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Risk Analysis of Nuclear Power Plants against External Events

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Risk Analysis of Nuclear Power Plants against External Events

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Sommario

This report presents the activities performed in the frame of LP1, Objective B (Advanced methodologies for the evaluation of accident consequences), task B2.1 of PAR 2013, ADP ENEA-MSE. The study addresses the risk assessment of nuclear power plants due to external events, such as the earthquake and tsunami, in the light of the Fukushima accident.

As a complement to traditional deterministic analysis, safety analysis is thus implemented with probabilistic models to estimate the relative risk within a "risk-informed" framework. To this end, a few illustrative applications of the proposed methods are presented as regards, respectively, the combination of external hazards modelling and the situation of incidental Station Blackout, caused by external events.

Note:

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Executive summary

The various gaps related to the current PSA (Probabilistic Safety Assessment) approach usage, as emerging from the Fukushima accident of Japan in 2011 are identified and analyzed, for their implementation in the PSA application and state of practice. These include the combination of external events as initiating events, the risk relative to sites with many units, the accident scenarios involving the performance of safety systems, such as for heat removal, for prolonged periods of time and the risk associated with spent fuel pools, to quote any.

The evaluation of the correlated hazards in the light of the external event PSA framework, is addressed specifically. To this aim some foundational notions to implement the PSA models to include external hazard combination, e.g., earthquake and tsunami as at the Fukushima accident, and the external hazard-caused internal events, e.g., seismic induced fire, are proposed and discussed for their incorporation within the risk assessment structure. The study is endowed with an illustrative example.

As part of risk-informed oriented probabilistic safety studies, deterministic calculations are also made to evaluate the response of the plant in relation to the function of containment and the "key times" in the face of certain situations such as accidental incident of prolonged Station Blackout.

A model of the containment code, such as MELCOR, is used to calculate the response of the system and to support the success criteria of probabilistic analysis and characterization of the response of the system (eg. temperature, pressure) and the critical times.

The analysis here aims to provide information for the development of any strategies, timeframes and alternative actions for accident management.

The issues emphasized within the present study are to be tackled to use the results of the PSA appropriately in future risk-informed decision making processes. In particular, risk assessment of external hazards is required and utilized as an integrated part of PRA for operating and new reactor units.

The risk-informed decision process requires, therefore, a strong interconnection between the deterministic and the probabilistic procedures in order to properly evaluate and examine the performances of the nuclear facilities (short and long term safety assessment). When dealing with the short term safety assessment, the use of a deterministic approach, supported by

conservative assumptions, is expected to lead to improved safety and a more rational allocation of the limited resources available.

To the aim, and in the light of the events that occurred in Japan, the vulnerability of fundamental safety functions system and components, like the outer containment building of the nuclear plant, has to be analysed in design conditions, like the exposure to external hazards, in the case of prolonged loss of power and cooling water supplies. Therefore the objective of the safety margin assessment, by deterministic approach, has been to evaluate the robustness of an existing plant in terms of design features and procedures against the impact of extreme events, such as the earthquake and the ensuing tsunami inundation phase, focusing on fulfilment of the fundamental safety functions.

The seismic and tsunami hazards are presented and discussed, in a comprehensive and systematic way, with reference to the acceptance criteria adopted for the Generation II reactor. The specific plant vulnerabilities and actions needed to improve plant and mitigation actions are identified and improvements for safety recommended.

List of acronyms

AC	Alternating Current
ADS	Accident Dynamic Simulation Methodology
AP1000	Advanced Passive
ATH	Acceleration Time History
ATWS	Advanced Transient Without Scram
BWR	Boiling Water Reactor
CCIE	Common Cause Initiating Events
CIRTEN	Consorzio Interuniversitario per la Ricerca Tecnologica Nucleare
DBA	Design Basis Accident
DC	Direct Current
DFE	Design Flood Elevation
DID	Defence-in-depth
DI&C	Digital I&C
DSA	Deterministic Safety Analysis
ENEA	Agenzia nazionale per le nuove tecnologie, l'energia e lo sviluppo economico sostenibile
EOP	Emergency Operating Procedures
EPS	Emergency Power Systems
ESBWR	Economic Simplified Boiling Water Reactor
ET	Event tree
FRS	Floor Response Spectra
FT	Fault tree
I&C	Instrumentation and Control
IE	Initiating Event
IDPSA	Integrated Deterministic and Probabilistic Safety Assessment
LOCA	Loss of Coolant Accidents
LOSP	Loss of Offsite Power
LPSD	Low Power and Shutdown
LWR	Light Water Reactor
NPP	Nuclear Power Plant
OP	Onahama Port Base Level
PGA	Peak ground acceleration
PRA	Probabilistic Risk Analysis, Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment, Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
SBO	Station Blackout
SSC	Structure System and Component
SFP	Spent Fuel Pool
SMA	Safety Margin Assessment

1. Introduction and scope

The Fukushima accident has highlighted some gaps in relation to the studies aimed at the analysis of the risk of nuclear installations, such as the analysis of the combination of external events as initiating events, the assessment of risk relative to sites with many units, the examination of accident scenarios involving the performance of safety systems, such as for heat removal, for prolonged periods of time and the risk associated with spent fuel pools, to quote any.
The proposed activity aims to:

• Identification and analysis of specific aspects regarding the probabilistic safety analysis, as highlighted by the accident of Fukushima.
Specifically, this part endeavours to address some significant issues revealed by the Fukushima accident in Japan in 2011, such as the analysis of various dependency aspects arisen in the light of the external event PSA framework, as the treatment of the correlated hazards. To this aim some foundational notions to implement the PSA (Probabilistic Safety Assessment) models related to specific aspects, like the external hazard combination, e.g., earthquake and tsunami as at the Fukushima accident, and the external hazard-caused internal events, e.g., seismic induced fire, are proposed and discussed for their incorporation within the risk assessment structure.

• As part of risk-informed oriented probabilistic safety studies, deterministic calculations are also made to evaluate the response of the plant in relation to the function of containment and the "key times" in the face of certain situations such as accidental incident of prolonged Station Blackout, caused by external event as typhoon and tornado.
A model of the containment code (MELCOR or alternatively MAAP) is used to calculate the response of the system and to support the success criteria of probabilistic analysis and characterization of the response of the system (eg. temperature, pressure) and the critical times.
The analysis here aims to provide information for the development of any strategies, timeframes and alternative actions for accident management.

The analysis should also be accompanied by an analysis of uncertainties and sensitivity with respect to the most significant phenomena to the failure of the containment, in terms of: 1) identification of the parameters that contribute most to the accident and the change results in the face of change these parameters, 2) quantification of the relative uncertainties.

The first part is organized by ENEA, the second is performed by CIRTEN.

2. Probabilistic Safety Assessment issues in the light of Fukushima accident

This section highlights some lessons learnt as coming out from the Fukushima accident for the development of a comprehensive PSA (Level 1).

The Fukushima accident of Japan in 2011 has discovered various gaps related to the current PSA approach usage for plant risk assessment. This makes some issues to be re-considered and/or implemented in the PSA application and state of practice: such issues are suitable to be classified into two major groups. The first group is related to the incompleteness of the current PSA practice including, for instance, PSA for extreme external events, site-wide risks (including multiple units and spent fuel pools), extended accident scenarios (including long-term station blackout, SBO, and loss of ultimate heat sink), implying, for instance, consideration for extended mission times and the development of full scope PSA. Conversely the second group is related to the combined hazards, which includes, for instance, the combined external hazards, e.g. earthquake and tsunami as at the Fukushima event and the external hazard-caused internal events, e.g. seismic induced fire. The identification of the dependencies between the external hazards and their modeling within the PSA framework represents an additional important point worthy of further investigation.

An assessment of the lessons learnt from the Fukushima nuclear accident for implementing PSA methodology will address some foundational notions related to a number of factors, as highlighted by the event:

- PSA for external hazards
- Dependency between events
    - between seismic events and tsunamis (and, more generally, between certain classes of external events)
    - external hazard-caused internal events, e.g. seismic induced fire
- Plant vulnerability to SBO (Station Black Out)
- Extended accident scenarios
- Multi-unit site risk issue
- Risk associated with spent fuel pools
- Consideration for prolonged mission times

- Performance assessment of passive systems and their role for the mitigation of external events implying the SBO
- The role of operator under extreme harsh conditions (human reliability)
- External hazard screening
- Re-assessment of DID (Defense In Depth), in terms of weaknesses and gaps between the different levels
- PSA application to all power plant statuses, e.g. low power and shutdown and consideration of all internal and external hazards: full scope PSA
- Emergency operating procedures implementation

All these issues are further set out in the following, as a "focused" update of the lessons learned and experience gained from Fukushima accident with regard to PSA state of practice: subsequently they should be conveniently addressed and worked out in the context of post-Fukushima follow-up activities, in order to implement their impact on nuclear safety. These should be incorporated into a well-structured frame in order to handle all possible risk contributors and their effects consistently and efficiently within one framework. For instance, in the future, the site risk including internal and external hazards will be required rather than just the risk of a unit.

In addition various approaches to resolve the coverage of some relevant issues, notably external hazards, site risk, and risk of spent fuel pool issues are outlined and suggested in this section.

2.1 PSA for external hazards

The tsunami caused by the massive earthquake facilitated the station blackout condition and subsequent core damage and gross containment failure. Hence the analysis points out the need of the tsunami and seismic PSA development and more in general PSA enlargement to take account of all external hazards. This is consistent with the requisite **of full scope PSA** development for NPPs, that is including all initiating events and all hazards (external and internal) and all plant states.

Indeed since the Fukushima accident, in which a catastrophic earthquake was followed by great tsunami greater than the design basis, extreme external events have emerged as significant risk contributors to NPPs. This accident shows that extreme external events have the potential to simultaneously affect redundant and diverse safety systems, and thereby induce common cause failure or common cause initiators.

Some standards set forth requirements for external event PSAs used to support risk-informed decisions for commercial NPPs, and prescribe a method for applying these requirements for specific applications [1-2].

External events covered within these standards include both natural external events and man-made external events, despite the fact that, actually, in many cases some extreme external hazards had been screened-out due to the low probability of occurrence estimated by deterministic and probabilistic hazard analysis in and nearby the plant sites. So up to now only seismic risk has been mostly considered as an extreme external event, in addition to flooding PSA in a few cases.

Most of the external events generally included within an external-events PSA are listed in the appendix of these standards, which is adapted from NUREG/CR-2300 [3].

The external event is a site specific issue, so each NPP should check the external event for its own site, so that it is essential to identify the extreme external events that can potentially affect the safety of NPPs for the evaluation of the site specific external hazards and risks. This applies to the seismic PSA: in the seismic PSA, a realistic seismic hazard evaluation by reviewing and estimating the several earthquake data sources is one of the most important tasks, and a significant source of uncertainty to the seismic risk of a plant.

A tsunami that follows a great earthquake became a big issue after the Fukushima accident: as in the previous case for the realistic tsunami hazard analysis and reduction of the uncertainty in the

tsunami hazard of a plant site, it is very important to identify the tsunami sources, and investigate the characteristics.

For the realistic assessment of a tsunami hazard curve and get more accurate results of the tsunami PSA, research on the tsunami hazard, tsunami fragility and system analysis should be performed.

2.2 Dependencies between certain classes of external events

External events can occur as single initiating events or as multiple external events, that is a combination of two or more external coincident external occurring more or less simultaneously. In addition they have the potential to result in an internal initiating event.

This emphasizes as relevant another issue related to unanticipated scenarios concerning correlated hazards. They include for example, combined external hazards, such as the earthquake and tsunami in Fukushima, and events hazards causing internal events, such as seismic induced fire and flood.

For instance, in case of induced hazards as formerly defined, the application of the conditional probability concept would imply the consideration for the dependencies between the events: this concerns essentially the assumption of dependency between the marginal distributions, to construct the joint probability distribution of the frequencies relative to the conditioning event and the conditioned event.

With this regard the present analysis is endowed with an illustrative example, aimed at the implementation of the proposed models in the probabilistic approach, as shown in Section 3.

2.3 Plant vulnerability to SBO

Plant vulnerability to SBO implies the complete loss of external and internal power supply, including the emergency on-site power supplies like the diesel generators.

The SBO plant susceptibility assessment should include both the LOSP (Loss Of off-Site Power) risk and the availability/reliability of backup power supplies: both aspects might depend on external hazards, like tsunami (according to Fukushima accident lessons learnt) and thus the relative evaluation would rely on site characteristics. On the one hand provisions at the design level are to be implemented in order to reduce this risk: this concerns both the reliability of off-site power,

which is usually assured by two or more physically independent transmission circuits to the NPP to minimize the likelihood of their simultaneous failure, and similarly, the reliability of on-site power which is enhanced by sufficient independence, redundancy and testability of batteries, diesel generators, gas turbines and the on-site electric distribution systems to perform safety and other functions even if a single failure occurs.

On the other hand emergency procedures aimed at AC power supply recovery are envisaged: these could include additional backup diesels available at the site, portable generators, etc. which can provide power to DC or instrumentation buses, or a limited amount of vital equipment. Dedicated gas turbine unit which is normally disconnected from the grid or a dedicated connection to a hydroelectric facility, are worth consideration as well.

In a further step, for completion, the assessment should consider as well the failure of additional back up equipment that can exist and be ready to be put into service in case of a station black out scenario to power some important equipment.

Ultimately, the capability of SBO systems to mitigate design basis and beyond design basis events needs to be re-evaluated and strengthened as necessary. Emergency preparedness equipment must be capable of addressing multiunit and SBO situations, and facility emergency plans must address prolonged SBO and multiunit events.

2.4 Extended accident scenarios

The Fukushima accident demonstrated that attention has to be paid to extended accident scenarios, like the long term SBO, which, in turn, implies an extension of mission time from 24 hours to 48 hours, or 72 hours or even one month, as proposed in the paragraph addressing the specific issue related to mission time.

In such cases that the event would not stabilize in 24 hours it is important to assure that each event is treated as long as necessary in order to secure a stable end state that will not generate new risks for core damage or radioactive release. Success criteria in terms of requirements for long term operation are to be set.

In fact, the long term SBO calls additional safety and reliability requisites for the plant such as the need for a stable, long term ultimate heat sink to perform the decay heat removal function and the need for AC on-site power supply recovery.

For instance, the first item gives way to the implementation of passive systems within the design to achieve simplicity in safety, albeit their modelling complexity, as highlighted in the related paragraph.

The second point demands the development of proper on-site AC recovery actions, to cope with an extended loss of backup AC power supply, as previously described.

2.5 Multi-unit site risk issue

The site risk issue is a very important one, especially in some countries as Korea and Japan where there are from 4 to 6 units per site.

The Fukushima Daiichi nuclear Power Station is a six-unit facility. Hydrogen explosions occurred in multiple units (Units 1 and 3) and the operating units (Units1, 2 and 3) affected Unit 4, which was defueled at the time of the accident, [4].

Concomitant reactor accidents at a site have been ignored in most of the current PSAs, because they were performed with the assumption that the event leading to core damage can only occur in one reactor at a time. Following the Fukushima accident, however, the issue of site risk is spreading over all the multi-unit sites, composed of two or more operating reactors. It is the reason that the independent risk for one reactor can be significantly underestimated by some missing scenarios associated with multi-unit site risk.

Research is to be performed with the main goal of development of site risk assessment methodology and models, including the extremely complex multiunit accidents and development of site-risk profile, based on all power modes, all hazards, including the extreme risk factors.

There are a variety of initiating events such as certain loss of offsite power events, loss of service water events, and seismic events that lead to concurrent event sequences on two or more reactor units on a site. The probability of multiple concurrent reactor accidents is significantly influenced by the use of shared and dependent systems, as well as common cause failures in redundant systems at the multiunit sites. There are several key inter-unit dependencies at a NPP which are likely be found to influence the development of an integral risk statement: some important examples are the electric power systems and the service water supply systems.

It is expected that multi-unit accident sequences make a significant contribution to multi-unit risk in comparison with the linear combination of single reactor accidents at each unit and therefore can not be dismissed.

The envisioned PSA models should include the concurrent initiating events, the single-caused-multiple events and the dependencies and common cause failures between multi-units.

With this respect, for instance, an approach was proposed to develop PSA models, including three types of initiating events (IEs) [5]: (1) IEs impacting the concerned units (loss of offsite power, seismic events, external flooding, tornado and wind, or a truck crash in switchyard), (2) IEs impacting the concerned units under certain conditions (loss of condenser vacuum, loss of service water, and turbine missile), and (3) IEs impacting each unit independently (loss of coolant, general transients, loss of component cooling, loss of one DC (Direct Current) bus, internal fire and flooding).

2.6 Risk associated with spent fuel pools

An assessment of the accidental risk of the spent fuel pool (SFP) against both internal events and external hazards is one of the emerging issues required to be incorporated into an integrated risk assessment framework. This issue was identified as one of the lessons learned from the Fukushima accident, and relevant activity is currently progressing worldwide. In fact in the Fukushima accident the hydrogen explosions in Units 1, 3 and 4 damaged their respective reactor buildings. The extent of the damage to the spent fuel is uncertain, but the Unit 4 appears to have been the mostly heavily damaged. Debris from the hydrogen explosions fell into the pools and may have mechanically damaged fuel and associated safety systems [4].

From the DBA point of view, the major safety-related issues for the SFP are closely related to (1) controlling the configuration of fuel assemblies in the pool without loss of pool coolants, (2) ensuring the pool storage space is enough to prevent fuel criticality due to chain reactions of fission products, and the ability for neutron absorption to keep the fuel cool.

The Fukushima accident has stimulated the need for in-depth research to enhance the safety of spent fuels stored in the SFP, and related regulation requirements.

For instance, an installed seismically qualified means to spray water into the SFP must be provided: the enhanced SFP water addition capability must include associated instrumentation and safety related power. Safety related instrumentation must be capable of withstanding design basis natural phenomena to monitor spent fuel parameters including water level, temperature, and radiological conditions.

Although the underlying accident phenomena and progressions in a SFP are different from the reactor case subject to high pressure and temperature, a similar framework with the reactor case can be formulated to assess accidental risk from the PSA point of view [6-7].

The accidental risk in SFP can be assessed by specifying the initiating events and relevant accident sequences leading to the uncover of spent fuel, making significant contributions to SFP risk (similar with the Level 1 PSA in the reactor case), followed by severe accident evaluation, to be dealt with in the following section.

In the following some aspects for the accidental risk assessment for the SFP are summarized [8].

Level 1 PSA for SFP

- Specification of Initiating Events and Frequency Analysis
  - Loss of pool cooling system (LOPC)
  - Loss of coolant inventory (LOCI)
  - Loss of offsite power (LOOP), due to plant-centered and grid-related events, and severe weather like typhoon
  - Station blackout (SBO)
  - Cask drop caused by human error,
  - SFP structural failure, due to seismic events (including concurrent tsunami)
  - Internal fire, due to pool structural failure and seismic initiator, cask drop, etc.
  - Man-made external attack such as aircraft impact, external fire and explosion

- Analysis of Accident Sequences Leading to Spent Fuel Uncovery
  - Based on conventional Event Tree/Fault Tree (ET/FT) Analysis
  - Level 1 risk surrogate metric: Frequency of Spent Fuel Uncovery

2.7 Consideration for prolonged mission times

Fukushima event underlined the necessity to reconsider mission time span for its extension in PSA studies: in fact the “usual” mission time of 24 hours has been proved to be unrealistic so that the required mission time should be not only longer than 24 hours as usual Level 1 PSA mission time, but be extended beyond the 72 hours corresponding to the grace period.

This implies a realistic consideration of the event under investigation, in terms of accident progress and safety systems timing intervention, to figure the relative degree of mission time increase, to be considered for the extended scenario (e.g., including long-term station blackout and loss of ultimate heat sink assessment) assessment. The main issues that need attention in mission times re-definition, concern, in particular, prolonged accidental situations, implying, for instance, the protracted losses of AC power and residual heat removal.

A more proper mission time should be considered, as well, in order to identify long term dependencies (especially dependencies to non-safety functions).
Fukushima accident progress over time justifies the consideration for the duration of mission times for safety systems and components longer than 24 hours in a realistic way and, more generally for prolonged mission times to a very large extent (up to one month for instance), as it has been demonstrated that 24 hours recovery concept as for internal initiating events is not good for some external events.

2.8 Role of passive systems relevant for the mitigation of external events

One of the important considerations in the treatment of external events is the possibility of disruption of external sources of electricity, cooling water, other essential supplies and possibly prompt operator action following an extreme external event. In such a situation, some innovative reactor designs take advantage of passive safety features provided within the protected reactor building or inner containment, disregarding the availability of external sources of supply of electricity, cooling water, etc. Several designs provide for physical presence of large thermal capacity heat sinks (e.g. AP1000, ESBWR) available to cool the reactor core without depending on availability of externally powered pumps within the containment or elsewhere.
In this context, several passive systems enable prolonged grace period to the operator during which the reactor is maintained in a safe state without any operator intervention. This, in essence, implies availability of a large heat sink within the reactor building, and its highly reliable uninterruptible thermal communication with the reactor core to facilitate continued removal of core heat for prolonged durations without any involvement of active systems or operator interventions (e.g. natural convection, radiation, and conduction cooling). This feature too, is highly relevant for some extreme external events when, on account of possible devastation outside the protected reactor building, it is quite likely that all the external sources of cooling water, electricity, and instrumentation air and ventilation system become non-available. In such scenarios, it is also conceivable that the operators may not be in a position to act in an efficient or effective manner.
This is quite relevant as far as specific situations are concerned (for example, a specific combination of initiating events), implying, for instance, wired system incorporating sensors or actuators or a control system relevant to safety are assumed to be disabled in a manner that

the desired safety function could not be performed in the absence of required signals or power supplies. Of particular relevance are the thermal hydraulic passive systems implementing natural circulation to accomplish the decay heat removal function.
However the performance assessment of passive systems is still an open issue and the methodologies prompted so far need to be qualified to provide credit to the model quantification and the achieved reliability figures among the community of research in nuclear safety [9].
This complexity stems from a variety of open points coming out from the efforts conducted so far to address the topic and concern, for instance, the amount of uncertainties affecting the system performance evaluation, the lack of both operational and experimental data and the difficulties in merging the probabilistic aspects with the thermal-hydraulic aspects of the problem.

2.9 The role of operator under extreme harsh conditions

A review of the operational experience should be performed in order to reveal some key insights into human performance during external events. In fact there are some specific lessons to be learned from Fukushima accident analysis regarding possible weaknesses in the response capabilities.
Specifically the operator fragility during external events has to be assessed to take into account the eventual increase of the operator failure rate under a particular stress condition, dictated by that particular event which posed a significant load on organizational response including issues with distributed decision making and coordination with outside organization
This concerns the operator capability to adequately navigate the situation in a calm and orderly manner with no delays in the immediate actions to tackle situations implying for instance lack of information or lack of communication mechanisms with local operators, under the impact of severe environmental (i.e. radiological) conditions.
In fact, in case of Fukushima accident, the procedures proved difficult for operators to be followed, since they didn't match the scenario and combination of failures not normally considered credible. Overall there was a significant difference between the context (security, available equipment, indications, lighting and environment, and communication means) and the assumptions in and training on the procedures.

In addition tsunami protection actions were expected to be performed (e.g., stopping pumps, closing doors, etc.) prior to the tsunami, however, these actions were not fully performed because the operators did not have sufficient time to perform these local actions in the time frame before the tsunami hit, nor were the specific actions well proceduralized.

2.10 External hazard screening

A screening analysis methodology applying the probabilistic approach, aimed at identifying and prioritizing relevant single or coincident external events and overlooking irrelevant hazards, is envisaged. In fact, in order to arrive at a manageable amount of potential events, some sort of selection criteria are needed.

The aim of probabilistic screening is to evaluate which events represent an acceptable risk. The rest of the events may require in-depth analysis in order to become acceptable with or without modifications of the plant or improvements in instructions and training.

In fact, in view of the low risk Level of nuclear power plants, even very rare external events may give significant risk contributions. Therefore, the intention is to create as comprehensive a list as possible of potential external events to be further studied.

The screening process should include the following steps:

- identification of potentially relevant single external events, in order to generate a list of potentially relevant external events (and their combination) that may have safety impacts on the plant.
- survey of information about the plant's resistance to external events and on retrieval of relevant information on both external events and the facility. This includes data, methods and experience relating to external events, or protection against them in the facility.
- probabilistic accident analysis, i.e. plant PSA model as well as the estimation of the frequency of occurrence of the event.
- screening analysis based upon screening criteria which are defined for simple and multiple external events. These criteria are applied in the screening analysis to exclude non-relevant external events from further analysis. A screening criterion could be defined on a frequency (in terms of a limit threshold definition) or consequence (e.g., the impact is such that cannot initiate an event sequence that could lead to core

damage) basis or a combination of both. Criteria are clearly required to give guidance to the analyst so that the screening process can be accomplished in a logical way through a well-structured subjective judgment.

- the outcome from the screening analysis consists of the external events that have been excluded upon the screening criteria adopted, and a summary of conclusions and recommendations regarding events that require deeper analysis.

2.11 Re-assessment of DID

The Fukushima accident emphasized some deficiencies related to plant design, construction, maintenance and operation.

In order to reduce the risk of such accidents in future, the defence-in-depth (DID) design must be fostered within a risk-informed framework to complement the safety components and systems requisites of redundancy, diversification and independence, to prevent and/or mitigate accidents and decrease the core damage frequency.

The reliability requirements for the individual components and systems requested to accomplish the preventive and mitigative safety actions are to be improved in order to prevent the risk of the “transition” of the accident from level 3 to level 4 of the DID scale, that is from DBA to BDBA (severe accident), implying the core degradation albeit confined within the containment.

2.12 Full scope PSA

The Fukushima Daiichi Unit 4 had its entire fuel inventory offloaded to its spent fuel pool. The events in Units 1, 2 and 3 led to a hydrogen explosion in Unit 4 that severely damaged its reactor building and possibly damaged the spent fuel pool and its included fuel [4]. Accounting for the potential for a defueled unit to suffer a severe hydrogen explosion and its impact on a defueled unit’s spent fuel pool is necessary.

An effort has to be made in order to solve and overcome the issues related to limited scope PSA. Indeed a limited scope of the PSA might fail to provide appropriate insights with some risk-informed decision makings and/or sometimes result in too excessive conservatism. Actually, since the PSA requires a lot of resources, in many cases, only the limited scope of the PSA is performed, that is, the internal Level 1 and limited Level 2 PSA for full power mode. Furthermore, the low power shutdown (LPSD) PSA requires much more resources than the full power PSA, since the LPSD PSA consists of many PSA models developed for different plant statuses of a nuclear power plant (NPP) during the overhaul. So, there are a limited number of LPSD PSAs in the world.

In addition, since the external PSA framework is not well established compared to the internal PSA framework, only some external Level 1 PSAs have been usually performed.

It is expected that a sound framework has to be envisaged helpful to resolve the issues regarding the limited scope of the PSA, and to reduce the required resources. This should allow the reduction of some inconsistencies that might exist between (1) the internal and external PSA, and (2) full power mode and LPSD PSAs, which could arise from the manual modification process of the internal PSA models for developing the external and/or LPSD PSA models.

2.13 Emergency operating procedures implementation

Onsite emergency response capabilities, including emergency operating procedures (EOPs), severe accident management guidelines, and emergency damage mitigation guidelines must be strengthened and integrated. More realistic training and exercises must be provided for all staff expected to implement these guidelines during an emergency.

This topic is strictly connected with the human reliability point, since it involves corrective actions to be taken by the operators. Nevertheless an analysis to find out eventual deficiencies in preventive and recovery actions and envisage consequent implementation of the emergency operating procedures to withstand external hazards is suggested.

In fact, for instance, recovery actions should be mostly the same as in basic PSA, but some actions specific to external events may need to be added in the PSA model.

In any case the accident put in evidence the requirement related to safety system robustness and diversity in responding to beyond design accidents such as station black-out for long duration and loss of ultimate heat sink and the consequent need to take effective measures to the unforeseen events.

3. Implementation of PSA models to estimate the probabilities associated with external event combination

As mentioned earlier in Section 2, the requirement to consider correlated hazards is emphasized by the Fukushima accident, as regards the combination of extreme hazards and the hazard-induced initiating events. In order to foster the importance of this aspects the simplifying assumptions of independence have to be avoided and implemented with appropriate models suitable to describe the correlation mechanisms, in terms of Common Cause Initiating Events (CCIE), such as:

- seismic hazard and tsunami, as events sharing the same source of origin
- strong winds and heavy rain, as phenomenological correlated events
- seismic hazards and seismically induced fire, as induced hazards

The present analysis is not site-specific, but aimed at a sort of "technology neutral framework", acknowledging the fact that the frequency assessment of correlated hazards should take into account all the available information (i.e. site-specific, regional, worldwide), as well as all correlations and uncertainties.

3.1 Combination of hazards approach

The easiest and "uncomplicated" way to assess the frequency of two or more external events occurring simultaneously would be to consider them as independent events, so that the overall frequency would be quite straightforward as the product of the single frequencies.

But actually the problem is more complicated, especially when dependency between the events cannot be ignored in the frequency assessment of the initiating event.

In fact, further analysis, as shown in the previous section, reveals that the single frequencies, actually, are not suitable to be chosen independently of each other, mainly because of the expected synergism between the different events under investigation: these synergistic effects trigger an accident sequence with the potential to challenge the system safety and performance at a more severe degree and extent than it would be if the single event were to be considered.

This conclusion allows the implementation of the initiating event quantification, by properly capturing the interaction between the single frequencies characteristics of the various events.

One approach to address the case of dependent external events is to is to estimate the joint *p.d.f.* (probability density function) of the frequencies, and then estimate the frequency based on the estimated joint *p.d.f.*.

Consider a simple case characterized by two events. Let's denote x_1 and x_2 the relative frequencies with distributions $f(x_1)$ and $f(x_2)$: if the events are dependent the following relationship holds:

$$f(x_1, x_2) \neq f(x_1)*f(x_2) \tag{1}$$

where the left term denotes the frequency of the combined events to be assessed, in the form of the joint *p.d.f.* of the single frequencies.

This expression extended to a number n of external events becomes:

$$f(x_1, x_2, \ldots, x_n) \neq f(x_1)*f(x_2)*\ldots*f(x_n) \tag{2}$$

For instance, in case of induced hazards as formerly defined, the application of the conditional probability concept implies the consideration for the dependencies between the events: this concerns essentially the assumption of dependency between the marginal distributions, to construct the joint probability distribution of the frequencies relative to the conditioning event and the conditioned event.

Therefore, in particular, in this work the concept of conditional probability is applied to determine the conditional density estimate.

At first we'll recall some definitions and characteristics of the conditional density function.

The conditional probability for events A and B (conditional probability of A occurring given that B occurs) is given by:

$$P(A/B) = \frac{P(AB)}{P(B)} \tag{3}$$

if $P(F)>0$

The expression for the conditional probability density function is

$$f(y/x) = \frac{f(x,y)}{f_x(x)} \tag{4}$$

defined for $x = f_x(x) > 0$

where $f(x,y) \geq 0$ is the joint density function of the variables x and y

where the marginal density f_x of x satisfies

$$f_x(x) = \int_{-\infty}^{\infty} f(x,y)dy \tag{5}$$

Then the conditional probability of y given x, is

$$F(y/x) = P(y < Y/x = X) = \int_{-\infty}^{Y} f(y/x)dy \tag{6}$$

In the following, the normal distribution is considered for its relative simplicity and familiarity to engineers.

It represents a good approximation in case the standard deviation is small as compared to the mean value.

$$f(x) = (1/\sigma\sqrt{2\pi})\exp -((x-\mu)^2/2\sigma^2) \tag{7}$$

The values of the cumulative distribution function are derived from the tables of the standard normal distribution N(0,1),

$$f(t) = (1/\sqrt{2\pi})\exp -(t^2/2) \tag{8}$$

after the transformation $t=(x-\mu)/\sigma$

3.2 Illustrative example

As an illustrative example, Table 1 shows the parameters of interest of the normal distributions, with reference to the case of the combination of two events (such as the earthquake and tsunami or strong wind and heavy rain).

Table 1 - Normal pdf characteristics

Parameter	Range(a-b, 1/year)	Characteristics (1/year)
x_1	3-7 E-01	μ = 5.0E-1 σ = 1.0E-1
x_2	2-6 E-01	μ = 4.0E-1 σ = 1.0E-1

It's worth noticing that the ranges defined by two standard deviations roughly cover the 95% confidence interval, considering that the two-sided 95% confidence interval lies at $\pm$ 1.96 standard deviations from the mean value.

The joint *p.d.f.* of two normally distributed variables x and y, is given by the bivariate normal distribution expression:

$$f(x,y) = 1/[2\pi\sigma_1\sigma_2 (1-\rho^2)^{1/2}] \exp - [s/(2 (1-\rho^2))] \tag{9}$$

Where

$$s = (x- \mu_x)^2/ \sigma_x^2 - [2 \rho(x- \mu_x) (y- \mu_y)]/ (\sigma_x \sigma_y) + (y- \mu_y)^2/ \sigma_y^2 \tag{10}$$

The expression for the bivariate normal density function in the standard form is:

$$f(x,y) = (1/2\pi(1-\rho^2)^{1/2}) \exp -((x^2 + y^2 -2\rho xy)/2(1-\rho^2)) \tag{11}$$

with Pearson's product moment correlation coefficient ρ

$$\rho = \sigma_{12}/(\sigma_1 \sigma_2) \tag{12}$$

A bivariate normal distribution is specified by setting an average matrix $\mu = (\mu_1, \mu_2)$, and a variance-covariance matrix $\Sigma = (\sigma_{ij})$ with σ_{11} = Var (x), σ_{22} = Var (y) and $\sigma_{12} = \sigma_{21}$ = COV(x,y), respectively as

$$\begin{bmatrix} \mu_1 \\ \mu_2 \end{bmatrix} \quad \text{and} \quad \begin{bmatrix} \sigma_1^2 & \sigma_{12} \\ \sigma_{21} & \sigma_2^2 \end{bmatrix}$$

Note that Σ is a symmetric positive matrix.

In the present case let's assume a correlation coefficient equal to 0.9, since the variables seem to be highly correlated: from (12) this is equivalent to a covariance value of 0.9. Thus the matrixes defined above assume the form of

$$\begin{bmatrix} \mu_1 \\ \mu_2 \end{bmatrix} = \begin{bmatrix} 5^* \\ 4^{**} \end{bmatrix} \quad \text{and} \quad \begin{bmatrix} \sigma_1^2 & \sigma_{12} \\ \sigma_{21} & \sigma_2^2 \end{bmatrix} = \begin{bmatrix} 1^* & 0{,}9^{**} \\ 0{,}9^{**} & 1^* \end{bmatrix}$$

* read as 5.0E-1
** read as 4.0E-1

*read as 1.0E-2
**read as 0.9E-2

However the evaluation of these quantities through expressions (9) and (11) requires numerical integration techniques. Thus another approach is followed if one takes into account the conditional distribution of y given that x = X. This is represented by another normal distribution:

$$f(y/x=X) = \text{Nor}\,(\mu_y + \rho(\sigma_y/\sigma_x)(x-\mu_x),\ \sigma^2_y(1-\rho^2)) \qquad (13)$$

With the correspondences x_1=y and x_2=x we can construct the joint probability mass function of the two variables. From the normal distribution parameters, one can determine the probability given that the variable x will fall in a given range. For example let's evaluate the probability value of the combined external events frequency (1/year), conditional on one single external event assuming a certain frequency value (e.g. x=4,1*10^{-1}/year). This point is illustrated in Figure 1 below, which refers to parameter values reported in table 1, so that the expected values E and variance Var of the normal pdf are respectively:

E (y/x= 4,1*10^{-1}/year) = 5,09E-1/year

Var (y/x= 4,1*10^{-1}/year) = 0,019E-1/year

The relative *p.d.f.* in the form f(y/x=4,1*10^{-1}/year) = Nor (5,09*10^{-1}/year, 0,019*10^{-1}/year) is represented in Figure 1.

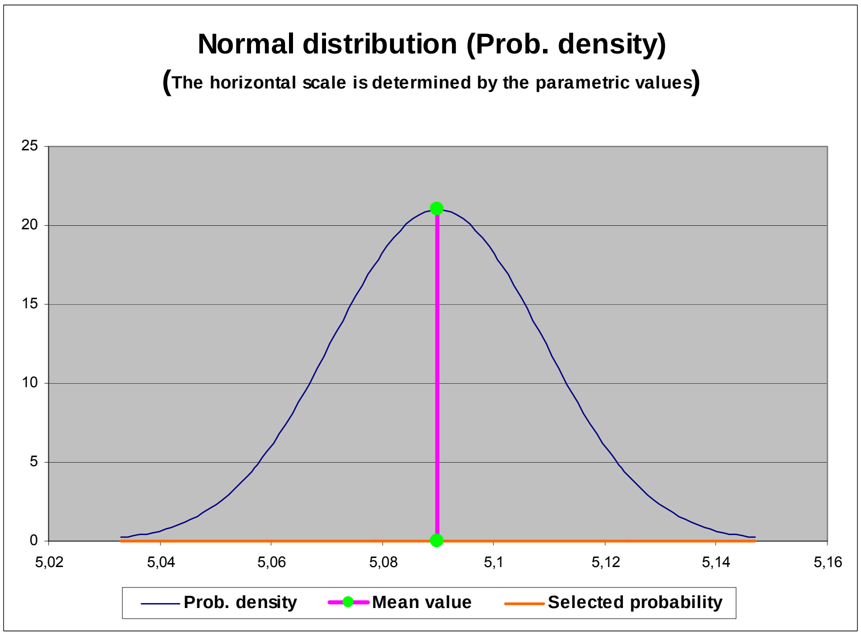


Figure 1: Conditional probability density function of events

The probability of the occurrence of both events, with frequency of, for instance, $5.1*10^{-1}$/year and $4,1*10^{-1}$/year respectively, as represented in the highlighted area of the Figure 2, is:

$P(y \leq 5,1 * 10^{-1} / x = 4,1 * 10^{-1}) = \Phi(0,53) = 0.7$

The previous analysis holds particularly as regards the case of induced accidents, where the probability of occurrence of an event is conditional upon the occurrence of another event.

This analysis may be extended to include more external events, by adopting multivariate normal distributions: obviously this adds a significant burden to the study.

It's worth noticing that this mathematical approach finds application, as well, as regards the development of the whole probabilistic safety analysis process, since models for PSA for external events typically consider a number of potential events that may challenge plant safety such as loss of AC power necessary to operate critical equipment and/or loss of capability to cool the nuclear reactor core.

As the characteristics of events that may challenge plant safety are identified, the capacity of the safety systems designed to protect critical functions is evaluated for the conditional probability of failure, thus resulting in an overall measure of the available protection with respect to the likelihood of the intervening event.

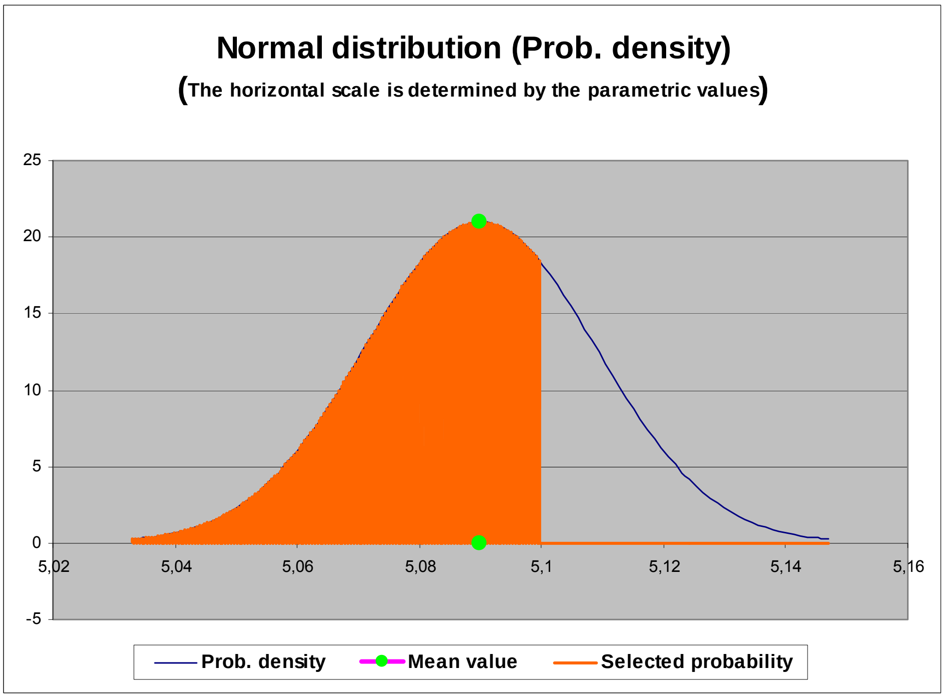


Figure 2: Probability of the occurrence of both events

Results are based and conditional upon the assumed distributions and the assumptions coming from a "rough" engineering investigation, without resorting to site-specific data bases for statistical inference, retaining the level of generality of the analysis, as formerly underlined.

4. DSA Implementation

4.1 Introduction

General and specific requirements for external hazards assessment are given in IAEA Safety Standard NS-R-3 [10]:

- Site characteristics that may affect the safety of the nuclear installation;
- Sites for nuclear installations shall be examined with regard to the frequency and severity of external events and phenomena that could affect the safety of the installation
- For an external event (or a combination of events) the parameters and the values of those parameters, characterizing the hazards, should be chosen so that they can be used easily in the design of the installation.
- In the determination of hazards, site specific data shall be used, unless such data is unavailable, data from other regions that are sufficiently relevant to the region of interest may be used along with appropriate and acceptable simulation techniques.
- Appropriate methods shall be adopted for establishing the hazards that are associated with major external phenomena. The methods shall be justified in terms of being up to date and compatible with the characteristics of the region.

The hazards assessment for existing NPPs should assess the design basis to determine if the current information of potential external hazards at their site is adequate and still applicable. This assessment may have to be undertaken on short term and long term basis: in the short term conservative estimates based on design basis review expert judgment and conservative assumptions may be used for parameter characterizing the hazards while in the long term a detailed analytical approach is required.

In the Seismic Margin Assessment (SMA) the success path is defined by which systems and components guarantee the safety of plant; the selection of these components should be based on results of analyses that should consider all the appropriate facility hazards and plant response.

For deterministic SMA the safety significant components are selected to assemble the success path: the components needed to ensure the performance of the fundamental safety functions including dependencies of support systems and interactions with non-safety related SSCs.

For probabilistic external events safety assessment the safety significant SSCs are those that are safety classified, their support systems and other non-safety classified that may interact with the safety classified SSCs and those that are credited to mitigate the loss of safety functions.

In most applications, these SSCs to be evaluated in a SMA are selected on the basis of the following minimum requirements [10]:

– Shutdown the reactor and maintain it in a shutdown state indefinitely (reactivity control);

– Remove decay heat during the shutdown period (decay heat removal);

– Maintain safety related monitoring and control functions concurrent with the LOSP and failure of SSC not credited to perform their design functions.

– SSC required providing containment and confinement functions (if requested by the regulatory body).

4.2 The Seismic Safety Margin

The seismic safety margin can be assessed by adopting two methods:

– Seismic probabilistic risk assessment (SPRA), described in the previous sect. 2;

– Seismic margins assessment (SMA).

As indicated in WENRA, 2011 document, the Fukushima accident revealed various gaps in the PSA (Probabilistic Safety Assessment) approach used to determine the plant risk assessment. This makes some issues to be re-considered and/or improved in the PSA application and state of practice: these include, for instance, PSA for external events.

A crucial initial finding in the post-accident assessment was that the tsunami risk, made by applying PSA approach, had been underestimated, therefore an integrated approach, like the IDPSA, linking/coupling the frequency of event to the consequences (capability to address the mutual interactions between the failures of the equipment and the response of the plant, e.g. transient behaviour of plant) must be taken into account.

IDPSA is a complementary tool that can provide a strong interconnection between deterministic and probabilistic procedures in order to correctly evaluate the performance of plant, as shown in Figure 3.

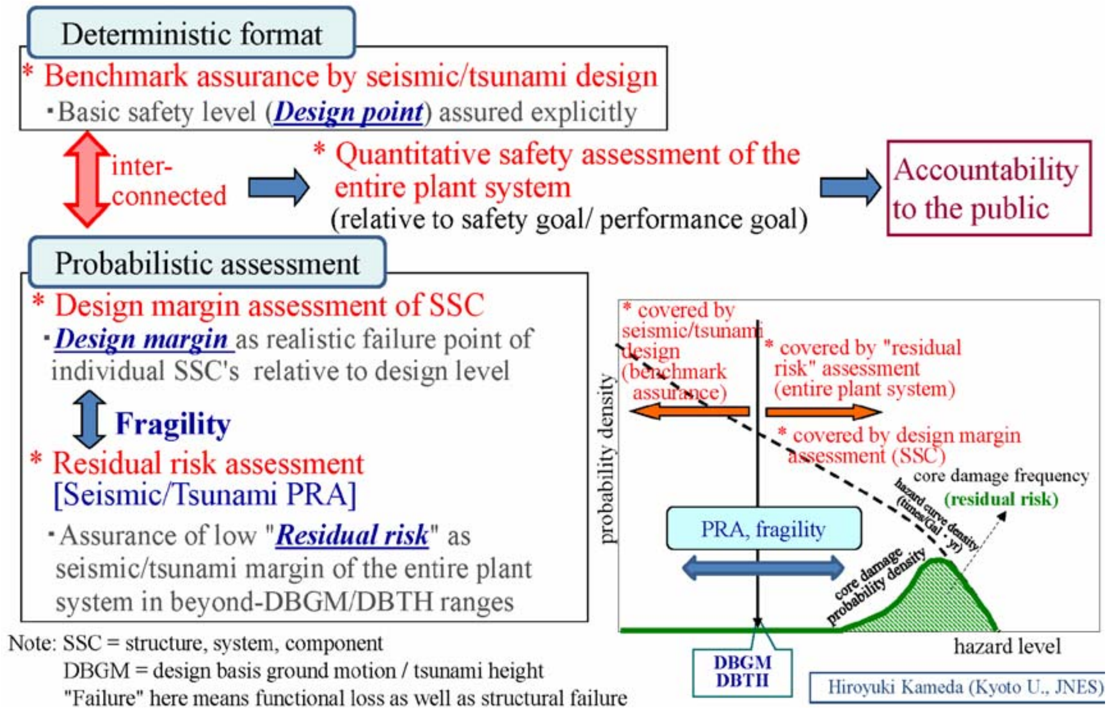


Figure 3: Scheme of seismic/tsunami safety assurance.

In order to analyse and calibrate the durability and robustness of the safety evaluation process, which are used by the nuclear utility to provide assurance of the safety of the public and the environment in the operation of Nuclear Power Plants (NPPs), post-accident assessments are of a meaningful importance since they have (allow) revealed scenarios that were not considered in the safety analysis.

This is the case of the post-accident assessment of the Fukushima accident. This accident is being studied with confidence that such issues will be uncovered and corrective actions taken to improve global safety.

From what is known to date, such a type of external event was the result of a combination of two external hazards initiated by an earthquake and the ensuing tsunami, specifically it has been the combination of:

1) an exceptional magnitude 9.0 earthquake, which caused the sudden loss of almost all the off-site power supply.

The effects on the site were measured in the basements of the six reactor units at between 0.33g and 0.56g peak horizontal acceleration: no evidence of any ground rupture on the site or of any liquefaction has been observed; although the on-going situation at the plant has prevented detailed inspection of many of the structures and systems, it is also clear from the limited images available from inside the plant that there was peripheral damage to items such as control room ceilings etc.

2) A tsunami that caused the flooding and the rapid inundation of site (Figure 4), resulting in both the loss of all the emergency power supply systems and heat sink. The extremely long wavelength (and consequently period) of tsunami waves means that the site remained inundated for a period of between 30 minutes and an hour following the main wave arrival.

(a)

(b)

Figure 4: View of the Fukushima Dai-chi reactor flood elevation (a) and inundation (b) (Source: Tepco©, 2011).

Based on TEPCO information, the heights of both the tsunami and the seawall to the baseline, level known as the Onahama Port Base Level (OP), known that the height of the sea wall (i.e. flood protection measures) was set at OP +5.7m, are: the general ground level adjacent to the

water's edge is at OP +4m; the ground level adjacent to the turbine building and the reactor building is at OP +10m (Reactor Units 1-4) and +13m (Reactor Units 5-6); the estimated height of the tsunami wave is at about OP +14÷15m.
The incoming wave completely surrounded the buildings on-site, and entered the buildings via ground level access doors. There are no details as yet over any protection measures that may have been available to prevent or limit the ingress of water into the buildings. The turbine hall and the reactor buildings have significant portions below ground level, and it is fair to assume that considerable volumes of water entered the lower portions of the buildings.

4.3 Case study

The earthquake and the tsunami hazards, till the 2011, were normally considered separately (seismic and flooding) during the design of a facility: in the case of the Fukushima accident they occurred sequentially and caused, as consequences of the unavailability of electrical power and the ultimate heat sink, the loss of the reactivity control, heat removal and containment integrity were lost (unmitigated accident progression).
While the hazard related to the earthquake are widely known, those ones associated with flooding events to be examined are:

1) Inundation, rise in water level at the site;
2) Hydrodynamic forces on structures;
3) Clogging of water intake and outlet due to sedimentation and debris.

These hazards, related to the inundation phase of the tsunami, must be evaluated, of course, when dealing with the safety assessment of NPP [10] and the design safety margin.
In this framework, although the outcome of the probabilistic method is the hazard curves which define the flood level function of annual frequency of exceedance, the outcome of the deterministic approach is the evaluation of the maximum flood level and its consequences; these results are used as input for design (to design protection against external flood) and for safety analysis.
Moreover to determine the deterministic SMA the safety significant components are selected to assemble the success path and, thus, to ensure the performance of the fundamental safety functions including dependencies of support systems and interactions with non-safety related

SSCs. In this case, only a short term assessment is valid; the components or structures to take into account are selected on the basis of conservative assumptions like [10]:

- Offsite Power is not available (power supply is provided only by the diesel generators);
- Station Blackout (emergency power is not available; all diesels failed to supply auxiliary and emergency power);
- Loss of normal heat removal path.
- Loss of Ultimate Heat Sink, etc.

With reference to the Fukushima accident, the main common technical issues arisen are summarized in Table 2.

In what follows, the technical issue, which are enclosed in the red box of the above mentioned table, related to the design against flooding have been considered. Specifically an applicable case aimed at the evaluation of the reliability of an existing plant subjected to a combination of an earthquake and the ensuing tsunami is presented and discussed.

Table 2 - Common technical issue from Fukushima accident [20].

Common issues	**Fact**	**Methodology**
Earthquake and tsunami source	Largest earthquake in history and much larger than forecasted.	Plate tectonics Long-term forecast, etc.
Large tsunami beyond design	Tsunami 2-3 times of design	Tsunami database Probabilistic tsunami hazard assessment Design-extension conditions
Multiple Hazard	- Earthquake plus following tsunami - Main shock plus many aftershocks	Link of Seismic PSA and Tsunami PSA
Planning and design against flooding	- Tsunami inundation and run-up with dynamic force - Damage to plant, equipment, buildings, etc.	- Design against flooding - Protection, redundancy against hydro-dynamic forces and fluidity - Water proof/ tightness - Prepare back up power

		- Diversity of heat sink - Power tie line, etc.
Robustness of off-site power sources	- LOOP - Switchyard lost function - Feed line tower fell down	- Robustness of switchyard - Off-site power grid
Spent fuel cooling	Loss of cooling function	Heat-sink and monitoring system
Accident management	- Hydrogen explosion - Under radiation working - Carry in pump and power - Huge amount of contaminated water arising from external injection	- H2 recombiner, igniter - Containment venting - Post-accident monitoring - Preparedness and training - Robustness of AM system, - Support system
Safety system design	- Common mode (tsunami) cause 4 unit accident - Prolonged station black-out and loss of ultimate heat sink	- Defence in depth design - Diversity, redundancy - Enough time grace in AM and supporting system - Seismically-induced events
Communication to public	Prompt evacuation direction	

5. Safety Assessment Methodology

Two types of methodologies are generally available for safety assessment against external hazards:

- Deterministic safety assessment aimed at evaluating the failure capacity (beyond design basis) of the success path.
  The success path is defined by the availability of SSCs required to perform the fundamental safety functions and to bring and maintain the NPP in a safe shutdown state.

- PSA aimed to evaluate the contribution to all possible accident sequences and scenarios induced by external IEs.

The S-PSA and SMA methodologies are mature after 30 years of development and applications. In most applications of the SMA procedures the components to examine are limited to SSCs which make up a Reactor Safety Shutdown success path including alternate redundant success path.
The implementation of such a procedure foresees mainly the:

– Definition of the performance criteria;
– Plant Walkdowns;
– SMA by Analysis (component and plant level);
– Identification of vulnerabilities and recommended resolution, etc.

The end results of the SMA include:

- Seismic safety margin at component level and plant level (this aspect ins investigated in the following with reference to the seismic event followed by a tsunami);
- Seismic vulnerabilities;
- Plant improvements (seismic upgrades of SSCs).

The prime objective of the assessment is to evaluate the robustness of the existing plant in terms of design features and procedures against the impact of extreme events (rare events of low probability of occurrence) focusing on fulfilment of the fundamental safety functions of criticality control, residual heat removal, and confining the radioactive material.

As an example, for what the concerns the flooding effects, the (major) impact on the safety of the plant that means on the outer containment building, that represents the last barrier against the radioactive release to the population and the environment, have to be studied: furthermore, the presence of water in many areas of the plant may be a common cause failure for safety related systems, such as the emergency power supply systems or the electric switchyard, with the associated possibility of losing the external connection to the electrical power grid, the decay heat removal system and other vital systems.

In consideration of that, the safety margins (obtained by performing steady state or transient analyses) have been determined, analytically or numerically; the results have to be compared to the plant design conditions in order to ensure that the operation of a nuclear power plant can be carried out with adequate levels of safety in all modes of operation and at all times. The basic concept is to determine, in a comprehensive and systematic way, the limiting values, which, if exceeded, could lead to an undesirable state (this acceptance criteria can be used also for the beyond design basis events). This methodology may allow to identify the specific plant vulnerabilities and to plan the actions needed to improve the plant safety (mitigation actions).

The assessment hereinafter presented (applicative case) has in its scope the consideration of the accident scenario originated by a rare event, like an earthquake coupled to a following tsunami, that may cause extensive damage to safety systems and challenge the fulfilment of the fundamental safety functions. It is considered that the accident scenario involves simultaneous failures of containments that could lead to fuel damage.

Finally as indicated in [10], the assessment should comply with the relevant safety standards, such as:

- NS-R-1 (to be replaced by SSR-2.1, in publication): Requirements for Safety of Nuclear Power Plants: Design [11];
- NS-G-1.8: Design of Emergency Power Systems for Nuclear Power Plants [12];
- NS-G-1.9: Design of the Reactor Coolant System and Associated Systems in NPPs [13];

- GSR Part 4:Safety Assessment of Facilities and Activities [14], etc.

The margin of compliance is the aspect of interest, for assessing which a sequential and progressive loss of the available supply possibility should be postulated and analysed deterministically, irrespectively of its probability.

6. Assessment Of The Fundamental Safety Functions

The assessment of the safety functions will consist of a verification of the lines of defence in depth following a postulated extreme event in the plant that will affect two safety system support functions: the power supply and the ultimate heat sink (this does not refer to the intermediary heat transport systems to remove heat from the core).

To be conservative the loss of the fundamental safety functions has to be considered in the evaluation of the plant performance (to examine the overall plant integrity).

In this regard, it is important to note, that when a plant has several units, the postulated accidental conditions would affect all units and facilities at once, and that degraded conditions could exist for the implementation of alternative power supply or cooling methods, particularly those that entail measures that have not been foreseen in the plant design.

In addition, the safety assessment should reflect the strengths of the plant design in terms of features relevant to cope with stepwise losses of supply sources and functionality: it is important to determine the limiting situations that could arise for accomplishing the safety functions when supplies fail (cliff edge effects) and the measures that are already in place or could be implemented to avoid reaching these situations (at least the core fuel damage).

The assessment of the fundamental safety functions has to account for the failures that could be originated by the extreme events postulated. Therefore in the light of the events occurred at Fukushima or any other extreme event that could be postulated at the site, the analysis should consider that offsite power supply could not be restored for a long period of time.

The available power supply sources (both the operational and the emergency systems) as well as the additional back up equipment need to be identified (despite of redundancy and diversity) and considered unavailable for a long period of time, as a result of the consequences of the occurrence of the extreme events on site and off site.

The "islanded mode" of plant, that as indicated in [10] identifies "*the automatic disconnection from of the plant generator from the grid and rapid reduction in reactor output to supply power for the NPP unit's own consumption*", can be considered in the short term safety assessment; the plant has to demonstrate the extreme event would not affect its capability/ fundamental safety functions.

Furthermore it is important to note a safety assessment must be based on the current status of the existing NPPs.

Finally to have an appropriate evaluation of the robustness of the plant and identification of the damage caused to the safety functions (the engineering system of the defence in depth) the most limiting scenario taking into account the overlapping effects of SBO, loss of ultimate heat sink and plant isolation, should be considered.
In what follows (par. 6.1), this scenario, consisting of an existing Generation II plant assumed to be affected simultaneously by an earthquake and tsunami events, is considered and analysed. This postulated scenario would allow to identify the possible limiting situations that could arise, and the additional measures that should be incorporated in the design to increase its robustness.

6.1 Safety Assessment Of An Existing Generation II Plant Subjected To Tsunami And Earthquake Events: Applicative Case

To the aim of the safety margin assessment [16] (deterministic evaluation), the dynamic behaviour of Gen. II reactor, 60 m tall and 42-45 m in diameter, hit by natural external events of a very low probability that can lead to very high consequences and significant potential for cliff edge effects, is studied according to the step 1 of the procedure showed in the previous Figure 3.
This assessment would also support the re-evaluation (adequacy) of the accident management programme to account for the potential weaknesses of plant conditions for extreme events.
The motivations to perform seismic safety evaluations of existing nuclear installations are many; in particular the IAEA guidelines are followed: " Systematic safety reassessments of the plant in accordance with the regulatory requirements shall be performed by the operating organization throughout its operational lifetime, with account taken of operating experience and significant new safety information from all relevant sources…"[10].
Above all, in the applicative case herein presented, they are associated to the new experienced high magnitude of the actual earthquakes and the observed performance of SSCs. These components have to provide confidence that there is no 'cliff edge effect'; i.e. if earthquake ground motion with or without ensuing tsunami occurs at the site, and to demonstrate that no significant failures would occur in the installation.
To determine and identify plant-specific vulnerabilities, due to external seismic and flood hazards, and other important insights, numerical simulations have been carried out with

adequate numerical tools taking into account the degradation mechanisms of the plant structures.

The main outcome of these analyses will be the specific response spectrum of the plant, (site condition) representing maximum potential hazard level. To take into account (and to cover) appropriately the uncertainties related to the evaluation process, a conservative procedure has been adopted.

6.1.1 Tsunami hazard

For earthquake induced tsunamis, the flooding hazard (run-off) can be assessed by using either a deterministic hazard analysis or a probabilistic hazard analysis.

In this evaluation, as already mentioned, a deterministic hazard analysis has been carried out.

The numerical simulation is performed based on the following steps:

a) select the largest historical tsunami in the near field and far field that could affect the site region;
b) identify (and validate) the corresponding run up height near the site and the hydrodynamic forces;
c) select the maximum and minimum water levels;
d) construct and validate a numerical model on the basis of records of observed historical tsunamis;
e) execute the numerical model including the propagation process for the selected tsunami.

When considering a tsunami event, the analysis of historical data (all past events, experienced on site, and initiating causes [17][18] is of meaningful importance to evaluate the elevation of water waves and the intensity of the hydro-dynamic forces exerted by the breaking water waves.

Indeed, the maximum flood for a given site may result not from the occurrence of one extreme event but from the simultaneous occurrences of more than one severe event, each of which is less severe than the resulting combined extreme event. In consideration of that and of the data summarized in Table 3, it is possible to demonstrate that 16 large tsunamis, caused by undersea earthquake, with amplitudes of at least 10 m occurred in past 513 years (indicated as in [17][18]), that means that one event once thirty years occurs in Japan. The experienced frequency resulted:

$$f = \frac{16}{513a} \cong 0.03a^{-1} \qquad (14)$$

This result allowed to determine the bounding values for the maximum water level at shoreline and, as a consequence, the run-up height, the inundation horizontal flood and the maximum water level at the plant site.

The flooding phase had been analysed in this applicative case: the inundation soon after the run-up of water waves near the shore: the water waves shorten in wavelength and increase in height amplitude [18] as a consequence of the reduced sea bed slope. At the shoreline, in fact the kinetic energy of the travelling sea waves is converted into potential energy determining the increase in height amplitude.

Table 3 - Large tsunamis with high amplitudes.

Date	Affected Region	Earthquake [1])	Tsunami [2])
11.03.2011	Japan	M = 9.0	23 m
04.10.1994	Kuril Islands	M = 8.3	11 m
12.07.1993	Sea of Japan	M = 7.7	31.7 m
26.05.1983	Noshiro	M = 7.7	14.5 m
07.12.1944	Kii Peninsula	M = 8.1	10 m
02.03.1933	Sanriku	M = 8.4	30 m
01.09.1923	Tokaido	M = 7.9	12 m
07.09.1918	Kuril Islands	M = 8.2	12 m
15.06.1896	Sanriku	M = 7.6	38 m
24.12.1854	Nankaido	M = 8.4	28 m
29.06.1780	Kuril Islands	M = 7.5	12 m
24.04.1771	Ryukyu Islands	M = 7.4	85 m
28.10.1707	Japan	M = 8.4	11 m
31.12.1703	Tokaido-Kashima	M = 8.2	10,5 m
02.12.1611	Sanriku	M = 8.0	25 m
20.09.1498	Nankaido	M = 8.6	17 m
Resulting Actual Design Basis		**M ≈ 7.5**	**> 10 m**

The impact of the water waves and debris at the plant site is primarily responsible for the massive and/or catastrophic damages of the in-site buildings, infrastructures, environment and

of fatalities: these consequences are related particularly to the flooding phase of the tsunami. The amplitude of waves may be much higher than local still-water depth [19].
An overview of historical tsunamis and casualties caused is also shown in Figure 5.

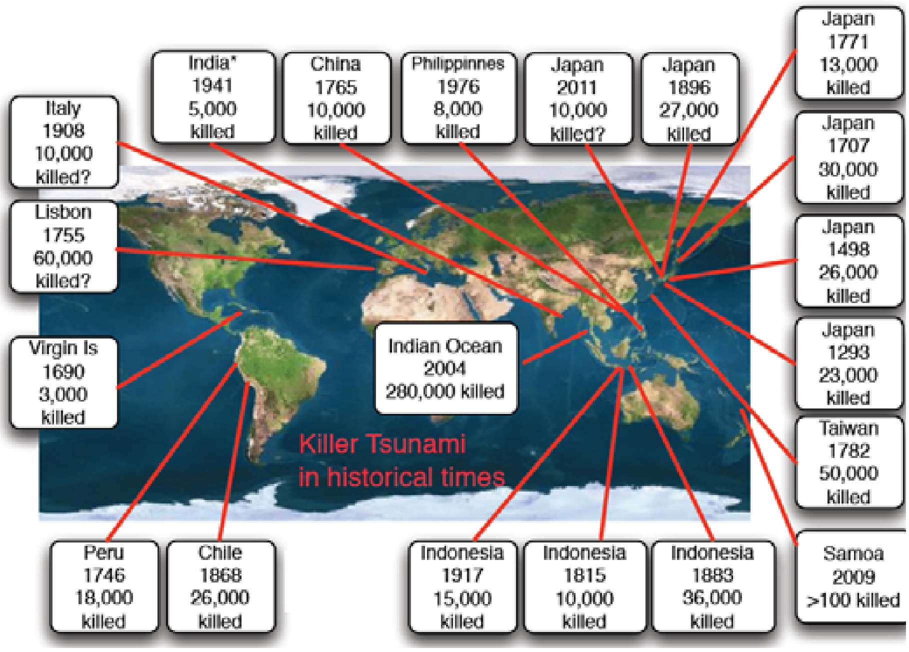


Figure 5: Overview of historical tsunamis.

The impact and lateral pushing of the waves and the destructive power of a large volume of water dragging debris and missiles inland, responsible of plant damages (like observed in Fukushima plants) must be considered.
These dynamic forces have been represented, in the simulations performed, like pressure loads acting on the outer walls of the plant containment building. The pressure loads are calculated on the basis of breaking-waves height according to the probabilistic studies and historical data (Table 3).

6.1.2 Modelling of the Containment

As described in [19] and [20] a typical Gen. II Containment Building, like the one represented in Figure 6) is 60 m tall and 42-45 m in diameter and is about 1 m thicker.

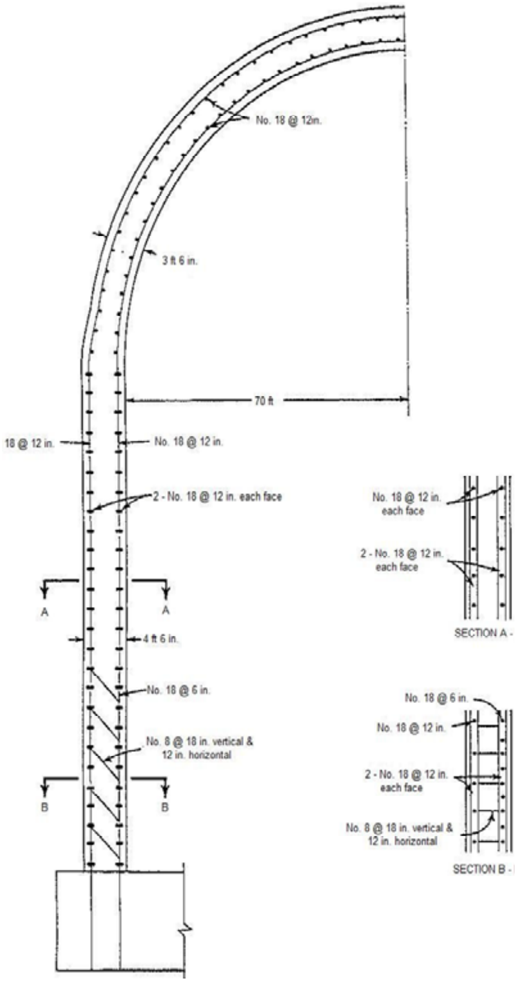


Figure 6: Distribution of steel bars in the containment wall [19].

It is characterized by heavy and thick reinforced concrete walls and designed to not exceed the allowable stresses, in agreement with the ACI Standard 359 [21],[22].

The determination and assessment of hazard weakness of such a type of plant entails unavoidably with:

- the structure configuration of the system or component in relation to its function;
- its potential failure mode (ductile or brittle, large displacement, vibration sensitivity, unacceptable function even though stress or displacement is within acceptable limits;
- generic performance during past similar hazards, if available it may allow to validate the model and methodology
- the actual support conditions of the system or component.

Based on the description of structure given in [20], a quite detailed model (Figure 7) has been set up and implemented, by MSC©Marc code [23], considering the same reinforcing steel bars distribution, horizontally and vertically oriented, as shown in Figure 6 (in agreement with ASCE rules and ACI standards requirements). The foundation was assumed based on a rock soil (“rigid foundation”).

The dynamic behaviour of containment building was simulated by assuming large deformation and implementing an updated Lagrangean formulation. The behaviour of concrete was assumed to be linear elastic up to the point of failure; progressive failure or damaging under increasing loads was also considered by implementing the maximum stress criterion. The reinforcement members of the concrete walls were made mainly of steel bars (distributed like in Figure 6) of A 615 grade 60; this material was assumed to be elastic-perfectly plastic behaviour.

The numerical model set up for the aim of this study is represented in Figure 7.

It was made of more than 68.000 solid element: the concrete structure was modelled by using SOLID-3D elements, the internal structures (i.e. the reactor vessel) 3-D thick shell and the steel reinforcement bars by means of discrete rebar elements that were set up by TRUSS-3D elements.

The mathematical model (direct integration method) and the degree of discretization have been chosen such that the natural behaviour of the structure, in the relevant frequency range, could be computed with good reliability.

To assess the vulnerability of the systems and components against analysed hazard, each safety system or component have a qualitative vulnerability rating which, when combined with the system safety significance, can provide the assessment of the relative risk associated with the external event induced hazards.

In this assessment the acceptance criteria will be based on the acceptable limit stress and displacement.

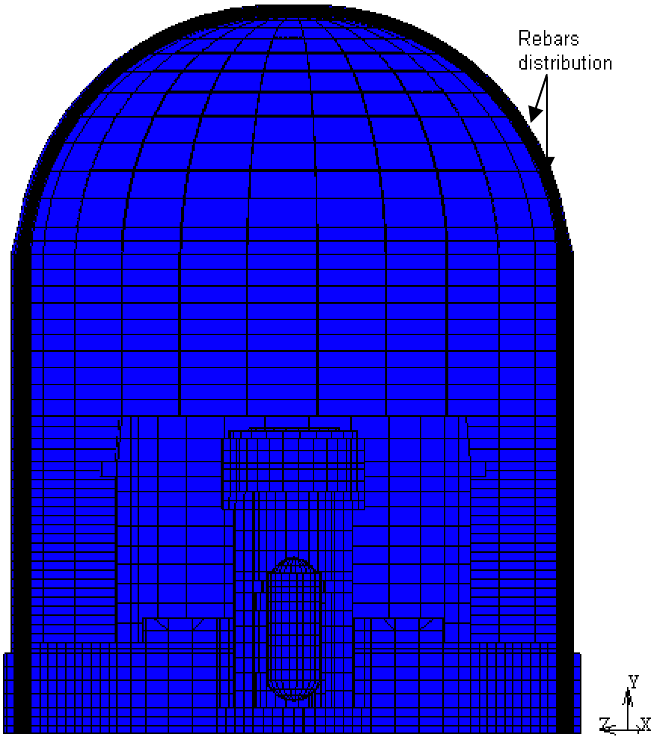


Figure 7: Containment model.

6.1.3 Input loadings

The definition of the level of earthquake for the SMA (the screening level does not correspond necessarily to a new design basis earthquake) is required at initiation of the evaluation, but it could be not dependent on the results of a Probabilistic Seismic Hazard Analysis-PSHA. The purpose is to define a screening level in the evaluation process: the earthquake plus tsunami magnitude is used to evaluate whether the existing nuclear installation can perform safely during and after the earthquake ground motion occurs at the site.

To provide seismic demand for SSCs, it is necessary to develop Floor Response Spectra (FRS), for each elevation in each important structure or components of the containment building, that represent the seismic input for each SSC's that requires seismic fragility calculation.

In developing realistic floor spectra, it is typical to use dynamic analysis for the structure and to account for non-linear effects by estimating the inelastic energy absorption capacity of each component: the floor response spectrum is so modified to account for how each component responds in frequency space.

These modifications account for several factors, such as damping, modal response combination, etc. all of which have variability which is included in the analysis and adjourned automatically in the numerical simulations (dynamic transient analysis).
To perform the transient analyses, aiming at the examination the safety assessment of the plant containment postulated, it was assumed the occurrence of an earthquake of 0.3 g peak ground acceleration (PGA) followed by a tsunami of breaking wave height of 20 m.
The seismic motion (of 24 s duration) is represented by means of three independent acceleration time histories - ATH (Figure 8) [24][25], two along the horizontal directions (Ax and Az) and one in the vertical direction (Avert).

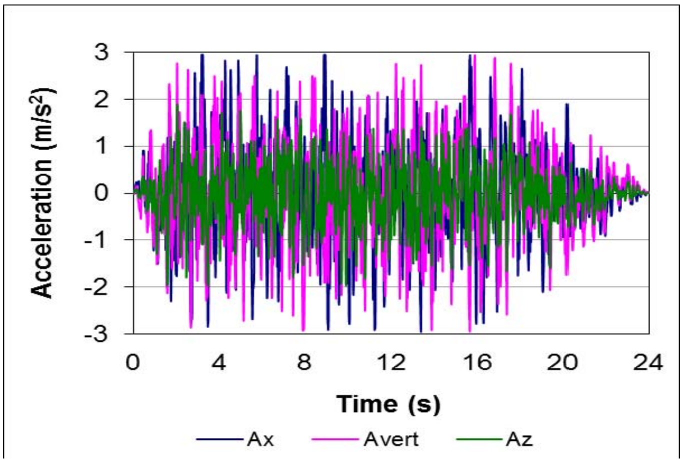


Figure 8: Input Seismic motion.

The inundation phase of tsunami is assumed to follow the earthquake; it is represented by means of pressure loads calculated on the basis of the following equation [19]:

$$P_{\max} = C_p \gamma_w d_s + 1.2 \gamma_w d_s \tag{15}$$

where γ_w is the water unit weight, equal to 10.05 kN/m^3 for the sea water; Cp the dynamic pressure coefficient, defined in table 5.4-1 of [19] the value of which depends on the category of risk associated with the extreme natural events considered, i.e. the flooding and earthquake. ds is instead the still water depth at the base of building impacted by the waves.
The directions of the application of the pressure loads are shown in Figure 9: they have been assumed to be orthogonal and tangential to the outer containment walls.

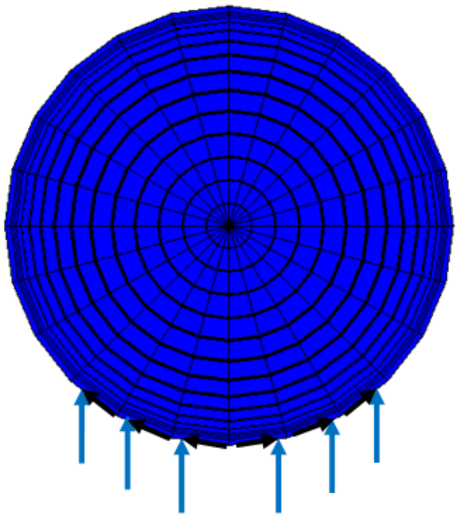


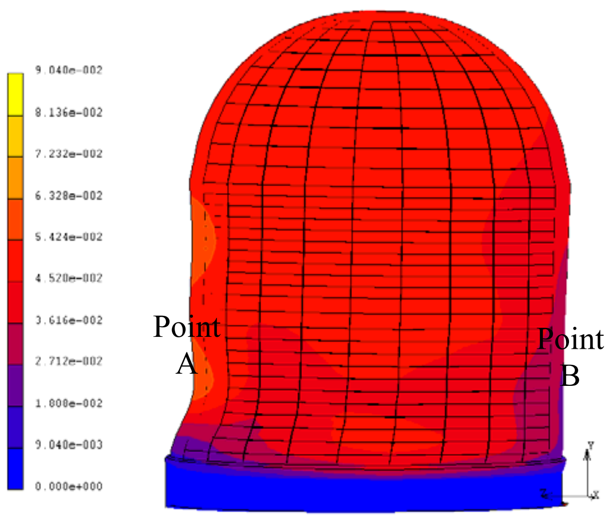
Figure 9: Directions of the pressure breaking-wave.

6.2 Analysis Results

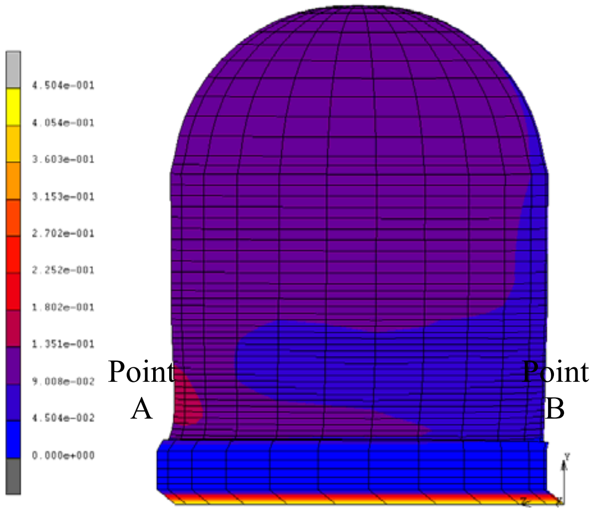
Analysing the obtained results it was observed an amplification of acceleration values along the containment height, as also foreseen in other study: for the structure analysed at the steam generators restraints (about +30 m from the ground level) they increased of about 30%.

It was also observed an overall wall displacement (Figure 10) greater on the surface containment subjected to the impact of water waves (though not linearly) (point A) than the opposite one (point B) Figure 10.

At point A, the maximum displacement, calculated at 4 s (after first peaks of accelerations) resulted about 30 cm (Figure 10 b) .



(a)



(b)

Figure 10: Overall displacement in the concrete body at the impact of the first breaking waves (a) and at 4 s transient (b).

The displacement values (about 30 cm) indicated that the containment wall is undergoing local damages. This has been confirmed by the stress values (Figure 10 and Figure 11): local stresses at point A are closer the allowable stress beyond which structural damaging/failure effects may come out. Also at point C located on the inner CB surface, the stress distribution indicated that the allowable limit is overcome.

Moreover, the impact area (point A) is under compression stress, while away (in longitudinal and/or circumferential directions) from it tensile stresses have been observed.

However, in general, the compressive stresses have been found to be higher and predominant compared to the tensile ones in the outer surface.

Finally as a consequence of the high local compression loads (point A), the containment structure develops outwards bulging (see Figure 10b).

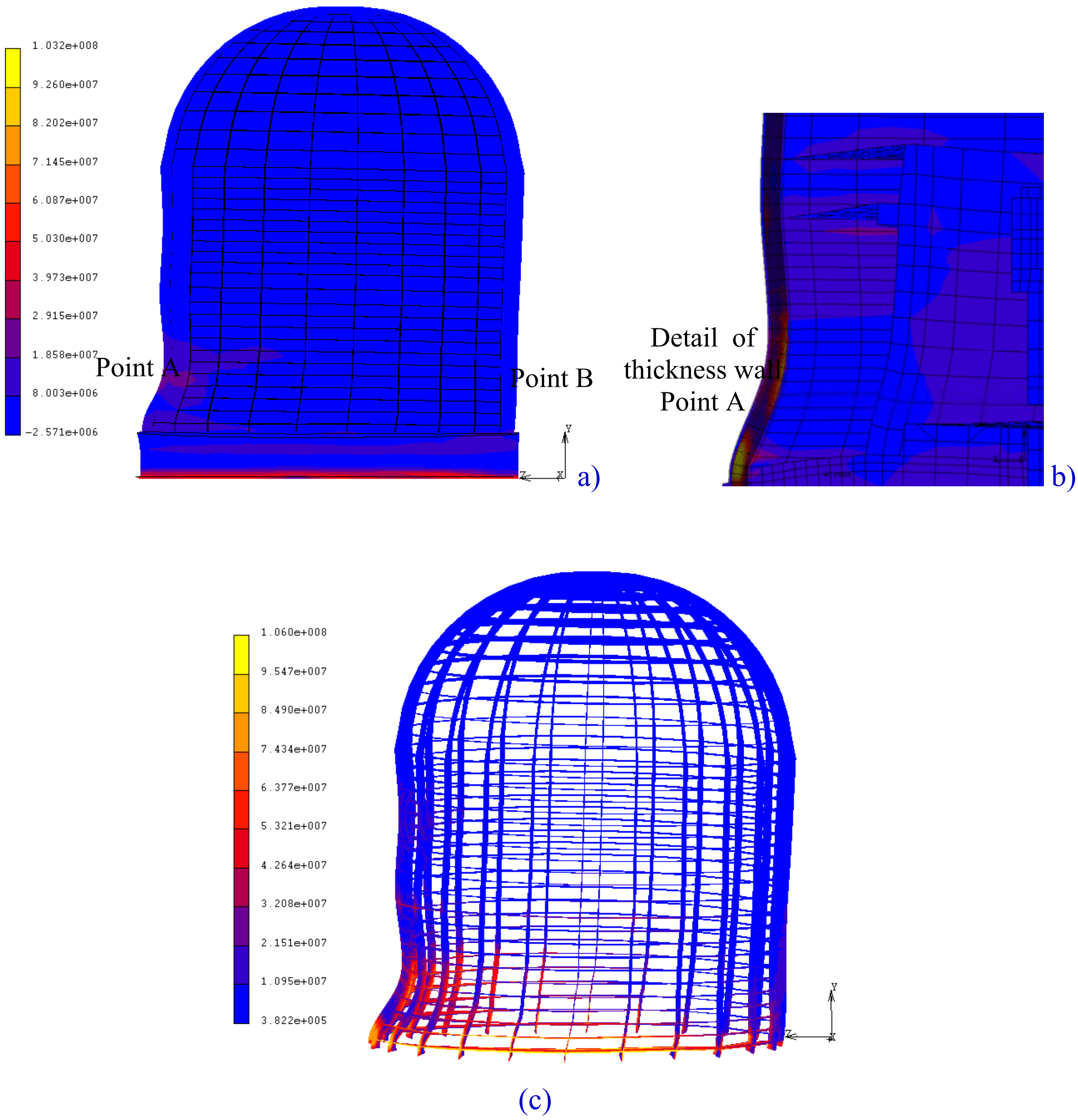


Figure 11: Stress behaviour in the concrete body at the impact of the first breaking waves (a), in the wall thickness (b) and reinforcing steel (c).

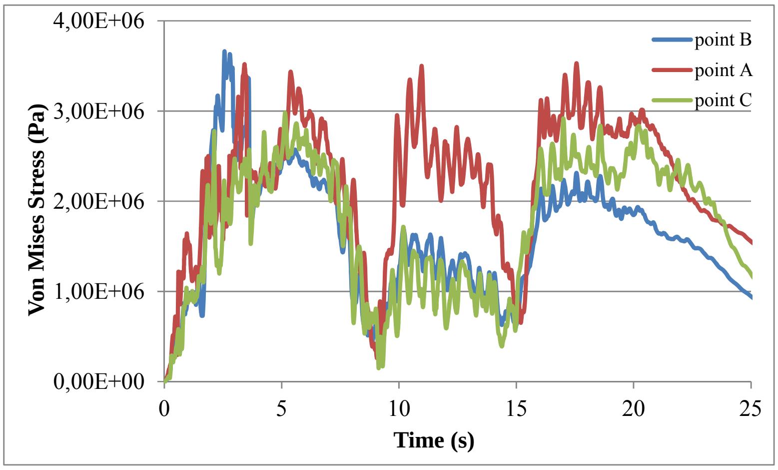


Figure 12: Von Mises stress in the concrete wall.

The stress values (with geometry and dimensions assumed) indicated/confirmed that the containment suffers local damages/failure phenomena (Figure 13): even if suffering void nucleation (cracking), the outer reinforced walls have/retain sufficient strength capability to withstand waves impact without jeopardize the overall plant integrity.

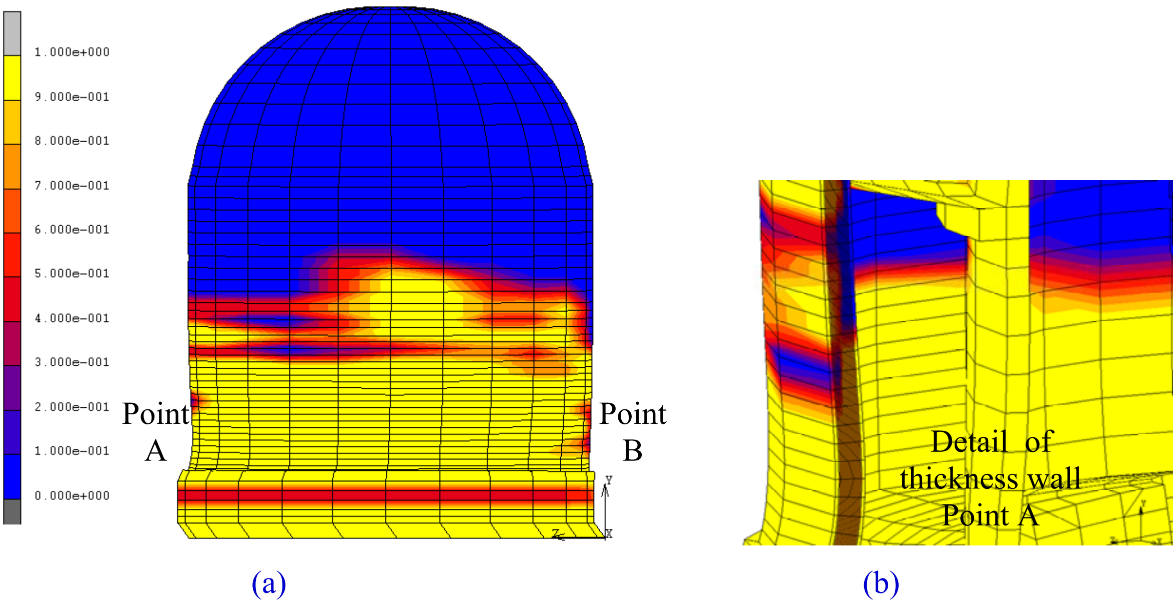


Figure 13: Damage 3 s after the impact of the first waves in the: outer surface (a) and wall thickness (b).

The most serious damages and the major potential issues appeared to be caused by the tsunami (i.e., failure of components) rather than the earthquake event, in agreement with what has been observed during the Fukushima accident scenario.

The results obtained indicated that the demand significantly exceeds the design demand in some part of the outer walls of the containment building.

The stress values exceeds the allowable limit stress, particularly in the area subject to tension: these vulnerabilities could result an early failure of containment functions including the containment integrity, the containment isolation, etc. However, it is worthy to note that the material degradation effects are limited by steel rebars.

On the basis of the results previously presented and discussed, it was possible to determine the weakness of the plant examined. The structure configuration of the containment building highlighted potential failure mode, with large displacement of about 30 cm which are unacceptable. The actual support conditions of the internal system or component, as implemented, seem within the acceptable limits (e.g. stress or displacement, anchorage capacity), in line with the acceptance criteria for the SMA.

In conclusion it is possible to say that the overall existing nuclear installation (away from the water waves impact area) can perform safely during and after the earthquake ground motion plus tsunami occurs at the site.

Moreover, with reference to the accident management, the actions needed to improve the safety of the plant are mainly the adoption of sea walls, shelter or items capable to avoid a direct impact of waves on the plant, the positioning of the power supply sources (operational and emergency systems) at a height much greater than the design flood elevation - DFE - so to prevent the malfunctioning and the non-availability at operation for a long period of time.

To provide seismic plus flooding demand for SSCs that requires fragility calculation, further analyses are necessary on the basis of the Floor Response Spectra (FRS), developed for each important equipment, and pressure loads (vs. DFE).

7. Conclusions

The issues emphasized within the present study are to be tackled to use the results of the PSA appropriately in future risk-informed decision making processes.

Focus should be on the risk itself, rather than just frequency, and all risk contributors are to be covered appropriately as far as possible in a consistent and exhaustive manner.

In order to solve some incompleteness issues, research on extreme external hazards, risk assessment of the spent fuel pool and site risk is required. These are the emerging issues after the Fukushima accidents.

In particular, risk assessment of external hazards is required and utilized as an integrated part of PRA for operating and new reactor units. In the light of the Fukushima accident, of special interest are correlated events, whose modelling is proposed in the present study, in the form of some theoretical concepts, which lay the foundations for the PSA framework implementation.

An applicative example is presented for illustrative purposes, since the analysis is carried out on the basis of generic numerical values assigned to an oversimplified model and results are achieved without any baseline comparison. Obviously the first step aimed at the process endorsement is the analysis of all available information in order to determine the level of applicability of the observed specific plant site events to the envisaged model and the statistical correlation analysis for event occurrence data that can be used as part of this process.

Despite these drawbacks that actually do not qualify the achieved results, the present work represents an exploratory study aimed at resolving current open issues to be resolved in the PSA, like topics related to unanticipated scenarios: the combined external hazards of the earthquake and tsunami in Fukushima, external hazards causing internal events, such as seismic induced fire.

These topics are to be resolved among the other ones as emerging from the Fukushima accident, in order to endorse and make more effective the risk assessment process.

With reference to the IPDSA approach, a numerical methodological has been adopted to assess the safety margin of an existing containment structure (even if simplified).

The treatment allowed to evaluate, with a good reliability, the effects caused by an earthquake ensuing by a tsunami, specifically the effects triggered during the inundation phase by the water waves (particularly by the breaking waves), which had been expressed in terms of pressure values.

The global dynamic performance of plant containment structure is examined by implementing detailed numerical analyses (with FEM model) assuming real geometry (steel reinforcement distributed according to the ASCE requirements), materials behaviour and constitutive laws. Analysing the obtained results it was observed that:

1) the acceleration values were amplified along the containment building height up to reach greater values especially in the dome section, where, it can reach 2-3 times the input PGA;
2) the containment wall is undergoing local high stress values (outer building surface-point A) exceeding in the impact area the allowable one beyond which structural damage and progressive failure may come out;
3) the material degradation effects are limited by steel rebars;
4) the overall displacement induced by the dynamic loads (earthquake plus tsunami inundation) resulted about 30 cm;
5) no loss of structural integrity was highlighted due to the positive action of the reinforcing steel rebars that provided the adequate stiffness and load bearing capability to absorb the seismic energy.

On the basis of the results previously presented and discussed, it was possible to determine the weakness of the plant examined. The structure configuration of the containment building highlighted potential failure mode, with large displacement of about 30 cm which are unacceptable. The support conditions of the internal system or component seem instead within the acceptable limits (e.g. stress or displacement, anchorage capacity).
Moreover, with reference to the accident management, the actions needed to improve the safety of the plant are mainly the adoption of sea walls, shelter or items capable to avoid a direct impact of waves on the plant, the positioning of the power supply sources (operational and emergency systems) at a height much greater than the design flood elevation - DFE - so to prevent the malfunctioning and the non-availability at operation for a long period of time.
To provide seismic plus flooding demand for SSCs that requires fragility calculation, further analyses are necessary on the basis of the Floor Response Spectra (FRS), developed for each important equipment, and pressure loads (vs. DFE).

References

[1] American National Standards Institute (ANSI), External Events in PRA Methodology, ANSI/ANS 58.21-2007, 2007.

[2] American Society of Mechanical Engineers (ASME), Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, AMSE/ANS RA-Sa-2009, 2009.

[3] US Nuclear Regulatory Commission, PRA Procedure Guide: A Guide to the performance of Probabilistic Risk Assessments for Nuclear Power Plants, NUREG/CR-2300, 1983.

[4] J. J. Bevelacqua, "Nuclear Regulation in the United States and a Possible Framework for an International Regulatory Approach", International Nuclear Safety Journal, Vol.1, No.1 (2013)

[5] K. N. Fleming, "On the Issue of Integrated Risk – A PRA Practitioners Perspectives", Proceedings of the ANS International Topical Meeting on Probabilistic Safety Analysis, San Francisco, CA., USA, Sep. 11-15, 2005.

[6] IAEA Safety Series No.118, Safety Assessment for Spent Fuel Storage Facilities, IAEA, Vienna, 1994.

[7] T. E. Collins and G. Hubbard, Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, NUREG-1738, U.S. Nuclear Regulatory Commission, January 2001.

[8] Joon-Eon Yang, Development of an Integrated Risk Assessment Framework for Internal/External Events and all Power Modes, Nuclear Engineering and Technology, Vol.44 No.5 June 2012

[9] Burgazzi L., Addressing the challenges posed by advanced reactor passive safety system performance assessment, Nuclear Engineering and Design, 241, pp. 1834-1841, May 2011.

[10] IAEA, A Methodology to Assess the Safety Vulnerabilities of Nuclear Power Plants against Site Specific Extreme Natural Hazards, 2011.

[11] IAEA, Safety of Nuclear Power Plants: Design, IAEA Safety Standard Series No. NS-R-1, IAEA, Vienna, 2000 (to be replaced by SSR-2.1 in publication).

[12] IAEA, Design of Emergency Power Systems for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.8, IAEA, Vienna, 2004.

[13] IAEA, Design of the Reactor Coolant System and Associated Systems in NPPs, IAEA Safety Standards Series No. NS-G-1.9, IAEA, Vienna, 2004.

[14] IAEA, Safety Assessment for Facilities and Activities, IAEA General Safety Requirements Part 4 No. GSR Part 4, IAEA, Vienna, 2009.

[15] WENRA, Report Safety of new NPP designs, WENRA RHWG 2013.

[16] American Society of Civil Engineers (ASCE), Structural Analysis and Design of Nuclear Plant Facilities, ASCE 58, 1980.

[17] G. Forasassi, R. Lo Frano, V. Baudanza, Preliminary evaluation of a severe flooding effects in an innovative SMR, 2nd International Technical Meeting on Small Reactors, Ottawa, Canada, November 7-9, 2012.

[18] US NRC, Tsunami Hazard Assessment at Nuclear Power Plant Sites in the United States of America, Final Report, March 2009.

[19] R. Lo Frano, G. Forasassi, Preliminary evaluation of the reliability of Gen II or III reactors in BDBE conditions Proceedings of the 2012 20th ICONE20-POWER2012, July 30 - August 3, 2011, Anaheim, California, USA.

[20] R. Lo Frano, L. Burgazzi, External event risk assessment: methodology and application, Proceedings of the 22nd International Conference on Nuclear Engineering - ICONE22, July 7-11, 2014, Prague, Czech Republic.

[21] ASCE Standard, Minimum design loads for buildings and other structures, ASCE/SEI 7-10, 2010.

[22] H. Ashar, B. Scott, J. F. Artuso and J. D. Stevenson. Code for concrete reactor Vessels and containments, ASME Boiler & Pressure Vessel Code, Sec. III Div. 2, 2001.

[23] MSC©Software, MSC.MARC user’s guide, 2010.

[24] US NRC R.G. 1.60, Design response spectra for seismic design of nuclear power plants, Rev. 1, 2010.

[25] U.S. Nuclear Regulatory Commission (NRC), 2006. Rev. 2, Regulatory Guide 1.92. Combining Modal Response and Spatial Components in Seismic Response Analysis.

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