mutual independence."* For a given level of the Defence-in-Depth, the Line of Protection is an "effective defence" (cf. IAEA SF1 (Ref. A3.3)) against a given mechanism or initiating event that has the potential to impair a fundamental safety function.

The term is perfectly synonym of "Layers of provisions" as indicated above [1] and within the IAEA SSR 2/1(Rev.1) (Ref. A3.4) and WENRA (Ref. A3.5) to describe the levels of Defence-in-Depth.

The term is used for any set of inherent characteristics, equipment, system (active or passive), etc., and any procedure, all being part of the plant safety architecture, whose objective is to accomplish jointly the mission required to a given safety function.

Following the principles of the DiD, for a given initiating event whose consequences are potentially unacceptable, for each challenged safety function, successive layers of provisions are implemented to control the abnormal condition, i.e. to maintain or to bring back the plant in safe conditions.

The subsequent levels shall be functionally redundant: *"Should one level fail, the subsequent level comes into play"* with performances that shall remain compatible with the safety objectives.

The performances of a DiD level (i.e. the corresponding layer of provisions, i.e. the corresponding LOP) shall be defined - within the context of a given sequence - versus deterministic and probabilistic success criteria for a given initiating event / sequence and for a given safety function.

Deterministic and probabilistic success criteria

The performances/success criteria of the LOP as a whole (i.e. the set of provisions contained by the LOP) can be fixed considering the safety objectives as defined by the F-C curve.

The figure A3.1 (Ref. A3.6) summarizes qualitatively the logic that can be used to determine the performances which are requested for the different levels of the DiD i.e. the deterministic and the probabilistic success criteria⁷⁰. For a given event, whose potential consequences are beyond that allowable (Spot "A"), that shall be managed by a given level of the DiD and the corresponding LOP, success criteria can be derived both in terms of physical performances ($A_{\text{failure}} \Rightarrow B_{\text{success}}$) and reliability ($A_{\text{failure}} \Rightarrow C_{\text{success}}$); these success criteria are essential to size (and eventually classify) the LOP's provisions with the corresponding level of the DiD.

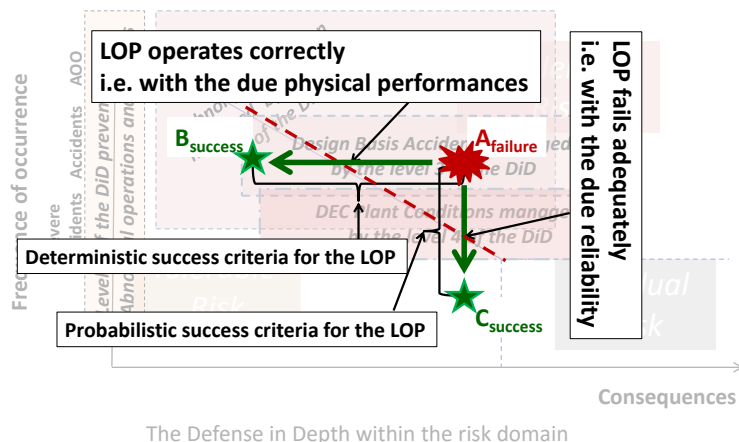
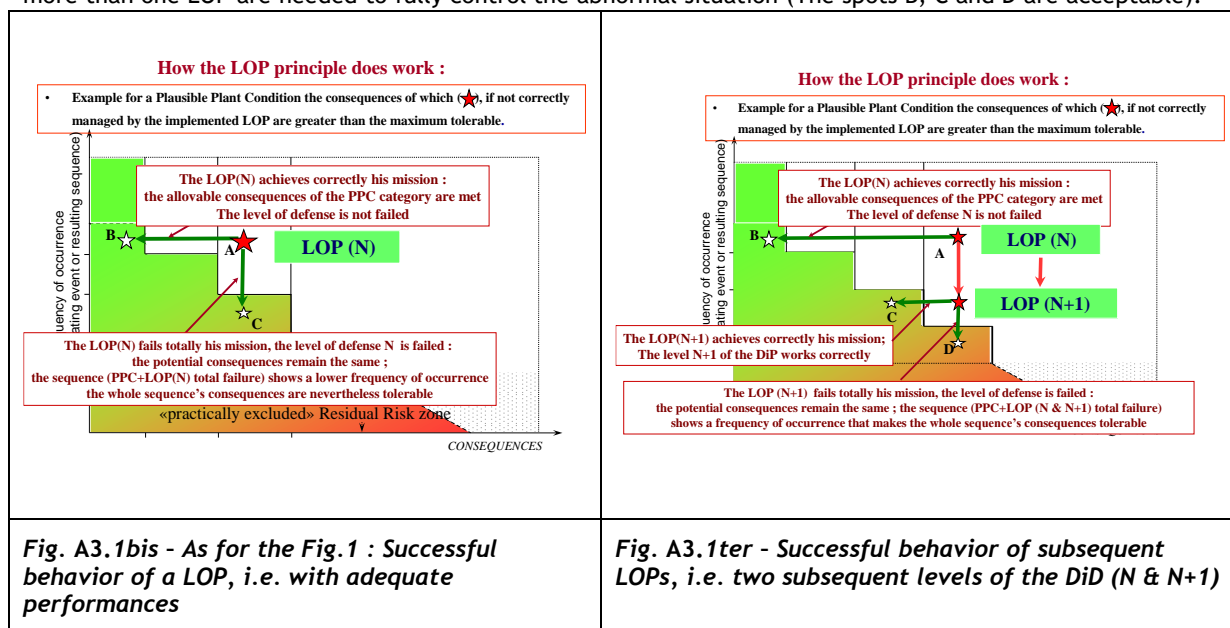


Fig. A3.1 - Deterministic and probabilistic success criteria: LOP behaviour for the management of risk

Fig. A3.1bis is analogous and details the content of Fig. A3.1; Fig. A3.1ter details specific situations where more than one LOP are needed to fully control the abnormal situation (The spots B, C and D are acceptable).



⁷⁰ The logic is analogous to that illustrated by the fig. 2 of the deliverable 30.4

References A3

1. Council Directive 2014/87/EURATOM of 8 July 2014 amending the Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations
2. Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems - Revision 1; Report GIF/RSWG/2007/002; November 24, 2008
3. IAEA Safety Standards Series N° SF-1: Fundamental Safety principles; 2006
4. IAEA Safety Standards Series No. SSR-2/1 (Rev. 1): Safety of Nuclear Power Plants: Design; 2016
5. WENRA RHWG Report: Safety of new NPP designs - Study by Reactor Harmonization Working Group RHWG; March 2013
6. G.L. Fiorini; S. La Rovere; P. Vestrucci: Peculiar role of the Defense in Depth and the probabilistic safety assessment in NPP performances optimization. ICAPP 2015 Paper N° 15421

Appendix 4 - Succinct analysis of the OPT IN IAEA TECDOC 1366 [8] - Event trees for the PSA

The following figures (A4.1 - A4.4) are extracted from the IAEA TECDOC 1366. They represent, in a simplified manner, the Objectives Provisions Trees for a MHTGR reactor. The objective is to show how one can construct the event trees analyzed with the PSA, from those constructed with the logic of the OPT.

N.B. The example remains fragmentary and could not be further explored because the whole study deserves to be updated (please consider the N.B. inserted within the figures).

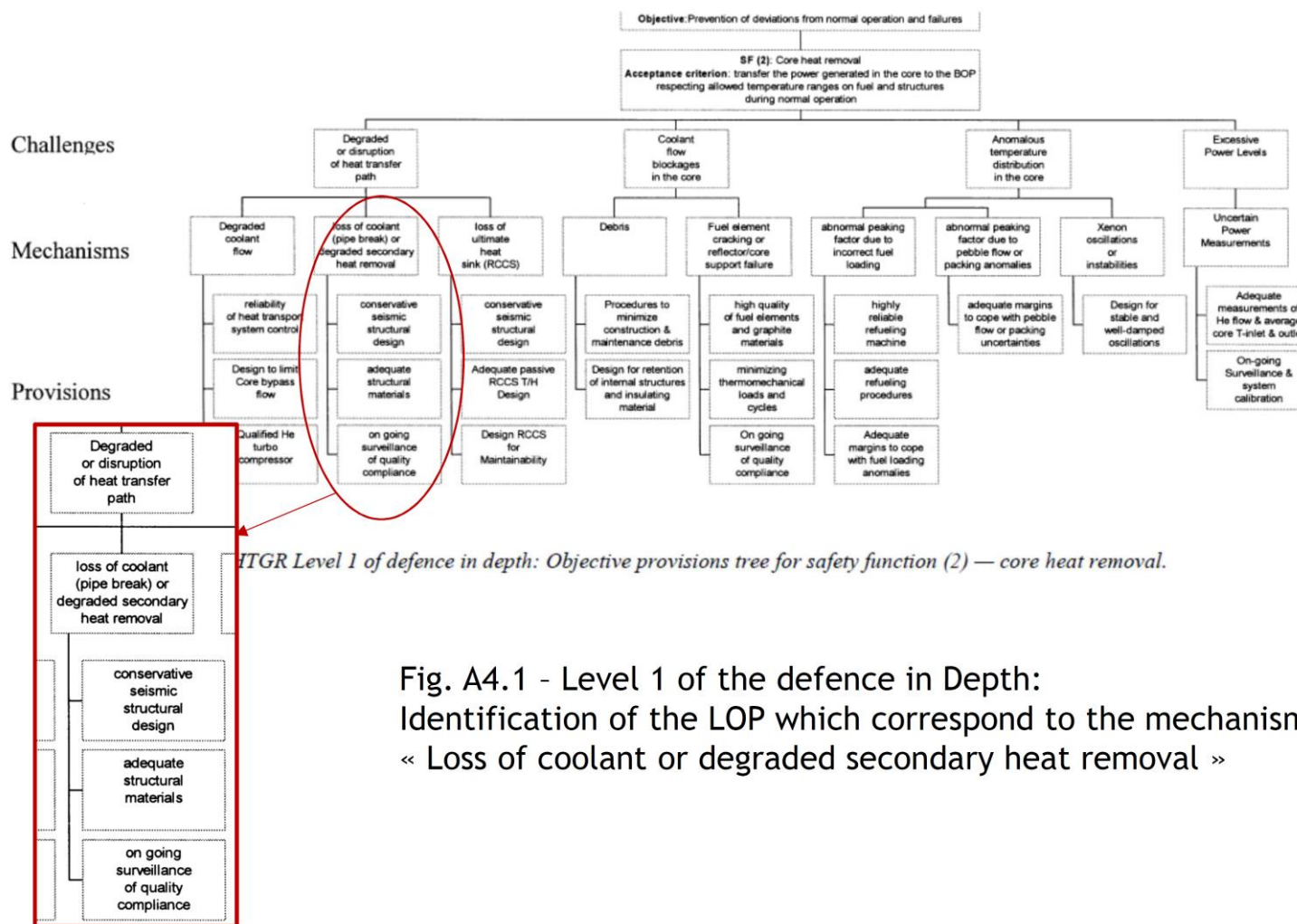
In each tree one can identify the sets of provisions which, in accordance with the logic of the OPT / LOP, achieve the missions required to control the considered accidental situations (Safety Function > Challenge > Mechanism).

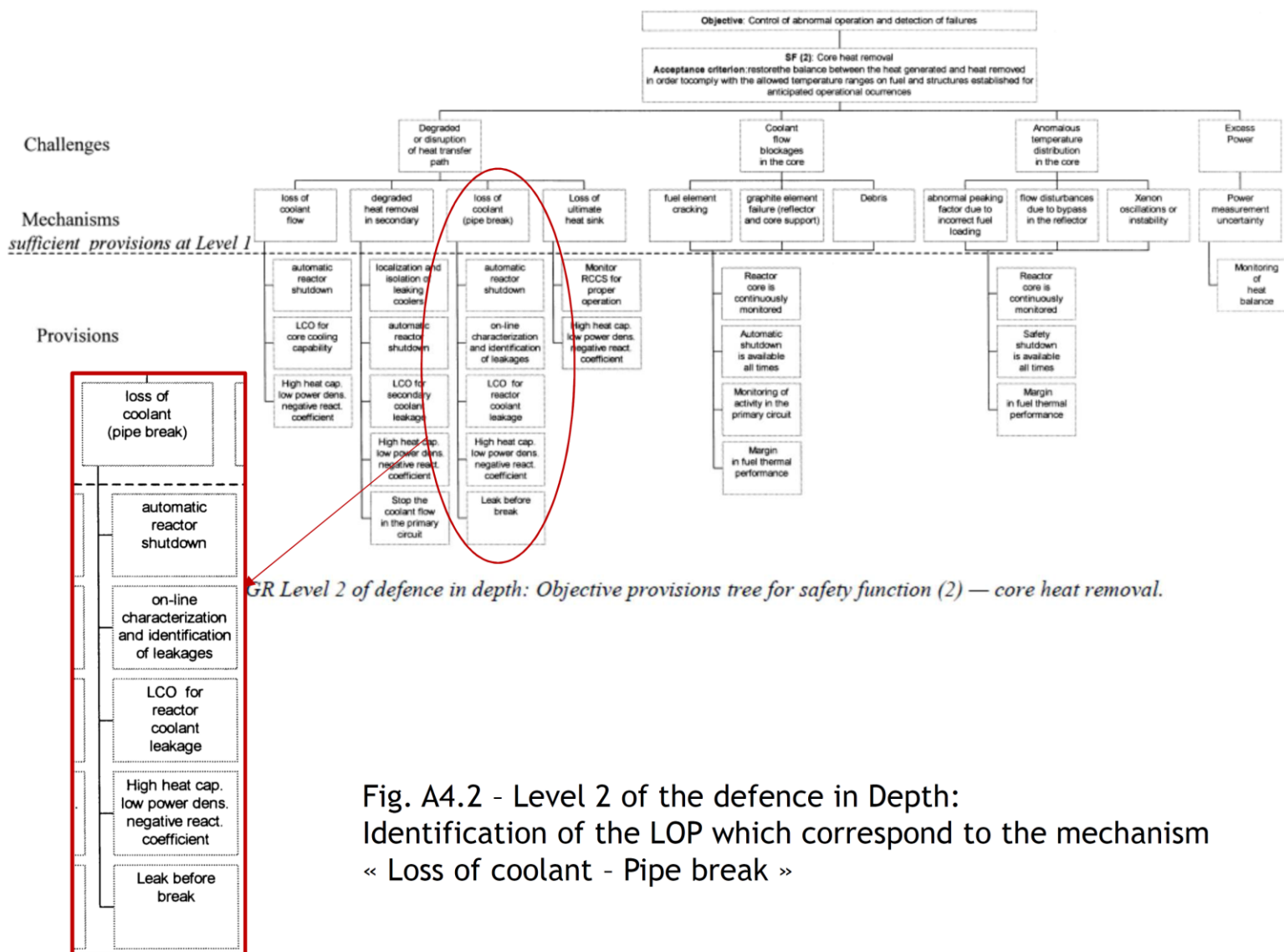
Within the figures, the lines of protection which are highlighted correspond to the degradation of the safety function “Decay Heat Removal” for the different levels of defense in depth (1 to 4). The example focuses on the challenge of “Degraded or disruption of the heat transfer path” and more precisely on the “loss of coolant” mechanism and the “total loss of the cold source”.

FIG. A4.5 summarizes the representation of the event tree constructed according to the logic of defense in depth: each node corresponds to the failure of a level of defense in depth. For PSA analysis, the probability of failure to be considered at each node of the event tree should be evaluated with a Fault Tree approach applied to each of the identified LOPs.

N.B. The following trees are taken from the IAEA TECDOC 1366

The « layers of provisions » which are identified for the « loss of coolant » conditions are highlighted





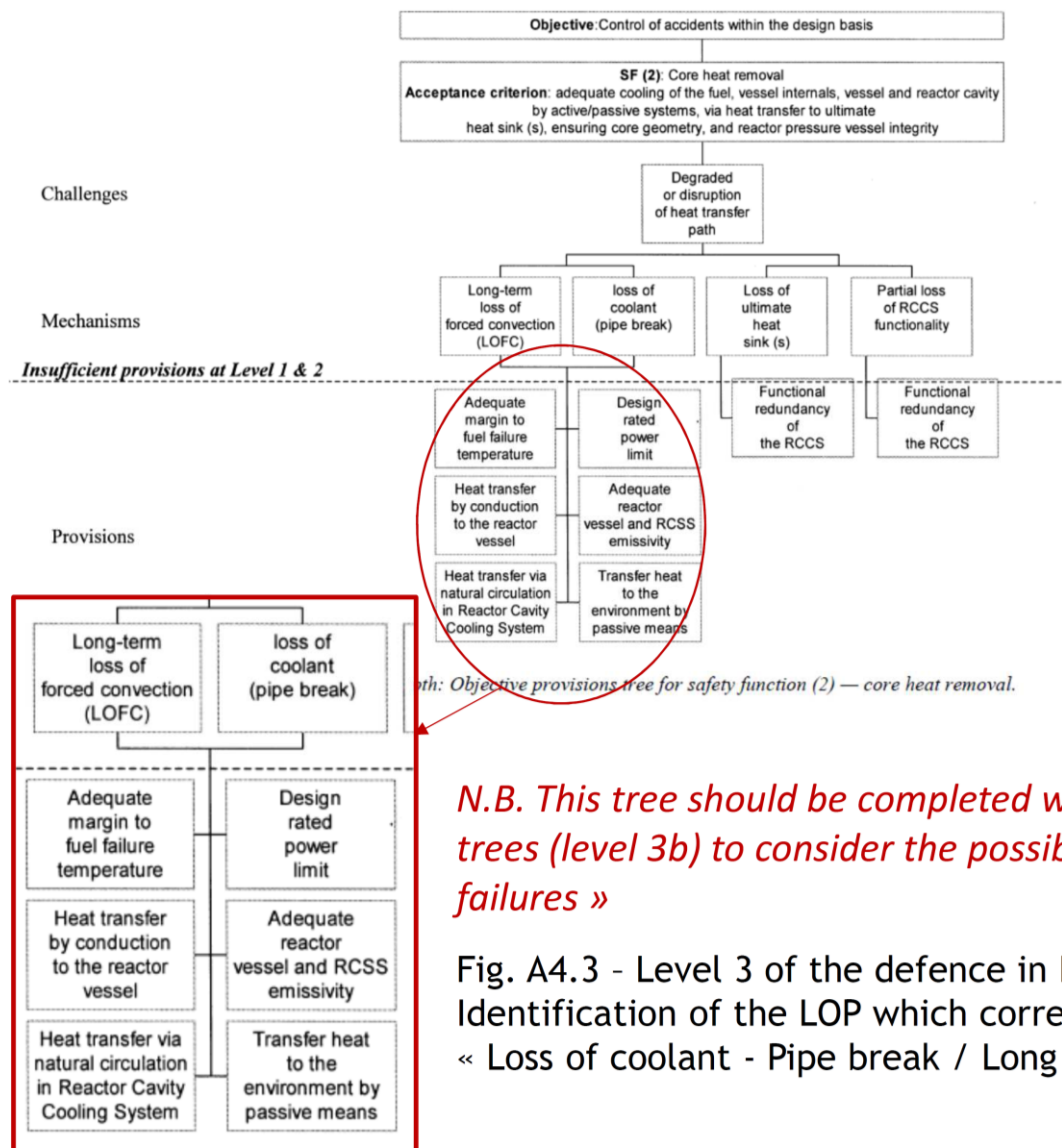
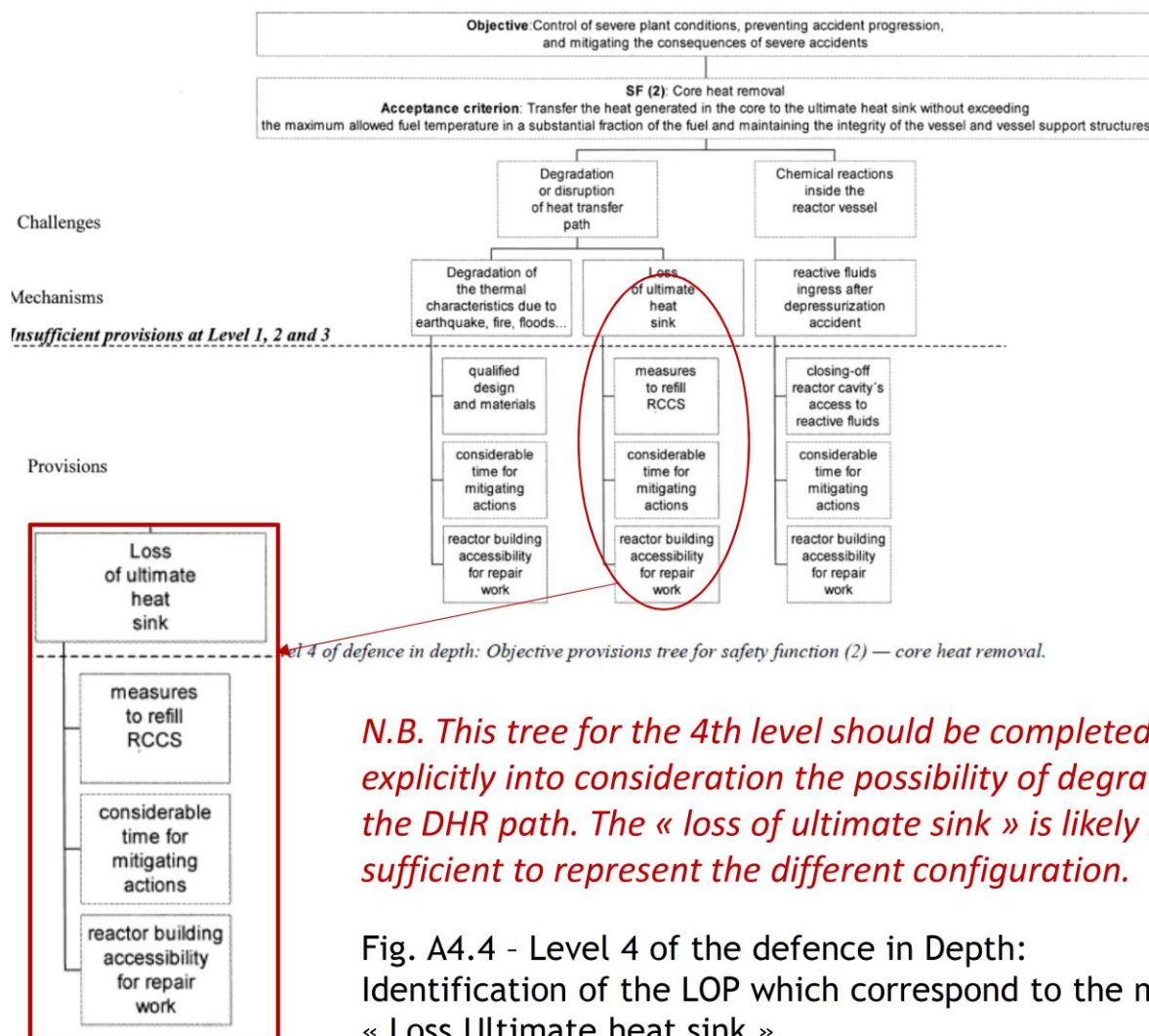


Fig. A4.3 - Level 3 of the defence in Depth:
Identification of the LOP which correspond to the mechanism
« Loss of coolant - Pipe break / Long term LOFC »



N.B. This tree for the 4th level should be completed to take explicitly into consideration the possibility of degradation of the DHR path. The « loss of ultimate sink » is likely not sufficient to represent the different configuration.

Fig. A4.4 - Level 4 of the defence in Depth:
Identification of the LOP which correspond to the mechanism
« Loss Ultimate heat sink »

Example of Event Tree organized following the structure of the Defense in depth.

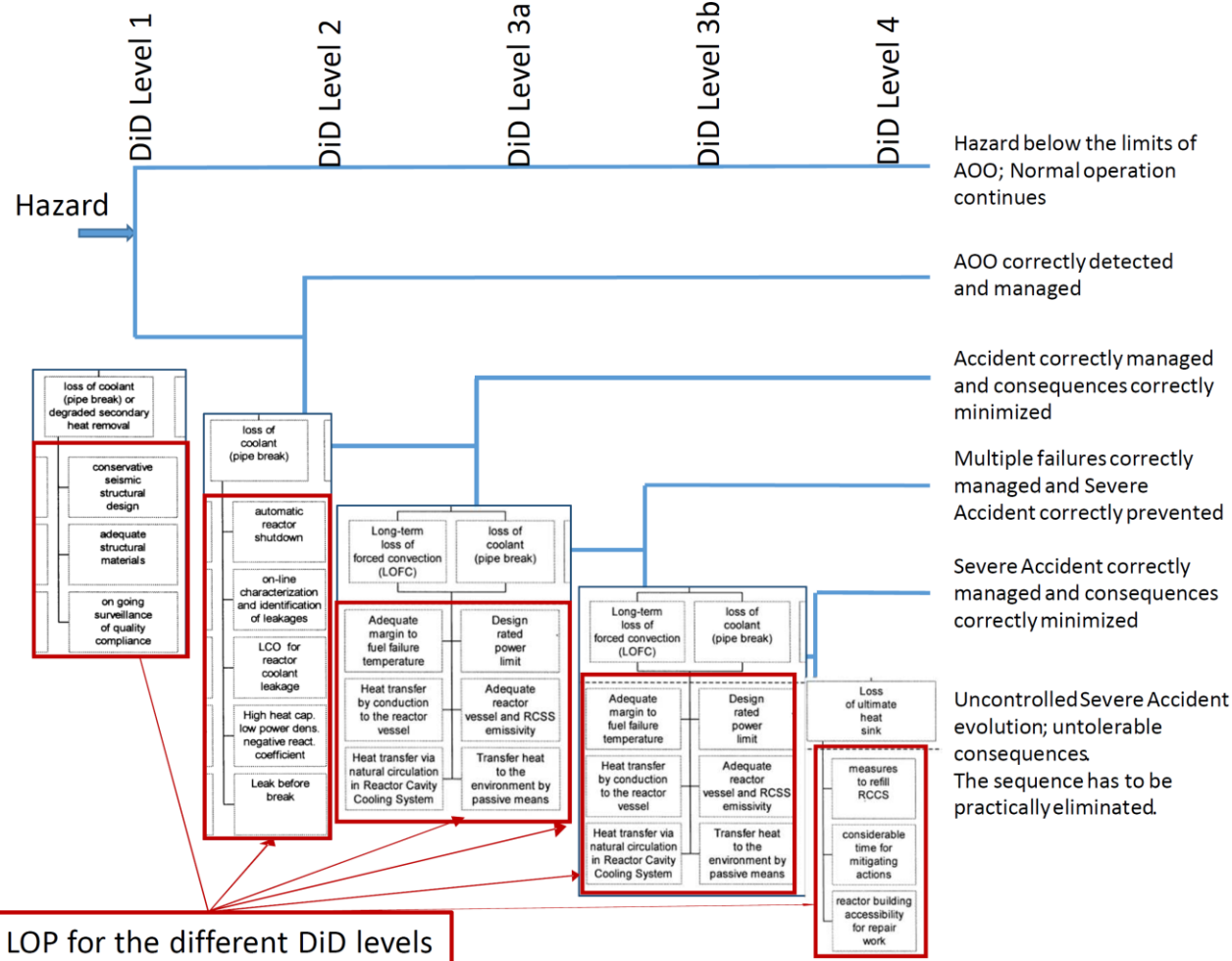


Fig. A4.5 - Event tree constructed according to the logic of defense in depth:
Each node corresponds to the failure of a level of defense in depth