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Application of risk-informed probabilistic and deterministic safety approach to estimate the risk of external events

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Sommario

This report presents the activities performed in the frame of LP1, Objective B (Accident consequences evaluation), task B3 of PAR 2014, ADP ENEA-MSE.
The study addresses the safety assessment of nuclear power plants due to external events, adopting a risk informed approach.
To this aim safety analysis is thus implemented with the evaluation of some specific probabilistic aspects, related e.g. to the multi-unit risk issue, as a complement to traditional deterministic investigations. To complement the hazard assessment, a deterministic investigation of an existing plant has been carried out. In doing that aging effects, as a consequence of loadings originated by the multiple accident event, have been considered in order to highlight possibly vulnerability.

Note:

This document has been prepared with the following main contributors:

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Table of contents

Executive summary

Alongside the analysis of the Fukushima accident in Japan in 2011 on a risk-informed perspective, the present study purports to:

- Identification and analysis of some specific relevant aspects of the probabilistic safety analysis, or PSA, as highlighted by the accident itself, which are critical to the safety.
  Specifically, some significant issues revealed by the Fukushima accident are addressed, such as
  - the analysis of the "level2" PSA, aimed at the evaluation of the source term;
  - risk assessment relative to sites with many units.

  To this aim some foundational notions to develop the PSA models related to specific aspects, are proposed and discussed for their incorporation within the risk assessment structure.
  The analysis is eventually accompanied by some case studies, to present the methodological options and to propose the solutions in order to address these issues, within a risk-informed approach.
  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.

- As part of the risk-informed approach also deterministic calculations are necessary to evaluate the dynamic response of existing installations, as regards the function of containment and main components of the reactor, against accidental situations arising from external events are performed. Moreover, the use of a deterministic approach for the safety assessment, supported by conservative assumptions, is expected to lead to improved safety.
  In this context, therefore, the residual safety margin of the containment system will be evaluated mainly, being the last defense barrier of the plant in terms of containment and confinement of radioactive releases to the external environment. In addition the performances of containment in relation to the effects of aging that could possibly affect the performance of the structural materials will be evaluated in consideration of the fact that the existing plant have been designed without addressing the extremely severe (and multiple) external events.

List of acronyms

AC	Alternating Current
ADS	Accident Dynamic Simulation Methodology
AP1000	Advanced Passive
ATWS	Advanced Transient Without Scram
BWR	Boiling Water Reactor
CCIE	Common Cause Initiating Events
CDF	Core Damage Frequency
CIRTEN	Consorzio Interuniversitario per la Ricerca Tecnologica Nucleare
DBA	Design Basis Accident
DC	Direct Current
DID	Defence-in-depth
DI&C	Digital I&C
DSA	Deterministic Safety Analysis
EDG	Electrical Diesel Generator
ENEA	Agenzia nazionale per le nuove tecnologie, l'energia e lo sviluppo economico sostenibile
EOP	Emergency Operating pPocedures
EPS	Emergency Power Systems
ESBWR	Economic Simplified Boiling Water Reactor
ET	Event tree
FT	Fault tree
I&C	Instrumentation and Control
LERF	Large Early release Frequency
IE	Initiating Event
LOCA	Loss of Coolant Accidents
LOSP	Loss of Offsite Power
LPSD	Low Power and Shutdown
LWR	Light Water Reactor
NPP	Nuclear Power Plant
PGA	Peak Ground Acceleration
PRA	Probabilistic Risk Analysis, Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment, Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objective
SAMG	Severe Accident Management Guidelines
SBO	Station Blackout
SFP	Spent Fuel Pool
SMA	Safety Margin Assessment

1. Introduction and scope

Alongside the analysis of the Fukushima accident on a risk-informed perspective and as a follow-up of the previous research in the same area, as referenced in [1], the objective of the present study is twofold:

- Identification and analysis of some specific relevant aspects of the probabilistic safety analysis, or PSA, as highlighted by the accident itself, which are critical to the safety.
  Specifically, some significant issues revealed by the Fukushima accident in Japan in 2011 are addressed, such as
  - the analysis of the "level2" PSA, aimed at the evaluation of the source term;
  - risk assessment relative to sites with many units.

  To this aim some foundational notions to implement the PSA models related to specific aspects are proposed and discussed for their incorporation within the risk assessment structure.
  The analysis is eventually accompanied by some case studies, to illustrate the proposed methodology oriented towards the implementation of its models in the probabilistic approach.

- As part of the risk-informed approach also deterministic calculations to evaluate the dynamic response of existing installations, as regards the function of containment and main components of the reactor, against accidental situations arising from external events are performed. In this context, therefore, the residual safety margin of the containment system will be evaluated mainly, being the last defense barrier of the plant in terms of containment and confinement of radioactive releases to the external environment.
  In addition the performances of containment in relation to the effects of aging that could possibly affect the performance of the structural materials will be evaluated.

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

2. PSA in the light of Fukushima accident: background

This section highlights some lessons learnt as coming out from the Fukushima accident for the development of a comprehensive PSA, to complement the findings of ref. [1].

Indeed, as pointed out in ref. [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.

Analysis in ref. [1] has been limited to level1 PSA (i.e. related to the core damage frequency assessment) aspects, while it is apparent that level2 PSA (i.e. related to the radioactive releases frequency assessment) facets as well are concerned, so that their treatment in the practice has to be addressed in a consistent way.

In addition to the identification of the features more pertinent to level2 PSA, the multi unit risk subject is treated in detail, as an important topic arising from Fukushima analysis, for its relevance to the licensing aspects: to this aim novel models are proposed for their development on the practical usage viewpoint.

Firstly lessons learnt pertaining to level2 PSA are illustrated, then focus is placed on multi-unit risk at a site subject.

3. Insufficiencies and gaps on level2 PSA: lessons learnt

This section describes some of the Fukushima Dai-ichi accident highlighted aspects pertaining to PSA Level 2, which, like in the previous Level 1 case, as identified in ref. [1], need to be dealt with in detail, in order to bridge the correspondent gaps. First of all, L2 PSA has to be extended to cover external hazards, in the frame of the full scope PSA development, which is being discussed in the following, as a separate item.
The list of significant issues to be conveniently worked out in order to implement the relative PSA approach application includes:

- Plant Damage States under external hazards
- Loss of containment function failure modes
- Accident phenomenology investigation
- Hydrogen explosion
- The role of operator under severe accident conditions and human reliability
- SAMG implementation
- Site risk issue
- Risk associated with spent fuel pools
- Consideration for prolonged mission times
- Role of passive systems relevant for the mitigation of severe accidents
- Re-assessment of DID, in terms of weaknesses and gaps between the different levels
- PSA application to all power plant statuses, e.g. low power and shutdown: full scope PSA
- Uncertainties evaluation

Discussion for each one follows.

3.1 Plant Damage States under external hazards

Modeling of external events PSA poses some additional issues, as compared to internal events. In the context of Level 2 PSA, one challenge is the identification and development of different additional PDS (Plant Damage States), whenever an external hazard initiator

produces conditions (combinations of failures) that differ from those produced by internally initiated accident sequences.

This should include the conditional component (such as the items used to mitigate the severe accident consequences as to remove heat from the containment and keep containment pressure and temperature within approved limits) failure probabilities, upon both a single or a combination of external hazards, the fragility assessment of the safety components, that is the failure probability at each level of the hazard, such as the seismic fragility of the containment.

Depending on the plant response to the strength of an external event, a set of different plant damage levels may need to be defined and analyzed.

3.2 Loss of Containment function failure modes

External events pose a threat to the plant integrity from direct impact like, for instance, in case of damage from missiles generated at another plant on the site or from military activity, and the plants are designed to withstand the effects of both man-induced events and natural phenomena (such as earthquake, tsunami-flooding, extreme wind, etc.).

All the possible modes of failure of the containment in the light of the external hazards, are to be examined in order to perform the source term evaluation, including the hydrogen explosion.

In particular the external events are to be considered as a risk factor for the containment strength and their impact in terms of the relative load on the containment performance has to be assessed to evaluate the plant resistance to the event. The duration of the effects of the external initiator and the load combination should be considered as well, to get a realistic picture of the accident.

To the aim the analysts benefit the probabilistic aspects of containment design against external hazards, i.e. probabilistic structural mechanics and reliability-based design of reactor containment structures.

For example fragility analysis is liable to be utilized to determine the probability of failure of containment structures, given the loads experienced because of the external hazard. The term fragility is taken from seismic analysis, but it applies equally well to any external initiator. Fragility is defined as the probability of failure as a function of the size of the input load.

Usually, the input load is parameterized in terms of a single variable (e.g. the pga for earthquakes).

3.3 Accident phenomenology investigation

Fukushima Dai-ichi accident calls for the research community to foster the investigation on accident phenomenology relevant to severe accident progress, such as the fuel rod – water interaction (FCI, fuel coolant interaction), the hydrogen build up and combustion, the fission product release, the molten fuel relocation and the MCCI (molten core concrete interaction).
This implies a high qualification requisite to be posed on the related in-core instrumentation for the diagnosis of severe accident and monitoring at deteriorated plant conditions, which proved to be unreliable or absent during the Fukushima Dai-ichi accident. These include measurement of temperatures in pressure vessel, containment and base-mat concrete, water levels in pressure vessel and containment, as well as hydrogen concentration and radiation measurements in containment and reactor building.

3.4 Hydrogen explosion

The risk of explosion of hydrogen, as a result of metal-water reaction during a postulated core overheat (that is, Zircaloy-steam interaction) or molten core concrete interaction, in the containment should be considered in Level 2 PSA.
The expected hydrogen hazard could be lowered through the implementation of passive preventive systems by providing the design with equipments inside containment to protect the plant and that automatically turn hydrogen gas into harmless water in the unlikely event of damage to the nuclear fuel.
These safety devices, referred to as hydrogen passive (*i.e.*, self-actuating or “autocatalytic”) “recombiners”, spontaneously recombine hydrogen and oxygen molecules, yielding steam and heat, which do not require electricity. Hydrogen recombiners are intended to maintain the hydrogen concentration in the containment below levels that can support a hydrogen explosion (*i.e.*, at about 4 % by volume and above).

In any way the Fukushima Dai-ichi accident has proved the relevance of hydrogen explosion as an important risk contributor for the reactor containment and hence to be carefully accounted for.

Moreover, in the spent fuel pools usually situated next to nuclear power plants, there are large numbers of additional fuel rods, used ones, disposed of as waste. There must be constant water circulation in the spent fuel pools. In what is labeled a "loss-of-water' accident in a spent fuel pool, the zirconium cladding of the fuel rods is projected as increasing the explosion hazard because of the hydrogen build-up.

3.5 The role of operator under severe accident conditions and human reliability

The operator performance has been challenged heavily by the Fukushima Dai-ichi events, especially as regards the stage relative to the core degradation and severe accident evolution, where the accident conditions and the impact of environment on operator actions became more and more severe over time.

These events fall into a special category of "cliff edge effects" where there were widespread effects on safety systems (primarily plant instrumentation) due to the resultant tsunami flood and subsequent evolution towards a severe accident.

Some possible performance factors from this extreme experience include the lack of information, when indications were severely impacted and instrumentation was damaged, so that operators had to operate on "best guess" of what was happening and had to make assumptions without relating to the local operators.

The lack of contingency procedures and pre-staged equipment impacted operator actions, so that operators had to operate outside the procedural space or formal training. For example, during containment venting, there were "a lack of contingency procedures for operating the vent system without power, as well as the lack of pre-staged equipment, such as an engine-driven air compressor", contributing to the delay in venting. The lack of contingency procedures to vent containment without power forced them to devise alternate, knowledge-based strategies.

In addition, during this phase the coordination with the local authorities for the evacuation plan becomes of outstanding importance, posing an extra load on the control room operator performance.

3.6 SAMG implementation

Analogously to Level 1, probabilistic analysis should be used not only for ascertaining risk levels, but also for emergency and risk management measures, and in particular for Severe Accident Management Guidelines (SAMGs).

A major important technical lesson learnt is the improvement of preventive and mitigative severe accident management measures, including hydrogen control and containment venting. Implementing the SAMGs, (in addition to the EOPs) into the PSA will indicate whether indeed risk is reduced by applying the various accident management strategies.

Severe accident management guidelines improvement imply the strengthening of emergency planning and preparedness, to withstand the impact of external hazards which can cause damage to the infrastructures and communication means, loss of equipment, stress and injuries on the personnel involved. As in the previous case of Level 1 this is related to human reliability aspects of PSA in a considerable way and PSA should be used aimed at the optimization of SAMGs. On the other hand the implementation of the guidelines concerning specific external events, or groups of external events with similar plant effect, in terms of suitable accident management measures and actions for both prevention and mitigation of severe accidents (as mobile equipment and resource and plant management from alternative control rooms and emergency centers) on account of the more severe conditions dictated by the external event, requires their appropriate modeling within the PSA framework and the and the relative methodology enhancement.

3.7 Site risk issue

The analysis of this issue both for existing multi-unit sites and proposed modular reactors has not adequately considered the risk of multi reactor accidents on the same site. Such accidents have been largely ignored in Probabilistic Risk Assessments that support most of the risk informed applications. This applies as well to the Level 2 (and Level 3) models for multiple reactor accidents.

If there were an accident involving core damage on more than one unit at a given site, the consequences from the damage from each reactor would in general be different as the same plant damage states and release categories resulting from the core damage would not

necessarily be the same. Important damage states from the standpoint of determining the risk of early health effects for the initiating events that impact both units involve the failure to isolate containment penetrations.

In the case of sequences involving station blackout following a loss of offsite power, or seismic induced loss of offsite power, the probability of failure to isolate small penetrations is high due to motor operated valves which fail open so that the probability of release category is high for these sequences.

This topic will be dealt with extensively next chapter 4.

3.8 Risk associated with Spent Fuel Pools

The SFP risk on the "Level 2 PSA" standpoint can be evaluated by assessing key phenomena leading to a severe damage of spent fuels [2] and relevant severe accident progression in an SFP (similar with the Level 2 PSA in the reactor case), and finally assessing the accidental risk based on the radiological source terms released to the SFP outside (similar with the Level 3 PSA in the reactor case), successively.

Table below [3] summarizes some relevant characteristics with reference to the probabilistic risk assessment at Level 2.

<u>Key Accident Phenomena</u>

- Spent fuel rods (decay heat sources with time, fuel heat-up and uncovery, severe damage, zirconium oxidation/ignition, fuel melting, etc.)
- Spent fuel assembly (radial heat transfer, fire propagation, fuel assembly collapse, etc.)
- Spent fuel storage rack
- Down comer next to the edge of the pool
- Base region beneath the racks (cooling air ingression into the fuel assembly, molten corium-concrete interaction, etc.)
- Spent fuel storage buildings (hydrogen, etc.)
- Effect of burn accident mitigation strategies
- Severe Accident Progression Analysis leading to SFP building failures and radiological source term releases
  - Tools MELCOR SFP version or MAAP SFP analysis model

• Probabilistic Accident Progression Analysis

- Based on Accident Progression Event Trees (APETs)
- Level 2 risk surrogate metric: Large Release Frequency (LRF)

3.9 Consideration for prolonged mission times

The considerations referred to Level 1 apply as well to Level 2 PSA, in light of the Fukushima Dai-ichi accident, which demands consideration for longer mission times, especially with regard to the residual heat removal issue, the assessment of the containment performance and the analysis of time-related aspects of the accident that influence the severe accident progression and containment loading.

This issue is relevant as well in relation to the use of passive systems designed to limit the consequences of severe accidents (see next paragraph).

3.10 Role of passive systems for the mitigation of severe accidents

Passive systems relevant to Level 2 PSA refer to different stages of the accident progress, that is the systems tasked to cool the debris bed after the core melt and the ones required to prevent the containment modes of failure due to overheating.

The first type refers to the late phase of the progression of a severe accident, associated with corium-melt discharge from the Reactor Pressure Vessel (RPV) and its relocation on the concrete base mat in the form of a debris bed consisting of liquid/particulate corium. The core debris generates decay heat and attacks the concrete base mat and the containment structures and continues to do so, until the coolability of debris bed is achieved, as, for instance, with bottom coolant injection which occurs passively.

The second category concerns the passive safety systems relevant to containment integrity, like e.g, Passive Containment Cooling System (PCCS) for AP1000 and PCCS condensers for ESBWR.

Performance of passive safety systems pertinent to severe accident will represent a new challenge owing to the amount of uncertainties, as e.g. the condensation and boiling heat transfer coefficients or the heat transfer coefficients under the presence of non-condensable

gases. Consequently difficulties arise to achieve a qualified reliability figure, since the scarcity of data and the little experience.

Due to the specificities of passive systems that utilize natural circulation (small driving force, large uncertainties in their performance, lack of data…), there is a strong need for the development and demonstration of consistent methodologies and approaches for evaluating their reliability. This is a crucial issue to be resolved for their extensive use in future nuclear power plants.

3.11 Reassessment of DID, in terms of weaknesses and gaps between the different levels

Analogously for level 1, a risk-informed DID framework is to be adopted to complement the requisites of safety components and systems redundancy, diversification and independence, to prevent and/or mitigate accidents and decrease the containment vessel failure, should the core degradation occur.

Therefore the defense-in-depth (DID) design needs to be strengthened in terms of performance and reliability requisites for the systems and severe accident management procedures requested to mitigate the severe accident progress up to the radioactive products release outside the containment to reduce the probability of the “transition” of the accident from level 4 to level 5 of the DID scale, that is from severe accident to the containment failure and consequent release to the atmosphere.

3.12 PSA application to all power plant status, e.g., low power and shutdown: full scope PSA

Severe accident assessment on the probabilistic standpoint needs to be enlarged to include the LPSD states and for all the initiating events, to overcome the “usual” practice to perform a limited Level 2 PSA for full power mode.

As already stated in the Level 1 section this would require a considerable endeavor among the community of researchers and practitioners in nuclear safety, to address the core degradation accident consequential to an external event in relation to all the aspects related to the consideration for all the plant states.

3.13 Uncertainties evaluation

All the possible uncertainties are to be included systematically in the PSA studies. In the light of Fukushima Dai-ichi accident they include primarily the phenomenological uncertainties increase with the progression of the accident, including the identification of the leak path which would provide a great uncertainty in the estimation of the radioactive material release and the estimation of the frequency of occurrence of rare external events.

With this regard, in fact, historical records on earthquakes, tsunamis, volcanism, etc. are very limited, so that extrapolation is inevitable and accordingly it gives large uncertainty.

Estimation of accident consequences showed to be very difficult to take into account the phenomena related to the hydrogen build-up, burning and transport and the unpredicted adverse effect of external initiators and severe accident phenomena on accident management operation.

An uncertainty analysis is needed to address the relevant uncertainties emerging from the study. These uncertainties are to be identified and properly evaluated in order to add credit to the probabilistic figures achieved so far and assess whether the accident has been correctly modeled on the probabilistic standpoint.

Finally, the analysis may point out in-depth analyses that need to be performed in order to eliminate or reduce major uncertainties in analysis and gain a higher confidence in the results hence making the decisions meaningful, as regards for instance on how additional protection is to be designed and implemented.

4. Multi-Unit risk

The events at the Fukushima nuclear power station draw attention to the need for consideration of risks from multiple nuclear reactor units co-located at a site. Currently, multi-unit site risk is neither formally nor adequately considered; this includes operating plant sites in either the regulatory or the commercial nuclear environment. The nuclear industry's integrated site risk solutions generally focus on only one facet of the Probabilistic Risk Assessment (PRA) at a time without considering other concurrent events. For example, the station blackout (SBO) has been investigated because of its site impact and interdependencies in the electrical systems. Similarly, the industry has looked at seismic events at a site. Although specific aspects of multi-unit risk have been looked at in an ad hoc fashion with greater detail, no integrated approach exists. While recognized as an important issue by the NRC and IAEA, very little progress has been made in understanding and measuring the safety significance of multi-unit risks and implications of their surrogate measures (i.e., CDF and LERF) in the context of multi-unit sites. Further, performing PRAs on multi-unit sites, one reactor at a time, yields misleading and optimistic risk insights. This study will discuss issues and propose strategies toward addressing formal integrated approaches to site-based risk assessment and examines the options and uses of novel risk metrics such as CDF, Large Release Frequency (LRF), Large Early Release Frequency (LERF) as related to the total site risk.

Main goal of the study is to bring forward methods to sum risk metrics for different hazards, multiple radioactive sources in the same plant and multiple plants on the same site, improved modelling of initiating events with multi-unit impact, in the context of risk aggregation process where all the risk factors are combined together to generate a value for the site as a whole.

Stemming from the current state of practice allows identifying the related issues and proposing suggestions for further improvements as well.

4.1 Background

As recognized in [1], the site risk assessment is a very important issue, 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, 3 and 4) and the operating units (Units1, 2 and 3) affected Unit 4, which was defueled at the time of the accident: it appears that the hydrogen in Unit 4 reactor building came from Unit 3 through an unexpected path. In addition core damage occurred in Units 1, 2 and 3. 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. 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 and radioactive sources, 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.

At present the Level 1 internal event PSAs include modeling of initiating events that originate on one of the other units (that is, not from the selected reference model unit). For example, a large secondary side line break is postulated to lead to a hostile powerhouse environment in the vicinity of both the unit on which the break occurs and the adjacent unit. For these events, the Level 1 internal event PSAs account for the possibility of severe core damage in the selected model unit after a secondary side line break that occurs on that same unit, or that occurs on another operating unit.

The Level 1 PSAs also model the possibility of severe core damage occurring after a common mode event that simultaneously impacts all units. For example, the internal events PSAs

model a loss of the bulk electrical system, or failures in the service water intake, which impact the entire plant. The internal fire PSAs analyze multi-unit fire scenarios as well as scenarios that only impact the reference model unit; the internal flooding PSA similarly considers flooding of rooms that contain common equipment as well as rooms for the selected reference model unit.

For external events, the risk of severe core damage from seismic events and high winds is estimated for a single reference unit.

For all such events that involve more than a single unit, the mitigating functions in the Level 1 PSAs are modeled for the selected reference model unit, but reflect the impact of the event on the other units. For example, the success criteria for common systems such as emergency power and water reflect the demand requirements on the system following a common mode event that affects all units. Similarly, the Level 1 PSAs account for reduced availability of shared systems such as inter-unit ties for instrument air or boiler feedwater following events that could affect the supplying unit.

Additional treatment of multi-unit issues is required in the Level 2 PSAs, given the differences in severe accident progression (e.g., potential consequential containment failure) for single-unit versus multi-unit accidents. In order to properly quantify the large release frequency per unit per year, it is therefore necessary to identify whether a given accident sequence from the Level 1 PSA represents fuel and core damage at a single unit or different combinations of multiple units.

For example, although a given initiating event (e.g., loss of off-site power) will initially affect all units in the station, the selected model unit in the Level 1 PSA could proceed to severe core damage as a result of either common mitigating function failures (e.g., site wide emergency power failure) or due to unit-specific failures (e.g., failure of unitized emergency equipment). Again PSA models should systematically consider dependencies on the systems levels, e.g. via shared support systems or buildings, as well as dependencies on the accident sequence level, e.g. via the impact of a severe accident in one unit on measures or systems in another unit.

However the present study aims at addressing for the most PSA level1 related issues.

It’s worth noticing that new risk metrics are to be envisioned since probabilistic safety goals used in association with current PRAs are based on those promulgated for new plants, that is

1.0E-5/ry for CDF and 1.0E-6 for LERF and are expressed per reactor-year and do not explicitly consider the implications of multiple units on a site.

4.2 Framework of site safety assessment

The Fukushima accident involving a combination of multiunit and multiple hazards highlighted the need for a holistic framework for risk assessment of a site which is capable of integrating the risk associated with all sources that can be released from a site.

In this context of site safety assessment, the risk assessment should include sensitivities to determine the extent to which multiunit considerations increase or decrease the risk associated with a specific nuclear installation site and thus the interaction between the units (be it from shared system, common cause, or interaction of responses) need to be addressed in a comprehensive framework. The quantification of such a risk at a site level allows the regulatory body to make risk informed decisions in their role as a regulator and protector of public health and the environment.

In the following the holistic framework for the risk assessment of a site with multiple units and possibly other co-located installations with nuclear inventory is presented. The framework has at its centre the reactor units which are challenged by the external events, the events cause one or more hazards which may challenge the safety of one or more reactor units on the site, the affected installations respond to the imposed challenges which in turn may or may not affect some other on- site installations, these interactions between installations continue till severe accident managements measures are brought in to play and further interactions continue to occur into the release phase from one or more installations. The risk quantification of this release as a measure of its impact on human and environmental health will provide the final response to the site level safety assessment.

Given this framework as the scope of the risk assessment many issues unaddressed before comes to focus. The treatment of a hazard on multiple units, the assessment of initiating events with the potential to jeopardize the safety of more than one unit, the identification of interactions and dependencies between units or interdependencies, as well as the metric for site wide risk and many such important factors need to be addressed within the context of this framework. Together with the treatment of the various aspects some mathematical

expressions are proposed as well, as a first effort to provide some modelling to be implemented in a probabilistic approach.

4.3 Initiating event analysis

Multiunit accident sequences may be caused by two classes of initiating events:

- Common-Cause Initiators (CCIs): Initiators that simultaneously challenge all of the units at the site. CCIs include initiators that are caused by external hazards (e.g., earthquakes, severe weather).

- Single-Unit Initiators (SUIs): Initiators that occur at one unit. SUIs generally include initiators caused by internal hazards such as internal events (e.g., loss of main feedwater, loss of coolant accidents), internal floods, and internal fires. SUIs may cause multiunit accidents due to cross-unit dependencies such as shared support systems, spatial interactions (e.g., internal flood and internal fire propagation pathways), common cause failures or operator actions.

Since SUIs only occur at one unit, multiunit accident sequences caused by SUIs must consider how accident sequences are initiated in the subsequent units (i.e., the units that did not experience the SUI). In order to distinguish among the types of multiunit accident sequences caused by SUIs, the following taxonomy has been used:

- Cascading sequence: A multi-source accident sequence caused by an SUI that causes core damage and release from the unit where the SUI occurred and also in one or more additional units.

- Propagating sequence: A multi-source accident sequence caused by an SUI that does not cause core damage in the unit where the SUI occurred, but causes core damage and release in one or more additional units.

- Restricted sequence: A single-source accident sequence caused by an SUI that only causes core damage and release in the unit where the SUI occurred (i.e., no other unit is affected).

In order to understand the development of the total site risk estimate, let's consider a three-unit site with units labelled Unit 1, Unit 2, and Unit 3. There are seven possible outcomes that involve release from one or more units, as listed below:

- Single-unit outcomes: Unit 1, Unit 2, Unit 3
- Dual-unit outcomes: Unit 1 and Unit 2, Unit 1 and Unit 3, Unit 2 and Unit 3
- Triple-unit outcomes: Units 1 and Unit 2 and Unit 3

Specifically, there are three single-unit outcomes, three dual-unit outcomes, and one triple-unit outcome. The various outcomes can be depicted on a diagram, as shown in Figure 1, where all of the outcomes that affect a specific unit are included within a circle [4].

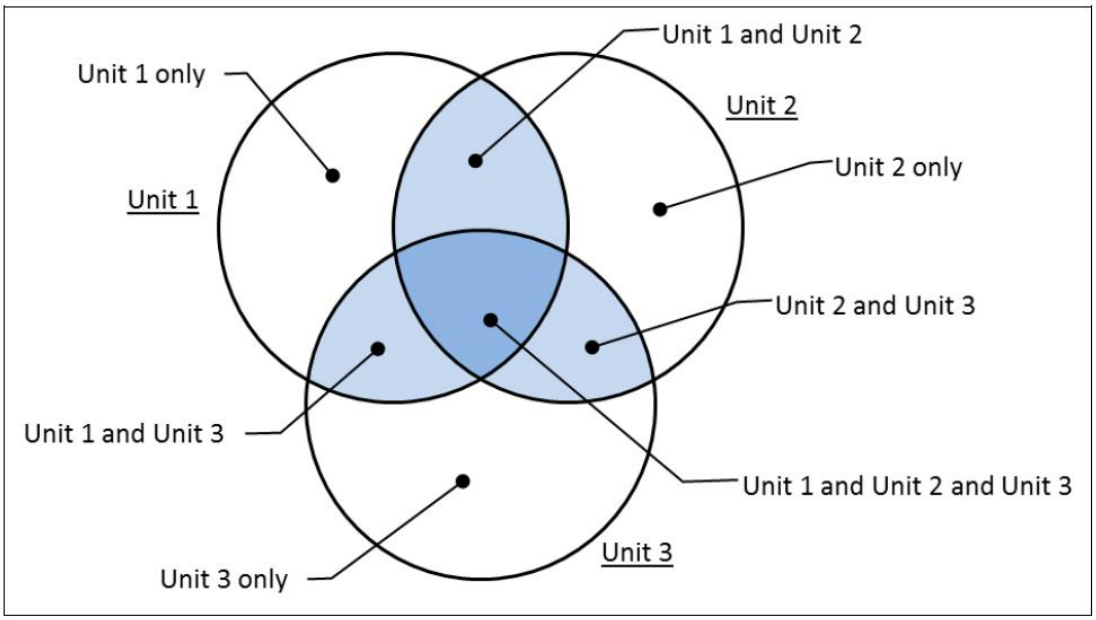

Figure 1. Diagram Depicting Multiunit Accidents

In general, for a site that has n units:
the number of outcomes that involve exactly k out of n units = $\binom{n}{k}$

Contribution from Common-Cause Initiators

Consider the occurrence of a CCI at a three-unit site with units labeled Unit 1, Unit 2, and Unit 3, and define the following events, as in Figure 1:

Event *U*1 = release from Unit1
Event *U*2 = release from Unit2
Event *U*3 = release from Unit3

From these fundamental definitions, we can have the following seven compound events:

P(*U*1/*CCI*) = probability of release from only Unit1
P(*U*2/*CCI*) = probability of release from only Unit2
P(*U*3/*CCI*) = probability of release from only Unit3
P((*U*1 and *U*2)/*CCI*) = probability of release from Unit1 and Unit2
P((*U*2 and *U*3)/*CCI*) = probability of release from Unit2 and Unit3
P((*U*1 and *U*3)/*CCI*) = probability of release from Unit1 and Unit3
P((*U*1 and *U*2 and *U*3)/*CCI*) = probability of release from Unit1 and Unit2 and Unit3

Therefore the total probability of having a release from the site as a consequence of a CCI is the sum of all these terms.

Contribution from Single-Unit Initiators

In order to estimate the contribution to site risk from SUIs, it is important to recognize that an SUI may occur in any unit, and that the occurrence of an SUI may result in cascading, propagating, or restricted sequences.
Consider the occurrence of an SUI at Unit1, *SUI1*, which is located at a three-unit site. Figure 2 illustrates the possible restricted (black arrow), cascading (blue arrows), and propagating sequences (red arrows) that result in core damage and release that are caused by the occurrence of *SUI1* [4].

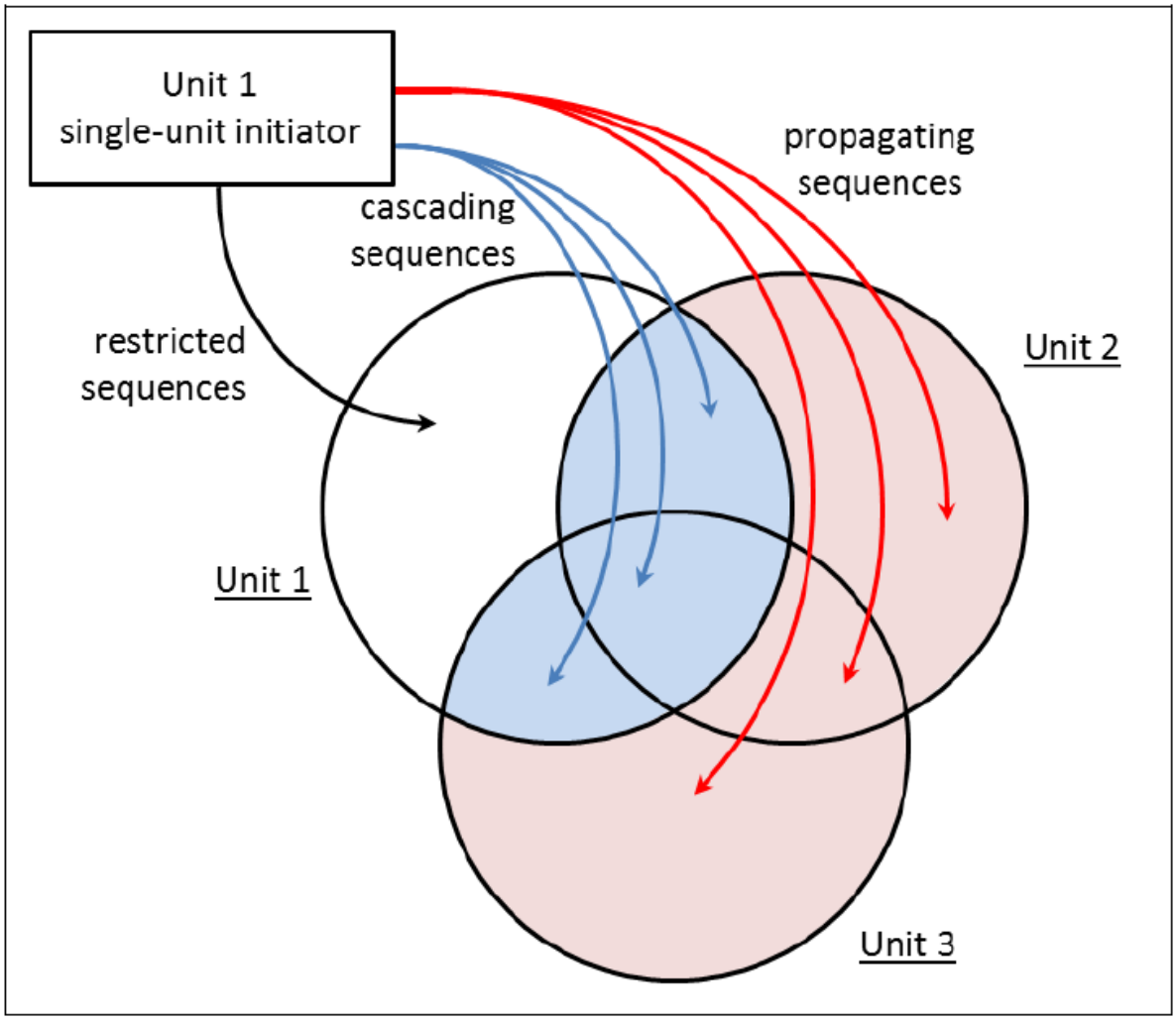

Figure 2. Restricted, Cascading and Propagating Sequences Caused by a Single-Unit Initiator

The contribution to site risk from this Unit-1 SUI is given by the sum of all the following terms:

P(*U*1/*SUI*1) = probability of release from only Unit1 *restricted*
P((*U*1 and *U*2)/*SUI*1) = probability of release from Unit1 and Unit2 *cascading*
P((*U*1 and *U*3)/*SUI*1) = probability of release from Unit1 and Unit3 *cascading*
P((*U*1 and *U*2 and *U*3)/*SUI*1) = probability of release from Unit1, Unit2 and Unit3 *cascading*
P((*U*2 and *U*3)/*SUI*1) = probability of release from Unit2 and Unit3 *propagating*
P(*U*2/*SUI*1) = probability of release from only Unit2 *propagating*
P(*U*3/*SUI*1) = probability of release from only Unit3 *propagating*

In order for an SUI to propagate into other units, there must be a sequence of events in the initiating unit (i.e., the unit where the SUI occurred) that causes an initiating event in one or more of the other units. As a result, the propagating probabilities are the product of the conditional probability that the subsequent unit(s) experience an initiating event given an SUI and the conditional probability that the subsequent unit(s) experiences core damage and release. In contrast, the cascading probabilities do not include the conditional probability that subsequent unit(s) experience an initiating event because it is assumed that, for cascading sequences, the conditional probability that subsequent unit(s) experiences an initiating event is identically 1.0.

There are similar expressions for the contributions to site risk from SUIs that occur at Unit 2 and Unit 3, and the total site risk due to SUIs is the sum of these three expressions.

Finally, the total site risk related to radioactive release can be found by summing the contribution from CCIs and the contribution from SUIs, as given by the previous expressions.

Examples of accident sequences triggered by CCI (case 1), or representing respectively propagating and cascading sequences (case 2 and 3) are illustrated in Figure 3 [5].

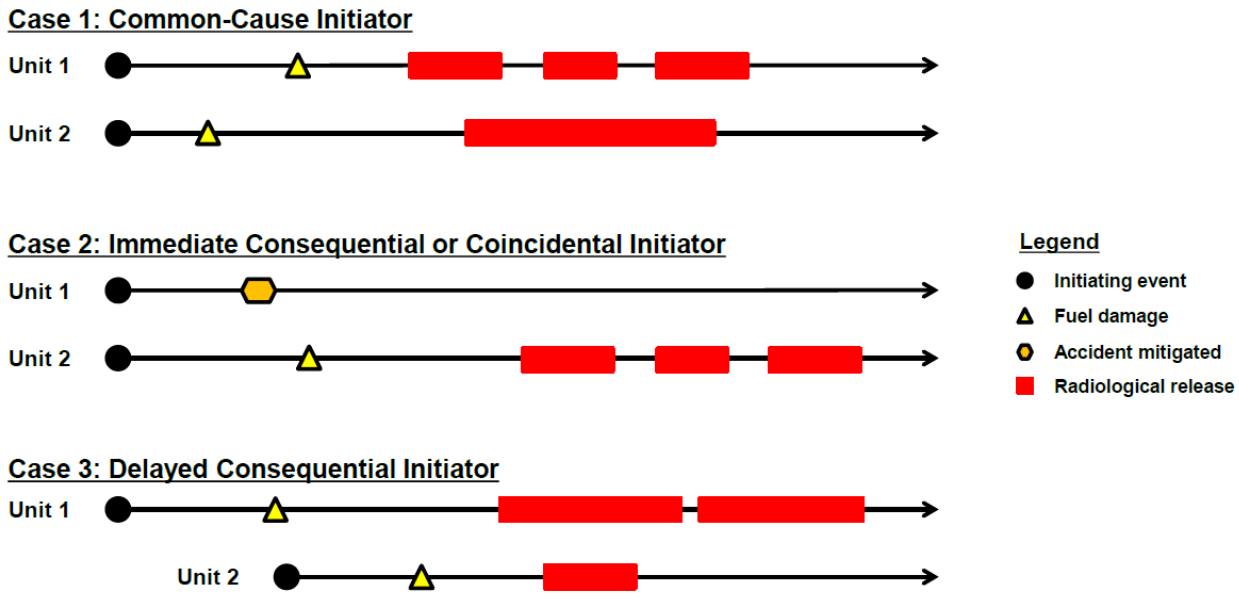

Figure 3. Site Risk Accident Sequences Examples

4.4 Interaction and dependencies

As illustrated by the Fukushima accident, the same hazard or hazard combination may lead to initiating events and accident sequences in multiple installations concurrently (common cause). An accident at one installation may affect the capabilities and compromise the resources available to support mitigating efforts in another installation. Hence the probability of preventing an accident in one installation cannot be assessed without considering the status of the other installations on the site. Consideration of interaction of structures, systems and components between the different installations, the response of the installation and its interaction with the response in individual installations, human reliability given these interactions and others that will result during the progression of an accident are essential interactions to be included in the holistic framework for site safety assessment.

In order to truly address a multi-unit PRA, one first must be able to understand all of the avenues in which units could be connected, which could be attempted through classification. This classification will allow multiple independent single unit PRAs to be integrated into a single multi-unit PRA.

Six main dependence classifications are identified: initiating events, shared connections, identical components, proximity dependencies, human dependencies, and organizational dependencies, as depicted in Figure 4, where a fishbone representation of categorization of inter-unit dependencies is used, [6]. Additionally, there will be a seventh classification of events that are completely independent.

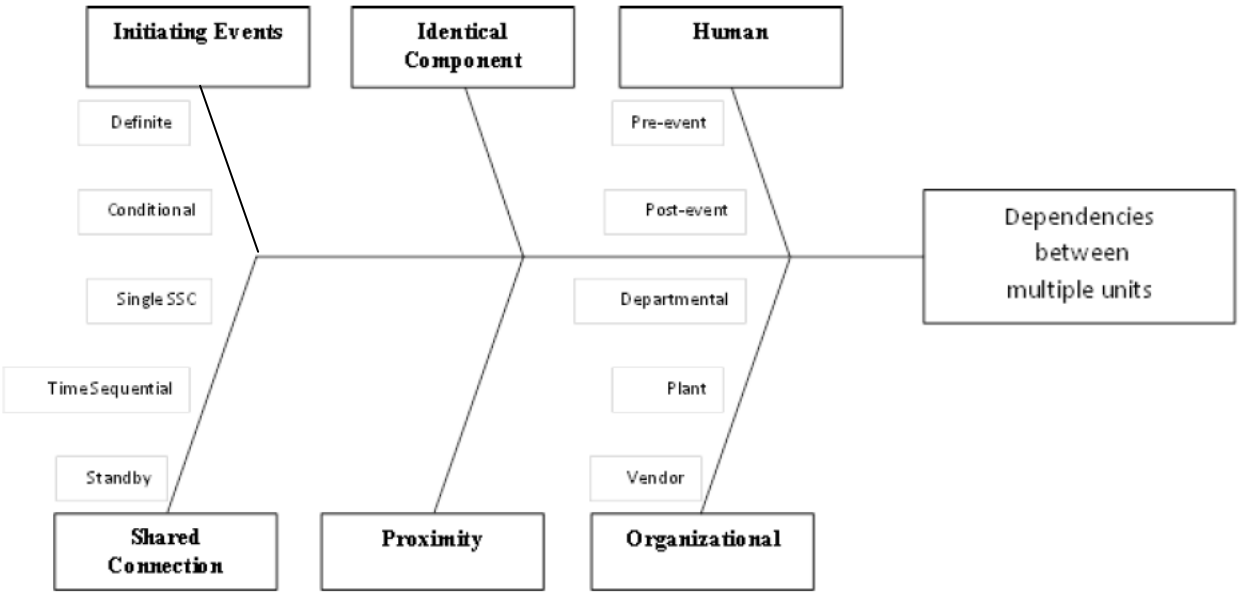

Figure 4. Classification of events

The first class, initiating events, represents those single events that have the capacity to affect multiple units of a nuclear power plant site. Not all initiating events that are incorporated into a typical nuclear power plant PRA will affect more than one unit at a site, although several have that capability. These initiating events can be divided into two subclasses, events that will always affect multiple units, referred to as “definite” events, and events that will only affect multiple units under certain circumstances, referred to as “conditional” events.

The second class, shared connections, refers to links (piping, cables, power divisions, etc.) between components that materially connect multiple units. These connections may be in three different sub-classes. The first is a single structure, system, or component (SSC). This occurs when multiple units rely on a single SSC for simultaneous support. Two examples are using the same plant exhaust stack or having a common header for safety injection. The second subclass is time sequential sharing or cross-connected SSCs. This is when an SSC is able to fully support any single unit; however, it is not capable of simultaneously supporting multiple units. This often occurs between electrical power supplies at nuclear power plants. The third subclass is standby sharing (e.g., crossties). Standby sharing occurs when multiple units share a standby or spare SSC that can only be used to support a single unit. This approach is commonly seen for safety systems such as emergency diesel generators and fire protection systems.

The third class, identical components, represents components that have the same design and operating environment for multiple units. This means that the components are designed, installed, and maintained identically and are operated in the same manner making them susceptible to traditional common-cause failures that are considered for single units. This not only includes conventional components, but also digital instrumentation and control systems and software.

The fourth class, proximity dependencies, can be manifested in several different ways. Proximity dependencies occur when a single environment has the potential to affect multiple units. This common environment could be either intentionally or unintentionally created. The proximity could be within a room, positions between or within systems, or occur because of the site layout. Additionally, conduits and doors may connect otherwise independent areas. If, for example, the chemical and volume control system for multiple units were in the same room, a fire or other event could affect multiple units. Likewise, if there was an explosion onsite and two units were located very close together, the same explosion could affect both units.

The fifth class, human dependencies, can also be manifested in a variety of ways. Human dependencies occur when a person's interaction with a machine affects multiple units. This could be an operator, a maintenance team member, a member of an installation crew, or the like. Human dependencies are split into two subclasses, pre-event and post-event actions. Human actions that occur before an event typically create latent conditions. For instance, in currently operating plants, the same maintenance team could perform the same task and create the same failure environment on multiple units. Human actions that occur after an event typically have immediate consequences. One example would be in small modular reactors where one operator would control multiple modules or units at the same time. If the operator is responding to a situation on one unit, the operator may not notice or be able to control an evolving situation on another unit.

The sixth class, organizational dependencies, has a number of different facets. Organizational dependencies occur when an organization somehow connects multiple units (through programs such as operating and emergency procedures, reliability assurance, surveillance procedures, training simulators, etc.), typically by some sort of logic error that permeates the organization. Although human and organizational dependencies are closely related, there is delineation between the two, which lies in the root cause of the failure. Human dependencies

are dependencies that are caused by the man-machine interaction, while organizational dependencies are often human actions caused by the culture of the organization. In this case, the organization could refer to a department at the plant, the plant itself, or the vendor that supplies components to the plant. These dependencies occur because the same logic or culture exists across an entire group, which affects multiple units and, at times, multiple sites.

The seventh class, independent events, represents those events that do not create a dependency between multiple units. This class only includes events whose occurrence and effect are limited to a single unit. Any events or SSCs that do not fall into the previously discussed categories would fall into this classification. For example, loss of coolant accidents would be an independent event. The majority of the SSCs for each unit would be in this category.

Table 1 points up the distribution of structures, systems and components sorted according to this categorization, [7].

Dedicated Single-Unit Structure, Systems and Components	Shared Multi-Unit Structures, Systems and Components
Safety DC Electrical and Essential AC Distribution System	Cooling towers, pond or other ultimate heat sink
Reactor Building or Bay	Turbine-Generator Building
Containment Vessel	Reactor Building
Decay Heat Removal System	Control Room
Emergency Core Cooling System	Spent Fuel Pool
Non-safety Control and Instrumentation System	Site Cooling Water System
Chemical Volume and Control System	

Table 1. Dedicated and Shared systems

Table 2 represents the matrix showing the classification scheme, [7].

Accident Sequence Classifications	Definition	Potential Systems Belonging to Classification
Initiating Events	Single events that have the capacity to affect multiple units	Loss of Offsite Power, Loss of Ultimate Heat Sink, seismic event (including seismically-induced tsunami), external fire, external flood, hurricane, high wind, extreme temperature
Shared Connections	Links that physically connect SSCs of multiple units	Reactor pool, chilled water system, BOP water system, spent fuel pool cooling system, circulating water system, reactor component cooling water system, high, medium and low voltage AC distribution systems
Identical Components	Components with same design, operations or operating environment	Safety DC electrical and essential AC distribution system, reactor module bay, containment, decay heat removal system, emergency core cooling system, non-safety instrumentation and control, chemical volume and control system, power conversion system
Proximity Dependencies	A single environment has the potential to affect multiple units	Reactors, ultimate heat sink, containment, non-safety DC electrical and essential AC distribution system, control room HVAC
Human Dependencies	A person's interaction with a machine affects multiple units	Shared control room, operator staffing more than one reactor
Organizational Dependencies	Connection through multiple units typically by a logic error that permeates the organization	Same vendor for safety and non-safety system valves, consolidated utility ownership of multiple nuclear power plant sites, decision-maker overseeing more than one reactor or more than one operator

Table 2. Classification matrix

It has to be noted that most of the existing PSAs already account for shared equipment and systems, as well as crosstie capability as allowed by design and procedures. If multi-unit considerations are taken into account in the PSA, and if a shared asset only has the capacity to support one plant at a time, then a shared availability factor should be incorporated into the system fault tree that reflects the probability that the other plant will not need the asset in order to meet minimal functional success criteria. The shared availability factor should include the human error probabilities of implementation actions, and hardware failure probabilities. Constructing an aid such as a table or matrix showing all possible combinations of available equipment may be useful (e.g., EDGs, alternative AC power, and service water pumps). It is necessary to review relevant system fault trees where operator action to cross-tie units is credited and to ensure the reasonableness of actual plant and operator response to an event (e.g., time available for operator response vs. feasibility of recovery actions under changing environmental conditions).

The existing human error analyses may extensively change in case of multi-unit site initiators, for example if the model considers the plant procedures dealing with shared diesel response to

loss of off-site power initiators for a multi-unit site. For example, it may be necessary to modify the values of performing shaping factors in accordance with the human reliability analysis methodology.

Inter-system common cause failures should be considered for components in systems that are shared between different plants. In case of multi-unit site initiators, it is necessary to review common cause component groups and probabilities.

The credited recovery action may be also, reviewed. For example, the recovery actions are less probable in a multi-unit LOOP than in a single-unit LOOP.

4.5 Risk metrics

Since current risk metrics (Core Damage Frequency and Large Early Release Frequency) don't capture the integrated site risk, nuclear reactor regulation based on single-unit safety goals is to be superseded by the options and uses of multi-unit risk metrics CDF and LERF as surrogates to QHOs' prompt fatalities and latent cancer deaths due to the total site risk.

To gain an accurate view of a site's risk profile, a measure of Core Damage Frequency (CDF) representing the site rather than the unit should be considered and estimated through a multi-unit PRA.

If there is release from more than one installation during the same accident then the emergency planning and severe accident management will be grossly impacted. Considering the fact that the large levels of radiation exposure will quickly saturate the dose levels of the responders and as a result the concurrent release from more than one reactor unit may exceed the linear sum of the consequence of individual reactors. Given this and the fact the frequency of the release at a multiunit site is related to the number of units on the site, the risk metric of core damage frequency (CDF) and large early release (LERF) is no longer an adequate metric for the risk assessment of multiunit sites. A more general set of risk metrics that would apply to all types of accidents similar to that at Fukushima would be those associated with a Level 3 PSA in which the risk of consequences to public health and safety are fully quantified. Thus a new or modified set of risk metric need to be developed which can rationally quantify the risk associated with multiunit sites involving non-reactor installations, as shown in table 3, [8].

Risk Metric	Applicability
Core Damage Frequency (CDF)	Level 1 Single-unit PSA
Large Early Release Frequency (LERF)	Limited scope Single-unit Level 2 PSA
Site Core Damage Frequency (SCDF)	Level 1 MUS PSA
Single Unit Core Damage Frequency (SUCDF)	
Multi-Unit Core Damage Frequency (MUCDF)	
Conditional Probability of Multi-Unit Accident (CPMA)	
Site Large Early Release Frequency (SLERF)	Limited scope Multi-unit Level 2 PSA
Release Category Frequency (RCF)	Full Scope Level 2 Single Unit PSA
Site Release Category Frequency (SRCF)	Full Scope Level 2 MUS PSA
Complementary Cumulative Distribution Function (CCDF)	Level 3 Single Unit PSA
Site CCDF (SCCDF)	Level 3 Multi-unit or Multi-facility PSA
Quantitative Health Objectives (QHOs)	

Table 3. Summary of Risk Metrics for Integrated Site Safety Assessment

The relationship between CDF and SCDF can be seen in the following equations. Equations 1through 5 demonstrate how using newly established risk metrics, the risk can then be integrated for the overall site, starting from the single unit risk assessments.

$$CDF = CDF_1 + CDF_2 \quad (1)$$

$$SCDF = SUCDF + MUCDF \quad (2)$$

$$SUCDF = 2CDF_1 \quad (3)$$

$$MUCDF = CDF_2 \quad (4)$$

$$SCDF = 2CDF_1 + CDF_2 = 2CDF - CDF_2 \quad (5)$$

Where:

CDF: Reactor CDF, frequency of core damage involving a specific reactor, per reactor year

SCDF: Site CDF, frequency of core damage on one or more reactors at the site, per site year

SUCDF: Site single unit CDF, frequency of an accident involving core damage involving a single reactor unit, per site year

MUCDF: Site Multi-unit CDF, frequency of an accident involving core damage involving multiple reactor units, per site year

CDF_1: Single unit core damage frequency, frequency of core damage on one reactor, per reactor year

CDF_2: Dual unit core damage frequency, frequency of core damage on two reactors concurrently, per site (pair of reactors) year

4.6 Human reliability

In current PSA models credit is taken for operator recovery actions and accident management for the recovery of the plant from a degraded state or core damage condition. As demonstrated in the Fukushima accident these activities can be severely restricted by releases at other installations. The human reliability analysis for single units does not take such a scenario into consideration.

In multiple-unit sites situation, specific human reliability analysis of the actions and activities to be taken by shared staff, especially in light of work-load and availability of staff, during a scenarios affecting several units should be performed.

In addition, for multiunit site the human reliability analysis needs to account for condition where the site is contaminated with radioactive material and accident management action need to be executed in this environment, adding another level of complexity to the safety assessment of multiunit sites.

4.7 Illustrative example

To illustrate the steps to follow in estimating the event frequencies for a multi-unit PRA, let's consider the case of loss of offsite power (LOOP) at a site with two identical reactor units. In traditional PRAs that are performed on each reactor separately, it is customary to analyse the initiating event frequencies on a reactor basis and for a multi-unit site, each unit is analyzed separately. In a multi-unit PRA it is necessary to resolve which events impact each reactor separately and independently and which impact both units concurrently. This requires careful analysis of the industry data which may come from an assortment of sites with different numbers of reactors on each site.

The example is quantified using data in table 4 that have been recently developed for U.S. nuclear plant PRAs. The event tree for this example is shown in Figure 5. LOOP/SBO event tree. This event tree models the occurrence of both multi-unit and single unit loss of offsite power events at a two unit site, and the response of the emergency diesel generators (EDGs) at each unit in a manner that is similar to the Seabrook PRA, [9].

When comparing these results against those of typical existing PRAs there are two key differences. One is that the frequency of a single unit LOOP is increased to reflect this is a site based frequency. The other is that there are different results for LOOP events and SBO events

involving single units and both units on this example two unit site. While the frequency of the dual unit SBO is significantly smaller than that for a single unit, it is sufficiently high to avoid screening out of a multi-unit PRA. Note that this example did not include the probability of non-recovery of offsite or onsite power, nor did it include other components such as breakers, fuel transfer pumps, and other components whose failure or unavailability could contribute to an SBO sequence at one or multiple units.

Model Parameter	Assumed Value	Basis
F_M = Frequency of site and region based events involving loss of offsite power at both units at a two unit site	2.39E-02 per site-year	"LOOP Event and Exposure Data" for region based and site based LOOP events
F_S = Frequency of reactor based events involving loss of offsite power	1.55E-02 per reactor-year	"LOOP Event and Exposure Data" for reactor based LOOP events
F_{Site} = Frequency of loss of offsite power at two unit site	5.49E-02 per site-year	$F_{Site} = F_M + 2F_S$ and above values
f_M = Fraction of loss of offsite power events at a two unit site involving loss of offsite power at both units	0.435	$f_M = \frac{F_M}{F_M + F_S}$ and above values
Q, EDG failure probability	$Q = \lambda_s + \lambda_r T$	Standard model for standby component
λ_s = EDG Failure rate for failure to Start or to load and run for 1hr	7.45E-03 per demand	NUREG/CR-6928 based on U.S. NPP service data
λ_r =EDG failure to run after first hour	8.48E-04 per hour	NUREG/CR-6928 based on U.S. NPP service data
T = mission time	23 hours after first hour	Model assumption
M = EDG maintenance unavalability	1.26E-02	NUREG/CR-6928 based on U.S.
		NPP service data
β = Fraction of EDG failures involving common cause failures shared with another EDG	.025	
β' = Fraction of EDG common cause failures involving failure of all 4 EDGs on both units	$\beta' = \frac{4n_4}{4n_4 + 2n_2}$	MGL model equation for non-staggered testing per NUREG/CR-4780
n_2, number of EDG common cause events with two component failures on one site	6	Seabrook PRA, one out of 7 common cause events of EDGs would impact all 4 EDGs on a multi-unit site
n_4, number of EDG common cause events with four component failures on two sites	1	

Table 4. Parameter Data for LOOP/SBO Example

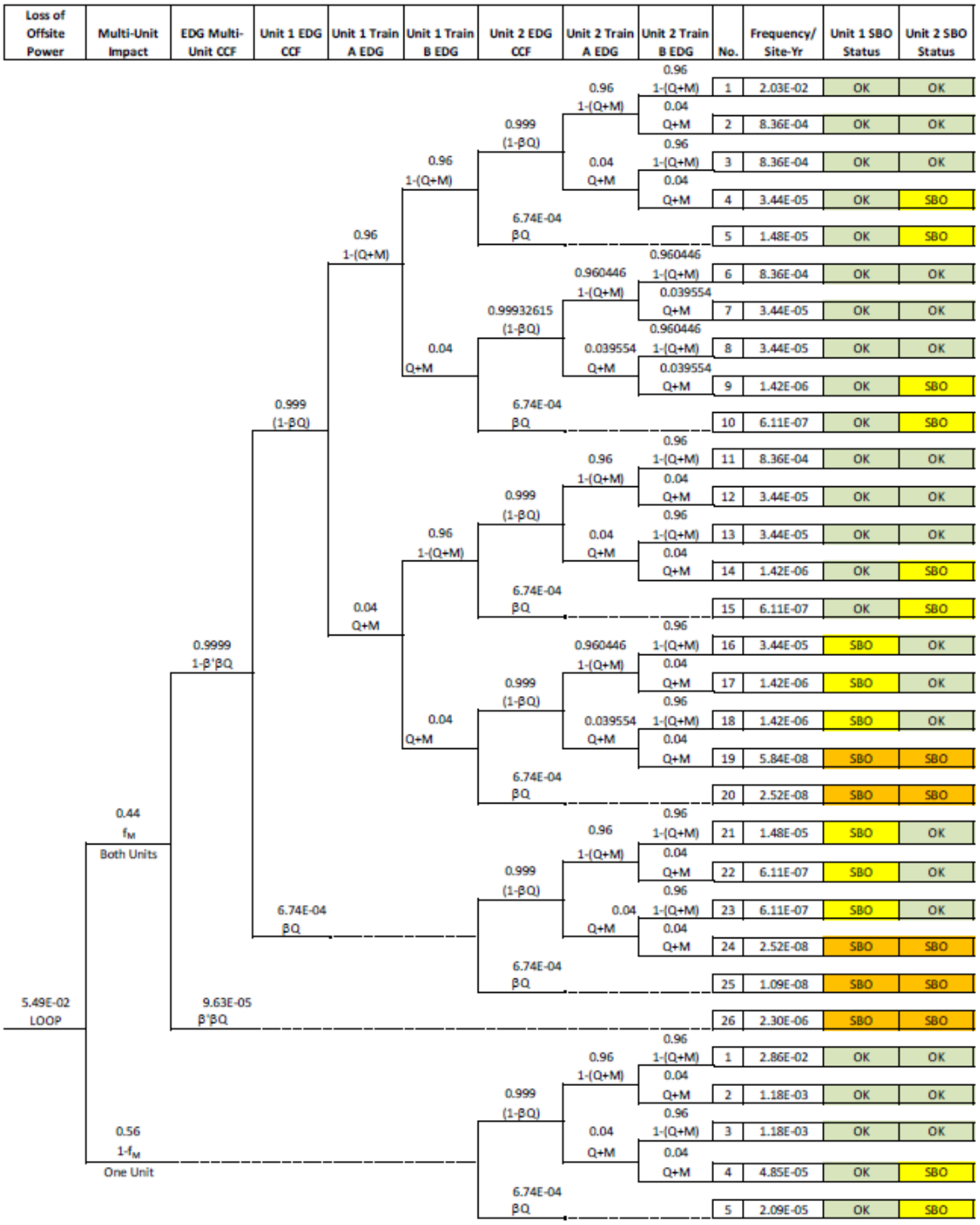

Figure 5. LOOP/SBO event tree

4.8 Conclusion and Perspective

The study has presented some possible solutions or methodological options in order to switch from a level 1unit PSA model to a model for the site to take into account the multi-unit dependencies.

The approach adopted for evaluating the risk for a site includes proposals of classification of initiating events and common systems and dependencies existing between the units on the site, as well as considerations of the human factor.

A study case was used to illustrate the proposed methodology. However, additional developments must be provided to cover the level 2 PSA, as well as level 3. Even when multi-unit aspects are taken into account in a level 1 PSA, some methodological problems arise which will need to be the focus of further development.

5. DSA in the light of Fukushima accident

Although in the past three decades a few nuclear power plants have experienced earthquake ground motions, strong earthquakes have occurred recently that have surpassed the original design value or evaluation levels and seriously affected the operation of plants, mainly in Japan.

The experience in this regard shows that operating plants were shut down immediately; afterthat they remained generally shut down for the period necessary to conduct safety assessments/evaluations: in most cases, no significant damage was identified even if in a limited number of cases, an upgrading of plant design was implemented to meet new design requirements. This is particolarly true for beyond design basis earthquakes.

Based on that the design of NPPs or its upgrading become a focal point in guaranteeing the safety of the installation (preventing any leakage to the environment and people).

The design of a nuclear power plant should provide for a sufficient margin of safety along with an evaluation of potential cliff edge effects for each natural hazard considered, to ensure that the values associated with such effects do not approach the design basis for external events.

In 2011, it was the first time that external hazards have significantly contributed to a nuclear accident: it challenges all layers of defence in depth. The overarching lesson to be learned is that an integrated approach is needed to protect nuclear installations against external hazards of similar magnitude.

The Fukushima Daiichi accident has emphasized the need for a critical re-examination of the margins of safety in the design and operation of critical facilities: safety margins should be re-assessed on a periodic basis, by taking into account the possibility of cliff edge effects.

The reassessments should also inform safety improvements, such as enhancing the existing design or providing additional safety functions.

The principal protection against seismic and tsunami hazards is provided by the development of an adequate design basis and the qualification of important safety related SSCs.

Hazard assessments using a deterministic approach allows to implicitly consider the uncertainties, by using as input data to the analyses the maximum historical events coupled with a margin of safety to compensate for incomplete knowledge.

6. Safety margin assessment

6.1 Introduction

As highlighted by the International Regulatory Bodies, to prevent the threats caused by an unexpected and extremely severe event like that of Fukushima Daiichi nuclear power plant, all the relevant technical aspects have to be considered and analysed deeply, [10]:

- Site characteristics that may affect the safety of the nuclear installations should be thoroughly investigated and fully assessed. They should be monitored throughout the lifetime of a nuclear power plant.
- Sites for nuclear installations shall be examined with regard to the frequency and severity of external events and human induced events and of 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. Tsunami hazard assessment should take into account recent advances in deterministic and probabilistic approaches, modelling, data gathering, data analysis, field investigations and other relevant activities.
- Evaluation of the effectiveness of defence in depth levels needs to consider an appropriate balance between deterministic and probabilistic approaches
- 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.

Moreover they assume a meaningful importance in the case of existing plants, which were constructed according to old requirements and rules. To consider adequately such a type of events (earthquake plus a resulting tsunami) a worldwide re-examination of the vulnerability of nuclear power plants was needed.

Under the traditional approach, the nature of such events was determined by estimation of the challenge at a particular site based on the historical record, for example, the largest known

flood or earthquake, and the plant was designed with the addition of a safety margin to take into account the uncertainties associated with such estimates.

As indicated in IAEA documents, unfortunately, the historical record may often not provide complete information as to the actual risk; therefore uncertainty may be large and the original margin could prove inadequate.

The situation may be complicated by additional factors, like the cliff edge effects, that may grow significantly the severity of the event, or by the threats that arise in combinations.

Fukushima was the first instance of a combination of extreme natural hazards initiating a nuclear accident, providing confirmation that such hazards can overwhelm a number of levels of defense in depth at nuclear power plants, including the containment building.

From that the need to investigate its dynamic behavior, when experiencing strong earthquakes in excess of its seismic design basis, with a deterministic approach (Seismic Margin Assessment - SMA).

For the SMA the important safety components are selected to assemble the success path: components needed to guarantee the fundamental safety functions (safely shut down of the plant) including dependencies and interactions with support systems and with non-safety related SSCs.

6.2 Seismic safety margin

The safety assessment of a NPP design and its re-evaluation should incorporate beyond design basis natural events- in this study consisting in 0.5 PGA earthquake followed by tsunami- in order to clearly identify and understand the failure modes of critical SSCs, with reference to their respective safety function. Moreover the identification of the possible failure modes of critical SSCs is a crucial step for the safety margin assessment.

The seismic margin assessment represents accordingly a method of assessing mainly though a deterministic approach the capability of nuclear power plants to withstand earthquakes beyond their design basis.

Deterministic method, providing the basis to analyse (and calibrate) the durability and robustness of SSCs, should be also supplemented by probabilistic methods, including the PSA.

The understanding of the integrated plant response to a natural event is recognized as extremely important in order to properly consider the potential accident sequences, the

interactions of equipment and human performance, and the effectiveness of various defence in depth features.

6.3 Main issues and lessons from the Fukushima Daiichi accident in relation to earthquakes and tsunamis

Based on the main issues and the lessons learnt from Fukushima event, as in IAEA 2012 report, when dealing with external natural hazards, there is a need to ensure:

- the design of nuclear plants includes sufficient protection against infrequent and complex combinations of external events, specifically those that can cause site flooding and that may have longer term impacts;
- 'dry site' layout concept should be adopted, where practicable, as a defense in depth measure against flooding as well as physical separation and diversity of critical safety systems;
- any changes in external hazards or understanding of them should be periodically reviewed for their impact on the current plant configuration
- aging of material is duly considered in order to amplify cliff edge effects that could impair the strength of structure.

This latter is widely discussed in the what follows.

6.4 Aging of structure

Nuclear power plants (NPPs) are designed, built, and operated to standards that aim to reduce the releases of radioactive materials to levels as low as reasonably achievable.

The safety-related reinforced concrete structures in these plants are designed to withstand loadings incurred during normal plant operation are generally not significant enough to cause appreciable degradation.

NPPs, however, involve complex engineering structures and components operating in demanding environments that potentially can challenge the high level of safety (i.e., safety margins) required throughout the operating life of the plant.

The material degradation effects may accumulate within the structures over time to cause failure under design conditions, or lead to repair. Ensuring that the structural capacity of the

reinforced concrete structures has not deteriorated unacceptably due to aging or environmental effects is essential in a plant because, if necessary, the replacement of most of the safety-related concrete structural components would be economically unfeasible.

All nuclear power plants will progressively undergo over time (till the end of plant life) the effects of structural aging; generally they determine changes in mechanical properties such as creep, modulus, and ultimate compressive and tensile strengths.

Structural aging, on the other hand, is the combined effects of changes in the time-dependent material properties, the prior physical changes resulting from the structure's past operating history, and the structure's new loading environment.

Concrete structures, like the containment system, are those mostly affect by aging even under operating conditions, and due to the high non linearity effects that characterizes it the structure's response will have to be performed using non-linear methods that consider the structure's changing physical conditions.

Three most important aspects of aging are:

1) effects due to expected time-dependent changes in material properties;
2) effects due to unexpected degradation in material properties, and
3) effects due to actual environmental and loading conditions encountered.

As to the first aspect of aging dependence, certain characteristics of material aging are generally beneficial, such as the increase in time of the ultimate strength and modulus. However, this also means that the stiffness of the concrete structure also increases over time, and for loading conditions that are functions of stiffness, such as thermal and seismic loads. This latter could lead to some non-beneficial effects because of higher stresses, which in turn could cause cracking and material degradation.

With respect to point 2), evidence of significant material property degradation is generally observed in unanticipated structural movement or unusual cracking trauma.

The third category of aging dependence highlights the importance of treating the true environmental and loading conditions, rather than considering only assumed design conditions that may or may not be encountered over the life of the structure.

Primary mechanisms that can cause a premature deterioration of concrete structures are related to the bonding between concrete and steel materials.

Degradation of concrete can be caused by adverse performance of either its cement-paste matrix or aggregate materials under chemical or physical attack. In particular the physical attack mechanisms include thermal exposure/cycling, abrasion/erosion, irradiation, and fatigue or vibration. Degradation of mild steel can instead occur as a result of corrosion, irradiation, elevated temperature, or fatigue effects.

Structural loads related to random event in nature may also determine variations in engineering material properties, and strength degradation mechanisms.

7. Safety assessment of an existing Gen. II containment under multiple event

To the aim of the safety margin assessment, the dynamic behaviour of Gen. II reactor hit by natural external events, was analysed. The motivation of this evaluation is associated to the new experienced high magnitude of the actual earthquakes and to the performance of SSCs, which have to demonstrate their integrity, when facing severe accident conditions.

To identify plant-specific vulnerabilities and other important insights, numerical simulations have been carried out with adequate numerical tools taking into account the aging/degradation mechanisms of the plant structures.

The duration of structural loads that arise from rare operating or environmental events such as accidental impact, earthquakes, and tornadoes, is short and such events occupy a negligible fraction of a structure's service life. Despite this, reliability, and performance at the desired level, of existing structures must be ensured (proven).

7.1 Modelling of the containment

The containment structure generally consists of a concrete basemat foundation, vertical cylindrical walls, and dome.

The basemat may consist of a simple mat foundation on fill, natural cut or bedrock, or may be a pile/pile cap arrangement. Its interior surfaces are lined with a thin carbon steel liner to prevent leakage. As in [11] and [12] a typical Gen. II Containment, like the one showed in Figure 6, is 60 m tall and 42-45 m in diameter and is about 1 m thicker.

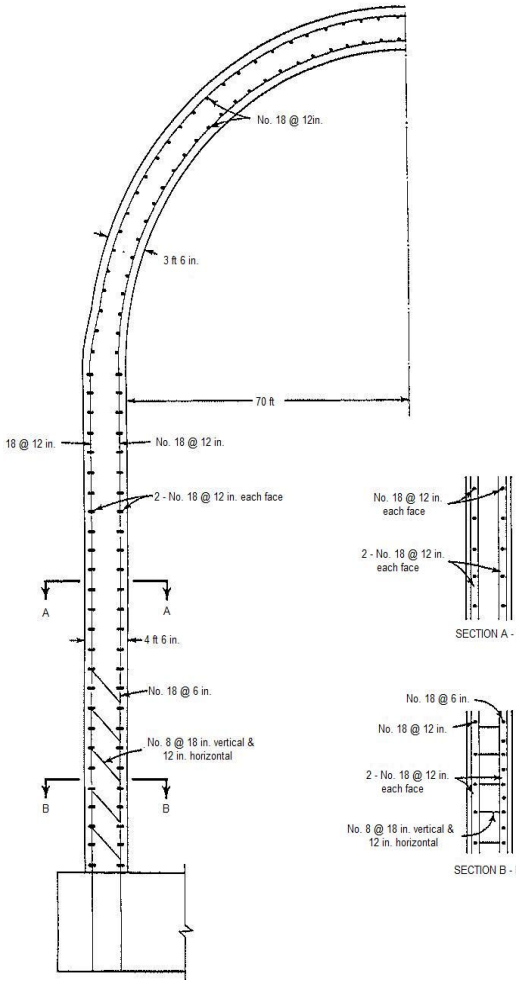

Figure 6. Distribution of steel bars in the containment wall [11].

The containment in PWR plants are typically constructed of reinforced concrete and tend to be more massive in nature than t in BWR plants because they typically support the reactor pressure vessel, steam generators, and other large equipment and tanks.

By design it is designed to not exceed the allowable stresses, in agreement with the ASME III Div. B and ACI Standard 359 [13],[14].

The assessment of hazard weakness of such a typical GEN II plant entails unavoidably with:

- the geometry of the system or component in relation to its function;
- the possibly failure mode (ductile or brittle, large displacement, vibration sensitivity, unacceptable function even though stress or displacement is within acceptable limits;
- the 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.

To the aim of this study, the evaluation of safety margin was performed with reference to the structure described in [12]. In doing that a quite detailed numerical model (Figure 7) was set up and implemented, by Marc© code [15], assuming the same geometry and reinforcement distribution shown in Figure 6. The foundation in the analysed case was assumed based on a rock soil ("rigid foundation").

Indeed, suitable material properties were considered in order to proper represent the effect of the aging on the containment performance.

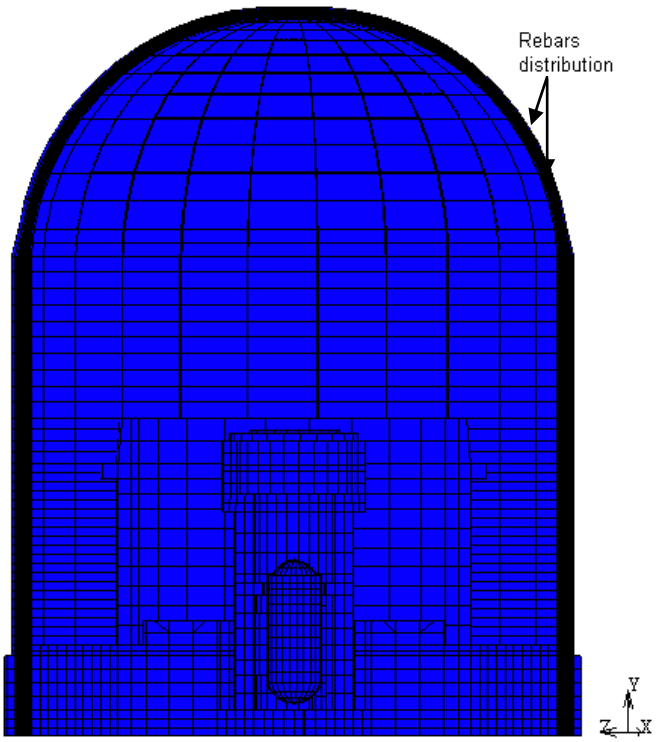

Figure 7. Containment model

7.2 Material properties

Nuclear safety-related concrete structures are composed of several constituents that, in concert, perform multiple functions (e.g., load-carrying capacity, radiation shielding, and leak tightness). Primarily, these constituents include the following material systems: concrete, conventional steel reinforcement, prestressing steel, steel liner plate, and embedment steel. Most of the reinforcing steel bars have to provide primary tensile and shear load resistance/transfer. They are made of carbon steel with deformations (lugs or protrusions) on

the surface to increase the adherence and bonding with concrete. These bars typically conform with ASTM A 615 or A 706 specifications; the minimum yield strength of these materials range from 270 MPa to 415 MPa (the 415 MPa material is the mostly common used).The steel reinforcement (distributed like in Figure 6) or rebar phase is treated as an elastic-plastic 3-D bar that derives its local stress response from the surrounding strain field.

The behavior of concrete is highly nonlinear, having low tensile strength, shear stiffness and strength that depend on crack widths, and a confinement-dependent compressive elasto-plasticity. A material constitutive model capable of simulating the behavior of reinforced concrete and capturing the effects of structural aging is provided.

The model treats reinforced concrete as a three-phase composite: plain concrete material as a three-dimensional continuum phase, steel reinforcement (rebar) as a uni-directional phase, and a rebar-concrete interaction phase.

The primary behavioral regimes considered for the concrete phase are:

- tensile cracking under multi-axial tensile stress field;
- compressive yielding; and
- crushing.

The behaviour was assumed linear elastic up to the point of failure, beyond which progressive failure and damages, as function of stress criterion, occur.

The concrete aging has been considered.

It analytically consists in a variation of the mechanical material properties as a result of (micro)structural changes caused by the loading conditions.

Form a numerical point of view, this aspect is introduced in the model by means of the CRACK DATA option, which allows to predicting crack initiation and simulate tension softening, plastic yielding and crushing. The converge option is set up such that these changes would not have to be detrimental to the point that reinforced concrete will not be able to meet its functional and performance requirements.

When cracking is initiated, the tensile stress normal to the crack surface, which is the principal plane, is reduced to zero and the stresses for the material point are recalculated in a subsequent iteration, thereby restoring equilibrium through stress redistribution to reinforcement and other points in the structure.

In addition under dynamic loading, the response of a concrete structure is strongly dependent on internal energy dissipation due to cracking: when cracks form, a large amount of energy loss occurs locally, anisotropically, at the crack locations.

Time-dependent material degradation mechanisms that can affect the long term performance of concrete structure as well as the temperature effects or the alkali aggregate reaction are not considered.

7.3 Numerical modelling

The numerical model set up for the aim of this study is represented in previous 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 evolution of cracks in a structure results in the reduction of the load carrying capacity. The internal stresses need to be redistributed through regions that have not failed. This is a highly nonlinear problem and can result in the ultimate failure of the structure.

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

The integration method used is the Newmark-Beta, whose generalized forms of integration (in u and v) are:

$$u^{n+1} = u^n + \Delta t v^n + \left(\frac{1}{2} - \beta\right)\Delta t^2 a^n + \beta \Delta t^2 a^{n+1} \tag{6}$$

$$v^{n+1} = v^n + (1-\gamma)\Delta t a^n + \gamma \Delta t a^{n+1} \tag{7}$$

Where n indicates the n-th integration step. By considering γ = ½ and β= ¼, the equation of the dynamic of the transient analysis becomes:

$$\left(\frac{4}{\Delta t^2} M + \frac{2}{\Delta t} C + K\right)\Delta U = F^{n+1} + R^n + M\left(a^n + \frac{4}{\Delta t} R v^n\right) + C v^n \tag{8}$$

Where M is the matrix of mass, K the matrix of stiffness and C the matrix of damping; R is the matrix of the internal forces.

In the performed transient analyses, it was assumed the occurrence of an earthquake of 0.5 g peak ground acceleration (PGA) followed by a tsunami of 20 m breaking wave height. The seismic motion (20 s duration) was represented by means of three independent acceleration time histories - ATH (Figure 8) [16][17], applied two along the horizontal directions (A_x and A_z) and one along the vertical one (A_{vert}).

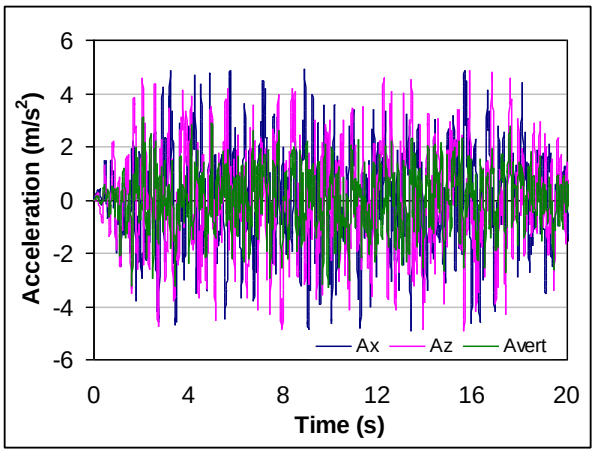

Figure 8. Input Seismic motion

For earthquake induced tsunamis, the flooding hazard (run-off) can be assessed by determining the maximum impact force and pressure provided by the elevated breaking waves. The impact and lateral pushing of the waves, and the destructive power of a large volume of water dragging debris and missiles inland (like observed at Fukushima plants) is the primarily responsible for the massive and/or catastrophic damages of the in-site buildings. The methodology adopted to calculate the maximum wave elevation and its associated pressure is described in [1]. All these dynamic forces have been represented in terms of pressure acting on the outer walls of the containment structure; the pressure was calculated as:

$$P_{\max} = C_p \gamma_w d_s + 1.2 \gamma_w d_s \quad (9)$$

where γ_w is the water unit weight, equal to 10.05 kN/m^3 for the sea water; C_p the dynamic pressure coefficient, which depends on the category of risk associated with the extreme natural events considered, i.e. the flooding and earthquake; d_s 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.

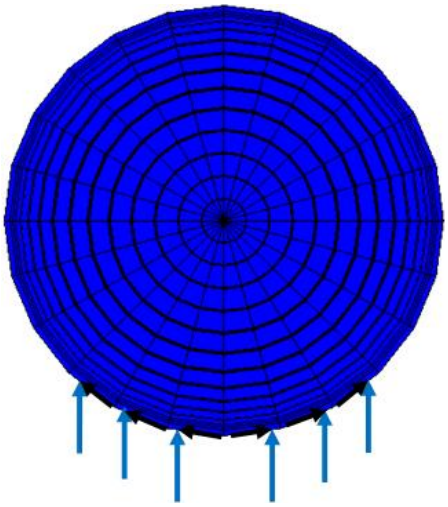

Figure 9. Directions of the pressure breaking-wave.

7.4 Results discussion

In respect to the outcomes of the study performed in framework of PAR 2012, the results obtained from simulating multiple events show ~ 35% acceleration amplification at the SGs upper restraints (about +35 m from the ground level), as in Figure 10.

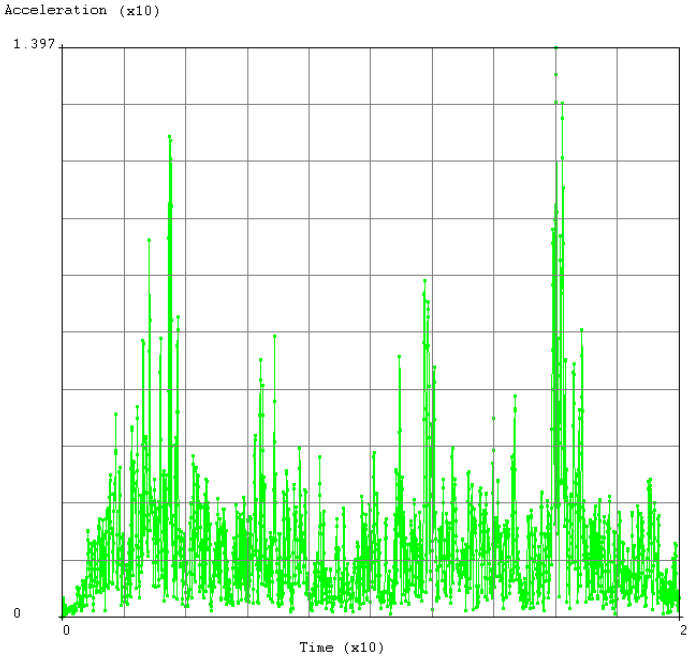

Figure 10. Acceleration at the SG upper restraints

Figure 11 shows the plot of the overall acceleration vs. time at the foundation mat, outer

containment wall, and rebar positioned at half the section: these location are identified respectively by the node 35664, node 19809, and node 20793.

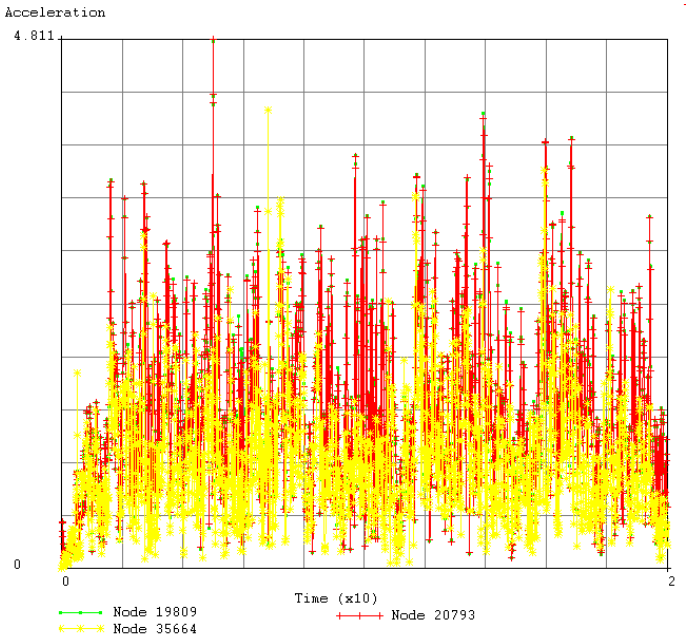

Figure 11. Acceleration at several location in the containment structure

It was also observed an overall wall displacement greater on the surface containment hit by the water waves (though not linearly) than the opposite one: the maximum relative displacement was about 35 cm (Figure 12). Such a large displacement indicates that the deterioration of concrete wall is occurring. As for the red curve it must consider that rebars did not plasticize, as it could be observed in Figure 13.

The containment wall resulted mainly compressed; the compression stress appears to be localized in zones where the ground motion plus the waves' hydrodynamic force are applied.

Tension instead appears at the inner surface of the containment wall, even if, in general, compression resulted higher and predominant than tension. Indeed, when the incremental load goes over the maximum level, stress overcomes the imposed limit in compression or tension and the concrete structure fractures suddenly.

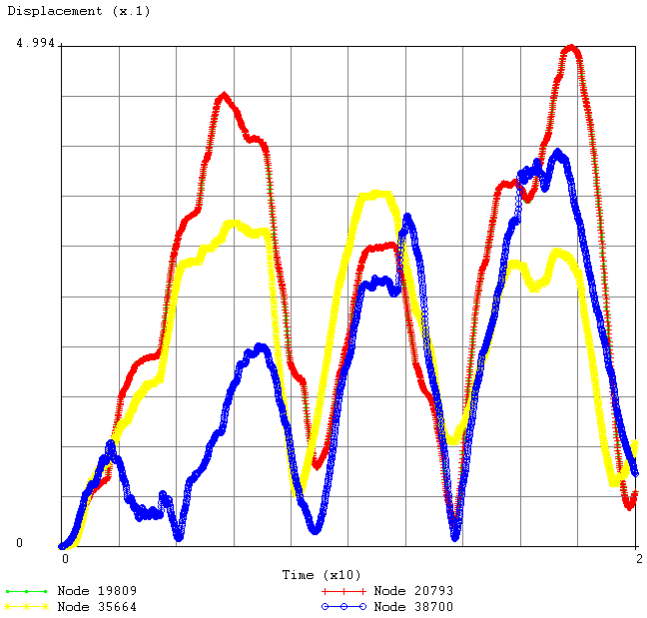

Figure 12. Relative displacement at several location in the containment structure.

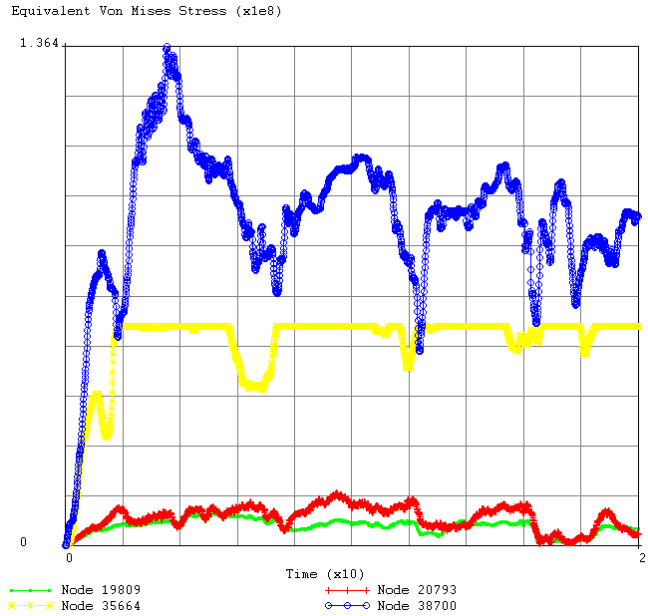

Figure 13. Von Mises stress at several location in the containment structure.

The relationship between stress and crack growth is stress-strain dependent. This gradual deterioration phase is consistent with the relationship shown in Figure 14: after an initial linear portion lasting up to about 30 - 40% of the ultimate load, the behaviour of concrete becomes non-linear, with large strains being registered for small increments of stress.

The non-linearity, appearing 2 seconds after the beginning of the dynamic transient, primarily indicates the coalescence of microcracks inside the concrete. The ultimate stress is reached when a large crack network is formed within the concrete, consisting of the coalesced microcracks and the cracks in the cement matrix. The stress-strain behaviour in tension is similar to that in compression.

By considering that the stress exceeds locally in the containment the allowable limit, it is confirmed the presence of structural damaging/failure. Nevertheless, no loss of the integrity appears because of the ductility of concrete (several orders of magnitude lower than steel) that still exhibits considerable deformation before failure.

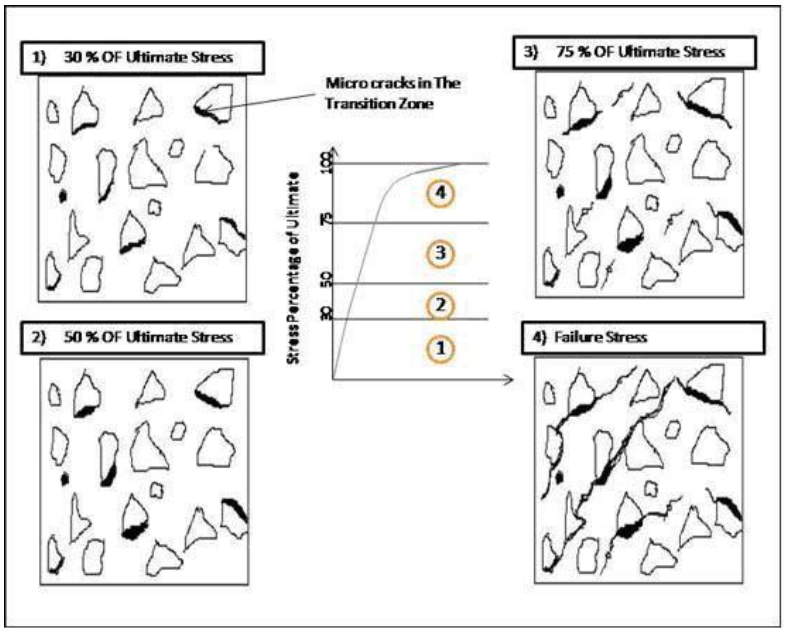

Figure 14. Stress-strain relationship for concrete.

The results indicate that the demand exceeds the design one in some part of the outer walls close to the foundation of the containment building. These zones along with the SG restraints and piping represent the weak points the plant examined (vulnerabilities to manage).

The deformation of structure contribute to dissipate the earthquake energy, as also expected.

Analysing the effects caused by the multiple accident scenario, it appears also clearly that the tsunami is more damaging than the earthquake event. This results is in very good agreement with the outcomes highlighted by the lessons learnt from Fukushima.

In conclusion it could be remarked that the aging, influencing the degradation process of concrete containment, which is suffering void nucleation and cracking, did not determine spalling, scabbing, etc. and therefore a loss of the containment and confinement functions of structure.

8. Conclusions

In this report the lessons learned on PSA Level2 as well as multiunit risk assessment using the implications from Fukushima accident are considered to identify and analyze gaps in the probabilistic risk analysis state of practice, to gain insights and to derive related conclusions and recommendations, including some discussion regarding the possibilities to address and solve them in future PSA studies.

An important challenge pertains to the capability of PSA Level2 aspects to meet the issues raised by the Fukushima event, as the adequacy of the currently adopted models, like for instance the newly identified plant damage states at the interface with Level1 PSA, the implementation of severe accident management measures, the full scope PSA development aimed at covering all the plant operating states, all the radioactive sources and addressing all the hazards.

To gain an accurate view of a site's risk profile, a measure of Core Damage Frequency (CDF) representing the site rather than the unit should be considered and estimated through a multi-unit PRA. In doing so possible unit-to-unit interactions and dependencies should be modeled and accounted for in the site CDF.

PSA models for multi-unit sites should systematically include relevant dependencies on the systems levels, e.g. via shared support systems or buildings, as well as dependencies on the accident sequence level, e.g. via the impact of a severe accident in one unit on measures or systems in another unit, into their PSA models. In addition, shared staff resources, mobile equipment, etc. have to be considered. This might require dedicated human reliability analysis. For adequately covering complex scenarios simultaneously affecting several units, site risk PSA models should be developed.

On these issues further developments are needed. To the aim, in the report, an effort has been performed to envisage some of them, which have been proposed for PSA modeling implementation and quantification improvement.

Hazard assessments has been also carried by using a deterministic approach in order to implicitly consider uncertainties by using as input data with suitable margin of safety to compensate them.

The Fukushima Daiichi accident has emphasized the need for a critical re-examination of the margins of safety in the design and operation of critical facilities, therefore in this study the

safety margins of an exixsting NPP has been evaluated by taking into account possibly cliff edge and aging effects caused by the loading conditions (20m height waves and 0.5 g PGA).

To identify plant-specific vulnerabilities and other important insights, numerical simulations were carried out with adequate numerical tools taking into account the aging/degradation mechanisms of the plant structures.

The behavioral regimes for the concrete took into account the tensile cracking under multi-axial tensile stress field, compressive yielding, and crushing: the behaviour was assumed linear elastic up to the point of failure, beyond which progressive failure and damages occur.

The critical analysis of the results obtained from the simulation of the containment building performance highlights that:

- accelerations, even if generally “damped” by the deformation/cracking of containment, are amplified of 35 % at the upper restraints of the SGs;
- a relative displacement large than 35 cm indicates that the deterioration of concrete wall is occurring;
- the deterioration of concrete is stress-strain dependent.: after an initial linear portion lasting up to about 30 - 40% of the ultimate load, the behaviour of concrete becomes non-linear, with large strains being registered for small increments of stress. The microcracks begin to form inside the concrete 2 s after the beginning of the transient, instant at which the concrete material begins to behave as non linear.
- some vulnerabilities appear in the structure to manage which improvement of the design will be needed.

Finally with reference to the accident management, the actions needed to improve the safety of the plant might consist in the adoption of elevated sea walls, shelter or items capable to avoid a direct impact of waves onto the plant, the adoption of seismic isolation to be positioned at a height much greater than the design flood elevation, of course, as well as the power supply sources (operational and emergency systems).

On these issues further developments are needed, even if, in this study, some of them, which have been proposed for DSA modeling have been envisaged.

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