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Probabilistic Safety Assessment Benchmarks of Multi-unit, Multi-reactor Sites

Report of a Coordinated Research Project

IAEA

International Atomic Energy Agency

PROBABILISTIC SAFETY ASSESSMENT BENCHMARKS OF MULTI-UNIT, MULTI-REACTOR SITES

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IAEA-TECDOC-2044

PROBABILISTIC SAFETY ASSESSMENT BENCHMARKS OF MULTI-UNIT, MULTI-REACTOR SITES

FINAL REPORT OF A COORDINATED RESEARCH PROJECT

INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The IAEA organizes coordinated research projects to facilitate cooperation in research and development between organizations in its Member States, including the development and validation of analysis methods or computer codes for design and safety analysis of nuclear power plants. Safety assessments of nuclear power plants have predominantly been based on deterministic and probabilistic approaches applied to a single unit. The risk at a site with multiple reactors has often been represented by summing up, or combining in a simplistic fashion, the risks from individual units, sometimes restricted only to internal initiating events. Such simplification has several limitations. For example, potentially complex interactions during a severe event are not included although they may have an impact on a multi-unit site; this is especially true with regard to external hazards, as was the case in the accident at the Fukushima Daiichi nuclear power plant. The proper assessment of the impact of shared equipment requires consideration of the entire site in a holistic way.

The need to consider effects from multiple units while performing a Level 1 probabilistic safety assessment (PSA) to determine core damage frequency was recognized in IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, published in 2010,prior to the accident at the Fukushima Daiichi nuclear power plant, and further elaborated in IAEA-TECDOC-1804, published in 2016. However, these publications provide limited guidance and information on how such analysis can be performed or evaluated in the case of sites with more than one reactor unit and/or reactor design. To address this gap, as well as the lack of information from ongoing national developments concerning multi-unit PSA (or MUPSA), the IAEA developed a methodology for multi-unit PSA, which is reflected in Safety Reports Series No. 110.

The objective of the present publication is to summarize the results of benchmark calculations developed under the coordinated research project on probabilistic safety assessment benchmarks for multi-unit, multi-reactor sites, conducted between 2017 and 2022. Quantitative insights for safety (or integral risk) in the context of MUPSA can be overlooked in single unit PSAs, and they can be different for different nuclear power plant sites. This publication therefore summarizes the background of the issue and provides both a description of the multi-unit sites that are the basis for the benchmarks, development and assessment of multi-unit PSA models, and the insights gained during the benchmark exercises. It is expected that future multi-unit PSAs will be compared against the benchmarks presented in this publication.

The IAEA expresses its appreciation to all participating organizations for developing and performing these original analyses and releasing the results to the international community. The IAEA officers responsible for this publication were M. Krause and T. Jevremovic of the Division of Nuclear Power.

Contents

1. INTRODUCTION

1.1. BACKGROUND

Nuclear power plants (NPPs) may consist of same or different reactor units' types, their designs, sizes or age, and they may all be located at a single site. The probabilistic safety assessments (PSAs) of NPPs estimates the risk arising from damage to a single unit at a time while the risk for a site with multiple reactor units is determined by simply summing up or combining the estimated risks from individual reactor units. This simplification in PSA has several limitations; potentially complex interactions during a severe event are not included while they may have an impact on a multiunit site; this is especially true regarding the external hazards, as it was in the Fukushima Daiichi NPP accident. The proper assessment of the impact of shared equipment requires consideration of the entire site in a holistic way.

The Coordinated Research Project (CRP) on Probabilistic Safety Assessment Benchmarks for Multiunit Multi Reactor Sites (2018–2022) brought together the experts from the IAEA Member States with mainly water cooled reactor technologies to utilize, test and further develop their current or planned PSA methods and assumptions by conducting and comparing results of meaningful multiunit multi reactor type probabilistic safety assessment (MUPSA) benchmark exercises.

The first practical MUPSA study can be referred as Seabrook PSA (mid–1980's, considering two units), which presented the integrated site risk in the form of the Farmer's curve[1]. Several methods have since been proposed and are being developed around the world to cover MUPSA considerations including risk–related safety goals and Level 1/ Level 2 MUPSA risk metrics and risk aggregation issues. These methods are not harmonized but have the same objectives. Therefore, a benchmark exercise is a useful means of fostering detailed technical discussions and facilitating mutual learning and improvements in the various methods. The results from this CRP were found to be useful to other advanced reactor types including small modular reactors (SMRs).

The need for consideration of multiunit effects while performing a Level 1 PSA (Level 1 refers to the calculation of core damage frequency (CDF)) was already indicated in the IAEA Safety Guide SSG–3 (2010, i.e. pre–Fukushima Daiichi NPP accident) [2] and further elaborated in the IAEA TECDOC–1804 Attributes of Full Scope Level 1 Probabilistic Safety Assessment for Applications in Nuclear Power Plants published in 2016. However, these publications provide very limited guidance on how such analysis can be performed or evaluated in the case of sites with more than one reactor unit and/or type, or how they need to be used in Level 2 PSA (Level 2 refers to the calculation of the frequency, magnitude and other relevant characteristics of the release of radioactive materials to the environment, e.g. calculation of large release frequency (LRF)). In addition, there is a lack of public information from ongoing national developments related to MUPSA.

Shortly after the Fukushima Daiichi NPP accident, the International Workshop on the Safety of Multiunit Nuclear Power Plant Sites against External Natural Hazard was organized by the IAEA and India Atomic

[1] Farmer's curve represents a frequency–consequence dependence for a single event scenario, while the complementary cumulative distribution function (CCDF) curve describes cumulative frequencies of accidents exceeding given doses from the entire spectrum of accident sequences.

Energy Regulatory Board and the Bhabha Atomic Research Centre (BARC) in Mumbai, India, 17–19 October 2012. The importance of multiunit considerations was strongly emphasized and the need for further research and development identified. One main conclusion from a 2014 CANDU Safety Association for Sustainability of Pressurized Heavy Water Reactor (PHWR) Safety Workshop, which focused on post–Fukushima Daiichi NPP accident R&D status and needs, was the need for improving PSA methodologies applied to multiunit, multi reactor type sites in an integrated way. A subsequent international workshop was held in Ottawa in November 2014 with many participants and a similar outcome [1].

1.2. OBJECTIVE

The objective of this publication is to summarize the national expertise and the results from newly developed MUPSA benchmark calculations. These calculations involved the development of MUPSA approach based on single unit PSAs (SUPSAs) available and used in Member States. Qualitative insights for safety, the goal of any PSA, in the context of MUPSA can be something not obvious from SUPSA and they can be different for different NPP sites. Evaluating these in the context of specific site layout or features was the main research aspect as detailed in this publication. The insights gained from the benchmarks are expected to further identify technology solutions towards reducing those risks that are prevalent to multiunit sites. The intended audience of this publication are professionals involved in and familiar with the terminology, development, conduct, and assessment of PSA.

1.3. SCOPE

The scope of this publication is to describe newly developed MUPSA relevant benchmarks and their specifics, corresponding calculation results and lessons learned. I development of MUPSA benchmarks is based on SUPSAs available to Member States' participants in the context of specific site configurations. This publication summarizes the methodologies developed and applied, the results obtained, and the lessons learned by the 15 participating organizations from 12 Member States.

Level 1 PSA is required to quantify the aggregate probability of severe damage to the reactor core from all plausible hazards, initiating events, and potential event sequences, while Level 2 is done to assess the overall risk of a release of radioactivity from the reactor containment. Risk, in general, is a multi-attribute quantity expressing hazard, danger or chance of harmful or injurious consequences associated with exposures or potential exposures. It relates to quantities such as the probability that specific deleterious consequences may arise and the magnitude and character of such consequences. In the case of a single unit analysis, the approach at both levels is well established and IAEA guidance is available at detail in Safety Guides SSG–3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [2], and SSG–4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [3]. When considering the potential additional risks from a multiunit site with numerous release sources and unit dependencies, the approach to include event or failure sequences involving more than one unit, and how to aggregate risks from various sources is explicitly covered in Safety Guides SSG–3 [2]. In terms of overall additional site risk to the public, workers and environment, the presence of several core damage events has much less impact than several radioactive releases. Noting that only at a Level 2 the relative impact of multiunit configurations could be meaningfully determined, the Level 2 MUPSA is also considered.

The Level 1 MUPSA consisted not only of calculating the aggregate probability of a core damage, but the probabilities of one or more cores (and possibly also spent fuel pools (SFPs)) being damaged, including coincident and consequential damage. Multiunit sequences are those that result in more than one source (core or SFP) being damaged. The Level 2 MUPSA analysis, analogously, identified sequences resulting in significant release from more than one source of radioactive material to the environment, and their relative contribution to all releases.

The important outcome of the benchmarks was not the absolute quantities of CDF or LRF, but the relative contribution(s) from multiunit event sequences, involving more than one unit, to the overall risks, compared to single unit event sequences.

1.4. STRUCTURE

This publication is divided into five Sections, including the Introduction in Section 1. The benchmark specifications are presented in Section 2. The Level 1 and Level 2 MUPSA results are described in Sections 3 and 4, respectively. The lessons learned from the benchmarks are summarized in Section 5. The conclusions are discussed in Section 6.

2. BENCHMARKS SPECIFICATION

2.1. ARGENTINA/CNEA

2.1.1 Site and plant information

The postulated hypothetical site for the benchmark is assumed to have a flat terrain with subtropical weather conditions near a river with a uniform population distribution surrounding the site.

The site includes two CAREM25–like SMRs with thermal power of 100 MWth each. The site has a single shared SFP for long term storage, which is considered as a source of radionuclide release. It is assumed that all units share the same main control room (MCR), as shown in Fig. 1. The CAREM25–like SMR has the design basis of CAREM25, including the strategy to control initiating events, but some design parameters and systems are postulated differently for the purpose of this benchmark.

The CAREM25 prototype SMR is a light water integral reactor with its design features available in IAEA's ARIS database [4]. The high energy primary system, core, steam generators, primary coolant and steam dome, are contained inside a single pressure vessel.

The core coolant circulation in the reactor primary system is achieved by natural circulation by placing the steam generators above the core. This integral reactor has several other innovative features, such as, but not limited to, passive safety systems, self–pressurization and reactivity control without boron in the coolant.

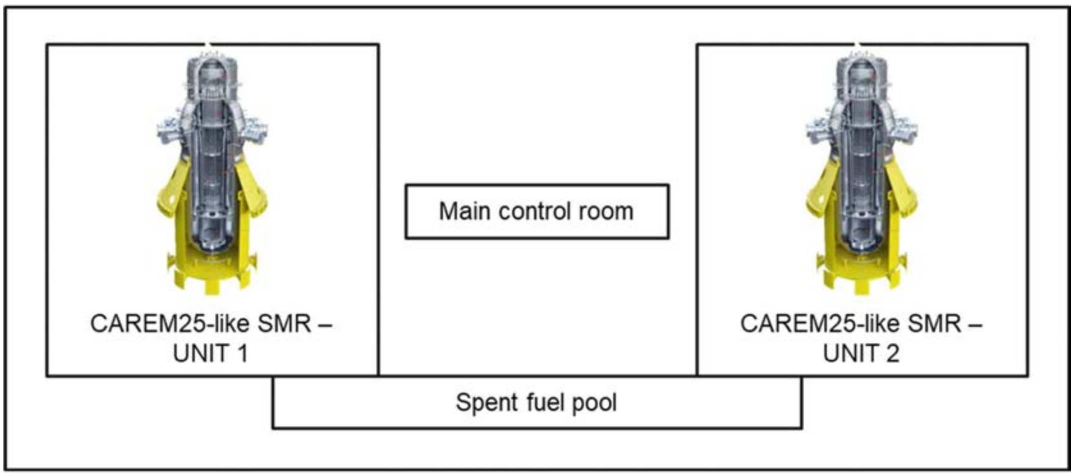

FIG. 1. Hypothetical multiunit site as a case study.

The strategy to control initiating events in CAREM25 SMR comprises two stages and main and diverse protection lines. After the occurrence of an initiating event, the first stage, Stage 1, is a grace period in which passive safety systems are required to fulfil their fundamental safety functions, namely control of reactivity and core cooling. If these systems succeed, then the reactor achieves a safe state. After the grace period, active systems and human actions are required during Stage 2 to achieve a final safe state or to extend the safe state. This strategy is also postulated for the CAREM25–like SMR in the benchmark analysis.

2.1.2 Shared systems

The benchmark assumes that each unit has its own passive and active systems (in the framework of defence in depth (DiD), Level 3A, Stages 1 and 2, respectively). The diesel generator system is a support system required for active systems in case of loss of offsite power (LOOP). This system is shared between both units and the SFP (DiD Level 3A, Stage 2). It can also be used in support of DiD Level 2 systems. The external water supply system (EWSS) is a support system for DiD Level 3B, Stage 2 (equivalent to design extension condition – DEC A), which provides water from different external sources, and is shared by two units and the SFP. The connection of systems of each unit and SFP with external water supply is shown in Fig. 2. The SFP requires active cooling during Stage 2 Iy its cooling system (DiD Level 3A) or the SFP injection DEC system (DiD Level 3B). In the case of a severe accident, each unit and the SFP have dedicated systems for their management (DiD Level 4; equivalent to DEC B).

The reactor pressure vessel (RPV) of each unit can be cooled by the RPV external cooling system (RPVECS) to remove the generated heat from the corium inside the RPV lower plenum to maintain its integrity (DiD Level 4). This system requires that operators manoeuvre in the field to connect it with EWSS. In the benchmark, RPVECS is shared between units 1 and 2, but the EWSS can feed only one unit at a time. Moreover, each unit has a passive autocatalytic hydrogen recombiner system (PARS), to limit hydrogen concentration in containment to reduce the possibility of deflagration or detonation. In unit 1 and 2, a containment venting from suppression pool system (CVS) is used to limit the containment pressure to maintain its integrity, and therefore to limit radioactive releases.

With respect to SFP, severe accident mitigation systems, the SFP spray cooling system is to limit the damage in fuel elements due to water level decreasing below their top in case of a hypothetical pool integrity loss.

This system has to be connected to EWSS as shown in Fig. 2. The EWSS can feed RPVECS connected to one unit and the SFP spray cooling system.

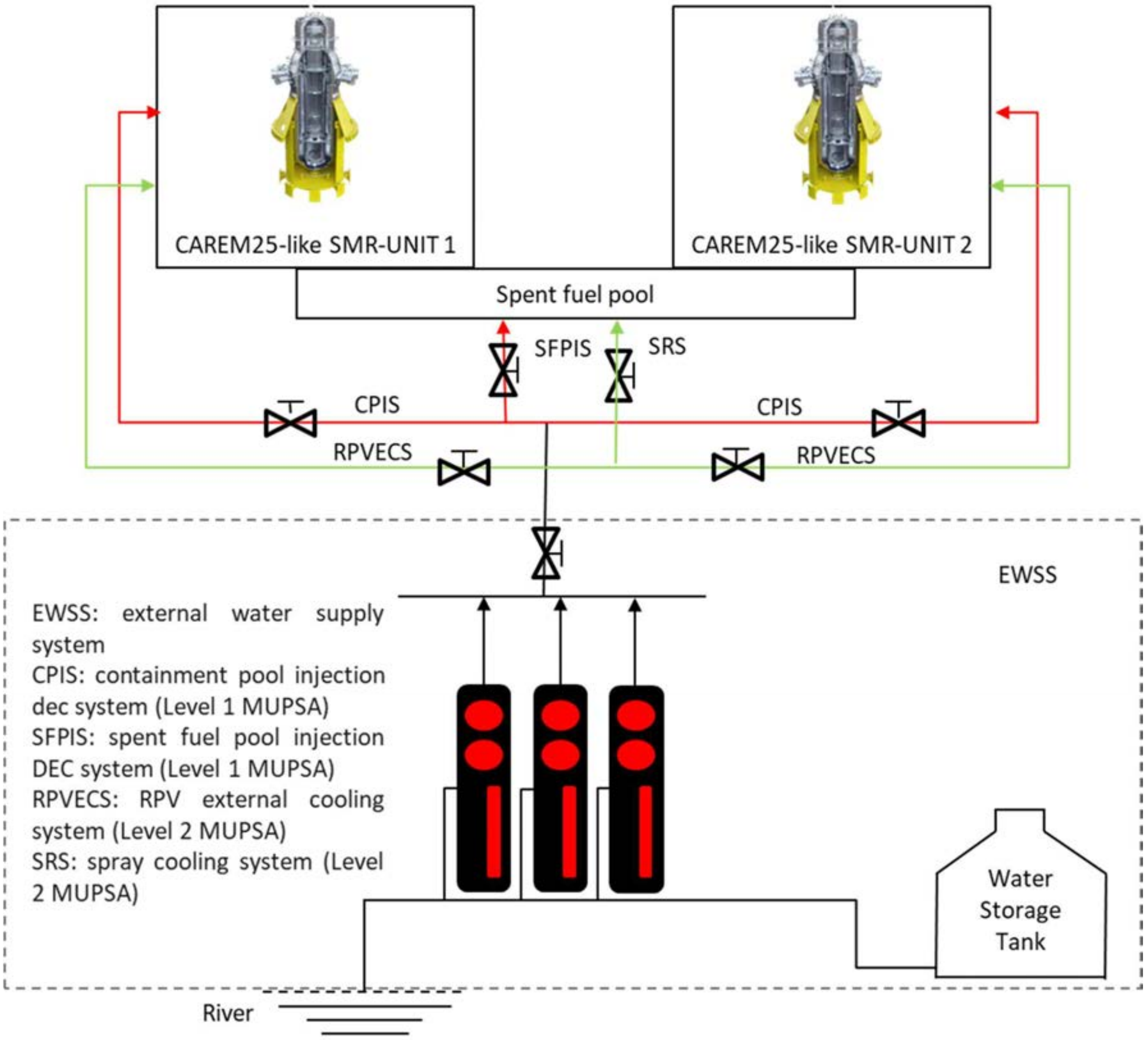

FIG. 2. Connection systems with external water supply system proposal for case study.

2.1.3 Analysis scope and safety goal

The scope of the benchmark was the development of MUPSA based on a hypothetical site with two SMRs and shared SFP and the development and assessment of individual radiological risk (IRR) for members of the public, defined as the probability of exposure for members of the public times the probability of fatality given that exposure, as a risk metric for the multiunit site. For Level 1 MUPSA, the mission time is 48 hr. For the partial Level 3 MUPSA (or Level 2+), the risk metric for multiunit site, based on the IRR used in the Argentinean acceptability criterion for licensing of NPPs, was adopted and analysed.

Methodological aspects of PSA development, including a risk metric for the multiunit site and the scope of available PSA for CAREM25 SMR, are based on the initiating event of LOOP for a full power operational state. Considering the design characteristics of CAREM25 and the strategy for controlling initiating events, event sequences (event trees) longer than 24 hr are analysed.

2.2. CANADA/COG

2.2.1. Site and plant information

The Pickering nuclear generating station (PNGS) near the city of Pickering, Province of Ontario. The site is on the north shore of Lake Ontario and has eight Canada Deuterium Uranium (CANDU) reactors, of which six are currently operating. The first four units at PNGS 'A' came on power between 1971 and 1974. The four units at PNGS 'B' came on power between 1983 and 1986. The units have a net output of 515 MWe each. Further operational details can be found in IAEA's PRIS database [5].

The reactors at PNGS are CANDU with thermal power of 1,744 MWth each. Each reactor has a closed heavy water primary heat transport system, which is separate from the secondary steam generator light water system. The moderator is heavy water that is contained within the calandria vessel. All of the primary systems are contained within the reactor building, which is part of the negative pressure containment system. All reactor buildings are connected through the pressure relief duct, which then connects to the vacuum building. The units also share the emergency coolant injection system as well as the other support systems. There is one irradiated fuel bay (IFB) for units 1 to 4 and one for units 5 to 8. There is also an auxiliary IFB for units 1 to 4 where the fuel from IFB goes after 4 years.

2.2.2. Analysis scope and safety goals

The existing PSAs in Canada cover Level 1 and Level 2 internal events for reactors at full power and shutdown states. The internal events PSA includes events that affect the representative unit as well as events that impact more than one unit, due to common structures and the sharing of support systems. PSA assessments are carried out for internal floods, internal fires, high wind and seismic. The whole site PSA project by COG included identification and assessment of plant operating states (POSs) that fall in between full power and shutdown as well as the identification, screening and qualitative assessment of non–reactor sources of potential release, which consisted of the IFBs and the used fuel dry storage (UFDS) facility. In Canada, two safety goals of interest are the severe CDF and the LRF. Both safety goals are defined on a per reactor, per hazard basis.

The objective of the benchmark was to share the knowledge and experience gained in the joint project carried out by the CANDU Owners Group (COG) in the development of MUPSA methodology and its application to PNGS. This project, known as whole site PSA, is based on SUPSAs and includes other POSs that fall between full power and shutdown and the assessment of non–reactor sources on the site.

2.2.3. Methodology/modelling techniques

The methodology for PSA starts by the identification and quantification of initiating events. This is then followed by the definition of accident progression and development of event trees. For each branch of the event tree, the safety function is then expanded into a fault tree. The fault trees and event trees account for operator actions both pre– and post–initiating event, common cause failures (CCFs), dependencies and recovery actions. The final step of the PSA is to integrate the fault trees and event trees through a high level logic that binds them into various fuel damage categories and to determine the cut sets that result in a site core damage frequency (SCDF) or LRF. The PSAs prepared by the Canadian utilities use the Electric Power ReseaIch Institute (EPRI) suite of software, CAFTA version 6.0b [6] and FTREX version 1.9.[2]

2.3. CHINA/INET

The Institute of Nuclear and New Energy Technology (INET) team focused its research efforts on the multi module PSA model for the high temperature gas cooled reactor pebble bed module (HTR–PM) reactor design, which is a demonstration reactor. The site is in Shidao Bay at the east end of Shandong Province, facing the Huang Hai Sea, as shown in Fig. 3. It is the first commercial high temperature gas cooled reactor (HTGR) with unit 1 achieving first criticality on 12 September 2021. General design features are available in IAEA's ARIS database [4] and operational details can be found in IAEA's PRIS database [5].

FIG. 3. HTR–PM site.

The HTR–PM adopts two nuclear steam supply system (NSSS) modules, rated at a power of 250 MWth, to feed one shared steam turbine generator. A single NSSS module is shown in Fig. 4 and consists of one reactor and one steam generator. The two reactors and the shared spent fuel sphere storage facility are shown in Fig. 5 and are considered as sources of potential large releases in the benchmark model. The INET team investigated Level 1 and Level 2 PSA together, because HTR–PM uses an integrated PSA framework (Level 1, Level 2, and partial of Level 3 are embedded in one model) unlike the traditional separate levels. This framework modification is a result of the characteristics of HTGR, since the core damage concept is no longer appropriate for HTGR. However, the integrated PSA model maintains the same kind of Boolean logic modelling framework as the traditional PSA. Most of the PSA elements are applicable with particular attention on the following aspects:

— Definition of risk metrics for site level risk assessment;
— Improvements to the current PSA models so that they can be appropriate for elaborating the multi module model;

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[2] FTREX is developed by the Korea Atomic Energy Research Institute (KAERI) and is available from EPRI for a fee under a licensing arrangement negotiated with KAERI. [https://www.epri.com/research/products/3002005280].

— Considerations to plant level operational states as opposed to unit level operational states;
— Modelling the role of shared resources including their positive and negative effects;
— Dependent failure analysis;
— Human reliability analysis (HRA) for multi module or multi source accidents;
— Risk quantification methods.

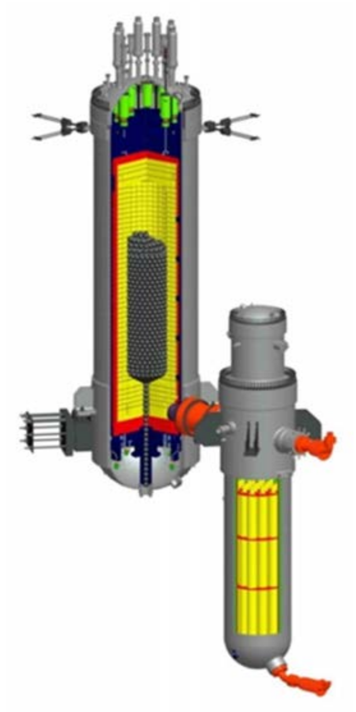


FIG. 4. HTR–PM NSSS module, [4].

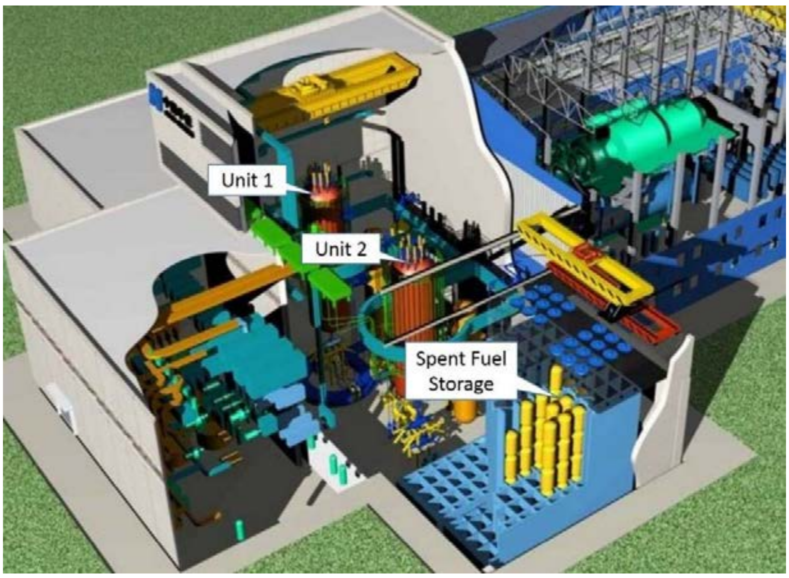


FIG. 5. General view of the HTR–PM, [4].

2.4. FINLAND/VTT

VTT’s part of the benchmark study is based on Nordic MUPSA project SITe Risk Of Nuclear installations (SITRON) [7]. Project partners included Risk Pilot AB (Sweden/Finland), Lloyd’s Register (Sweden) and IFE Halden (Norway). In the project, two pilot studies were performed for Swedish NPP sites: Forsmark and Ringhals. In both cases, only two units were analysed: Forsmark units 1&2 and Ringhals units 3 and 4. The Forsmark units are boiling water reactor of Asea–Atom design, and the Ringhals units are pressurized water reactors (PWR) of Westinghouse design. The thermal power of each unit is around 3,000 MWth. The use of the NPP units started in 1980’s. The Forsmark site is located in the eastern coast of Sweden.

The site includes a third boiling water reactor unit, which is however quite independent from the other two units. The Ringhals site is on the south–western coast of Sweden. The site also includes two more units, which will be decommissioned soon. Operational details for these reactor units can be found in IAEA PRIS database [5]. In both cases, the units are almost identical, have several common systems and structures, and are located close to each other. All the units have complete Level 1 and Level 2 PSAs covering all initiating events and POSs. The Forsmark units have common SUPSA. In the pilot studies, multiunit initiating events and dependencies were identified and analysed qualitatively. The quantification however focused only on multiunit LOOP scenario.

2.5. GHANA/GAEC

A conceptual two–unit same–design VVER type reactor site was selected for the benchmark. The two units are assumed to be located on a coastal site, in line with preliminary NPP siting results in Ghana, which involved discovering preferred site(s) to house Ghana’s NPPs. Candidate areas and potential sites were selected based on very specific criteria and the work is ongoing carried out by a combined team of scientists and engineers from the Ghana NPP organization and the Ghana geological survey to arrive at candidate sites. Shared systems in this conceptual two unit NPP site include electrical power supply system, diesel generators, cooling water system and outdoor switchgear. Two SFPs were assumed to be located on the site and the characteristics of the reactors are those of the VVER 1200 design.

2.6. HUNGARY/NUBIKI

The following is the description of the Hungary’s benchmark specifics including the description of the site and plant.

2.6.1. Site and plant information

The site of the Paks NPP is in Hungary, about 114 km south of Budapest and 5 km south of the town of Paks, as shown in Fig. 6. It lies 1 km west of the Danube River; hence the cooling water of all units is ensured by the same source, i.e. the Danube river. The elevation of the site is 97.15 m above the Baltic Sea level. Within a 3 km radius directly around the site, there are the operational site itself, spare area, fishing lakes, forests, and connecting roads. The wider area around the site, within 30 km radius, contains mostly agricultural fields with scattered villages and towns. The Paks site accommodates four operating reactor units, and the construction of two additional units is in preparation (as of 2023).

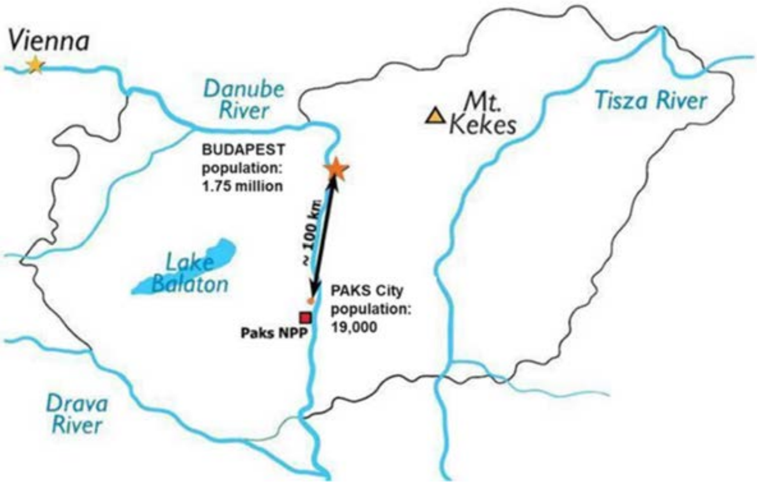


FIG. 6. Location of Paks NPP.

The individual reactors are installed in twin–unit buildings, each with two reactors. A view of the site units is shown in Fig. 7. All four units are of a VVER–440/V–213 power reactor type, cooled and moderated with light water; each reactor unit has the same thermal output power of 1,485 MWth. With the individual electric capacity of 500 MWe, the total site capacity is therefore ~2,000 MWe. The nominal power of 500 MWe is reached after power uprating from the original 440 MWe. The units were first connected to the grid from 1982 to 1987. Although the original 30 years of the NPP lifetime has expired for all the units, the Hungarian Atomic Energy Authority issued a lifetime extension permit for an additional 20 years. Operational details can be found in IAEA's PRIS database [5].

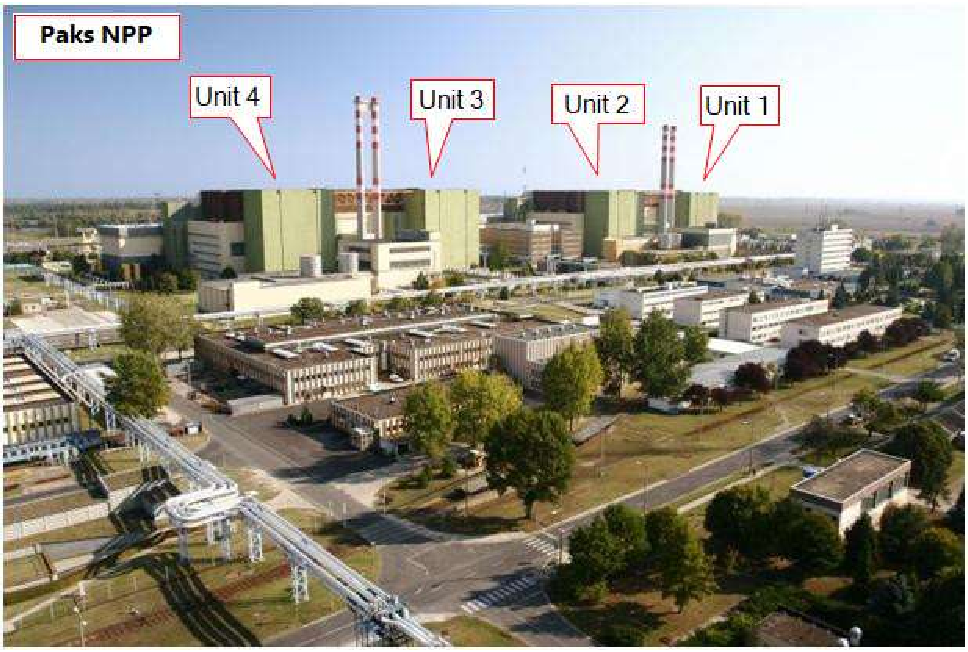


FIG. 7. General view of Paks NPP, [8].

2.6.2. Analysis scope and goals

The scope of the study was a development of a MUPSA model. Quantification of site level risk metrics was a desired result of the analysis, although emphasis was also laid on better understanding of a NPP vulnerabilities in comparison to relying merely on the separate single unit risk assessments for NPP. The MUPSA model was developed for the four VVER–440/213 reactor units of NPP Paks, Hungary. The reactor cores and the fuel stored in the SFP located adjacent to the reactors were considered as sources of potential large releases of radioactivity.

The developments started with Level 1 PSA, using the RiskSpectrum PSA software. Particular attention was paid to the following aspects:

- Definition of risk metrics for site level risk assessment;
- Improvements to the unit specific PSA models so that they can be appropriate for elaborating the multiunit model using the event tree linking approach with fault tree conversion of accident sequences;
- Considerations to plant level operational states as opposed to unit level operational states;
- Modelling shared resources including their advantageous and disadvantageous effects;
- Dependent failure analysis;
- HRA for multiunit or multi source accidents;
- Risk quantification methods.

The existing Level 2 PSA for NPP Paks (based on analysis results achieved from calculations performed by the MAAP5–VVER software) and the findings from the Level 1 MUPSA was the most important input to this analysis phase. The goal was to determine analysis methods that are considered applicable to a Level 2 MUPSA of the plant. Trial applications of the proposed method were made including event trees, similar to containment event tree (CET) in a single unit, single source Level 2 PSA, and corresponding fault tree modelling. Although a full blown Level 2 PSA was not done due to current uncertainties, unknowns, and time and resource limitations, attempts were made to perform initial pilot analysis for LOOP considering twin–units, both at–power operation, that can subsequently be used for a full scope Level 2 MUPSA in the future.

2.7. INDIA/AERB

The following is the description of the Hungary's benchmark specifics including the description of the site and plant.

2.7.1. Site and plant information

The benchmark site is Kakrapar, located in the state of Gujarat, India. This is an inland site located about 29 km downstream of Ukai dam. The site consists of two operating and two under construction units. Unit–1 was commissioned in 1993 and unit 2 in 1995. The plant site lies in seismic zone III per seismic zoning map of India. Operational details can be found in IAEA's PRIS database [5]. The plant layout is a twin unit and the main plant building consists of two reactor buildings. For each reactor, a reactor auxiliary building is provided adjacent to the reactor building to accommodate reactor auxiliary systems. In addition, a natural draft cooling tower and an induced draft cooling tower is provided for each unit. Other buildings such as turbine building, emergency control building, service building, spent fuel storage building, waste management building, cooling water pump house, plant water pump house, demineralized water plant and switchyard are common to both units. A complete physical separation is provided between the safety related systems of the two units. Fire water is the only common system but can feed the emergency feed requirements of both units.

2.7.2. Analysis scope and goals

The overall scope of the benchmark includes demonstration of an integrated approach to address both external and internal events that can affect single/multiple units at typical Indian PHWR site and a mathematical formulation to estimate the SCDF for the benchmark site. The source is reactor core and analysis is performed for a full power operation of both plants. The analysis is performed for both internal and external events applicable to the benchmark site. Full scope PSA for Level 1 PSA except seismic is available. Seismic PSA is being performed with plant specific seismic analysis. Level 2 PSA is not included in this benchmark, but it was updated according to Severe Accident Management Guidelines (SAMG).

2.8. INDIA/BARC

The following is the description of the Hungary's benchmark specifics including the description of the site and plant.

2.8.1. Site and plant information

The site under consideration is Tarapur, which is situated in the west coast of Maharashtra, India. The site lies in the moderate seismic zone of India. Being a coastal site, seawater is used on a once through basis for condenser cooling and process water cooling. In this benchmark, the source of radioactivity is considered to come from the reactor core of twin hypothetical advanced reactors. The plants under consideration are two hypothetical advanced heavy water reactors (AHWRs) with capacity of 920 MWth each along with one shared spent fuel facility (SFF). It is a pressure tube type, heavy water moderated, boiling light water cooled reactor relying on natural circulation for core cooling during normal and shutdown conditions (as shown in Fig. 8) and further design details of the AHWR can be found in the IAEA's ARIS database [4].

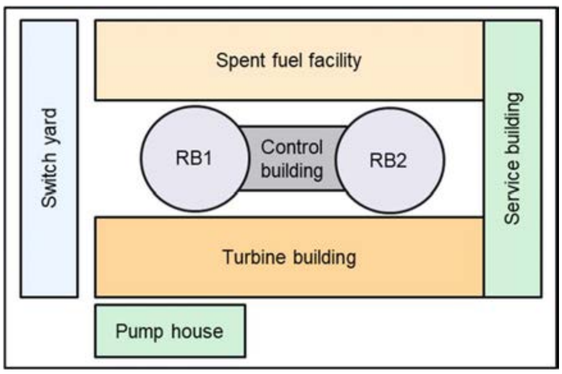


FIG. 8. Multiunit AHWR site layout (RB 1, RB 2: reactor building 1, 2).

2.8.2. Analysis scope

The PSA scope includes sources of radioactivity as reactor core in two hypothetical AHWRs and spent fuel in one common SFF. For the nuclear reactor, the operational state of the plants is considered at full power, and the considered initiating events are both internal and external events (seismic).

2.9. PAKISTAN/PAEC

The scope of the PSA benchmark study for multiunit Chashma site was to develop and apply MUPSA approach based on SUPSA studies, focuses on the assessment of multiunit accident(s) at Chashma site from external hazards. Special consideration is given to the multiunit LOOP events, station blackout (SBO) and resolution of CCF treatment for both intra–unit and inter–unit effects. As depicted in Fig. 9, two institutions from Pakistan collaborated in developing this benchmark model.

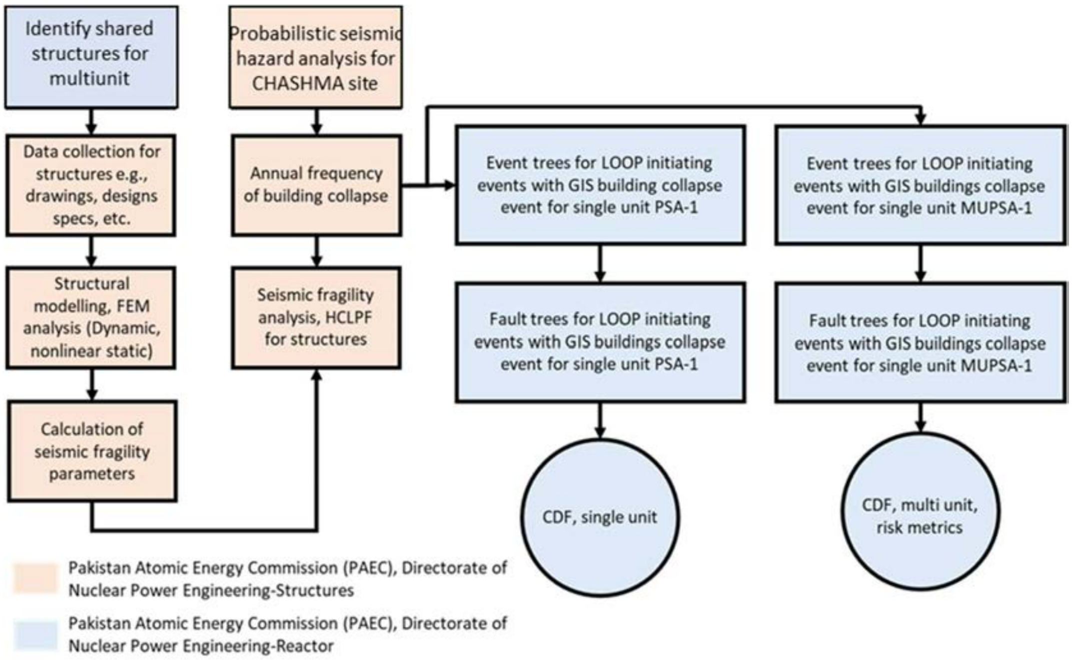


FIG. 9. Synergistic approach between two participating institutions from Pakistan.

2.9.1. Site and plant information

The Chashma NPP (CHASNUPP) multiunit site is located at about 10 km from the Chashma Barrage on left bank of the Indus River in Mianwali district, about 32 km to the south of Mianwali, 280 km to the southwest of Islamabad and 1,160 km northeast of Karachi. The site is in an area classified as arid to desert and is characterized by section of sand dunes, sparsely vegetated hills, sand soil and bare rocky mountains. The site location is shown in Fig. 10. The site area lies in the northwest region of the Thal Doab, which forms the area of Punjab Plain between the Indus and Jhelum rivers. The area is bounded in the northeast by the Chashma Jhelum Link Canal, in the southeast by the Thal Main Line Lower Canal and in the northwest by the 1.1 km wide Indus River. The distance to the main riverbed of the Indus is 10 km. and the average ground elevation at the site is 200 m. No population centre exists within 1.3 times the low population zone boundary around the NPP.

The Chashma–Jhelum Link Canal, which takes off from the bank of the Indus River at the Chashma Barrage, is the main source for the supply of cooling water. It is an unlined earthen canal with a capacity of 615 m3/s (21,700 cfs) and a full supply level of 194.2 m (637 ft) at the head regulator. Extensive field investigations in the CHASNUPP site area revealed uniform soil conditions and overburden alluvium all the way down to the maximum explored depth of 100 m.

There are many important seismic sources near site like Khisor, Marwat, Kalabagh and Bhittani faults etc. The maximum potential magnitude assigned to the most important structure, Khisor–Kundal is 6.8. This structure considered capable is reassigned a pessimistic value of magnitude 6 by Pakistan Atomic Energy Commission (PAEC) experts and endorsed by IAEA Mission (1992). The largest earthquake in the recent seismic history was the Bhakkar earthquake occurred on 1 May 1982. This earthquake had a magnitude of 5.5 and a focal depth of 3.5 km, occurred 75 km from the site. The earthquakes of this magnitude have not been reported as having an adverse effect on the site. Muzaffarabad earthquake 2005 (magnitude of 7.6, focal depth of 13km) is the most disastrous earthquake in Pakistan. CHASNUPP site is about 290 km from its epicentre. Seismic recordings at site gave zero period acceleration as 0.0325 g, 0.022 g and 0.017 g in free field. The horizontal ground acceleration for the safe shutdown earthquake (SSE) is 0.25g. The operating basis earthquake (OBE), or seismic level 1 earthquake, has a value half of SSE i.e. 0.125g, which was taken for OBE. The liquefaction analysis has already been carried out for the CHASNUPP area with a more severe ground motion. The minimum safety factor for free field case against liquefaction is 1.59 at a depth of 12 m while the values at greater depths increase monotonically.

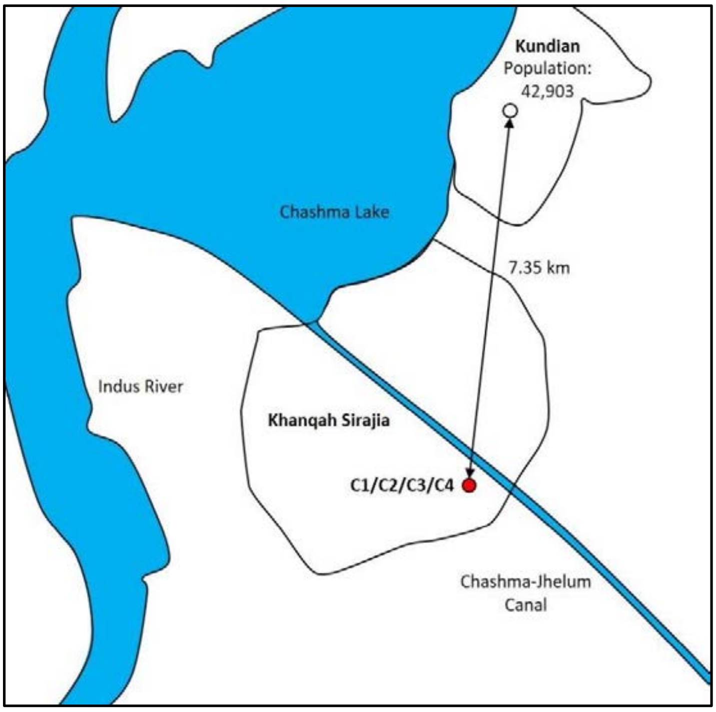


FIG. 10. Location of multiunit Chashma site.

All units on CHASNUPP site are of 300 MWe class two–loop PWR type NSSS supplied by China National Nuclear Corporation, as shown in Fig. 11. The C1 was commissioned in 2000 and is the oldest among all. C2 was commissioned in 2011, while C3 and C4 were commissioned in 2016 and 2017, respectively. Operational details can be found in IAEA's PRIS database [5].

The shared structures/buildings between C1 and C2 and between C3 and C4 are as follows:

— Shared between C1 and C2: intake structures, switchyard, circulating cooling water pumping station and drainage structures;
— Shared between C3 and C4: intake structures, Alternate alternating current (AAC) power supply, switchyard, raw water purification system, fire protection water supply system, drainage structures and boiler house;
— Shared between C1, C2, C3, C4: 132 kV offsite power supply system and dry fuel storage facility (under construction).

FIG. 11. Chasnupp 1, 2, 3 and 4 on multiunit Chashma site, [9].

2.9.2. Analysis scope and safety goals

Overall scope was to conduct seismic fragility and risk assessment of different important multiunit structures at the Chashma NPP site and calculate the annual frequency of structures unacceptable performance as input in MUPSA model for the Chashma site. The case structures are the one which are shared between C1 and C2, between C3 and C4 and among all four units. Moreover, risk assessment of structures housing radiological sources is conducted for input in MUPSA models since these structures have potential of large releases of radioactivity. The shared structures and the one housing radiological sources are important in

MUPSA Level 1 and Level 2, respectively. The outcomes of probabilistic seismic hazard assessment (PSHA) of Chashma site is used as well in evaluation of individual structure seismic risk and impact of cumulative seismic risk of all case structures on multiunit sites using a suitable risk matrix. The expected outcomes/achievements from the proposed CRP are as follows.

— Development of a family of seismic fragility curves for shared structures and structures housing radiological sources (e.g. SFP, SFP building, containment, dry fuel storage facility);
— Calculation of fragility parameters and high confidence of low probability of failure values;
— Fragility of unacceptable performance;
— Probabilistic seismic hazard curve of the Chashma site;
— Seismic risk of individual structures (annual frequency of unacceptable performance);
— Utilization of seismic fragility and risk outcomes in MUPSA models for Chashma site.

The scope was to perform seismic fragility and risk assessment of switchyard buildings shared between C1 and C2 units, and between C3 and C4 units. Two gas insulated switchgears (GIS) buildings i.e. GIS 220 and GIS 132 are shared between C1 and C2 and similarly two GIS buildings are shared between C3 and C4, as shown in Fig. 12. The LOOP event with seismically induced collapse of GIS buildings is considered as initiating event for the multiunit CDF calculation based on MUPSA 1 model. The outcomes from the seismic fragility assessment of the shared GIS buildings are same as mentioned in the overall scope. Convolution integration of both fragility curves and hazard curve is done to calculate the total annual frequency of the initiating event at the selected ground acceleration range. Finally, single unit CDF, SCDF and multiunit CDF are calculated. Single unit PSA Level 1 models are available for each unit at Chashma site. The scope of the PSA is limited to Level 1 bounding to internal initiating events occurring at power operation case (excluding internal fires and internal floods), which including analysis of the following initiating event groups: small break loss of coolant accident (LOCA), large LOCA, steam generator tube rupture, LOOP, general transient, loss of main feedwater, loss of component cooling water, loss of instrument air, loss of direct current (DC) power and others. The SUPSA shows that LOOP is the top contributing event resulting in core damage. Therefore, in the current study, LOOP with seismically induced collapse of shared switchyard buildings is considered as initiating event for the MUPSA Level 1 model developed for this benchmark.

(a) (b)

FIG. 12. Reinforced concrete frame structure of C1–C2 shared GIS buildings: (a) GIS 220 building, (b) GIS 132 building.

The scope of MUPSA encompasses the selection of radioactive sources (reactor core, SFP, interim/dry storage facility), POSs (full power, low power, shutdown), initiators (internal events, internal hazards, external hazards) and PSA end states (core(s)/fuel damage, radioactive releases, offsite radiological consequences) as shown in Fig. 13.

The reactor cores of all four NPPs, operating at Chashma site, are considered as radioactive sources for this benchmark. The current configuration of four NPPs is such that three NPPs are at full power mode/state while remaining NPP is in shutdown mode/state due to refuelling outage. However, current analysis considers all four NPPs at full power for the sake of simplicity. Only external hazard is considered for selection of initiating event. Finally, the core damage is considered for MUPSA Level 1.

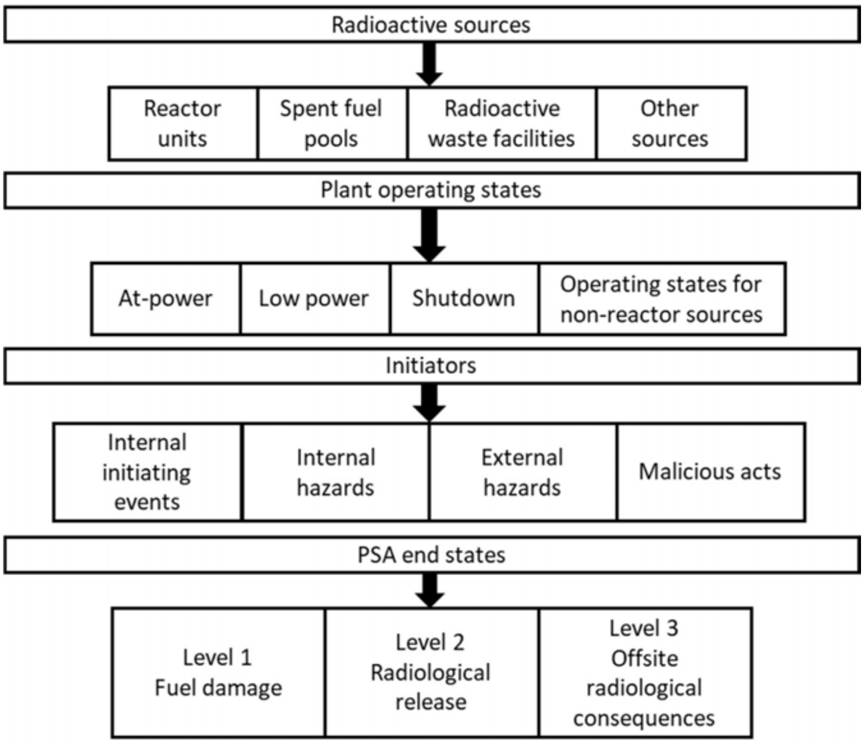


FIG. 13. Scope of the PSA model.

The risk and reliability analysis software RiskSpectrum is used to carry out MUPSA. It includes tools for fault tree and event tree modelling and analysis, risk monitoring, HRA and failure mode and effect analysis. In this benchmark, RiskSpectrum Version 1.1.0.0 is used.

2.10. REPUBLIC OF KOREA/KHNP

The following is the description of the Hungary's benchmark specifics including the description of the site and plant.

2.10.1. Site and plant information

Figure 14 shows the benchmark sites, Kori and Saeul, located in southeast coastal regions across Gijang–gun and Ulsan city in Korea. Kori is the first NPP site in Korea, which contains six units. Saeul site is 2.5 km apart from Kori unit 1–4, and 1.5 km apart from Shin Kori unit 1&2 in Kori site as shown in Fig. 14.

Although Kori and Saeul sites are physically separated and independently operated, a multiunit risk impact from both sites is assessed because they could be affected concurrently by external hazards such as but not limited to typhoons and seismic events.

There are five operating units and one permanent shutdown unit at Kori site, and one operating unit and three constructing units at Saeul site. The details of the units are described in Table 1 and further operational details for all units on these sites can be found in IAEA's PRIS database [5].

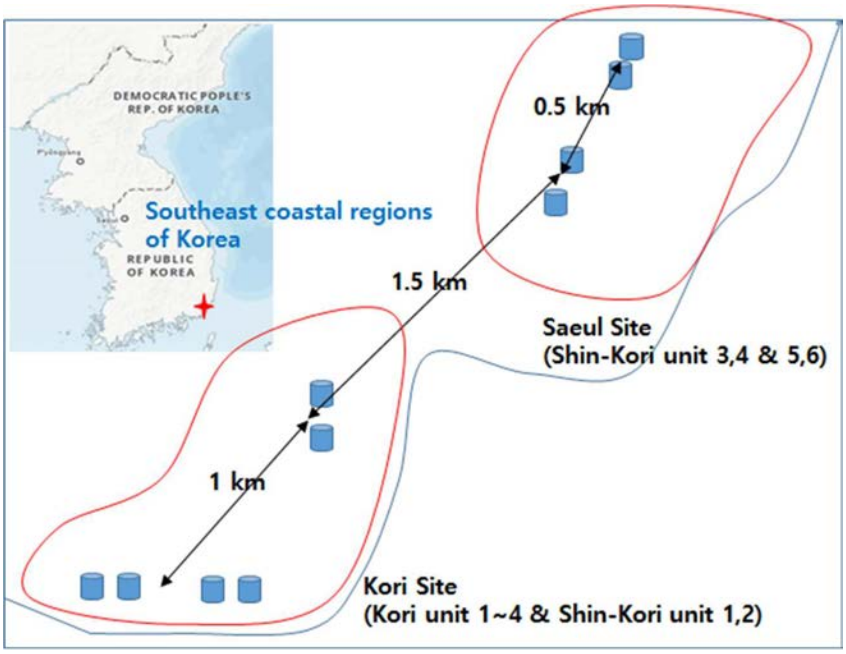


FIG. 14. Location of the Kori and Saeul NPP sites.

TABLE 1. NPP SUMMARY OF KORI AND SAEUL SITES

Site–Unit	Reactor Type	Status	Capacity (Mwe)	Commercial Operation
Kori–1	PWR (W/H type)	Permanent Shutdown	587	Apr. 1978
Kori–2		Operational	650	Jul. 1983
Kori–3			950	Sep. 1985
Kori–4				Apr. 1986
Shin–Kori–1	PWR (OPR1000)		1,000	Feb. 2011
Shin–Kori–2				Jul. 2012
Shin–Kori–3	PWR (APR1400)		1,400	Dec. 2016
Shin–Kori–4		Under Construction		–
Shin–Kori–5			1,455	
Shin–Kori–6				

2.10.2. Analysis scope

To identify the multiunit risk and derive insights, Korea Hydro & Nuclear Power Co., Ltd. (KHNP) has performed the MUPSA project and participated in this CRP to get better understandings and derive best practices related to MUPSA Level 1 and Level 2.

By participating in this CRP, the KHNP expected to develop best practice guidance for conducting Level 1 and Level 2 MUPSA with providing various modelling approaches to selecting multiunit initiating events, considering operating status of all units at sites, the assumptions and engineering judgements on technical considerations. Our approaches could be the best practices, especially, for the sites including many units, and for quantifying huge fault tree models.

After establishing the new legislation of Accident Management Program in 2016, the SUPSA models for all units in Korea have been updated and newly developed to cover Level 1 and 2 PSA for the operating units. Level 3 PSA is, however, required only for new units. Considering the scope of a SUPSA, KHNP determined the scope of MUPSA to cover Level 1 and 2 PSA at all operating modes for reactor sources.

KHNP used AIMS–PSA (Rev. 1.2e) software (similar to CAFTA or SAPHIRE [10]) to develop and manage MUPSA models with fault tree reliability evaluation expert (FTREX) as a quantification engine, which were developed by KAERI. In case of quantifying seismic PSA models, which include non–rare events, the FTeMC (fault tree top event probability evaluation using Monte Carlo simulation) software is used as well as BeEAST software, which applies binary digital diagram approach [11]. In addition, Splitter and integrator for total estimation of site risk (SiTER) software is used for convenience to deal with cut sets for extracting information of multiunit scenarios and for deleting double counted cut sets.

2.11. REPUBLIC OF KOREA/HANYANG UNIVERSITY

The MultiUnit Risk Research Group (MURRG) is conducting research to develop a multiunit risk assessment methodology as well as to develop site risk assessment (SRA) full scope model for regulatory purposes at Level 1, Level 2, and Level 3 PSA based on a probabilistic approach and systematic knowledge management.

With the support from Nuclear Safety and Security Commission in Republic of Korea, the site risk verification technology and suggestions of multiunit safety goals are under study in MURRG, and Hanyang University in Seoul. Eight organizations including Institute of Nuclear Safety (KINS), Korea Atomic Energy Research Institute (KAERI), Kyung Hee University, Chung–Ang University, Sejong University, Chosun University, and KEPCO International Nuclear Graduate School (KINGS) have been participating in this project since June 2017. As a part of the development of the verification technology, MURRG is conducting coordinated research to develop an integrated SRA model including internal and an external event for nine units on the Kori site.

The reference site for the site risk assessment is Kori which is the same sate that the utility KHNP had chosen for the reference site and Saeul site. Except Kori 1 unit which was shutdown permanently in 2017, the nine units SRA model is supposed to be developed by the end of 2021. Full power, low power and shutdown operation of Kori 2, Shin Kori 1 and Shin Kori 2 units would be reflected for mode combinations while the rest of reactors were assumed to be at full power operation.

Based on Level 1, Level 2, and Level 3 SUPSA models for both internal and seismic events, the integrated Level 1, Level 2 and Level 3 SRA model considering the dependency between the units would be under development to assess the site risk induced by single unit and multiunit accidents. Multiunit initiating event analysis, inter–unit CCF, shared component analysis, seismic correlation estimation, severe accident phenomena analysis for each reactor type and source term analysis using the MELCOR code, radiation health effect assessment using the MACCS code, HRA using the standardized plant analysis risk HRA (SPAR–H) method [12] are being investigated.

The risk assessment for SFP is excluded from this study. The detailed elements of this study are shown in Fig. 15 and Fig. 16.

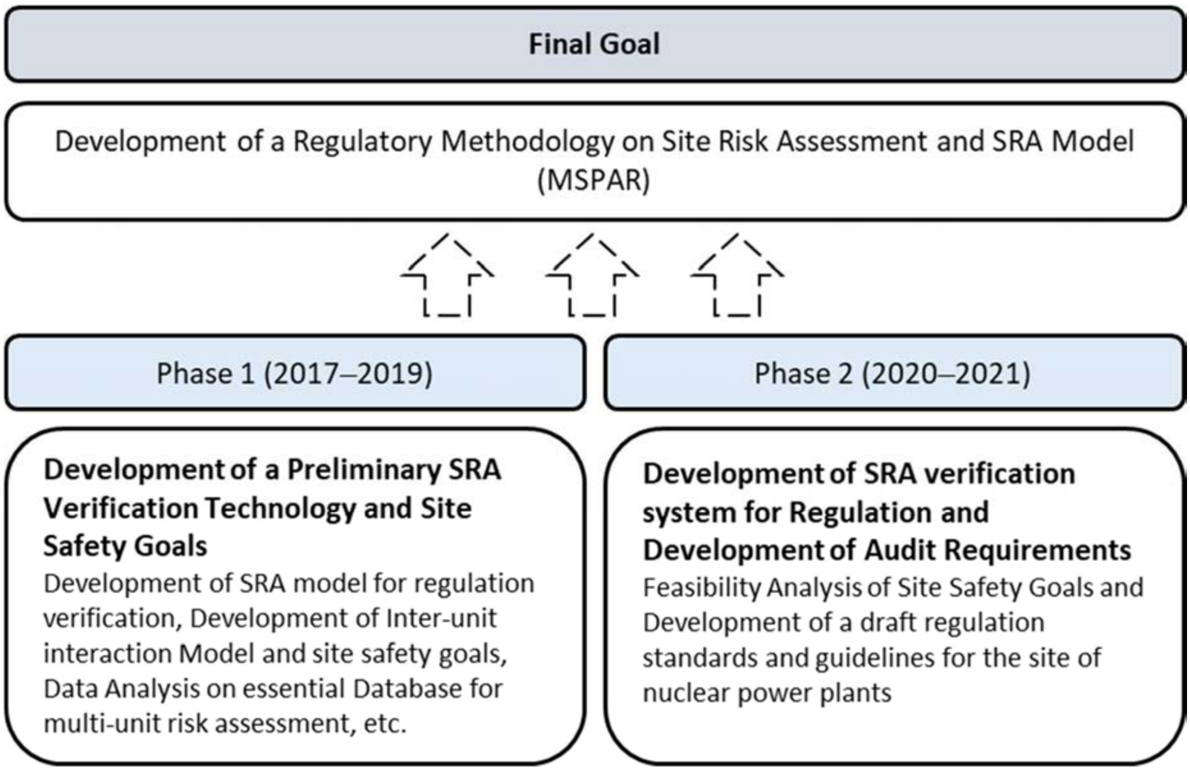


FIG. 15. Scope of MURRG project.

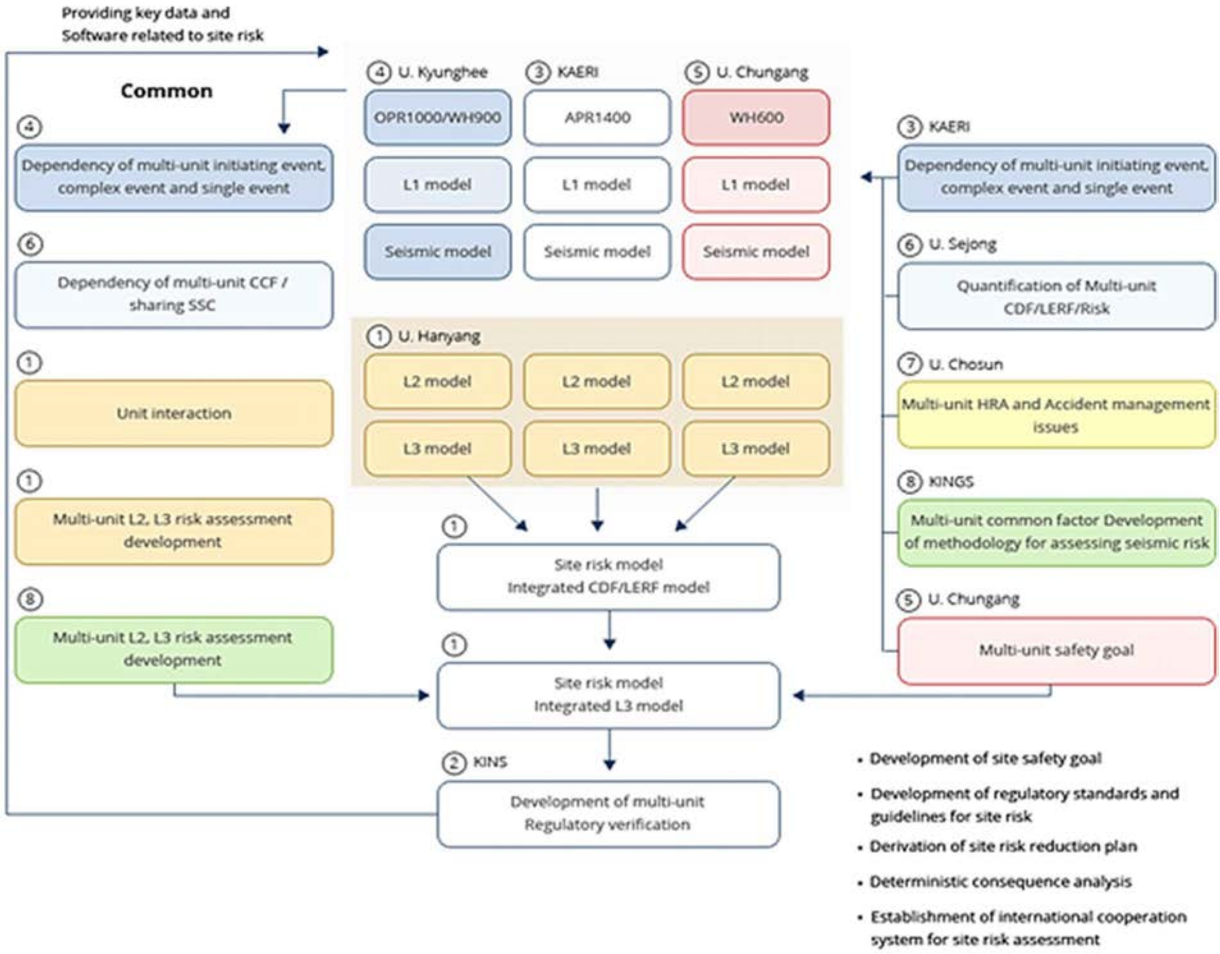


FIG. 16. Configuration of MURRG research elements.

2.12. ROMANIA/CNCAN

The following is the description of the Hungary’s benchmark specifics including the description of the site and plant.

2.12.1. Site and plant information

Table 2 and Fig. 17 provide site and plant information. Cernavoda NPP site is on a branch of the Danube and Black Sea canal close to Cernavoda city. It had initially five units, of which one is now considered only as the bunker type emergency room (under construction on the structures unfinished of unit 5) and four remaining units: two in operation (commissioned in 1996 and 2007) and two with delayed project under evaluation for restart in the next 10 years. There are no external events challenging the site and the geology, hydrology and seismology conditions align with the internationally standards and national regulations.

All of the reactor units are the CANDU 6 type reactors with the following characteristics:

— 700 MWe (2,070 MWth);

- Units are independent: water intake is common but lines to units separate, connection to grid with separate lines;
- Other sources considered in the benchmark model:
  - Spent fuel bays (for each unit) (SFB);
  - Dry spent fuel storage (DICA) one on site;
  - Tritium removal facility (CTRF) one on site.

Operational details can be found in IAEA's PRIS database [5].

TABLE 2. SUMMARY OF SOURCE COMBINATIONS IN NPP CERNAVODA BENCHMARK

Group	**Description**	**Group Objectives**	**Sub–group**	**Subgroup Description**
I	One unit + other sources	Impact of other sources on SUPSA and safety assessments	B1	1 unit operation+ 1 refurbishment + two commissioning + intermediate spent fuel storage (ISFS) + CTRF + SFB
			B2	1 unit operation+ 1 refurbishment + two decommissioning + ISFS + CTRF + SFB
			B3	1 unit prolonged operation + 1 refurbishment + two decommissioning + ISFS + CTRF + SFB
			B4	1 unit prolonged operation + 3 decommissioning + ISFS + CTRF + SFB
II	Two units + other sources	Impact of other sources on two units and possible impact of missed CCF / IEM for two units hazard review	A1	2 units operation + 1 commissioning + ISFS+CTRF + SFB
			A2	2 units operation + 2 decommissioning + ISFS + CTRF + SFB
			A3	2 units operation + 3 commissioning + ISFS+ CTRF + SFB
III	Three units + other sources	MUPSA tasks – missed IEM, CCF (including two software approach) + other sources for 3 units	C1	3 units operation+1 refurbishment + ISFS + CTRF + SFB
			C2	2 units operation +1 prolonged operation +1 refurbishment + ISFS + CTRF + SFB
			C3	2 units operation +1 prolonged operation +1 decommissioning + ISFS + CTRF + SFB
IV	Four units	MUPSA specific tasks – test of small event–tree/fault–tree approach	D1	2 units operation + 2 prolonged operation + ISFS + CTRF + SFB
V	No unit only other sources	No MUPSA	E1	1 unit refurbishment + 3 decommissioning + ISFS + CTRF + SFB
			E2	4 units in decommissioning + ISFS + CTRF + SFB

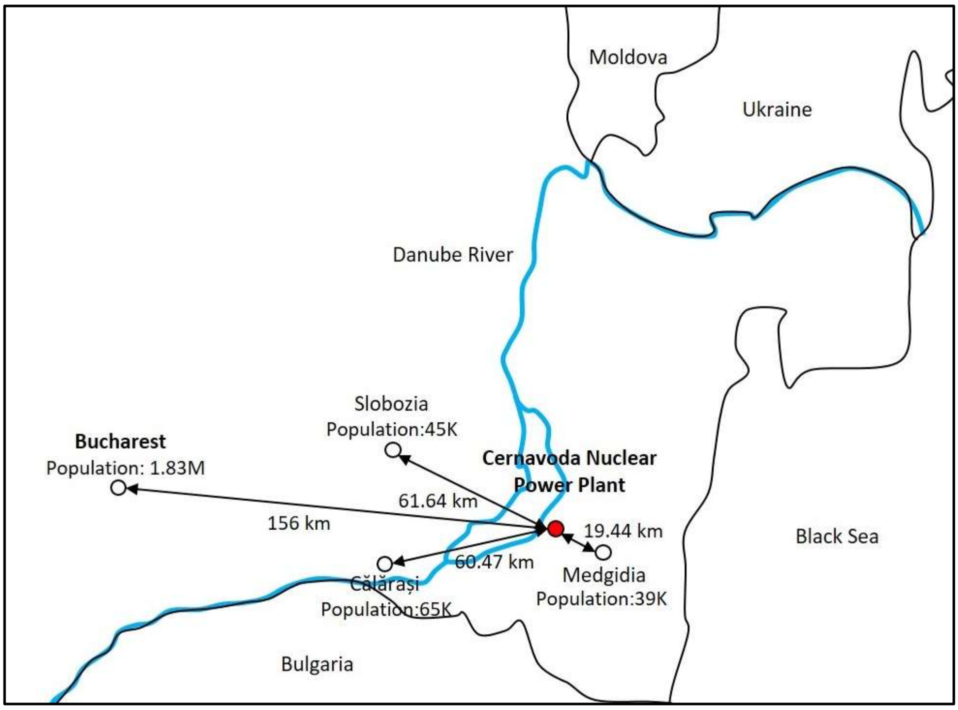


FIG. 17. Cernavoda NPP site.

2.12.2. Analysis scope

The technical rationale in choosing Level 2 is that existing results for CANDU 6 show that the core damage states (CDS) for MUPSA do not change plant reaction, as the units are independent and therefore the grouping/binning process to obtain external plant release categories (EPRCs) documented in CANDU approaches is not changed. As a result, from CANDU 6 independent reactors perspective only the PSA Level 2 has significance and therefore was considered for the project.

The following units and sources were considered in the benchmark:

a) Reactor:

— Two reactor cores (old units refurbished);
— Two new units.

b) Four SFP, one for each unit;
c) One DICA;
d) One CTRF.

The main objectives of the benchmark analysis are as follows:

— Define the main issues for CANDU 6 and related site sources if multiunit is considered;
— Support safety assessment of site with inputs for case studies on MUPSA;

— Consider combination of case studies lifetime oriented for site and build simplified two codes (RS & CAFTA models);
— Consider interface /connection MUPSA with its possible applications:

  - Risk monitoring;
  - SAMG;
  - Emergency planning

— Train a team of new generation PSA analysts; on the job task oriented;
— Focus on PSA Level 2 and only some tasks in Level 1 (CCF, hazard screening);
— Consider site impact for medium and long term evolution;
— Investigate new methodologies for MUPSA/multiunit multisource safety assessments;
— Coordinate with other international/national projects in which we are part.

The benchmark is aimed at fulfilling target connected with MUPSA that could support the site safety assessment. The PSA models for SUPSA Level 1 and Level 2 for units 1 and 2 were available to be used as a starting point for this analysis.

2.13. RUSSIAN FEDERATION/JSC A

For the benchmark, full scope Level 1 and Level 2 MUPSAs for Balakovo NPP are developed. Fuel in both reactors and SFP is in the scope of the assessment. The following limitations of the assessment are accepted but could be relaxed if the results suggest extension of the assessment:

— Assessment is focused initially on two units (Balakovo NPP units 1 and 2) and according to the results obtained can be extended to consideration of all units at the site;
— It is assumed that only Balakovo NPP unit 1 is in shutdown mode; other units are at full power.

The choice of the units is as follows:

— Balakovo NPP are the most common VVER–1000/320 designs widely applied in Russian Federation, Ukraine and Chezh Republic. Therefore, insight from the assessments will be useful for many existing and planned NPP sites with this type of reactor designs;
— For Balakovo NPP comprehensive and accepted by the regulators Level 1 PSA model is available for all power operation modes and all hazards. However, this model is available only for fuel in the reactor. Level 2 PSA model is available, but only for internal initiating events and for power operation. In 2019 the Leven 2 PSA model is extended and thus allows the use of insights for multiunit Level 2 PSA assessment;
— Assessment to be performed within the CRP is essential for better understanding of the full risk, associated with the unit operation. The availability of the PSA models allows performing multi source PSA relatively easy.

Therefore, the major prerequisites for multiunit and multi source PSAs are met. JSC Atomenergoproekt is the designer of VVER–1000/320 and develops and maintains the PSAs

2.13.1. Site and plant information

The Balakovo NPP site is located in the Balakovo region of the Saratov area on the left shore of the Saratov artificial lake, 140 km away from Saratov–city.

The NPP site is located 8 km to the north–east from Balakovo–city boundary. Sanitary protective area of 3 km in radius is free from housing. Four identical units of VVER–1000 units with the V–320 reactor type are located at the site as shown in Table 3 and Fig. 18. Operational details can be found in IAEA's PRIS database [5].

TABLE 3. COMMISSIONING OF BALAKOVO NPP UNITS

Unit	Start of construction	Initial start–up	Start of power operation
Balakovo–1	01.12.1980	28.12.1985	23.05.1986
Balakovo–2	01.08.1981	08.10.1987	18.01.1988
Balakovo–3	01.11.1982	25.12.1988	08.04.1989
Balakovo–4	01.04.1984	11.04.1993	22.12.1993

FIG. 18. Balakovo NPP units, [13].

Balakovo is the nearest to the NPP settlement, the population of which is over 100,000. There are no other settlements within 100 km area with similar number of inhabitants. Directly in the site area absolute elevations are within limits of 32–35 m.

The population size of the area around the NPP limited by 30 km radius is 218,200 people. Currently the population density within the 30 km area makes 77.2 persons/km2 (99.5 persons/km2 in future). The following impacts of natural and anthropogenic origins are related to NPP safety being typical for the site:

— Shock waves impact in case of explosions at industrial and transport facilities located near the NPP: Large industrial facilities are located mainly in the Balakovo industrial area that is at 8–12 km south–west of the NPP. There are no industrial, transport and other enterprises that could have an unfavorable effect on the NPP within the NPP immediate vicinity.

- Wind impact: The probability of tornado passing through NPP industrial site at the place of the reactor location is 5.6×10^{-7} per reactor per annum. The tornado wall rotation maximum horizontal velocity is 60 m/s, pressure differential between the periphery and centre of the tornado cone is 4.4 GPa. Tornado passing length is 13.5 km, while its path width is 135 m and travel speed is 15 m/s. The engineering structures of buildings of the reactor compartment, spent fuel storage, reactor auxiliary building remain stable under these loads.
- Extreme temperature effects: Design maximum air temperature with frequency 1.0×10^{-4}/yr is 37 °C, and the minimum temperature with the same frequency is –32 °C. Engineering structures stability under these extreme temperatures is not disturbed.
- Hydrologic phenomena: The NPP site is located on the left–hand shore of the Saratov artificial lake, 3 km to the north of Novoye Natal'ino village. The site planning elevation is 34 m. The source of the service water supply is the Saratov artificial lake, on the shallow coastal part of which a self–contained water basin–the NPP cooler– has been created by means of sandy dams in wash. The water basin – NPP cooler draw off lower than the NHL elevation is not envisaged. In case of the Saratov HPP dam failure and water fast abatement at the dam outside the water basin – NPP cooler level will goes down from 30 m to 28 m within approximately 50 days. On the basis of the accepted concept of creation of the ultimate possible flood in the Saratov artificial lake the NPP site range line its design level is considered to be 33.5 abs.
- Seismic impact level assessment: The final assessment of the Balakovo NPP site seismic hazard taking into account local engineering–geological and hydrogeological conditions is accepted as follows:
  - Design earthquake: 6 points;
  - SSE: 7 points.

- All buildings, structures, equipment and elements of safety classes 1 and 2 are designed for SSE impact. Each unit at Balakovo NPP is VVER–1000/320 standard VVER unit 3,000 MWt/h thermal power (1,000 MWt/h electrical). The following systems are included into the reactor plant structure:
  - Reactor coolant circuit;
  - Reactor control and protection system;
  - Primary circuit pressure maintaining system;
  - Primary circuit overpressure protection system;
  - Hydro accumulator system.

The reactor coolant circuit structure, shown in Fig. 19, includes nuclear power reactor VVER–1000 of vessel type with pressurized water of V–320 type, four circulation loops, each of which consists of the steam generator PGV–1000M, reactor coolant pump GTsN–195M and main circulation pipelines of nominal diameter 850 mm (Dn 850) that connect the loops equipment with the reactor. The main safety features of the unit are summarized in Table 4. All systems listed in Table 4 have no common parts between the units.

For the successful implementation of the safety functions and the functioning of the safety systems, each unit has three channels of essential service water for cooling the responsible consumers of front–line systems (group A). This service water is also the final absorber of residual heat generation of the reactor during cooling down of the unit through the primary circuit. In addition, each unit is equipped with three diesel generators.

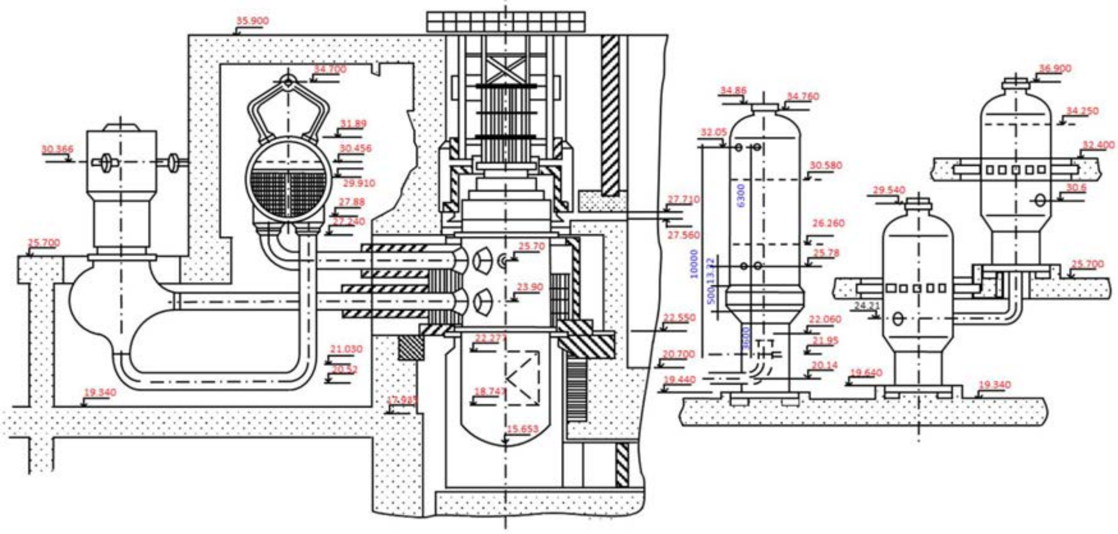


FIG. 19. Diagram of main equipment of VVER–1000/320.

TABLE 4. MAIN FRONT LINE SYSTEMS AND SAFETY FUNCTIONS OF VVER–1000/320 UNITS

Safety function	Front–line system
Residual heat removal	— Main feedwater–steam line system (3 Main Feedwater electrical driven pumps, 4 steam dump station to condenser, 3 condensate pumps, 3 circulation water pumps); — Auxiliary feedwater system (3 auxiliary feedwater pumps, 4 steam dump station to condenser, 3 condensate pumps, 3 circulation water pumps); — Emergency feedwater system (3 emergency feedwater pumps, 3 demineralized water tanks 1000 cub.m each, 4 steam dump stations to atmosphere, 8 steam generator safety valves (two on in each steam generator); — Planned cooling down system (3 low pressure ECCS pumps operating though planned cooldown line, each train equipped with heat exchanger); — Spend fuel pool cooling system (3 fuel poll cooling pumps, each train equipped with designated heat exchanger).
Maintaining primary water inventory	— High pressure emergency core cooling system (3 trains operating initially from sump filled with water though dedicated heat exchanger); — Low pressure ECCS (3 trains operating initially from sump filled with water though dedicated heat exchanger); — Hydro accumulators system (4 HAs 60 cub.m each).
Primary pressure protection system	Pressurizer safety valves system (3 pressurizer safety valves operating automatically and manually with ability to be kept opened for feed and bleed purposes)
Secondary pressure protection system	— Steam generators safety valves system (2 safety valves at each steam generator operating automatically and manually with ability to be kept opened); Steam dump station to atmosphere (4 in total in each steam line connected to steam generator).
Maintaining secondary side integrity	(a) Fast acting isolation valves, each on steam line connected to steam generator; (b) Feedwater isolation valves.
Maintaining containment pressure	Spray system (3 trains operating initially from sump filled with water though dedicated heat exchanger).

2.13.2. Features of the NPP important for MUPSA

The most important NPP information for the multiunit considerations is the information on possible dependencies between different units at the site. Units are not equipped with cooling towers; for condenser cooling the cooling pond is used. For cooling of essential equipment (including main cooling pumps) the service water system (SWS) is used that is cooled in specially dedicated spray ponds. It was identified that all units at Balakovo site have common parts of systems important to safety. The most important are as follows:

— Spray ponds and common parts of pipes of SWS: each service water train is connected to three spray ponds, in which the final heat removal to the atmosphere takes place. Each train of service water has a common part in the suction lines with similar train of all other units at Balakovo site. Each end–to–end pipeline is connected to four power units. Removing out of service of their gravity or pressure part requires shutting down the entire service water train at all four units, which leads to the inoperability of one safety train of all front–line systems at all four units;
— Diesel generator stations contain diesel generators and service water pumps. Since diesel generators buildings contain compartments with a diesel and compartments with service water pumps for two adjacent power units, if one building of the diesel power plant is lost (for example, due to a fire / explosion, external influence), the following initiating events can arise:
    o Loss of two channels of service water in unit 1;
    o Administrative shutdown due to loss of one safety systems train in unit 2;
    o Loss of chemically demineralized water storage tank;

Damage of open switchgears (200 kV of unit 1, 500 kV of units 2, 3, 4)

2.14. TUNISIA/STEG

The Tunisian Government decided on 3 November 2006 to conduct a feasibility study on the development of nuclear power production. The Tunisian Company of Electricity and Gas (STEG) leads the pre–feasibility studies. Participation in this CRP is aimed to support the development of technical capabilities in Tunisia needed for the future NPP project in PSA by developing a PSA model for Tunisian NPP.

2.14.1. Site and plant information

The Skhira site, shown in Fig. 20, is located 6 km south of the city of Skhira (Governate of Sfax), characterized by high banks of 5 m but eroded. It is a low activity zone with favourable agricultural accessibility, 10 km from the renovated industrial zone, TRAPSA storage bins, the Skhira oil terminal, and the SIAPE plant. A railway and the GP1 road are at a distance of 2 km.

Figure 20 presents a schematic of the Skhira site geographic location.

The selected reactor technology for the site is 3 units of ACP100, shown in Fig. 21. Main design parameters can be found in the IAEA ARIS database [3]

The ACP100 is an integrated PWR design developed by China National Nuclear Corporation to generate 125 MWe, based on existing PWR technology with passive safety systems designed to cope with consequences of the accidents. In case of transients and postulated design basis accidents the natural convection cools down the reactor. The ACP100 integrated RCS design enables the installation of major primary circuit components within the RPV.

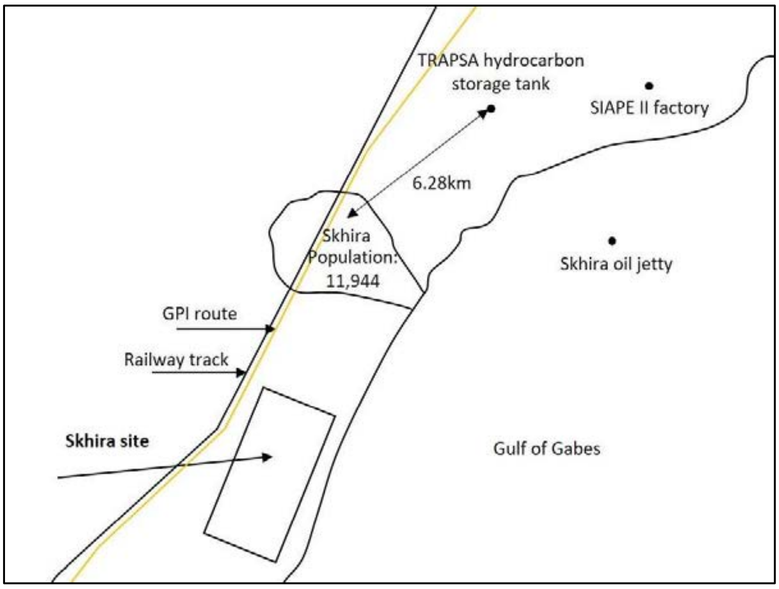


FIG. 20. Location of Skhira site.

FIG. 21. ACP 100.

2.14.2. Analysis scope

The scope of the analysis presented is Level 1 and Level 2 PSA on the Skhira site, in which the design and operation of three ACP100planned units in Tunisia are analysed to identify the sequences of events that can lead to core damage; the CDF is therefore estimated. The model is focused on the reactor core and does not include other sources of radioactive materials. The benchmark analysis includes the internal initiating events, i.e. the events that are caused by random component failures and human errors, while the external initiating events such as earthquakes or floods are not included. The full power operation as is considered to present the most significant risk when compared to other operation and shutdown modes. During the phase one and among 19 sites initially identified, only two sites have been selected and approved by the authorities. The criteria of selection were to avoid:

- Areas with low availability of heat sink cooling;
- Areas with high seismicity;
- Areas with high housing density;
- Military zone;
- Touristic zone.

2.15. UKRAINE/ENERGORISK

The benchmark aims to develop and quantify a stylized MUPSA model with meaningful modelling of single non–seismic hazard that affects operation of several units at Zaporizhzhya NPP (ZNPP). Each unit has separate full scope PSA model, developed during 2012□2018 and includes integrated Level 1 and Level 2 PSA models for all internal events, internal fires, floods and external hazards (except for seismic) for all operational modes (nominal and low power and shutdown modes) at reactors and SFP.

2.15.1. Site and plant information

The ZNPP site is located in the Kamenka–Dniprovska district of the Zaporizhzhya region on the left bank of the Kakhovka water reservoir (Dnipro River). The district centre, the town of Kamenka–Dniprovska, is positioned 12 km from the ZNPP site, 52 km from the regional centre, the city of Zaporizhzhya, and at distance of 5 km from the satellite town of Energodar. The local relief of the ZNPP site is flat, with alternating sand hummocks and hollows. The site levelling elevation is at 22 m. There are six power units operating at the ZNPP site with VVER–1000/320 reactors with the total electric power of 6,000 MWe. The general layout of the ZNPP site is shown in Fig. 22. Operational details can be found in IAEA’s PRIS database [5].

The main equipment of ZNPP units 1–6 is:

- Water cooled water moderated pressurized power reactor VVER–1000/320;
- Turbine unit K–1000–60/1500–2;
- Electric generator TVV–1000–4.

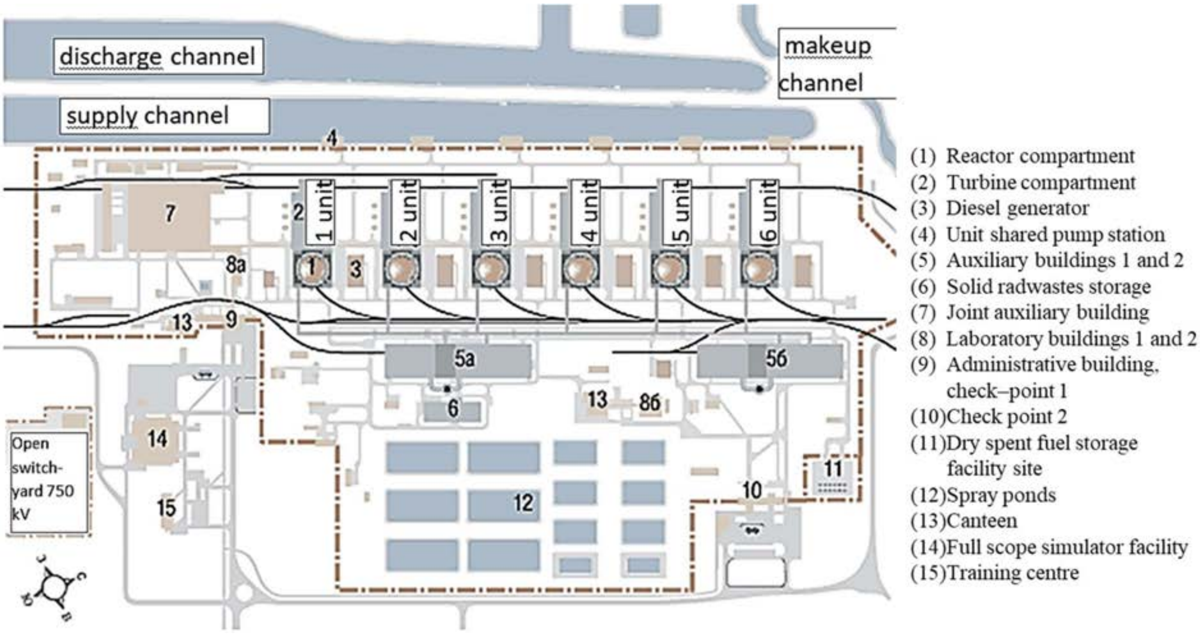


FIG. 22. ZNPP site layout [15].

Each reactor facility is equipped with VVER–1000/320 reactors. Layout diagram of main equipment of VVER–1000/320 is shown in Fig. 19. Reactor facility equipment is contained within the pre–stressed leak–tight reinforced concrete containment. The containment is a hollow cylinder with the spherical dome and flat bottom. The containment is made of reinforced concrete with 1.2 m thick wall while the dome section is 1.1 m thick. There is a leak–tight of 8 mm metal lining on the internal side of the containment.

The spent fuel assemblies unloaded from the core are stored in the SPF's racks. Before placing to a storage, the fuel assemblies are subject to fuel cladding leak testing. The testing results determine if a spent fuel assembly is placed in the rack slots or in a sealed canister. The SFP is adjacent to the reactor (housed inside the containment); it consists of three compartments used to store the spent fuel assemblies, as well as a well (an area for loading of transport casks with spent fuel assemblies and unloading of fresh fuel casing). Dividing the SFP into three compartments allows for maintenance in one of them while spent fuel assemblies are placed into the remaining two. The well stands separately from the fuel storage area, which permits installation of a fresh fuel casing into the dry well. The SFP is connected with upper part of the reactor cavity by a refuelling channel for transport of fuel assemblies.

The SFP is filled with of at least 16 g/dm3 concentration of boric acid solution. Water in each SFP compartment circulates through the SFP keeping the water temperature within a permissible limit of no more than 70°C (with the reactor core completely unloaded). This value is established to prevent boiling of the cooling water and fuel melting due to decay heat. Additionally, it ensures a protective level of water in the SFP during spent fuel storage and provides radiation protection for maintenance personnel.

In an emergency situation there is a possibility for a makeup and fuel cooling in SFP compartments with a 16 g/dm3 boric acid solution from pumps of the containment spray system. In this case makeup and water cooling in SFP proceeds as follows: containment sump – emergency cooldown heat exchanger –

containment spray pump – SFP compartment – SFP flow through the transport opening to the refuelling pool – water flow through the lower elevation of the containment to the containment sump.

2.15.2. Analysis scope

The scope includes:

- Selection of non–seismic hazard (or combination of events) to be modelled;
- Development of new stylized Level 1 and Level 2 MUPSA models for full power operation, considering all technical elements of PSA; site facilities shares (electrical interconnections, support systems, common buildings); other dependencies (proximity, human, operational, organizational), as well consideration of mutual impact of reactor/ SFP at single unit;
- Quantification of CDF for damage combinations (i.e. damage frequency for 2/5, 3/5, 4/5, 5/5 reactor cores) and fuel damage frequency (FDF) for damage combinations (i.e. damage frequency for 2/6, 3/6, 4/6, 5/6, 6/6 SFP).

2.16. UKRAINE/SSTC NRS

The following is the description of the Hungary's benchmark specifics including the description of the site and plant.

2.16.1. Site and plant information

The RNPP site shown in Fig. 23 consists of four VVER units of different designs; two are VVER–440/213, and other two are VVER–1000/320. RNPP site is located in the north–west of Ukraine on the border of the Rivne and Volyn regions.

FIG. 23. Location of reactor units at RNPP site, [14].

The designing of the NPP has started in 1971. RNPP is the first NPP in Ukraine with VVER reactors. Four power units with total electric power of 2,835 MWe are operated at RNPP site: two with VVER–440/213 (power units 1, 2) and two with VVER–1000/320 (power units 3, 4). The first two power units with VVER–440/213 were commissioned in 1980 and 1981, and unit 3 with VVER–1000, in 1986. RNPP–4 was commissioned in October 2004. In April 2006, RNPP–4 was put into commercial operation. After unit 4 startup, the annual NPP electricity production exceeds 17 billion kWh. Operational details can be found in IAEA's PRIS database [5].

On 10 December 2010, at State Nuclear Regulatory Inspectorate of Ukraine Board meeting, it was decided to extend lifetime of RNPP–1 and 2 for 20 years, upon condition to perform safety review for these power units each 10 years. RNPP–3 lifetime was extended for 20 years by relevant State Nuclear Regulatory Inspectorate of Ukraine decision.

2.16.1.1. Units 1 and 2

Total electric power of RNPP 1 and 2 is 835 MWe. The main equipment of RNPP 1 and 2 is:

— VVER–440/213;
— K–220–44 turbine installation;
— TVV–220–2AUZ electric generator.

According to the design principles, the reactors of RNPP 1 and 2 have two sides. Reactor of each unit are located in separate reactor buildings, and turbine generators are located in the common turbine hall. Systems and equipment of VVER–440/213 ensure:

— Safe and reliable reactor operation under all normal operation modes envisaged by the design;
— Operation control, state control of the main reactor equipment;
— Reactor integrity in all emergency modes envisaged in the design;
— Integrity and reliable cooling of fuel assemblies in the reactor core under all operation mode envisaged in the design.

Reactor design provides protection against radiation impact on NPP operating personnel and the environment in accordance with current health and safety standards in all reactor operation modes envisaged in the design (including expected transients and design–basis accidents).

The cooling system of VVER–440/V–213 consists of six circulation loops. Each loop includes: main circulation pump; reactor coolant pump; steam generator; main gate valves to ensure disconnecting of any loop from the reactor. Each of the six circulation loops consists of a hot and a cold leg: by the hot leg, the coolant is transferred from the reactor to steam generator; by the cold leg, the coolant cooled in steam generator is transferred to reactor coolant pump suction side and then to the reactor. On the cold and hot legs of each loop, main gate valves are installed, which are used to isolate a loop with a damaged component in case of an accident (steam generator leak, main circulation pump leaks in the isolatable part, etc.).

The main objective of the core cooling function is to prevent fuel damage due to overheating. For this, heat removal systems and equipment are provided in all operation modes. Heat is removed from the core under normal operation of power units with VVER according to the following scheme: core – primary coolant – steam generator – secondary coolant – final absorber – atmosphere. During accidents, as well as after reactor shutdown, residual heat is removed according to the same scheme, and if it is impossible to remove heat by steam generator, it is removed by low or high pressure injection system to the essential service water system (ESWS).

Spray systems purpose is to perform safety functions in case of initiating event and any independent single failure of one active component that has mechanical moving parts or additional human error to prevent release of radioactive products outside the NPP site. In loss of integrity of a primary side, release of radioactive products into the environment is prevented by the containment system, in which negative pressure is maintained due to operation of the ventilation systems. To limit radioactivity release during the accidents, the design provides confining safety systems (components): systems (components) designed to prevent or limit the spread of radioactive substances and radiation beyond the boundaries provided in the design.

Cooling towers with a capacity of 100,000 m^3/h each are available as auxiliary heat sink from turbine condensers and heat exchangers. Spray ponds are used to remove heat from essential loads.

An auxiliary system for emergency feedwater supply to steam generators is provided for feedwater supply to steam generator in case of CCF of all design main emergency feedwater supply systems. The system was mounted and put into service in 2010. The auxiliary system for emergency feedwater supply to steam generators is located in a separated building; demineralized water storage tanks are located outside the building on separate foundations and are connected to auxiliary system for emergency feedwater supply to steam generators building by an underground channel. The system is common to RNPP 1 and 2 and consists of two independent subsystems; each is designed for its own power unit.

The reactors of units 1 and 2 have a system of sealed reactor building rooms. The RNPP 1 and 2 units are identical to the VVER–440/213 design, have identical space layouts and do not have any fundamental differences impacting assessment of power unit vulnerability to external extreme natural hazards, safety function fulfilment and severe accident management.

Spent fuel assemblies unloaded from the reactor core are stored in the racks in the SFP. The storage system for spent fuel assemblies is located in reactor building provided with necessary rooms and equipment to accept and store spent fuel assemblies. The SFP is located within the steam generator box and is connected with the reactor by reloading channels to transport fuel assemblies. Spent fuel storage and refuelling is performed under a protective water layer. The SFP is equipped with fuel compact storage system and racks in which spent fuel assemblies, screen cartridges, tight canisters for defective fuel assemblies and special tools are placed. In the event of an emergency spent fuel unloading from one power unit, if necessary, additional use of SFP of the second power unit is envisaged.

2.16.1.2. Units 3 and 4

Total electric power of RNPP 3 and4 is 2000 MW. The main equipment of RNPP 3 and 4 is:

— VVER–1000/320;
— K–1000–60/3000 turbine installation;
— TVV–1000–2UZ electric generator.

The reactors of RNPP 3 and 4 are standard VVER–1000/320 design. The systems and equipment of VVER–1000/320 provide:

— Safe and reliable operation of the reactor under normal operation modes as per design;
— Operation management, control of state of the main reactor equipment during operation;
— Preserving integrity of reactor structure under all emergency modes envisaged by the design;
— Integrity and reliable cooling of reactor core fuel assemblies under all reactor operation modes envisaged by the design.

Reactor equipment is enclosed in a pre–stressed sealed reinforced concrete containment with a shape of a hollow cylinder with a torus–spherical dome and a flat bottom. Thickness of the reinforced concrete containment wall is 1.2 m (in the cylindrical part) and 1.1 m (in the dome part). Inside, there is a sealed metal lining with a thickness of 8 mm. Primary coolant system includes the reactor and four circulation loops, each consisting of PGV–1000M steam generator, 195M reactor coolant pump, and main circulation pump. The main circulation pump is composed of pipes with an internal diameter of 850 mm and a thickness of 70 mm. Each primary loop is conditionally divided into cold and hot legs. By the hot leg, the coolant from the reactor enters steam generator, and by the cold leg, from steam generator, it returns to the reactor through reactor coolant pump. The design of all loops is similar that ensures equal hydraulic resistance of the loops and, consequently equal coolant flow rate.

Spent fuel assemblies unloaded from the reactor core are stored in racks in SFP. The storage system for spent fuel assemblies is located in reactor building provided with necessary rooms and equipment to accept and store spent fuel assemblies. The SFP is located in containment and consists of three compartments designed directly to store spent fuel assemblies and a well: area to load TK–13 transport container with spent fuel assemblies and unload a cask with fresh fuel assemblies. The SFP division into three compartments allows maintenance in one of them, while placing spent assemblies in others and draining a compartment under maintenance. The well is structurally separated from fuel assembly's storage area, which allows installing a cask with fresh fuel in the dry well. The SFP is directly adjacent to the reactor cavity and is connected to it by a refuelling channel to transport fuel assemblies. The design of NPP units with VVER–320 provides emergency makeup and cooling of fuel in SFP compartments by boric solution with a concentration of 16 g/dm3 from spray system pumps. Emergency makeup and cooling of fuel in SFP is carried out according to the following scheme: sump tank – emergency cooling heat exchanger — spray pump — SFP compartment — SFP trans–fill through a transport opening into the cavities of the wet refuelling pool — water drainage through the lower Level of the containment into the sump tank. RNPP 3 and 4 are designed according to the identical VVER–1000/320 design, have identical reactors, planning, space and layout solutions. However, for unit 4, the design additionally provides a unit uninterruptible power system including two channels with autonomous diesel generators. Power of each diesel generator (5,600 kW) is selected to ensure startup for loading when power is lost for two units. The following possibilities of power supply under SBO are implemented at RNPP site for any power unit:

— Power supply from one of power units at the site transferred to essential loads;
— Power supply to 6 kV standby power supply lines of power units from diesel generator of any safety system of any power unit at the site or from the unit standby diesel generator station;
— OnsiteSNF storage facilities are not presented at RNPP site.

Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs (C(I)SIP) is under implementation at RNPP units (as of the time this publication is written). It was developed to perform safety improvement activities within implementing the long–term state safety improvement strategy for Energoatom NPP units. C(I)SIP measures aimed at safety improvement of power units and equipment modernization are implemented, as a rule during annual scheduled outage. In total, implementing 212 C(I)SIP measures are envisaged for RNPP. In addition, within C(I)SIP a set of measures of the National Action Plan Based on Stress–Tests is being implemented to prevent beyond design basis modes possible due to natural phenomena associated with the improvement of severe accident management (so–called post–Fukushima Daiichi NPP accident measures). The main purpose of implementing these measures is to prevent and eliminate a beyond design basis accidents

2.16.2. Analysis scope and safety goals

The benchmark studied the significance of multiunit effects in PSA Level 1 and Level 2 for Rivne NPP (RNPP) site with the goal to develop a general methodology/approach for MUPSA and elaborate on the background for further development of national regulatory guides. The overall objective is to prepare the framework and process for consideration of the multiunit effects in PSA. The specific goal is to define the list of aspects significant for multiunit effects at the RNPP site (hazards, common systems and buildings, mutual impacts, etc.), elaborate the approach to their consideration in the existing PSA Level 1 and Level 2 models, and modify the PSA models to assess their influence on CDF/LRF. The benchmark steps adopted for developing MUPSA model for RNPP site are shown in Fig. 24. The integrated PSA models have been developed and agreed for RNPP units considering all possible internal and external initiators with respect to the reactor core and SFP, and all power unit states, as shown in Fig. 25. SUPSAs are developed using SAPHIRE software/tool [9], which is also used for the MUPSA benchmark.

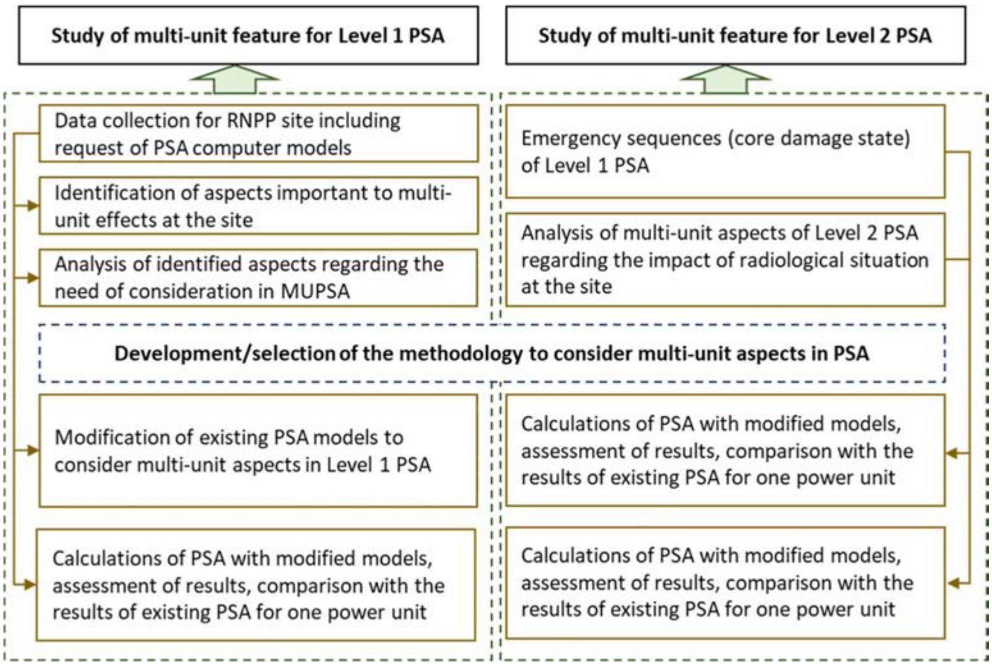


FIG. 24. MUPSA benchmark steps for RNPP site.

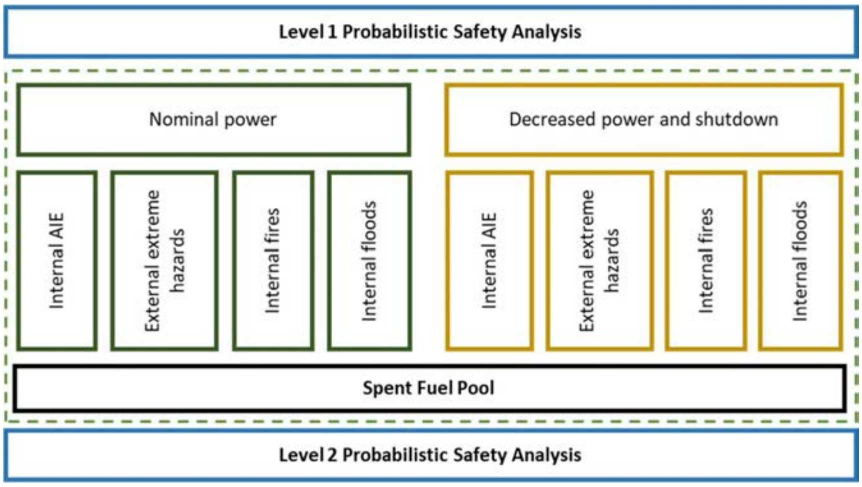


FIG. 25. PSA approach for RNPP units 1–4.

2.17. UKRAINE/ENERGOATOM

The following is the description of the Hungary's benchmark specifics including the description of the site and plant.

2.17.1. Site and plant information

The South Ukraine NPP (SUNPP) site, shown in Fig. 26, is located in the Arbuzinka district in the Mykolayiv region of Ukraine, on the left bank of the Yuzhny Bug river. Operational details can be found in IAEA's PRIS database [5].

FIG. 26. SUNPP site, [15].

The SUNPP is located on the left bank of the Yuzhny Bug river, at a distance of ~159 km from its estuary. From south to north, the territory of the SUNPP site is crossed by the Tashlyk water reservoir. From north–west to south–east the Yuzhny Bug river flows 60 km through the 30 km area. The nearest main building (unit 1) is 2.7 km away from the river bank. There is no woodland around the NPP site. There are no gas or oil pipelines, factories or chemical plants within the 30 km area of the SUNPP. The closest town, Yuzhnoukrainsk, is 2.5 km away from the site. The town of Voznesensk and several urban settlements and villages are within 30 km from the NPP site. The nearest big city, Mykolayiv, is located 112 km from the SUNPP. In the seismicity study of SUNPP site the maximum possible value of the peak ground acceleration (PGA) is 0.093g. However, according to the regulator requirements for equipment seismic qualification this value is increased by 30%, to 0.12g.

There are three VVER–1000 units in operation at the SUNPP. The units provide an electrical power output of 3,000 MWe. The unit 1 is VVER–100/V302 reactor facility, unit 2 is VVER–1000/V338 reactor facility and unit 3 is VVER–1000/V320 reactor facility. Each VVER–1000 safety system consists of three independent trains, each independently capable of ensuring the performance of design functions, and that is, the redundancy ratio for safety systems being assumed to be 3×100%. Reactor equipment is housed in the pre–stressed leak tight reinforced concrete containment having the shape of a hollow cylinder with a spherical dome and flat bottom. SUNPP units 1 and 2 are of identical design with similar reactor facilities, layout and arrangement; the reactor units therefore have no significant differences that may affect assessment of their vulnerability to extreme external natural events, performance of safety functions or severe accident management. The differences between the designs of V302/338 and V320 are in the design of containments, the absence of the main gate valve in the V320 reactor, and the presence of the additional non–safety common unit diesel generators; this all taken into account in the MUPSA model

2.17.2. Analysis scope

A limited scope Level 1 MUPSAs was developed and recommendations for the multiunit CDF and LRF assessment for national industry were specified.

2.18. UAE/KHALIFA UNIVERSITY

The following is the description of the Hungary's benchmark specifics including the description of the site and plant.

2.18.1. Site and plant information

The Barakah Nuclear Energy Plant is in Al Dhafra of the Emirate of Abu Dhabi on the Arabian Gulf, approximately 53 km from the city of Ruwais, as shown in Fig. 27. The NPP's four APR1400 type nuclear reactors will once fully operational, supply total of 5,600 MWe up to 25% of the UAE's electricity needs. Operational details can be found in IAEA's PRIS database [5].

Construction of the Barakah Nuclear Energy Plant commenced in July 2012, following the receipt of the Construction License from the Federal Authority for Nuclear Regulation and a No Objection Certificate from Abu Dhabi's environmental regulator, the Environment Agency, Abu Dhabi. However, the standard PWR model is used as the reference plant of the PRA pilot model. The site of the NPP could have from 1 to 4 units of PWR reactors in the pilot model. The PRA of a single unit with multiunit features is studied (AAC sharing and electrical Emergency Diesel Generator (EDG) crosstie); the schematic is shown in Fig. 28.

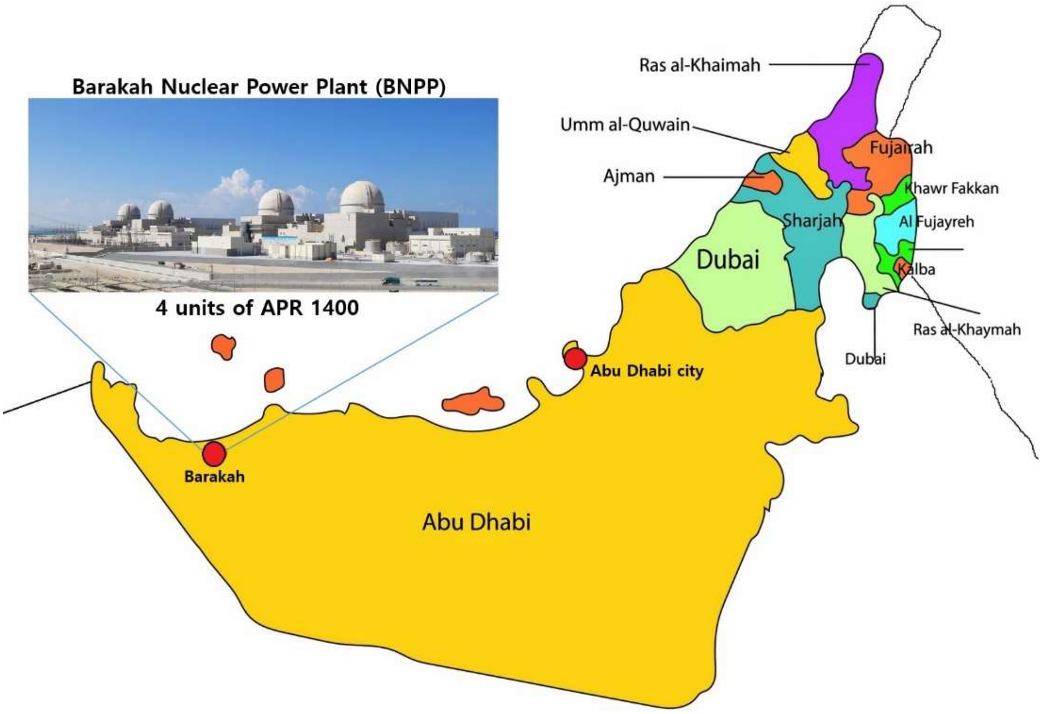


FIG. 27. Barakah nuclear energy plant site.

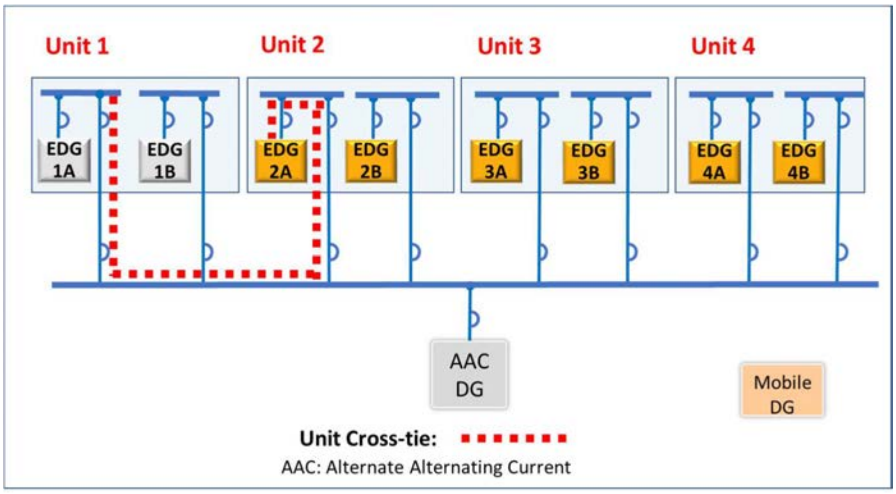


FIG. 28. Reference Barakah nuclear energy plant of the PRA pilot model for electrical crosstie.

2.18.2. Analysis scope and safety goals

The NPPs in one site may share structures, systems, and components (SSCs) either in normal operation or on demand. For instance, the sharing of the electrical power source as a crosstie of EDGs could reduce the SBO frequency for the site in extended SBO. However, the crosstie is associated with risks that can limit the shared systems from performing their intended functions, which need to be managed.

A methodology for risk analysis and management of shared electrical power in NPPs was developed to investigate electrical cross– tie in extended SBO event as a risk of dependency in a multiunit site.

The impact of CCFs was examined in a case study to quantify the risk of sharing of electrical cross–was modelled in different cases including the risk management process to reduce the crosstie risk of multiunit in different accident scenarios of SBO. Related analysis such as (sensitivity, importance measures, risk and cost analysis) supported the risk informed decision making process in selecting between mitigation options and reducing the SBO risk. Following are the objectives:

- Identify the associated issues of the electrical crosstie option to cope with SBO in systemic approach and to review the literature on the issues of sharing in multiunit;
- Develop a probabilistic approach to quantify the risk of crosstie and conduct a risk/cost analysis for the sharing of AC power based on developed criteria;
- Carry a study of Risk Management of multiunit in different accident scenarios that related to LOOP and SBO (optimization in using AC power to prevent multiunit core damage).

This methodology provides a comprehensive probabilistic approach that incorporates the latest data of multiunit LOOP, which specify the conditional probability of all plants at a site experiencing a LOOP, given a LOOP at the specific plant being analysed, and addresses the following:

- CCF between shared components and systems including common SBO initiating event. The CCF is examined in a simple example to confirm its significant impact on total risk;
- Factor of occupancy of shared components and systems;
- Human actions needed and involved human failures;
- Time factor to achieve the success criteria.

3. LEVEL 1 MUPSA

The PSA Level 1 may be of different scope starting from internal events PSA that considers only damage of the fuel in the reactors and ending with full scope PSA. Full scope PSA in wider term considers all fuel located at the site, not limited to fuel in the reactor and SFP but also fresh fuel and spent fuel in long term storage located at the plant site. Full scope PSA analyzes all POSs: power operation at different power levels, refueling states, states due to unplanned maintenance and repairs. It also considers all potential initiating events that may challenge plant safety leading to fuel damage. These initiating events can be caused by equipment random failures or human errors (called internal initiating events), but they may be also induced by internal hazards (e.g. fires and floods inside the plant) or external hazards, both natural (seismic, strong winds, external fires and floods) or human–induced events (such as but not limited to aircraft crash, transportation accidents, chemical materials releases, explosion in nearby facilities or gas lines). The recommended scope of PSA Level 1 in the IAEA Safety Guide SSG–3 Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [2] is limited to full scope

PSA in terms of initiating events and operational states; however, recommendation for consideration of fuel in SFP is only limited for low power and shutdown modes when fuel is removed from the reactor as defined in paragraph 1.13 of SSG–3 [2]. This limitation may have relatively low impact on PSA Level 1 results, but even more important for PSA Level 2, where release of radioactive materials are considered from all sources. This aspect is only slightly touched in paragraph 2.3 of SSG–3 and in relation to PSA Level 2 and Level 3, where it is recommended to "address the impact of radioactive releases from other sources of radioactive materials on the site, such as spent fuel and radioactive waste while assessing the total risk from the plant to the public near the site."

Multiunit considerations are mentioned in SSG–3 in relation to fire or flood spreading from one unit to another and are not at all addressed in SSG–4 Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [3]. However, TECDOC 1804 [16] provides a set of technical attributes for multiunit PSA, which were considered while developing this document.

3.1. RISK METRICS AND SAFETY GOALS

Both SSG–3 [2] and SSG–4 [3] do not provide specific safety goals. The SSG–3 refers to the following risk metrics in para 2.11:

a) Probability of failure of particular safety functions or safety systems;
b) Frequency of core damage (PSA Level 1).

It also highlights following objectives for core damage frequency, established in [17]:

a) 1×10^{-4} per reactor–year for existing plants and;
b) 1×10^{-5} per reactor–year for future plants.

SSG–3provides further discussion on potential risk metric for Level–2 in para 2.17: "A large release of radioactive material, which would have severe implications for society and would require off–site emergency arrangements to be implemented."

With reference to INSAG–12, SSG–4 [3] provides in paragraph 2.18 "the objective for large off–site releases requiring short term off–site response as 1×10^{-5} per reactor–year for existing plants." The SSG–4 does not specify a numerical value for a large off–site radioactive release for future plants, but states: "Another objective for these future plants is the practical elimination of accident sequences that could lead to large early radioactive releases, whereas severe accidents that could imply late containment failure would be considered in the design process with realistic assumptions and best estimate analyses so that their consequences would necessitate only protective measures limited in area and in time." Finally, SSG–3 mentions in paragraph 2.20 a safety goal applicable to Level–3 PSA: Health effects to members of the public. It refers to the target for the risk of death of a member of the public that is taken in some member states to be 1×10^{-6} per reactor–year. Neither SSG–3 nor SSG–4 provide any risk metrics and safety goals for multiunit sites. The definitions of risk metrics that are used for single units and can be used for multiunit are provided in TECDOC–1804 [16], and even though it does not discuss any numerical values for safety goals that can utilize such metrics these metrics are quite comprehensive and for convenience are cited here:

— **CDF/FDF, annual average (CDF_{AVE}, FDF_{AVE}):** represents “the frequency of core/fuel damage quantified in a single reactor PSA that is averaged over the time dependent variations that may be exhibited during the course of a year due to plant configuration changes, removing equipment from service to perform tests or maintenance, and the occurrence of plant initiating events which may in fact vary over the course of a reactor lifetime. Periodic updates of this risk metric over the course of the plant lifetime provide a slow version of a time dependent risk monitor that reflects broad trends in plant and SSC performance that are reflected in the plant data as well as any permanent changes that are made in the design, maintenance and operation. CDF_{AVE} is expressed in a per reactor–calendar–year basis.” [3]

— **Multiunit CDF/FDF, annual average ($MCDF_{AVE}$, $MFDF_{AVE}$)**: represent “the frequency of an accident involving core/fuel damage on two or more reactor units quantified in a multiunit PSA that is averaged over the time dependent variations that may be exhibited from combinations of reactor unit configurations. $SCDF_{AVE}$ is expressed in a per site calendar year basis.” [3]

— **Site CDF/FDF, annual average ($SCDF_{AVE}$, $SCDF_{AVE}$)**: represent “the frequency of core/fuel damage on one or more reactor units quantified in a multiunit PSA that is averaged over the time dependent variations that may be exhibited from combinations of reactor unit configurations.” [3] The $SCDF_{AVE}$ is expressed in per site calendar year.

— **Large early release frequency, annual average ($LERF_{AVE}$)**: represent the frequency of a large early release that is averaged over the time dependent variations that may be exhibited during the course of a year, which may in fact vary over the course of a reactor lifetime.

— **Site large early release frequency, annual average ($LERF_{AVE}$)**: represent “the frequency of a large early release due to an accident involving releases from one or more reactor units that is averaged over the time dependent variations that may be exhibited during the course of a year, which may in fact vary over the course of a reactor lifetime.” [3]

— **Complementary cumulative distribution function, annual average ($CCDF_{AVE}$)**: represent the annual frequency of exceedance of consequences quantified in a single reactor Level 3 PSA for different consequence metrics such as early fatalities, latent cancer fatalities, property damage costs, etc. The CCDFs are aggregated to account for all the release categories and associated single unit accident sequences modelled in a Level 3 SUPSA.

— **Site complementary cumulative distribution function, annual average ($SCCDF_{AVE}$)**: represent “the annual frequency of exceedance of consequences quantified in a multi reactor unit Level 3 PSA for different consequence metrics such as early fatalities, latent cancer fatalities, property damage costs, etc. CCDFs are aggregated to account for all the release categories and associated single unit and multiunit accident sequences modelled in a multiunit Level 3 PSA.” [3]

As it can be seen from these definitions the difference is only in several units involved in risk assessment; however, this difference may lead to major difference in safety goals concerning the NPP, multiunit or site risk. The safety goal that are based on multiunit or site risk are not established in the IAEA Member States, but it seems evident that public risk needs to be assessed considering all source of radioactivity at the site and safety goals that are based on the overall site risk are expected to be established. However, to establish safety goals that are based on the site risk, the technic how this risk needs to be assessed needs to be developed. This publication is aimed to provide information on how to perform the MUPSA, which is the first step in assessing the site risk. The following sections provide the relevant discussion based on performed benchmarks.

3.1.1. Argentina/CNEA

The radioactive sources considered in the benchmark are the reactor core of each unit and the spent fuel in respective pool. It is considered that both units are at full power operational state. According to the scope of this MUPSA study, the multiunit initiating event that is modelled is LOOP. The proposed risk metric of Level 1 MUPSA is multiunit FDF. This is an intermediate metric, considering that the proposed risk metric for MUPSA is the IRR, which is explained in Section 4.2.1. A safety goal for IRR as a metric could be defined based on Argentine acceptability criterion for licensing NPP and research reactors.

3.1.2. Canada/COG

Table 5presents PSA quantitative safety goals for the Canadian utilities operating CANDU reactors. In Canada and more specifically Ontario, multiple units exist within a common containment structure and many important systems are shared or interconnected. This has highlighted the need for whole site PSA and the adoption of a more holistic and site based approach to provide a more complete picture. By incorporating both programmatic and quantitative elements, a broad perspective on the adequacy of activities necessary to ensure an adequate level of safety for a multiunit site is provided. The reasonableness of site risk is demonstrated by means of various programmes that (a) are in place for all aspects of operation, (b) comply with applicable regulatory requirements, (c) collectively assure NPP safety, and (d) manage risk to be reasonably low.

TABLE 5. PSA QUANTITATIVE SAFETY GOALS

Safety goal (application)	Average risk (per year)		Instantaneous risk (per year)
	Administrative safety goal	**Safety goal**	**Safety goal**
Large off–site release[3] (per unit)	10^{-6}	10^{-5}	3×10^{-5}
Severe core damage[4] (per unit)	10^{-5}	10^{-4}	3×10^{-4}

3.1.3. China/INET

The existing PSAs for HTR–PM were done by INET, covering internal events for reactors and spent fuel storage facility at full power and shutdown mode. Preliminary PSA assessments are also carried out for internal floods, internal fires, and PSA based seismic margin assessment. This benchmark is focused on internal event PSA for reactors and spent fuel storage facility, while a detailed assessment for internal fire and seismic PSA were not included in this benchmark. The current safety goals for the other light water reactor (LWR) NPPs in China are defined by CDF and large or early release frequency (LERF) on a per–reactor basis. However, these two metrics are not suitable to describe the feature of HTGR. For the HTR–PM, it has been agreed to use a specific risk metric, that is, the cumulative frequency of accident sequences with the estimated consequence exceeding 50 mSv at the site boundary in terms of individual effective dose that is < 1×10–6/reactor year. This goal is defined on a per reactor basis. This benchmark is intended to see if it is appropriate for the whole plant. HTR–PM PSAs adopt a different framework from the traditional

[3] Large release refers to the release of fission products containing greater than 10^{14} Bq of ^{137}Cs to the environment.

[4] Severe core damage refers to loss of core structural integrity.

PSAs, e.g. Level 1, Level 2 and Level 3. Namely, HTR–PM PSAs combine Level 1 and Level 2 development into one integrated model, starting from the initiating event directly to the release category. The related source terms of each release category are then quantified to get their dose estimates to check whether they can meet the 50 mSv metric or not. Thus, HTR–PM PSAs are here considered as PSA Level 2+.

3.1.4. Finland/VTT

In Finland and Sweden, the safety goals are unit specific. In the Nordic SITRON project [7], proposals for site level risk metrics are made. The main risk metric for Level 1 MUPSA is the site CDF/FDF. Multiunit CDF is also calculated in this benchmark.

3.1.5. Ghana/GAEC

Multiunit risk may be defined as a frequency of one or more core damage states [18]. This definition implies the union of core damage states of unit 1 and unit 2, as shown in Fig. 29.

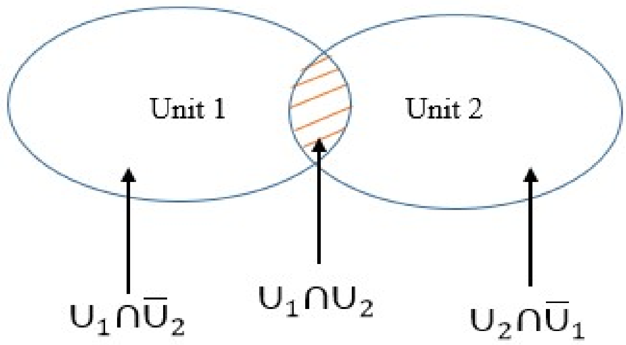


FIG. 29. SCDF concept.

The assumption is that the core damage states occur simultaneously. The union of minimum cut sets (MCS) of units 1 and 2 represent the SCDF. The SCDF estimation considers the probability per year resulting from all inter–unit dependencies $P(CD^{(i)})$ as well as the conditional CDF of a unit given condition, C_j, expressed as $P(CD^{(i)}|C_j)$. A further assumption underlying this approach is that all reactor units on site are subject to the same operating profile within the period for which the safety analysis is conducted. The concept of frequency used in this context is based on the Poisson model. The SCDF can be expressed mathematically as in Eq. (1–5):

$$P\left(\cup_{i=1}^{2} CD^{(i)}\right) = \sum_{i\leq 2} P\left(CD^{(i)}\right) - \sum P\left(CD^{(1)} \cap CD^{(2)}\right) \quad (1)$$

A simpler and more conservative estimate of the SCDF can be obtained using Boole's inequality:

$$P\left(\cup_{i=1}^{2} CD^{(i)}\right) \leq P(CD^{(1)}) + P(CD^{(2)}) \quad (2)$$

If a condition C_j exists that may affect a subset of reactor units 1 to k, each term of Eq. (2) is written as:

$$P(CD^{(1)}) = \sum_j P(CD^{(i)}|C_j)P(C_j) \quad (3)$$

For multiunit (two units in our case), the SCDF under condition C_j can be expressed as:

$$P(\cap_{i=1}^{k} CD^{(i)}|C_j) = P(CD^{(1)}|C_j)P(CD^{(2)}CD^{(1)} \cap C_j) \quad (4)$$

The total annual probability of core damage for k units under all conditions C_j is:

$$P(\cap_{i=1}^{k} CD^{(i)}) = \sum_j P(\cap_{i=1}^{k} CD^{(i)}|C_j)P(C_j) \quad (5)$$

Both marginal annual probability of core damage and conditional probability of core damage need to be estimated for a unit given a set of specific conditions. Initiating events such as LOOP may lead to abnormal plant conditions requiring safety systems to be activated to return the plant to a normal condition. The large offsite release safety goal states that the aggregate of frequencies, LRF of all event sequences that can lead to a total release from the site to the environment of more than 10^{14} Bq of ^{137}Cs need to be less than $N\times10^{-5}$ per site year (N is the number of units on the site). The SCDF goal states that the aggregate of frequencies of all event sequences that can lead to significant core degradation in any of one or more reactors on the site to be less than $N\times10^{-4}$ per site year. In addition to the above metrics which are quantitative, there are qualitative safety goals that need to be established. These can be defined broadly as follows:

The public have to be provided a level of protection from likely consequences of accidents at NPP sites such that the likelihood of additional risk to the life and health of individuals is significantly reduced; and the potential for extensive societal disruption due to a nuclear incident need to be practically eliminated, so as to not significantly add to the other societal risks to which the public is normally exposed [19]. A key point to note in view of the above safety goals is the fact that PSA is a limited methodology in estimating risks associated with NPP sites. Therefore, PSA results are not risk measures but instead provide an indication of risk to be used to complement other deterministic approaches as well as provide insights into plant vulnerabilities.

3.1.6. Hungary/NUBIKI

The Level 1 PSA for NPP Paks includes the quantification of CDF in the reactor and FDF in the SFP separately for each of the four reactor units. To quantify risk at site level, the frequency of single and multiple core damage and fuel damage sequences have to be known and aggregated correctly. The fuel damage can be regarded as a generic term, and core damage sequences, most commonly quantified in a Level 1 PSA, thus representing a subset of the entire space of fuel damage conditions. The widely applied definition of site level SCDF/SFDF was found appropriate for use in the MUPSA of NPP Paks, hence, the frequency of core/fuel damage associated with at least one large radiological source was aimed to be quantified. However, calculating the cumulative annual frequency of accident sequences leading to core or fuel damage at exactly two/three/four units (or even more sources) was also targeted, since these metrics are an important basis of getting valuable risk insights. There are no multiunit or multi source safety goals available in Hungary at present, so the results were not compared to any predefined goal or criteria.

3.1.7. India/AERB

The quantitative safety targets for CDF for new reactors is 1×10^{-5} per reactor per year. Currently, there is no risk metric defined for multiunit multi source. However, multiunit CDF is proposed as a risk metric for Level 1 PSA of multiunit site. As India has adopted unitized concept, the safety systems of multiple units are independent and physically isolated and hence less susceptible to multiunit core damage.

3.1.8. India/BARC

The risk metrics proposed to use in the present study are site CDF (SCDF) and site release frequency (SiRF). Currently India does not have PSA based safety goals. Only risk metrics based on CDF and LERF are regulatory requirement for single unit. For multiunit, no metrics are devised based on PSA. SCDF and SiRF are proposed as part of R&D framework on MUPSA.

3.1.9. Pakistan/PAEC

The risk metrics for MUPSA Level 1 are single unit CDF, multiunit CDF and SCDF. Single unit CDF is defined as frequency of a reactor accident involving core damage on one and only one reactor unit per site calendar year. It is highlighted here that single unit CDF is not the same as a specific unit's CDF. The CDF normally reflects the estimated frequency of core damage per reactor–calendar–year associated with a particular unit on the site. The single unit CDF is the sum of all CDFs involving single core damage. Multiunit CDF is defined as frequency of an accident involving core damage on two or more reactor units concurrently per site calendar–year. Finally, SCDF is defined as frequency of a reactor accident involving core damage on one or more reactor units concurrently per site calendar–year. Therefore, the traditional/conventional PSA Level 1 study provides CDF which is based on single unit and could be utilized/used in the estimation of risk metrics for MUPSA Level 1. The concept of risk metrics is depicted in the Fig. 30.

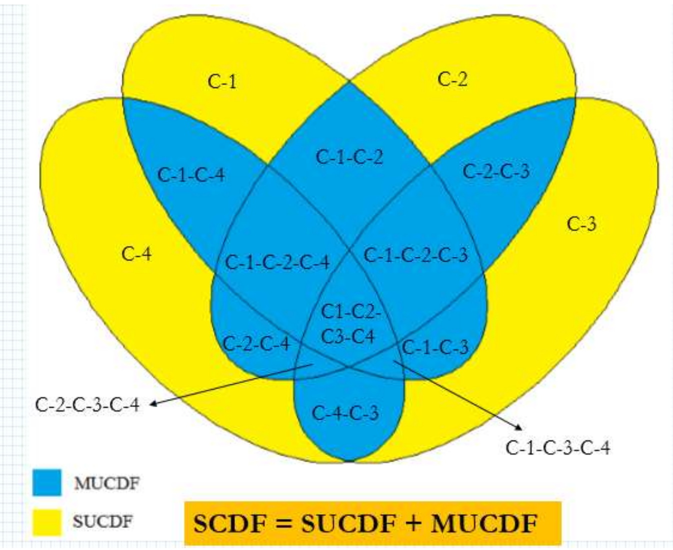


FIG. 30. Risk metrics for Chashma site. C–1, C–2, etc. refer to Chashma units that experience core damage.

Each circle in the figure represents CDF due to single unit and is based on single unit/base unit PSA Level 1 study. The overlapping areas represent multiunit CDF which is based on dependency analysis of a multiunit NPP site. The dependency of multiunit site may have various perspective e.g. multiunit initiating event particularly external hazards, shared SSCs, identical components, proximity, human resources, organizational level etc. Therefore, multiunit CDF is a better risk metric in comparison to CDF as it captures dependencies of a particular site and design. Further, CDF and multiunit CDF are used in the calculations of single unit CDF and SCDF. In present study, concurrent damage of reactor cores of all four units is considered in computation of multiunit CDF. The risk metrics considered in this study for MUPSA Level 1 are tabulated in Table 6.

TABLE 6. RISK METRICS

Risk Metric	Applicability
CDF	Level 1 single unit PSA
Single unit CDF Multiunit CDF SCDF	Level 1 multiunit PSA

For all shared GIS shared structures the annual frequency of unacceptable performance of structure (i.e. seismic risk) is calculated. The seismic risk metrics of shared structures corresponding to family of fragility curves is given in Table 7.

The outcome of the annual frequency of the unacceptable performance (GIS buildings collapse seismic risk) are used as input for the MUPSA–1 models for the event of multiunit LOOP due to seismic induced shared GIS buildings collapse. Since both projects are coupled, the ultimate PSA scope and goals are same. All 04 NPPs are assumed to be at full power. The MUPSA–1 risk metrics used for the current study is tabulated in Table 7.

TABLE 7. ANNUAL FREQUANCY OF UNACCEPTABLE PERFORMANCE (SEISMIC RISK)

Buildings	**AF w.r.t. Median fragility**	**AF w.r.t. Mean fragility**	**AF w.r.t. 5%POE fragility**	**AF w.r.t. 95%POE fragility**
GIS 220 C1C2	1.39×10^{-4}	5.89×10^{-4}	2.50×10^{-3}	3.12×10^{-6}
GIS 132 C1C2	2.71×10^{-5}	1.63×10^{-4}	7.01×10^{-4}	3.34×10^{-7}
GIS 220 C3C4	7.47×10^{-6}	6.22×10^{-5}	2.67×10^{-4}	5.47×10^{-8}
GIS 132 C3C4	1.14×10^{-5}	8.50×10^{-5}	3.66×10^{-4}	9.97×10^{-8}

3.1.10. REPUBLIC OF KOREA/KHNP

Even though there is no safety goal or performance goal for multiunit risk in Korea yet, we currently evaluate multiunit CDF, which consists of concurrent CDF from two units, from three units, …, from all units, as well as Site CDF. These risk metrics could be changed if new regulatory requirements for multiunit risk are established as different metrics in near future.

Figure 31 describes the concept of multiunit CDF and site CDF, where U_n means that a core damage accident occurs at unit 'n'. $\overline{U_n}$ means that a core damage accident does not occur at unit 'n'.

$$Multiunit\ CDF = U_1U_2\overline{U_3} + U_1U_3\overline{U_2} + U_2U_3\overline{U_1} + U_1U_2U_3 \quad (6)$$

$$Site\ CDF = U_1\overline{U_2}\,\overline{U_3} + \overline{U_1}U_2\overline{U_3} + \overline{U_1}\,\overline{U_2}U_3 + Multiunit\ CDF \quad (7)$$

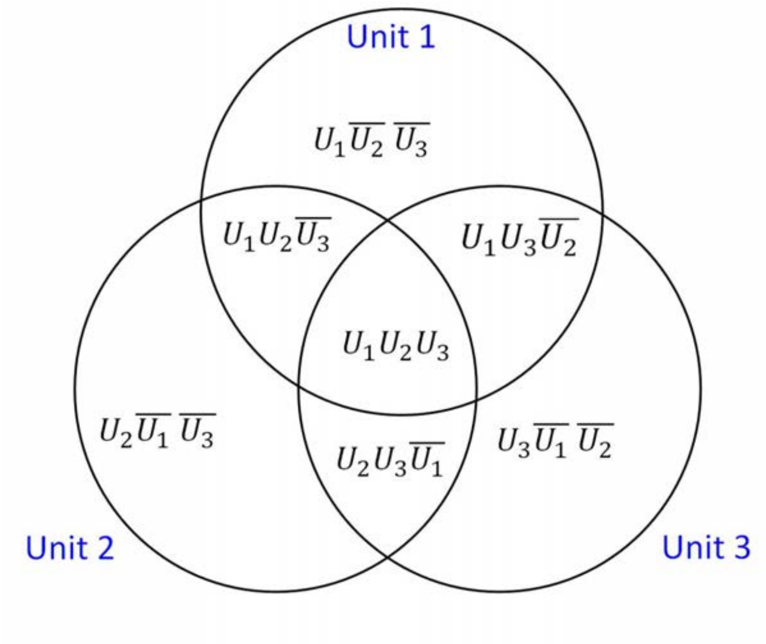


FIG. 31. CDF decomposition of three units for multiunit risk.

3.1.11. Republic of Korea/Hanyang University

The MURRG utilizes the risk metrics basically proposed by IAEA-TECDOC on Technical Approach to Probabilistic Safety Assessment for Multiple Reactor Units [20] for Level 1 PSA as shown in Table 8. National safety goals for Level 1 SRA or Level 1 MUPSA are not yet proposed. The concept of single unit initiator and common cause initiator (CCI) is implemented for initiating event analysis for SRA and site risk metrics assessed (Fig. 32) assuming multiunit risk induced by single unit initiator is negligible.

TABLE 8. RISK METRICS PROPOSED BY IAEA FOR LEVEL 1 PSA [20]

PSA	**Risk metrics (Abbreviation)**
Level 1 SUPSA	CDF
Level 1 MUPSA	SCDF Single unit CDF Multiunit CDF

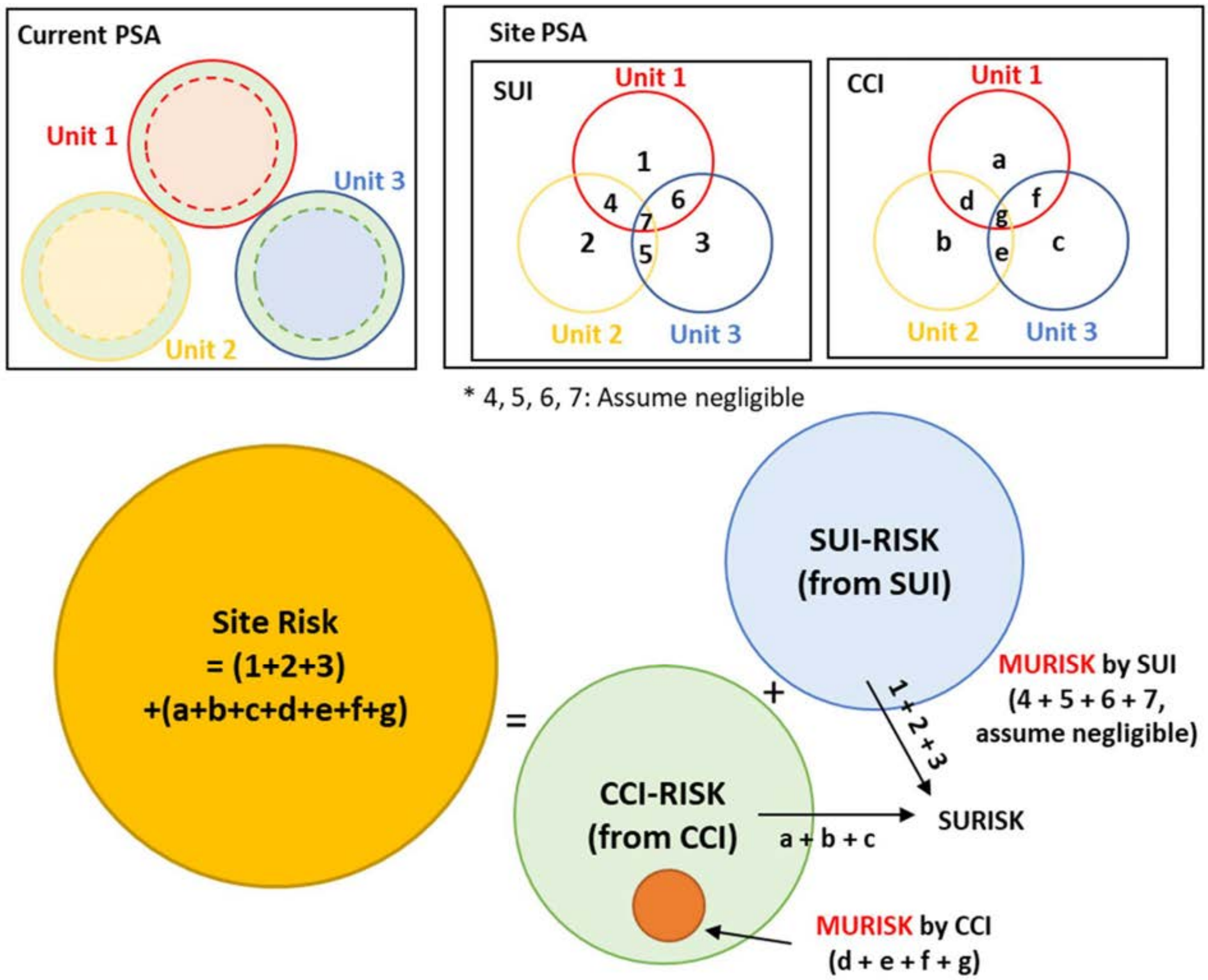


FIG. 32. Composition of site risk.

3.1.12. Romania/CNCAN

The SUPSA Level 1 for Cernavoda NPP units 1 and 2 contains six CDS, defined in Table 9. For each, the CDF were calculated for internal and external events and are shown in Fig. 33 and in Table 10.

The Level 1 PSA is performed for both operating Cernavoda NPP units; the results for all operating stages, including external (seismic) and internal events, show a CDF value of 3.3×10^{-5} events/year for unit 1 and 3×10^{-5} events/year for unit 2. These results are three times less than the internationally accepted target of 1×10^{-4} event/years for operating plants [17]. To support operational decisions with input from PSA, risk monitor applications are developed based on plant specific PSA models for on–line and off–line users.

The Cernavoda NPP risk monitor is based on the equipment out of service application as developed by EPRI. The use of equipment out of service for risk–informed decision making is reviewed and approved by CNCAN. For both Cernavoda units, the risk monitoring results show that the medium annual cumulative recorded CDF is lower than the average PSA Level 1 CDF.

TABLE 9. CDS FOR CANDU–6 CERNAVODA SINGLE UNIT

	Barriers/Challenges		CDS	Typical Events		CDS	Typical Events	
				Severe reactivity transients	Loss of all heat sinks for external LOCA events		Loss of all heat sinks for transients	Loss of all heat sinks for external LOCA
Safety and mitigating systems lost	Cooling	LOCA external	CDS0: Early (Rapid) Loss of Core Structural Integrity			CDS1: Late core disassembly		**X**
	Reactivity	Fast Reactivity Increase		X				
	Reactivity control	Failure to Shutdown		X				
	Heat Sinks	FEEDWATER						X
		SDC (shutdown cooling)						X
		EWS						X
		ECCS						X
		Moderator						X
	Cooling	LOCA external			X		X	
	ECCS	ECCS injection			X		X	
	Loops	No loop isolated			X			
	Loops	Loop available					X	
	Engineered safety features for decay heat removal	Calandria tube integrity maintained			X		X	
Engineered safety features for severe accidents lost	Engineered safety features for decay heat removal	Decay heat removed by radiation from fuel to pressure tubes	CDS2: Moderator system as a heat sink for two loops		X	CDS3: Moderator system as a heat sink for one loop	X	
		Decay heat removed to moderator inventory			X		X	
	Fuel status	Moderator cooled by RCW			X		X	
		Extensive fuel sheath failures in both loops			X			
	Fuel status	No fuel melting			X			
	Cooling	LOCA external			X			
	ECCS	ECCS injection	CDS4: LOCA with fuel failures		X	CDS5: LOCA without fuel failures	X	
	Fuel status	No fuel failure					X	
		Limited fuel failure			X		X	

The licensee performed a Level 2 PSA for both Cernavoda units 1 and 2. The fault tree analyses for containment systems demonstrate that the unavailability of 1×10^{-3} years/year, imposed by the design standards, is met. The results show a LERF below 1×10^{-6} years/year. The annual cumulative CDF as well as the containment systems performance are monitored and reported quarterly to the Romanian nuclear regulatory authority. The results confirm that the probabilistic safety goals related to core damage and radioactive release frequency are met. Based on the results from Level 1 for CDS, a grouping/binning process is performed to derive a set of EPRCs for all POSs. The categories from zero to six are grouped as LERF and are calculated for all modes of operation. For some dominant sequences a PSA Level 2 calculation is made for all POSs.

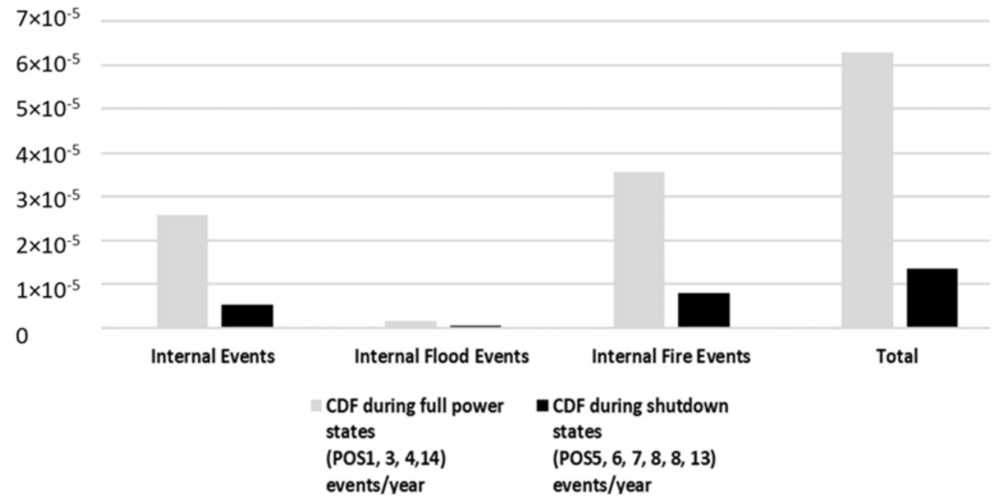


FIG. 33. Single units CDF, risk metric for PSA Level 1, CDF for CDS as per PSA for single unit.

TABLE 10. CDF FOR CANDU–6 CERNAVODA SINGLE UNIT

Contributor	**CDF during full power states (POS1, 3, 4,14) events/year**	**CDF during shutdown states (POS5, 6, 7, 8, 8, 13) events/year**
Internal Events	2.57×10^{-5}	5.27×10^{-6}
Internal Flood Events	1.53×10^{-6}	4.12×10^{-7}
Internal Fire Events	3.56×10^{-5}	8.06×10^{-6}
Total	6.28×10^{-5}	1.37×10^{-5}
Overall CDF	7.65×10^{-5}	1.50×10^{-5}

3.1.13. Russian Federation/JSC A

3.1.13.1. Plant operation and damage state

All POSs are considered in the analyses. This is important for the VVER units where on–line maintenance is not allowed and for about a month one safety state is in maintenance during shutdown. The following plant damages states (PDS) are analysed in the framework of the CRP:

— Damage of the fuel in the reactor core of one reactor. The criteria for fuel damage is 1200 ºC of the fuel cladding;
— Damage in the fuel in the SFP in one unit. The criteria for fuel damage is the level in the SFP below top of the fuel elements;
— Simultaneous damage of the fuel in the reactors of two units. Simultaneous means that core damage at the second unit occur before severe accident at the first unit is stabilized;

— Fuel damage in the reactor and fuel damage in the SFP occurring simultaneously at one unit. Simultaneous means that core damage at the SFP occurs before severe accident in the reactor is stabilized.

3.1.13.2. Risk metrics

In accordance with PDSs considered in the analysis the following risk matrixes are defined in the assessment:

Level 1 PSA

— CDF in the reactor for single unit;
— FDF in the SFP for single unit;
— CDF in two units occurring simultaneously (multiunit CDF);
— FDF in the reactor and in SFP occurring simultaneously at one unit.

Level 2 PSA

— LRF after core damage in the reactor of single unit (LRRF);
— LRF after fuel damage in the SFP for single unit (LSRF);
— LRF after core damage in two units occurring simultaneously (LRMU);
— Large release after fuel damage in the reactor and in the SFP occurring simultaneously at one unit (LSRF).

Other risk metrics are not considered.

3.1.13.3. Safety goals

There are no site related safety goals. These are the safety goals established for single unit:

— Total probability of all accident sequences leading to severe fuel damage during one calendar year is below 1.0×10^{-5};
— Total probability of accident sequences leading to large accidental release[5] during one calendar year is below 1.0×10^{-7}.

3.1.14. Tunisia/STEG

The risk metric, quantified by the risk measure, represents a risk feature or risk property such as but not limited to consequence, transition between two states, or an indicator derived from another risk measures. Risk measures are used for the representation, discussion, and interpretation of the PSA results. For the risk measures such as the CDF, conditional failure probability of a system, or basic event importance for CDF to be used, the risk model has to support the respective risk metrics. The proposed risk metric of Level 1 MUPSA is multiunit FDF. In the analysis the full power operation is considered to be the most risk

[5] Large accidental release – release requiring actions outside of the boundary of emergency planning zone.

significant mode compared to other operation and shutdown modes and therefore has been considered in this study for each unit.

3.1.15. Ukraine/Energorisk

3.1.15.1. Risk metrics and damage states

Only nominal power operation is considered in the study. It was assumed that during calendar year at least one unit at the site is in shutdown mode for maintenance. Therefore, normal power operation of 5 out of six reactor facilities and six out of six SFPs is modelled. The following plant damages states are analyzed:

— Damage of the fuel in the reactor core of one reactor. The criteria for fuel damage is 1200°C of the fuel cladding;
— Damage in the fuel in the SFP in one unit. The criteria for fuel damage is the level in the SFP below top of the fuel rods;
— Simultaneous damage of the fuel in the reactors of several units. Simultaneous means that core damage at the second and next units occur before severe accident at the first unit is stabilized;
— Fuel damage in the reactor and fuel damage in the SFP occurring simultaneously at one unit. Simultaneous means that fuel damage at the SFP occurs before severe accident in the reactor is stabilized.

The following risk metric are considered in the Level 1 PSA study:

— Single unit CDF for each ZNPP reactor;
— Multiunit CDF occurring simultaneously at two ZNPP reactors;
— Multiunit CDF occurring simultaneously at three ZNPP reactors;
— Multiunit CDF occurring simultaneously at four ZNPP reactors;
— Site CDF, i.e., Multiunit CDF occurring simultaneously at five ZNPP reactors;
— Single unit FDF for each ZNPP SFP;
— Multiunit FDF occurring simultaneously at two ZNPP SFPs;
— Multiunit FDF occurring simultaneously at three ZNPP SFPs;
— Multiunit FDF occurring simultaneously at four ZNPP SFPs;
— Multiunit FDF occurring simultaneously at five ZNPP SFPs;
— Site FDF, i.e., multiunit CDF occurring simultaneously at six ZNPP SFPs.

The following risk metric are considered in the Level 2 PSA study:

— Single unit LRF for each ZNPP reactors;
— Single unit LRF occurring simultaneously at reactor and SFP at single unit;
— Multiunit LRF occurring simultaneously at two ZNPP reactors;
— Multiunit LRF occurring simultaneously at three ZNPP reactors;
— Multiunit LRF occurring simultaneously at four ZNPP reactors;
— Multiunit LRF occurring simultaneously at five ZNPP reactors;
— Single unit LRF for each ZNPP SFP;
— Multiunit LRF occurring simultaneously at two ZNPP SFPs;

— Multiunit LRF occurring simultaneously at three ZNPP SFPs;
— Multiunit LRF occurring simultaneously at four ZNPP SFPs;
— Multiunit LRF occurring simultaneously at five ZNPP SFPs;
— Multiunit LRF occurring simultaneously at six ZNPP SFPs.

3.1.15.2. Safety goals

Only safety goals applicable for single unit in terms of CDF and LRF are stated in Ukraine, no safety goals related to multiunit or site ate established. Top level regulatory document General Provisions on Safety (2008), defines safety goals for operating and future NPPs as follows:

— CDF $\leq 1.0\times10^{-4}$ /y for operating plants and $\leq 1.0\times10^{-5}$ /y for new plants;
— LRF $\leq 1.0\times10^{-5}$ /y for operating plants and $\leq 1.0\times10^{-6}$ /y for new plants.

3.1.16. Ukraine/SSTC NRS

Taking into account dependency analysis MUPSA study for RNPP is limited with normal power operation state and the fuel in the reactor core of units 1, 2 is considered. Dependency analysis showed that there is no important inter–connection between units 3, 4 RNPP and connection of units 1, 2 with units 3,4.

3.1.16.1. Risk metrics and damage states

The analysis is performed taking into account the following limitations:

— Reactors of RNPP–1 and RNPP–2 are the objects of analysis, and fuel in SFP is not considered within this study;
— Internal initiating events, which were taken into account and modelled in existing Level 1 PSA models of RNPP–1,2 reactors, are considered as initiators of accident sequences leading to severe fuel damage in the reactor core;
— Analysis was performed for full power operation (decreased power and shutdown are not within the analysis scope).

The following risk metric are considered in the Level 1 PSA study:

— Single unit CDF for each RNPP reactor, unit 1 and 2;
— Multiunit CDF at two RNPP reactors, unit 1 and 2.

3.1.16.2. Safety goals

The SUPSA probabilistic safety goals in terms of CDF and LRF are stated in level regulatory document in Ukraine General Safety Provisions for NPP (2008). The safety goals for operating and new NPPs:

— CDF does not exceed 1×10^{-4} reactor/yr for operating plants and CDF does not exceed 1×10^{-5} reactor/yr for new plants;
— LRF does not exceed 1×10^{-5} reactor/yr for operating plants and LRF does not exceed 1×10^{-6}

reactor/yr for new plants.

The safety goals related to multiunit or site are not established in Ukraine.

3.1.17. Ukraine/Energoatom

3.1.17.1. Risk metrics

In accordance with PDSs considered in the analysis the following risk metrics are used in the assessment:

For MUPSA Level 1

— Single unit CDF;
— Multiunit CDF in two units occurring simultaneously.

For MUPSA Level 2

— LRF after core damage in the reactor of single unit;
— Multiunit large release frequency after core damage in two units occurring simultaneously.

3.1.17.2. Safety goals

Quantitative safety criteria for the units under operating are as follows:

— Severe CDF does not exceed 1.0×10^{-4} reactor per year;
— LRF does not exceed 1.0×10^{-5} per reactor per year.

There is no site related probabilistic safety criteria available in Ukraine.

3.1.18. UAE/Khalifa University

To get a more realistic estimation of CDF, the AIMS–PSA code and the PRA pilot model (KAERI) is used for the development of the Khalifa university PRA model for the evaluation of the extended SBO with EDG crosstie option. The KAERI simplified PRA pilot model is used to demonstrate the SBO event in different cases; the reference plant is OPR1000 (KEPCO, 2019). The development of the Level 1 PRA starts with identifying the initiating events that could impact the front–line systems and lead to the core damage. Before developing the event tree of SBO, the LOOP event is modelled, which could lead to the SBO event if EDGs are failed. The LOOP event occurs if generator and reactor trip are induced by the electrical issue of offsite transmission network or switchyard or generator that leads to loss of all AC powers supporting safety class 4.16 kV switchgear from the switchyard. The LOOP has an impact on the main safety functions and on the availability of vital systems. The risk metrics considered in the study are the CDF and the CDF associated with SBO event. The relevant safety goals from the United Arab Emirates Regulator FANR are in [21]; it is indicated in Article (6) Probabilistic Safety Targets – Evaluation Criteria to be:

— “The NPP should be designed, operated and maintained so as to limit its overall CDF to < 10^{-5} /yr

(mean value from the PRA1 considering internal and external events and all modes of operation);
— The NPP should be designed, operated and maintained so as to limit its overall LRF to < 10^{-6}/yr (mean value from the PRA considering internal and external events and all modes of Operation)."

3.1.19. Summary

From information provided by Member States participating in CRP the following can be summarized:

a) All Member States do have safety goals and risk metric for single unit. These goals and metrics cover both Level 1 and Level 2 PSA. The most commonly used risk metrics are CDF, LERF or LRF that generally correspond to risk metrics discussed above;
b) All Member States have no or very limited experience in developing multiunit PSA;
c) All Member States do not have safety goals and Level 1 PSA risk metric related to multiunit PSA. All Member States have no or very limited experience (Finland/VTT) in developing multiunit PSA.

In the CRP most participants:

a) Use multiunit fuel or core damage frequency (multiunit FDF, multiunit CDF or single unit CDF) as the risk metric for Level 1 PSA that are similar to risk metrics discussed in Section 3.1;
b) Consider sequences with fuel damage in at least two units to be modelled in MUPSA;
c) Consider full power operation for both units;
d) Analyze LOOP scenario as multiunit initiator.

The following participants add specific consideration in the CRP:

a) Argentina/CNEA and Russian Federation: Considers two sources of radioactive releases: fuel in the reactor and fuel in SFP. Special risk metric for fuel damage in the reactor and SFP in single unit was used by Russian Federation;
b) Canada/ COG: highlighted the need for whole site PSA for Ontario NPP as multiple units exist within a common containment structure and many important systems are shared or interconnected;
c) Argentina/CNEA: propose the IRR as a risk metric. A safety goal for IRR could be define based on Argentine acceptability criterion for licensing NPP;
d) India/BARC proposed to use SiRF as the Level–2 MUPSA risk metric;
e) Pakistan/Chasnup suggest distinguishing risk metrics single unit CDF and SCDF, where single unit CDF is defined as frequency of a reactor accident involving core damage on one and only one reactor unit and SCDF is a frequency of a reactor accident involving core damage on one or more reactor units concurrently per site calendar–year;
f) Pakistan/PAEC consider seismic hazard as the main MUPSA initiator;
g) Ukraine/Energorisk considers several metrics for 6 units at ZNPP site that are based on multiunit CDF and multiunit LRF depending on the number of units involved in the accident.

All participants involved in CRP confirm that MUPSA is a logical step in the further extension of risk assessment process and multiunit or site risk safety goal need to be established.

3.2. IDENTIFICATION AND SCREENING OF MULTIUNIT INITIATING EVENTS

This section describes initiating events for the multi unit models per participating organization.

3.2.1. Argentina/CNEA

Taking into account the scope of available PSA for CAREM25 reactor, the initiating event of LOOP for full power operational state was postulated for the present case study and analysed.

3.2.2. Canada/COG

3.2.2.1. Initiating events for reactors

There are primarily three elements of this task:

— Initiating event identification overview;
— Selection of initiating events;
— Screening analysis for internal and external hazards.

The reactor fuel is the major source of radioactivity in a generating unit in a nuclear power station. Initiating events leading to event sequences in which fuel failures can occur are, therefore, of primary interest because if fuel failures occur, there is a potential for radioactive releases. This, therefore, suggests the following definitions of initiating events for internal event PSAs for reactors at power:

a) An initiating event is a malfunction of the generating unit operating at power, that would lead to fuel failures in the absence of a reduction in reactor power. This can be due to the malfunction alone or a malfunction in a specific plant configuration;
b) An initiating event is a malfunction of the generating unit operating at power, that results in a reactor shutdown. Without successful decay heat removal after the reactor shutdown, fuel failures would occur. This can be due to the malfunction alone or a malfunction in a specific plant configurations;
c) For NPPs with more than one reactor unit, an initiating events that can affect more than one unit at the same time (e.g. LOOP), or arise in one unit and lead to an accident in another unit (e.g., large main steam line break).

By following the above steps, a complete list of potential initiating events is created. Similarly, the major malfunctions examined for reactors in a shutdown PSA are events which cause:

— A loss of normal decay heat removal;
— A loss of heat transport system (HTS) inventory or compromises to the HTS integrity;
— A loss of reactivity control;
— Failures of support systems leading to any of the above;
— Purious plant signals leading to any of the above.

3.2.2.2. Initiating events for non-reactor sources

Irradiated fuel bay: the same hazard types are considered as for the reactor PSAs. All the internal events impacting the IFBs can be grouped into the following bounding consequence categories for PSA:

— Loss of IFB Heat Sink (resulting from e.g. random IFB cooling and support system failures, human errors, reactor hazards that may impact IFB cooling system equipment operation);
— Rapid/Slow Loss of IFB Inventory (resulting from e.g. random IFB piping breaks, loss of inventory make–up to compensate for evaporative losses, damage from heavy load drops etc.);
— Impact of reactor events on IFB and vice–versa.

Used fuel dry storage (UFDS): for an accident to result in a major release of activity, a large quantity of fuel is involved and exposed to severe temperature excursions. Otherwise, the releases become quickly limited to a small amount of noble gases escaping from the free inventory of a few failed fuel elements. Thus, to release ^{137}Cs, which is the radionuclide of concern for the LRF in a PSA, the fuel would need to be melted. The fuel in the DSCs no longer generates enough heat to require active cooling. The DSCs are arranged in the dry storage buildings to ensure sufficient air flow to keep the containers cool, so the temperature would need to be raised by an external source.

The identification of events for PSA for the UFDS makes use of this condition. A plant walkdown is conducted to confirm that the list of selected hazards is appropriate. To develop the desired list of hazards that are to be considered in a PSA, a screening assessment is required to produce a list of hazards appropriate for the individual stations, from a starting list of internal and external hazards that are potentially of interest. As per the screening methodology, first a qualitative assessment is performed for the hazards' impacts on the station's safety operation. If the hazard cannot be screened out by this assessment, then a quantitative assessment is carried out. The event frequency, SCDF, LRF or conditional core damage probability (CCDP) are used as quantitative screening parameters.

3.2.3. China/INET

The INET team re–categorizes the initiating events according to the NSSS modules affected directly. Three subcategories are summarized: Type D, Type C and Type S, as shown in Table 11.

Type D initiating events refer to the events that will always affect multiple modules, e.g. LOOP or loss of ultimate heat sink. With respect to the internal events, typically we can find support system initiating events in this category. Most of the external events (e.g. seismic events, tornados) would also be classified into this type, as they usually affect all the modules at the same time.

Type C initiating events include the events that may affect multiple modules under certain circumstances. Most of the internal hazards (e.g. internal fire and internal flood) and some of the external hazards (e.g. aircraft impact) this category, as they generally have local effect on the candidate module, but may affect other modules under certain instances.

Type S initiating events cover the events whose effects are limited to the candidate module, such as primary pressure boundary break or steam generator tube rupture. The re–categorization results in Table 11 are only for internal events with the plan to be updated when internal hazards and external events are analysed.

TABLE 11. RECATEGORIZATION OF INITIATING EVENTS BY MULTIMODULE FEATURE

No.	General category	No.	Initiating event group	Subclass
1	Primary pressure boundary break	1	Small primary pressure boundary break φ<10mm	S
		2	Large PB break φ≥10mm	S
		3	Small non–isolable pressure boundary break φ<10mm	S
		4	Large non–isolable pressure boundary break φ≥10mm	S
		5	Vessel rupture	S
2	Water ingress	6	Steam generator tube rupture, small break	S
		7	Steam generator tube rupture, medium break	S
3	Transients	8	LOOP	D
		9	General transient	C
		10	Main blower failure	S
		11	Loss of component cooling water	D
		12	Loss of one train of emergency safety bus	D
		13	Loss of one train of uninterruptable power supply	S
		14	Loss of residual heat removal system	S
		15	Loss of the common part of the confinement ventilation system	D
		16	Loss of main feedwater, condensation water or feedwater pipe rupture	D
		17	Loss of vessel support structure cooling system	S

3.2.4. Finland/VTT

Given that the SUPSAs are complete, initiating event analysis can be based on the initiating events of SUPSAs. The initiating events need to be categorized into multiunit events, single unit events and partial multiunit events. All external hazards can be regarded as potential multiunit initiating events. Partial multiunit events can impact one or multiple units, e.g. depending on the causes. LOOP is a typical example of a partial multiunit event. Partial multiunit events need to be divided into multiunit events and single unit events. There is also a possibility that an accident starts in one unit by single unit initiating event and propagates later to another unit. An example of such propagation is spreading of fire. Propagating events however were not analysed in the Nordic SITRON project [7]. The analysis is very case specific and requires plant walkdowns. Multiunit initiating events can be screened based on the related CDFs in SUPSAs. For the pilot studies, the multiunit LOOP was selected as the only initiating event to limit the scope.

3.2.5. Ghana/GAEC

3.2.5.1. Multiunit initiating events classification

Initiation events and hazards are events that disturb Safety SSC, and hence affect the safe operation of NPPs. In worst cases, the occurrence of these events leads to nuclear accidents that may result in core damage and hence radiation exposure to the environment and the public. These initiating events and hazards are analysed when performing PSA at a site with one unit or multiunit NPPs. External hazards can be categorized into definite and conditional hazards (Table 12).

TABLE 12. LIST OF EXTERNAL HAZARDS [22]

Definite External Hazards	Conditional External Hazards
Earthquakes	Aircraft crash
Tsunamis	Explosions
External floods	Lightning
External fires	Fouling or clogging in intake tunnel
High wind hazards like cyclones	

TABLE 13. LIST OF INTERNAL INITIATING EVENTS [22]

Definite Internal Initiating Events	Conditional Internal Initiating Events
LOOP	Loss of emergency service water
Loss of ultimate heat sink	Loss of feed water
	Loss of DC bus
	SBO
	Turbine missile
	Loss of instrument air

The hazards that will always affect multiple units are called definite hazards and those, which only under certain circumstances affect multiple units, are called conditional hazards. Initiating events can be classified into definite and conditional internal initiating events (Table 13). Initiating events can also be classified into Internal Independent initiating events. Internal Independent initiating events have effects on only single unit NPPs and have no effects on multiunit NPPs on the site. That is the occurrence of Internal Independent initiating events in one NPP does not affect other NPPs on the site. Examples of Internal Independent initiating events include LOCAs, Transients, etc. [22].

3.2.5.2. Screening criteria

Qualitative screening criterion involves primarily the exclusion of initiating events caused by factors for which modelling assumptions are not applicable. These factors include Organizational events, precautionary actions and violation of technical specifications. Such factors may be identified by analysing events reported by a licensed NPP operator such as the Licensee Events Report submitted to a regulatory authority.

3.2.6. Hungary/NUBIKI

All initiating events considered in the PSA of NPP Paks have been subject to a screening analysis concerned with identification of areas in need of modelling in MUPSA. In summary, it was found that the following categories of initiating events need to be subject to modelling multiunit (and multi source) effects in PSA:

— Loss of power due to onsite causes;
— LOOP;
— Internal hazards included in the SUPSA: fire and internal flooding;
— All external hazards included in the SUPSA: seismic, high winds, extreme snow, ice formation, external events endangering cooling water intake from the river Danube and others;
— Any single unit initiating event that indirectly causes a transient (e.g. forced shutdown) at other units (domino effect).

3.2.7. India/AERB

Initiating events are the starting point for identifying vulnerabilities and for conducting PSAs. As for SUPSA, a comprehensive list of initiating events is initially prepared for a multiunit site. To develop an initial set of multiunit initiating events, it is important to understand the types of coupling or interconnections between the units. Due to these interconnections, initiating event at multiunit site can be concurrent or event at one unit can cascade or propagate to other units.

- Initiating event at a multiunit site can trigger an accident sequence simultaneously at multiple units (Common Cause Initiators like Grid Failure, seismic induced events, external flood etc);
- Sequence in one unit may cause an initiating event in other units at the site which are otherwise safely operating or shutdown state.

    - Event progression (prior to core damage) in one unit may affect the other unit and may initiate or change the event progression in other unit which otherwise is in a controlled state;
    - Event progression after core damage in one unit may also affect the other unit (e.g. a reactor accident that prevents accessibility to other unit key components and systems for prolonged period of time).

Based upon the above, multiunit initiating events can be classified as those:

- Restricted to single unit;
- Directly affect multiple units;
- Affect other units due to proximity;
- Cascade/propagate to other units;
- External events;
- Correlated hazards.

3.2.7.1. Approach for multiunit initiating events at Kakrapar site

To identify applicable multiunit initiating events at Kakrapar site, an engineering evaluation of units at Kakrapar was performed to explore the potential of an event affecting or initiating an accident in the other unit. The following are the design characteristics of multiunit at Kakrapar site:

- Located at Downstream of Ukai dam;
- Seismic Zone III;
- Ultimate heat sink;
- Two independent paths for decay heat removal;
- Unitized concept and physical separation;
- Diesel engine driven fire water pumps;
- Fire water system is the only common system.
- Isolated inter unit crossties for fluid systems;
- No inter unit crossties in electrical systems;
- Independent start up transformers for each unit;
- Ventilation systems are linked only at stack;

— SSE qualified 3*100% diesel generator for each unit;
— Pump house, switch yard and control room are important buildings common to both units.

Operating experience of Indian PHWRs was reviewed for occurrences of multiunit events. The Indian PHWRs operating experience for multiunit event is:

— Fire event at Narora Atomic Power Station (1993);
— Kakrapar Flood Event (1994);
— Tsunami at Madras Atomic Power Station (2004);
— LOOP at Rawatbhata Rajasthan site (2012).

Shared systems are also reassessed for additional initiating event to address any failure that may result from connection from other unit.

3.2.7.2. Initial list of multiunit initiating events at Kakrapar site

The initial list of multiunit initiating events for Kakrapara site are:

— LOOP;
— Turbine missile;
— Loss of ultimate heat sink;
— Fire events (due to proximity);
— Internal flood event (due to proximity);
— Loss of ultimate heat sink due to extreme flood event;
— Seismic Induced LOOP;
— Seismic Induced LOCA;

Loss of ultimate heat sink due to seismic event.

3.2.8. India/BARC

Engineering analysis is used for screening of multiunit initiating events based on shared systems/ resources. Some of the initiating events common to both the AHWR units considered during the seismic events are LOOP, main steam line break outside the reactor building, active process water system failure, SWS failure. In general, initiating events are categorized as follows:

— Internal events – affecting individual units (includes internal fire, flood etc.). Excludes class IV failure from grid. For spent fuel also – internal events (no shared system contribution).
— External events – affecting more than one unit/facility. Seismic, tsunami, external flood, are likely to affect all the units. Class IV grid failure too would affect all the units.
— Shared system (dependency) – affecting more than one unit/facility.

Based on the above categorization, the following four categories of initiating events have been identified for this benchmark study:

— Category 1: Initiating events affecting only individual units/ source resulting in site release
— Category 2: Initiating events which can affect both reactor units (turbine building structural failure leading to main steam line break (MSLB–ORB) during seismic event)
— Category 3: Initiating events which can affect one reactor unit and SFF (active process water system failure)
— Category 4: Initiating events which can affect both reactor units and SFF simultaneously (Seismic, LOOP, Pump house structural failure resulting in SWS failure).

3.2.9. Pakistan/PAEC

Seismic event at Chashma Site is considered as the potential external hazard amongst other external hazards (e.g. flood, high wind) considering the Chashma site specific features. There are many important seismic sources near Chashma site like Khisor, Marwat, Kalabagh and Bhittani faults etc. The maximum potential magnitude assigned to the most important structure, Khisor–Kundal is 6.8. This structure considered capable and was reassigned a pessimistic value of M=6 by PAEC experts and endorsed by IAEA Mission (1992). The 1982 Bhakkar earthquake had magnitude of 5.5 and a focal depth of 3.5 km, occurring 75 km from the site. This earthquake has not produced adverse effect on the site. The horizontal ground acceleration for the SSE is 0.25g. The OBE or SL1 earthquake was determined on the basis of seismicity during the last 100 years as OBE is determined for a return period of 100 years. As ground acceleration at CHASNUPP from the 100 years record did not exceed 0.10g, a value half of SSE=0.125g was taken for OBE.

The liquefaction analysis has already been carried out for the CHASNUPP area with a more severe ground motion. The minimum safety factor for free field case against liquefaction is 1.59 at a depth of 12 meters while the values at greater depths increase monotonically. Moreover, for a non–free field case, a minimum safety factor of 1.51 was evaluated for a soil column at depths ranging from 10.4 meters to 13 meters. Hence, in both free field and non–free field condition adequate factors of safety against liquefaction exist for the SSE. The SSE or SL2 earthquake is determined with deterministic approach covering all earthquake sources within 150 km radius around the site. The horizontal ground acceleration for the SSE is 0.25g. The OBE or SL1 earthquake is determined based on seismicity during the last 100 years as OBE is determined for a return period of 100 years. As ground acceleration at Chashma site from 100 years record did not exceed 0.10g, a value half of SSE=0.125g was taken for OBE. The tsunami hazard is not applicable at Chashma site as it is far away from Arabian Sea. A detailed flood study of Chashma site reveals that all safety related systems will remain dry in case of the worst case scenario i.e. flooding due to breaking of all upstream dams (existing and planned) and superposition/addition of probable maximum precipitation concurrently with dam breaks. The external flood is less important for Chashma site. Similarly, no high winds are observed/recorded in Chashma area and thus it is less significant. The SUPSA study shows that LOOP is the top contributing event resulting in core damage. The LOOP due to seismically induced collapse of switchyard buildings (multiunit LOOP) is selected as multiunit initiating event in this study. The SUPSA shows that LOOP is the top contributing event resulting in core damage. Seismically induced LOOP is considered as multiunit initiating unit for the present study. Switchyards buildings are shared among C1 & C2 and C3 & C4 NPPs. As a result of seismic event, the shared switchyard buildings collapse resulting in multiunit LOOP (both 220 kV and 132 kV lines are unavailable and credit of recovery is not considered). The LOOP due to seismically induced collapse of switchyard buildings (multiunit LOOP) is selected for Chashma site.

3.2.10. Republic of Korea/KHNP

After reviewing all initiating events considered in SUPSA, internal initiating events, internal flooding events, and internal fire events, which occur independently in multiunit were screened out because multiunit risk due to independent initiating events is negligible based on a previous research project performed by KAERI where the multiunit risk impacts from the case that an independent initiating event occurs in other unit within 72 hr after an initiating event firstly occurs in a unit were evaluated. Even though assuming component failures and human errors are completely dependent, the multiunit risk impacts were identified as not significant as shown in Table 14.

TABLE 14. PORTION OF MULTIUNIT CDF TO SINGLE UNIT CDF

# of Damaged Cores	Internal Event		Internal Flooding		Internal Fire	
	Comp.Dep.[6]	Comp.Ind. [7]	Comp.Dep.	Comp.Ind.	Comp.Dep.	Comp.Ind.
2	3.57%	< 0.0001%	1.02%	< 0.0001%	0.63%	< 0.0001%
3	0.10%	< 0.0001%	0.01%	< 0.0001%	0.003%	< 0.0001%
4	0.002%	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%
5	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%
6	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%	< 0.0001%

From these insights, we considered that only the initiators, which cause concurrent reactor trips in multiunit, could have an impact on multiunit risk. Therefore, a seismic event of the scope of SUPSA was only selected as a multiunit initiating event for MUPSA. In addition, we reviewed all our historical reactor trip events to identify a candidate for a multiunit initiator, which can cause reactor trips in multiunit at the same period. As a results, we identified that some external hazards such as typhoon, heavy snow, marine lives, etc. could cause concurrent reactor trips in multiunit from our operating experiences such as:

— Multiunit LOOP due to typhoon and heavy snow;
— Multiunit general transient due to typhoon, lightening, and system faults;
— Multiunit loss of circulating water due to marine lives, which causes loss of condenser vacuum

From these insights, the focus can be on seismic events and other concurrent multiunit initiating events when developing the MUPSA models.

3.2.11. Republic of Korea/Hanyang University

The SRA need to be able to consider single unit accident and the simultaneous accident occurred at multiple units. Therefore, in this project, the internal initiating events that would be probable to cause those accidents were classified into single unit initiators and CCI. The rest of the internal events were considered as SUI only. To identify CCIs, operation data and accident records of Republic of Korea were reviewed and loss of condenser vacuum, LOOP, and General Transient were chosen. Even though loss of ultimate heat sink

[6] Comp.Dep.: assume that the same components with the same function are totally dependent among units, that is, completely dependent inter-unit CCFs are considered in quantification.

[7] Comp.Ind.: assume that the same components with the same function are independent among units, that is, inter-unit CCFs are not considered in quantification.

(LOUHS) in multiple units has not occurred, it would be analyzed as it may occur by marine organism. Seismic event analysis is under development considering it as a common cause initiator that would give impact to every unit on the site.

3.2.12. Romania/CNCAN

Initiating events are grouped by the end states of the sequences generated by them, which are obtained after the results from PSA Level 1 are binned into to EPRCs, defining the LERF specific for PSA Level 2 CANDU (14 release categories). The release categories were split into three groups, and are shown in Table 15: REL1 for the EPRC 7–11; REL0 for the EPRC 12 to 14 and REL2 for EPRC 0 to 6 The initiating event for single unit (IES) PSA are derived for the whole site and sources: the reactor (two reactor core –old units refurbished and two new units); four SFB separate for each unit, one DICA, one low radioactive storage and CTRF. The possible cases, A, B, and C, differ mainly by the number of operating, aged and commissioning units. The cases are defined by the single unit characteristics (IES for reactor and SFB) as defined for operating units, aged operating units and units in commissioning.

TABLE 15. RELEASE CATEGORY GROUPS FOR EPRC DEFINITION (CD: CORE DAMAGE)

Release category	EPRC	Description
REL2 – high release Equivalent LERF from PRA Level 2 Cernavoda NPP	EPRC0	Early large releases as result of containment isolation failure and severe core damage (CDS0, CDS1)
	EPRC1	Early releases as result of severe core damage, between 0 and 6 hr and severe core damage (CDS1)
	EPRC2	Early releases as results of severe core damage, between 6 and 24 hr and severe core damage (CDS1)
	EPRC3	Late releases as results of severe core damage, between 24 and 72 hr and severe core damage (CDS1)
	EPRC4	Initial containment by–pass + EPRC1. The releases are due to severe core damage (CDS1)
	EPRC5	Initial containment by–pass + EPRC2. The releases are due to severe core damage (CDS1)
	EPRC6	Initial containment by–pass + EPRC3. The releases are due to severe core damage (CDS1)
REL1 Medium release	EPRC7	Containment by–pass and severe core damage (CDS1)
	EPRC8	CDS2 and CDS3 and containment isolation failure
	EPRC9	CDS2 and CDS3 and failure of containment heat sinks
	EPRC10	CDS4 and containment isolation failure
REL0 Low release	EPRC11	CDS4 and failure of containment heat sinks
	EPRC12	CDS5 and containment isolation failure
	EPRC13	CDS5 and failure of containment heat sinks
	EPRC14	Containment by–pas + MSSV closed

The release category considered is represented by the column LRF. For the multiunit situation cases to be considered for the model include multiunit initiating event for reactor, SFB and other sources and the units grouped in operating, aged or in commissioning. Therefore, MUPSA includes the development of generated scenarios for any single unit in case of multiunit initiating event. Case A was chosen as a conservative case for this benchmark.

The initiating event list considered the new SAMG results, latest safety analyses results, as well as operating experience. The initiating event review was performed in parallel with a systematic emergency planning review. The latest international requirements related to DEC A & B and small immediate release and COG results/approaches were considered. Results of an internal company project for the review of the operator model, leading to reconsideration of the HRA tasks in PSA, as well as operator review was also included in the updated lists of initiating events.

The new HRA reviews are considering the refinement of the operator actions, identification of the critical ones and the time windows for reaction, as represented in Table 16. As the operator action is a PSA task under more scrutiny after Fukushima Daiichi NPP accident, this HRA review is an important step in the evaluation of the recovery actions in the dominant sequences of the PSA and it is subject for further sensitivity analysis in this project, too.

TABLE 16. HRA REVIEW – MAIN ASPECTS

Difficulty	**Characteristics of Execution Groups**	**Actions in this Group**
Low	Time window several hours, simple local actions OR time window 20–40 min, simple push of few buttons in SCA/MCR	Unblocking of containment filtered vent in the long term OR ECCS injection into PHT system (limited flow or not cooling)
Medium	Time window 30–60 min, alignment of equipment in simple configuration	Start dousing re–circulation
High	Time window 30–60 min, alignment of equipment in complex configuration	To be credited as recovery actions and to be included in the SAMG
Difficulty	**Characteristics of Diagnosis/Decision Groups**	**Actions in this Group**
Low	Time window several hours, clear and written guidance	Unblocking of containment filtered vent in the long term
Medium	Time window 20–40 min, clear and written guidance	Emergency staff follows and implement relevant SAMGs
High	Time window only few minutes or no clear written guidance	To be credited as recovery actions and to be included in the SAMG

Difficulty (Diagnosis/Execution)	**Median Error Rate**	**Factor**	**Mean Error Rate**
Low	10^{-3}	5	1.60×10^{-3}
Medium	10^{-2}	5	1.60×10^{-2}
High	10^{-1}	5	1.60×10^{-1}

The IES model is defined for the case A. The list of IES is with a sample for one unit and is presented in Table 17; however, it is developed for four units and other sources in its full length.

TABLE 17. EXTRACTS OF CASE A – SINGLE UNIT MODEL – SAMPLE OF ONE UNIT

Initiating Event SUPSA	**Frequency**	**Description of the SUPSA Initiating Event (IES)**
IES_U1_MSLBM	2.26×10^{-5}	High energy line break main steam above MCR
IES_U1_LESC	4.15×10^{-2}	Loss of end shield cooling flow
IES_U1_LESCB	4.13×10^{-4}	Loss of end shield cooling inventory
IES_U1_ICLOCA	5.35×10^{-4}	In core LOCA
IES_U1_VSLOCA	3.16×10^{-3}	Very small LOCA
IES_U1_TLCLI	9.69×10^{-5}	Total loss of class I power supply
IES_U1_TLCLIV	3.10×10^{-2}	Total LOCLIV power supply
IES_U1_GT	8.66×10^{-1}	General transient
IES_U1_LKI	3.71×10^{-2}	Leak inside containment
IES_U1_LOIA	6.36×10^{-3}	Loss of IAS –decrease of IAS pressure below 862 KPa (g) at main distribution header
IES_U1_TLCLIII	3.14×10^{-4}	Loss of both 6 KV Class III buses
IES_U1_FIRB005BC	1.68×10^{-4}	Cable fire of control cable trains ODD/EVEN (B) and EVEN (C) in room R–005
IES_U1_FIS326EVAC	1.68×10^{-4}	Fire damage to any control panel causes MCR evacuation
IES_U1_FIS013	2.67×10^{-4}	Fire in the cables access tunnel S013
IES_U1_SDDFIS327ELPLB	4.37×10^{-4}	Fire of one cabinet (13 of them) in room S327 (DG2&4 and CLIV lost)
IES_U1_SDDFIRB107B	4.34×10^{-4}	Fire in the reactor building room R–107
IES_U1_SEISM_Z2	9.88×10^{-4}	Seismic (ground acceleration 0.2–0.3 g)
IES_U1_SEISM_Z3	1.46×10^{-4}	Seismic (ground acceleration 0.3–0.4 g)
IES_U1_SEISM_Z4	4.08×10^{-5}	Seismic (ground acceleration 0.2–0.3 g)
IES_U1_PLC	7.17×10^{-3}	HTS leak into annulus gas system (PT leak)
IES_U1_PHPT	1.76×10^{-1}	Partial loss of PHTS pumped flow
IES_U1_TLCLII	1.85×10^{-4}	Total loss of class II power supply
IES_U1_PHTSLP	7.32×10^{-3}	PHTS low pressure with no pressure control
IES_U1_LDI	4.29×10^{-2}	Loss of deaerator inventory
IES_U1_EFF	8.80×10^{-4}	End fitting failure
IES_U1_SGTR	2.03×10^{-3}	Single steam generator tube rupture (outside containment)
IES_U1_SLOCA	6.50×10^{-4}	Small LOCA

The list of initiating events for MUPSA is presented in detail in Table 18 for case A.

The list considers the fact that units 1 and 2 are aged and the units 3 and 4 are new. It is also considered that the site type of fires generated by aircraft crash depend on the critical impact route and the reactor disposal on site.

The correlation factors for the plant ageing are presented in Section 4.4.8.

TABLE 1. LIST OF MULTIUNIT INITIATING EVENT FOR CASE A

Initiating Event MUPSA (IEM)	Frequency of IEM	Initiating Event MUPSA (IEM)	Frequency of IEM
IEM1_U1_GT_1	8.66×10^{-2}	IEM11_U1_S3	1.46×10^{-5}
IEM1_U2_GT_1	8.66×10^{-2}	IEM11_U2_S3	1.46×10^{-5}
IEM2_U1_GT_2	8.66×10^{-3}	IEM11_U3_S3	1.46×10^{-5}
IEM2_U2_GT_2	8.66×10^{-3}	IEM11_U4_S3	1.46×10^{-5}
IEM2_U3_GT_2	8.66×10^{-3}	IEM12_U1_S4	4.80×10^{-6}
IEM3_U1_GT_3	8.66×10^{-4}	IEM12_U2_S4	4.80×10^{-6}
IEM3_U2_TLCLIV	8.66×10^{-4}	IEM12_U3_S4	4.80×10^{-6}
IEM3_U3_TLCLIV	8.66×10^{-4}	IEM12_U4_S4	4.80×10^{-6}
IEM3_U4_TLCLIV	8.66×10^{-4}	IEM17_U1_GT	1.00×10^{-7}
IEM6_U1_FIRE_1	1.00×10^{-5}	IEM17_U2_GT	1.00×10^{-7}
IEM6_U2_FIRE_1	1.00×10^{-5}	IEM17_U1_GT	1.00×10^{-7}
IEM6_U3_FIRE_1	1.00×10^{-5}	IEM17_U4_GT	1.00×10^{-7}
IEM6_U4_FIRE_1	1.00×10^{-5}	IEM14_U1_AC1	1.00×10^{-6}
IEM7_U1_FIRE_2	1.00×10^{-6}	IEM14_U2_AC1	1.00×10^{-6}
IEM7_U2_FIRE_2	1.00×10^{-6}	IEM14_U3_AC1	1.00×10^{-6}
IEM7_U3_FIRE_2	1.00×10^{-6}	IEM14_U4_AC1	1.00×10^{-6}
IEM7_U4_FIRE_2	1.00×10^{-6}	IEM14_U1_AC1	1.00×10^{-6}
IEM8_U1_FIRE_3	1.00×10^{-6}	IEM14_U2_AC1	1.00×10^{-6}
IEM8_U2_FIRE_3	1.00×10^{-6}	IEM14_U3_AC1	1.00×10^{-6}
IEM8_U3_FIRE_3	1.00×10^{-6}	IEM14_U4_AC1	1.00×10^{-6}
IEM8_U4_FIRE_3	1.00×10^{-6}	IEM15_U1_AC2	1.00×10^{-7}
IEM4_U1_LOCLIV_1	3.01×10^{-3}	IEM15_U2_AC2	1.00×10^{-7}
IEM4_U2_LOCLIV_1	3.01×10^{-3}	IEM15_U3_AC2	1.00×10^{-7}
IEM4_U3_LOCLIV_1	3.01×10^{-3}	IEM15_U4_AC2	1.00×10^{-7}
IEM4_U4_LOCLIV_1	3.01×10^{-3}	IEM15_U1_AC2	1.00×10^{-7}
IEM5_U1_LOCLIV_2	3.01×10^{-3}	IEM15_U2_AC2	1.00×10^{-7}
IEM5_U2_LOCLIV_2	3.01×10^{-3}	IEM15_U3_AC2	1.00×10^{-7}
IEM5_U3_LOCLIV_2	3.01×10^{-3}	IEM15_U4_AC2	1.00×10^{-7}
IEM5_U4_LOCLIV_2	3.01×10^{-3}	IEM9_U1_GT	1.00×10^{-2}
IEM10_U1_S2	3.88×10^{-5}	IEM9_U2_GT	1.00×10^{-2}
IEM10_U2_S2	3.88×10^{-5}	IEM9_U3_GT	1.00×10^{-2}
IEM10_U3_S2	3.88×10^{-5}	IEM9_U4_GT	1.00×10^{-2}
IEM10_U4_S2	3.88×10^{-5}	IEM13_U1_EW	4.80×10^{-6}
IEM11_U1_S3	1.46×10^{-5}	IEM13_U2_EW	4.80×10^{-6}
IEM11_U1_S3	1.46×10^{-5}	IEM13_U3_EW	4.80×10^{-6}
IEM11_U1_S3	1.46×10^{-5}	IEM13_U4_EW	4.80×10^{-6}
IEM11_U1_S3	1.46×10^{-5}		

3.2.13. Russian Federation/JSC A

The main assumption in the development of MUPSA is that simultaneous occurrence of initiating events is only credible due to failures/damage to elements of common systems or external hazards. The list of initiating events that can affect several units of Balakovo NPP is presented in Table 19. Independent occurrence of initiating events in different units is believed to be extremely unlikely (has frequency in the range of 10^{-7}–10^{-12}). This can be illustrated by considering two independent initiating events with frequency of 0.1/yr each, with first to occur at the first unit resulting in severe accident with core damage.

TABLE 19. INITIATING EVENTS CREDIBLE FOR MULTIUNIT PSA FOR BALAKOVO NPP

Initiating event	**Description**	**Comment**
IEM– multiunit LOOP	Loss of of–site power of two or more units	Causes of the initiating event – Typical LOOP causes; – Damage to onsiteswitchyard (internal hazards); – LOOP due to external hazards (including seismic).
IEM– multiunit ASSWS	Administrative shutdown of several units due to loos of one service water train	Initiating event caused by failures in common part for all units of one or more service water trains (e.g. plugging of through–out pipe or leakage in these pipes). This initiating event can result from internal and external reasons (e.g. seismic).
IEM– multiunit AS– CDWS	Administrative shutdown of two units due to loss of demineralized water	This initiating event can occur only if all demineralized water tanks will be unavailable. This again can happen due to seismic hazards or fire in the demineralized water tanks area. This initiating event is extremely conservative, but still require attention.
multiunit ASSWS/2 SWS	Combination of two events: loss of two SWS trains at unit 1 and administrative shutdown of unit 2 due to one safety train failure.	This initiating event caused by damage of diesel generator building for example due to internal fire\flood or external hazard (including deflagration of the fuel in fuel storage building.
Multiunit 2SWS	Loss of two SWS trains	Initiating event caused by failures in two service water trains in the parts common for several units (can be the consequence of external hazard).
multiunit 3SWS	Loss of three SWS trains	Initiating event caused by failures in all service water trains in all units (can be the consequence of external hazard that affect all SWS trains – external flooding or extreme temperature).
Multiunit LNHR	Loss of normal heat removal at several units	This initiating event caused by the reasons that made cooling pond ineffective: – Extreme environmental temperature potentially leading to high temperature or freezing of water in the cooing pond; – Decrease of the quality of water in cooling pond (e.g. water grass, etc.) leading to plugging of filters of circulation water.
LOOP– induced	Loss of of–site power induced by the disconnection of one operating unit from the grid	This specific initiating event comprises from two events occurred on different units: – One unit experienced any initiating event that has led to reactor scram; – Reactor scram at this unit has led to disturbance in the external grid and loss of external grid for all other units.

Typical CCDP is below 1×10^{-4}. If the second unit experiences another event with the same frequency during the severe accident, the frequency of the initiating event occurring at another unit after core damage at the first unit will be less than $0.1 \times 1\times10^{-4} \times 1\times10^{-2} = 1\times10^{-7}$ 1/y. Assuming very high level of dependency of core damage at the second unit given core damage at the first unit (e.g. 0.1) the frequency of core damage at both units will be below 1×10^{-7} 1/y, which is negligible compared with the frequency of core damage at any one single unit. If duration of the severe accident at the first unit is limited to one month, during which the dependency between two units can be reasonably postulated, this frequency will be even lower (reduced by factor of 12). The only exception is LOOP induced by reactor scram at one of operation units. Sudden disconnection from the grid of one 1000 MW unit may cause disturbance in the external grid that will lead to consequential failure of the grid.

3.2.14. Tunisia/STEG

3.2.14.1. General considerations for the screening approach

The screening for an extended PSA is based on the following assumptions:

- Risk is (or will be) described by L1 and L2 PSA;
- Risk measures for reporting PSA results for the unit and the site (if applicable) may differ; although it depends on the PSA application, it needs to include the following:
  - CDF/FDF as the main L1 PSA results;
  - As a minimum for L2 PSA, the LRF and the early release frequency measures;
  - Preferably for the L2 PSA, the frequencies of an appropriate number of release categories to obtain a meaningful calculated risk profile.

A practical approach is to progressively introduce into the PSA model the risk sources and to select relevant internal initiating events and hazard scenarios, as follows:

a) Start with screening the internal initiating events and with the PSA model development;
b) Continue with internal hazard scenarios and their integration into the PSA model;
c) Extend the model taking into account the external hazard scenarios;
d) Complement the model with combinations of hazards and correlated hazards;
e) Complete by extension to multiunit and multi sources.

This approach assumes that a PSA model for a specific hazard scenario will benefit from the available internal events PSA. Each hazard scenario represents an initiator for an initiating event that is directly challenging some of the safety functions. For all operating states and all relevant sources on the reactor units site it is expected:

a) To identify possible initiating events, hazard scenarios, and combinations thereof;
b) To analyse the plant response and suitably group the initiating events or hazard scenarios into a representative group;
c) That for each representative group, analysis consists of:
   - Qualitative plant response;
   - Quantitative assessment of the likelihood of the scenario, and its consequences for NPP.
d) To define a set of initiating events and hazard scenarios for an extended PSA analysis.

Screening of the events is needed not just because in practice it is impossible to analyse all potential sequences after each and every initiating event, but to also efficiently make the use of available resources. Accordingly, the screening assessment is based on fairly simple analysis, referring to existing studies, often qualitative and based on the engineering judgement. Realistic assumptions can be used as well as bounding assessments. There are two different objectives:

a) Screening can justify that several initiating events may be omitted from the analyses, or that they may be binned into specific groups and therefore are not analysed individually. This approach is relevant if it is necessary to demonstrate that the plant under consideration complies with a certain quantitative objective;
b) Screening can identify the most relevant initiating events supporting the most efficient use of available resources.

3.2.14.2. General considerations for initiating events selection

The Fukushima Daiichi NPP accident initiated growing efforts in many countries to assess the MUPSA relevant issues, one of which is regarding the many combinations of accident sequences. Initiating event identification and selection (screening in or out) for an extended PSA are based on the PSA objectives helping to define:

- Aspects of risk of relevance for PSA model to provide results;
- Risk measures of relevance to interpreting the PSA results;
- Values of risk criteria relevant to risk measures to be compared to the extended PSA results;
- Suitable scope, level of details, and level of conservatism for the PSA;

These objectives may differ for a NPP during design or operation phases.

3.2.14.3. Selection of initiating events and hazards in MUPSA

In order to select a set of initiating events and hazards (internal or external) to be considered in a PSA, progressive and iterative steps are often taken to include some of the following:

STEP–1: Selection of initiating events for the internal events PSA (one NPP, all reactor states)

- Development of a comprehensive list of initiating events that can challenge the NPP safety functions;
- For each POS, the NPP response analysis and grouping of initiating events that have similar impact on the NPP;
- Estimating the occurrence frequency of the grouped initiating events;
- Bounding probabilistic analysis to select the initiating events to be considered in detail in the internal event PSA (using qualitative or quantitative screening criteria);
- Provide a list of internal events that can be justified as internal events for the PSA.

STEP–2: Selection of hazards scenarios for internal / external hazards PSAs (with no correlations between hazards, limited to one NPP, all reactor states)

- Development of a comprehensive list of internal / externals hazards which can challenge some NPP safety functions and lead to hazards scenarios;
- For each POS, NPP response analysis and grouping of hazards scenarios that have similar impact on the NPP;
- Estimating the occurrence frequency of the grouped hazards scenarios;

— Bounding probabilistic analysis to select the hazards scenarios to be considered in detail in each hazard PSA (using qualitative or quantitative screening criteria);
— Provide a list of hazard scenarios can be justified for each hazard PSA.

STEP–3: Selection of combinations of hazards (with all correlations between internal / external hazards / initiating event, limited to one NPP, all reactor states)

— Identification of possible hazard scenarios / internal events combinations;
— For each POS, NPP response analysis after grouping of combinations that have similar impact on the NPP;
— Bounding probabilistic analysis to select the combinations to be considered in the extended PSA (using qualitative or quantitative screening criteria).

STEP–4: Selection of combinations of hazards for a site extended PSA (with all correlations between internal / external hazards / initiating event, all NPPs on a site, all reactor states)

Based on the previously described approach, around 40 initiating events are generated and then grouped into 10 groups:

1. Loss of flow accident/failure of the primary cooling system pumps;
2. Partial loss of flow accident/failure of one primary cooling system pump;
3. LOOP;
4. Loss of secondary cooling;
5. Reactivity insertion accident;
6. Small LOCA/LOCA in primary cooling system;
7. Core bypass due to primary cooling system pipe rupture inside the pool or spurious opening;
8. Large LOCA due to multiple steam tube ruptures;
9. General transients;
10. Flow blockage.

3.2.15. Ukraine/Energorisk

All initiating events due to unit specific independent causes (loss of different safety–significant SSC, internal fires and floods) has been screened out. The following categories of initiating event are considered as potential contributors to core (or fuel) damage simultaneously at several units:

— Seismic;
— LOOP;
— Administrative shutdown or transients leading to reactor scram of non–affected units after severe accident at one unit;
— Strong tornado, other external hazards leading to loss of essential service water at several units.

For methodology development purposes only one initiating events, loss of ESWS was modelled.

3.2.16. Ukraine/ SSTC NRS

The following aspects have been accepted as criteria for the selection of internal initiating event to assess mutual impact of RNPP 1 and RNPP 2 interconnections and the calculation of multiunit CDF:

- Initiating event on one of the units affects the existing interconnections of the units;
- Occurrence of the initiating event leads to the possibility of supplying the cooling medium, working medium or power supply to the systems of RNPP–1 and RNPP–2, which are used in the reactor transfer to a safe state;
- Initiating event on one of the power units due to simultaneous occurrence of additional failures of components (equipment, technical means) of adjacent unit leads to the initialization of initiating event at an adjacent unit1).

To identify an internal initiating event according to selection criteria, existing interconnections between RNPP 1 and RNPP 2 were analysed. Besides, the results of identification and grouping of initiating events at reactor full power operation were under analysis. As a result, initiating event T8, loss of ESWS was selected for further assessment, as it complies with the accepted selection criteria.

The initiating event T8 is the result of a failure of three channels of the ESWS. Failure of one ESWS channel occurs in the absence of water supply from three out of four pumps of the channel2), namely:

- Failure of first ESWS channel of units 1 and 2 occurs in the absence of water supply from three out of four pumps: 1NTO–1, 1NTO–2 (unit 1); 2NTO–1, 2NTO–2 (unit 2);
- Failure of second ESWS channel of units 1 and 2 occurs in the absence of water supply from three out of four pumps: 1NTO–3, 1NTO–4 (unit 1); 2NTO–3, 2NTO–4 (unit 2);
- Failure of third ESWS channel of units 1 and 2 occurs in the absence of water supply from three out of four pumps: 1NTO–5, 1NTO–6 (unit 1); 2NTO–5, 2NTO–6 (unit 2).

Therefore, initiating event T8 is an inter–unit event, which leads to the loss of ESWS at both power units. Frequency of initiating event was calculated using fault tree method for the case of failure of three ESWS channels considering service water pumps of adjacent units, including with the account of CCFs.

3.2.17. UAE/Khalifa University

To address the multiunit factor of the initiating event, the initiating events of LOOP are divided into two initiating events as follows:

1. Single unit LOOP – single unit LOOP (in case the LOOP only occurred in unit 1);
2. Multiunit LOOP – multiunit LOOP (in case the LOOP occurred in unit 1 and other units).

The single unit LOOP and multiunit LOOP data is incorporated into the PRA model to reflect on the impact of multiunit LOOP on the shared AC power either the AAC diesel generator or EDG crosstie. The single unit SBO and multiunit SBO events are the events that transferred from single unit LOOP and multiunit LOOP respectively with the unavailability of the impacted unit's EDGs.

The multiunit initiating events of multiunit LOOP and multiunit SBO have been addressed in the event sequences where concept of sharing and occupancy factors are applied between units.

Figure 34 shows the event sequences of the single unit SBO for unit 1 of two units' site, while Fig. 35 shows the event sequences of multiunit SBO for unit 1 at the multiunit site.

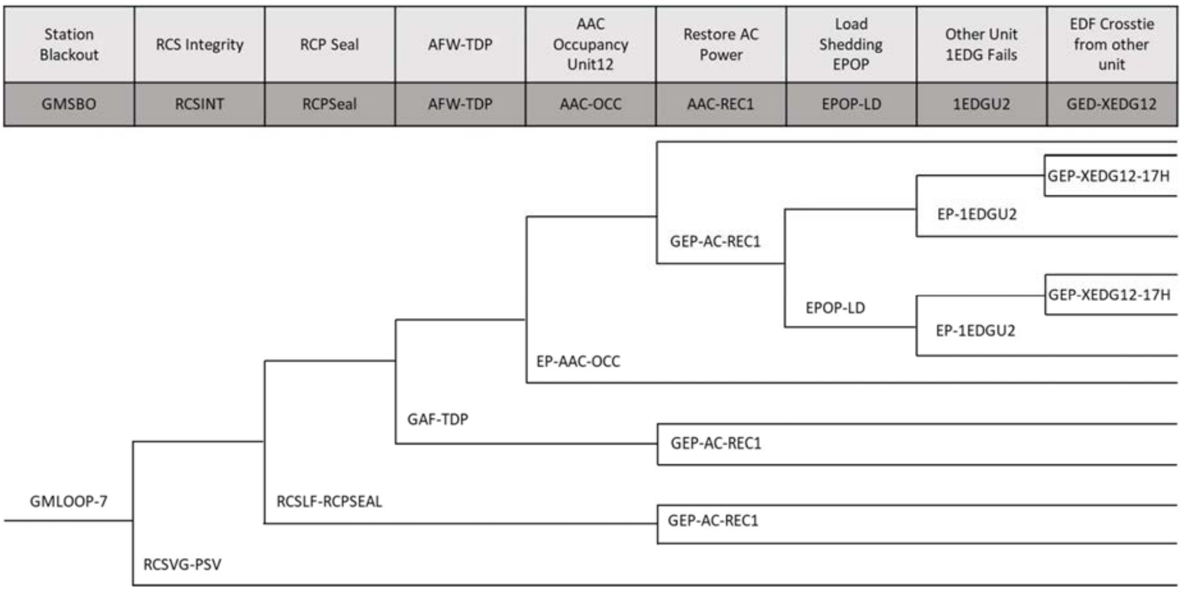


FIG. 34. Event sequences of the single unit SBO event for unit 1 of two units' site.

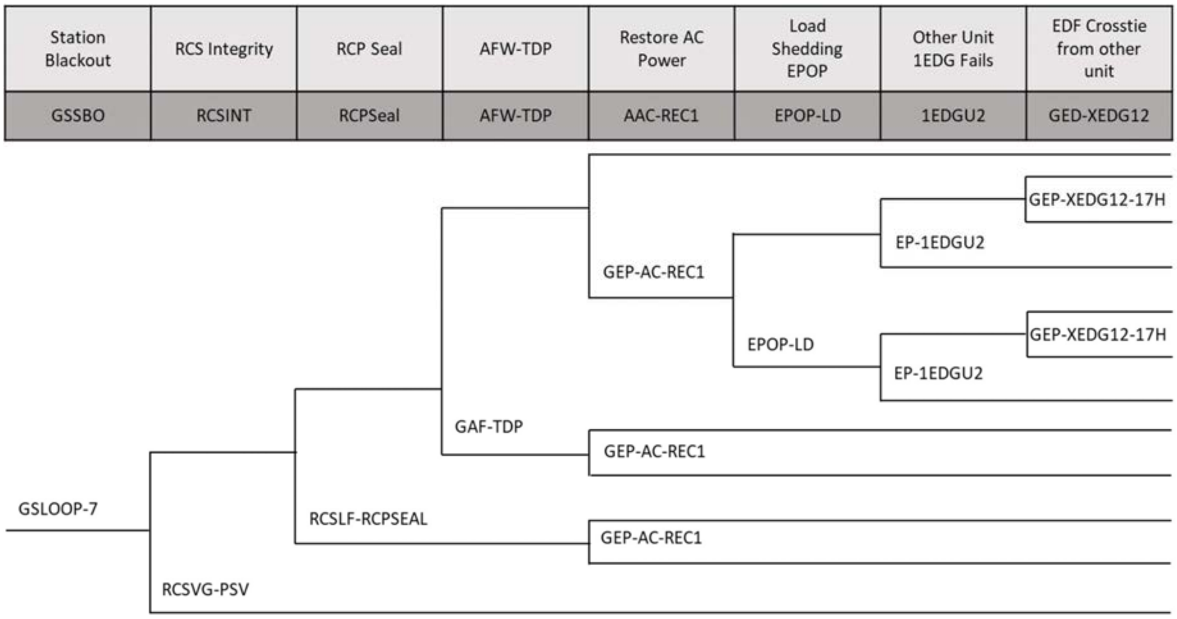


FIG. 35. Event sequences of the multiunit SBO event for unit 1 of two units' site.

3.3. OVERALL MODELLING APPROACH

This section describes modelling approaches adopted by participating organizations.

3.3.1. Argentina/CNEA

The MUPSA study is developed following quality planning and procedures applied for SUPSA, defined in Nuclear Safety Department – CNEA. The general procedure is described in the next steps.

3.3.1.1. Step 1 — definition of the objective of the study

To develop Level 1 MUPSA event trees derived from LOOP as initiating event that can affect two units of CAREM25–like SMR and SFP.

3.3.1.2. Step 2 — definition of scope, hypothesis and general considerations

Regarding the scope for Level 1 MUPSA, it has been only considered the fundamental safety functions of control of reactivity and cooling of reactor and SFP. The considered radioactive sources are the cores of both units and fuels in SFP.

The mission time has been defined as 48 hr. The grace period for each unit CAREM25–like SMR is 36 hr. After that time, active systems are required to achieve the Final Safety State or DEC systems to extend the Safety State. As consequence, the extension of mission time to 48 hr allows identifying sequences that end in core damage after the grace period. Considering the objective of this research, the event trees headers are modelled as basic events or by simplified fault trees. CCFs are evaluated qualitatively. In the framework of HRA, some human actions are analysed to study methodological issues, considering CREAM method. These results are not included in the quantitative model; an uncertainty analysis was not performed.

3.3.1.3. Step 3 — event tree development

Definition of initiating event control strategy: in the first place, the control strategy of the initiating event is defined. For CAREM25–like SMR, risk reduction systems are considered in DiD Level 2 to control anticipated operational occurrences. DiD Level 3 of has as objective the control of postulated single initiating event and postulated multi failure events, to prevent severe accident conditions. Then, DiD Level 3 is divided in sub–level 3A and 3B. Moreover, two stages are considered. Stage 1 objective is to achieve and maintain the Safe State by means of passive safety systems. Stage 2 has as objective to achieve and maintain the Final Safe State (Sub–level 3A) or to extend the Safe State (Sub–level 3B). For this case study, the following strategy for units 1 and 2 has been implemented:

- First, DiD Level 2 systems would be required;
- In case of failure of those systems, passive systems are considered for DiD Level 3A, Stage 1;
- In case of failure of Level 3A systems, Level 3B systems will be required;
- Once safe state is achieved during Stage 1, active systems will be required for Stage 2. For this stage, Level 3B DiD, (DEC B) systems are considered, in case of active systems unavailability.

Dependencies analyses: Physical and functional dependencies are one of the main elements evaluated in MUPSA. The physical dependencies in each unit systems were modelled in simplified fault trees linked to the headers. Inter–unit dependencies were mostly identified in Stage 2. The DEC systems that are required for Stage 2 in DiD Level 3B are autonomous, implying that they are not affected by LOOP. The EWSS, which is autonomous, is required to supply water to DEC systems and is shared among both units and the SFP. As design criterion, the capacity of EWSS allows to provide water only to one unit and SFP at the same time. From an operational point of view, it is needed to define a strategy about how to provide water to DEC systems. This is reflected when event tree is constructed. The EDG system is used as support to systems that perform safety functions in DiD Level 2 and Stage 2, DiD Level 3A. The EDG system is shared by the three radioactive sources, but it has the capability of supplying to both units and SFP at the same time.

Event tree development: Considering the preview steps, the event tree derived from LOOP has been developed, integrating in the same tree both units and the SFP respective headers. The developed sequences are only those that imply fuel damage in at least two sources. Each sequence is analysed defining the final state of the considered radioactive sources, considering the stage in which fuel damage occurs. To simplify huge event trees, functional headers have been defined.

3.3.1.4. Step 4 — deterministic analysis

It is considered that the deterministic analysis used as support for Level 1 PSA, for a single unit, can be used for Level 1 MUPSA. In the framework of Level 1 PSA there is no interaction between unit 1 and unit 2 and with the SFP. In that sense, the final unit state definition is based on deterministic analysis made for CAREM25 Preliminary Safety Assessment. Moreover, other deterministic analyses have been made as support for event sequences development after grace period.

3.3.1.5. Step 5 — event tree quantification

For the event tree quantification, screened values were used as frequency of initiating event and unavailability of headers or basic events were used in simplified fault trees. The considered frequency of multiunit LOOP was 2.5×10^{-2} yr^{-1}.

3.3.2. Canada/COG

It is central to highlight that, given the extensive sharing of safety–related systems, including shared containment, the Canadian utilities' per unit based PSAs have always addressed multiunit effects and hence, MUPSA. For each type of hazard (internal events, fire, flood, etc.), the detailed PSAs are used to estimate SCDF and LRF on a per hazard, per unit basis. One of the units is chosen as the reference one and the risk metrics are estimated for that unit. As there are few design differences between the units in a station, the result of the risk metrics for the reference unit are representative of the risk metrics for the other units. The initiating events include those that:

- Occur on the reference unit and affect only the reference unit;
- Occur on an adjacent unit and affect the reference unit as well as the adjacent unit; and
- Affect all units simultaneously.

For each initiating event, and event tree is developed, and a fault tree is established for each of the safety functions defined in the event tree. While the focus of the PSA is the reference unit, the event trees and the fault trees account for multiunit dependencies:

— Common initiating event can affect the reliability of the safety functions on all units and affect the reliability of inter–unit safety functions; failures associated with the common service water intake can cause an initiating event and thus affect the reliability of the unitized, shared and inter–unit emergency service water supplies;
— PSA takes into account the number of units participating in the sequence; extra emergency service water pumps may be required to operate following an initiating event affecting multiple units compared to an initiating event affecting a single unit;
— Range of post–operator actions required to be performed in a sequence affecting multiple units may be greater than the range of actions required to be performed in a single unit sequence, which may increase the probability of failure to perform the required actions either as a result of increased complexity or increased time pressure.

Through systematic review of the cut sets, the SCDF results are split into single, dual or four units' accidents.

The regulator requires the assessment of other states where the reactor is expected to operate for extended periods of time. As part of the Canadian multiunit whole site PSA work, such states have been comprehensively assessed. Based on this work, the overall conclusion is that the other reactor operating states can be dispositioned or covered by the full power and shutdown PSAs and hence, the risk associated with those other operating states is low. With respect to other sources of radioactivity, comprehensive work was performed to assess the associated risk. Based on this work and plant walkdowns, various non–reactor sources of radioactivity were screened out as being insignificant risk sources, except for two sources identified for further study: the IFBs and the UFDS facility.

For the IFB since there is no additional containment, fuel uncovery is assumed to lead directly to LRF. The IFB risk assessment involved both deterministic and probabilistic considerations. Analysis of losses of heat sink at the IFBs indicated that the time to reach boiling was greater than 72 hr. For any hazards that cannot be screened out, a bounding assessment was performed for the impact on whole site risk for the following consequences:

— Hazard that may lead to loss of the IFB heat sink, loss of inventory and if not mitigated to the IFB fuel uncovery;
— Hazard that may lead to high radiation fields, which may produce conditions that challenge fuel cooling in the reactor units, due to for example habitability issues for the operator to monitor reactor operation.

Since there is no additional containment for the DSCs, a direct containment bypass or failure is always assumed in case of failure of a UFDS. At the Canadian NPPs, once the used fuel has resided in the bays for a minimum of ten years, the residual decay heat is sufficiently low to allow this fuel to be moved to dry storage.

3.3.3. China/INET

Overall modelling framework of HTR–PM MUPSA is described in Fig. 36. The overall modelling approach is event tree and fault tree linking. Shared SSCs are explicitly modelled in the event trees and fault trees. Both inter–unit and intra–unit CCFs are considered.

The MUPSA for HTR–PM does not apply core damage state as the output of event trees. Release categories are concluded at the end of event tree branches. Each release category is then analysed to determine its source term and dose estimates.

Event tree and fault tree models are developed by using RiskSpectrum PSA software. Dose consequence assessment is done by the software Advanced Radioactive Consequence Assessment Toolkit (ARCAT) which is developed by INET for the purpose of multiple source releases. As mentioned above, HTR–PM MUPSA adopts event tree and fault tree linking approach to do the event sequence analysis. The main concern is MUPSA event trees may be very complex and very big. During the CRP, our modelling approach evolves gradually.

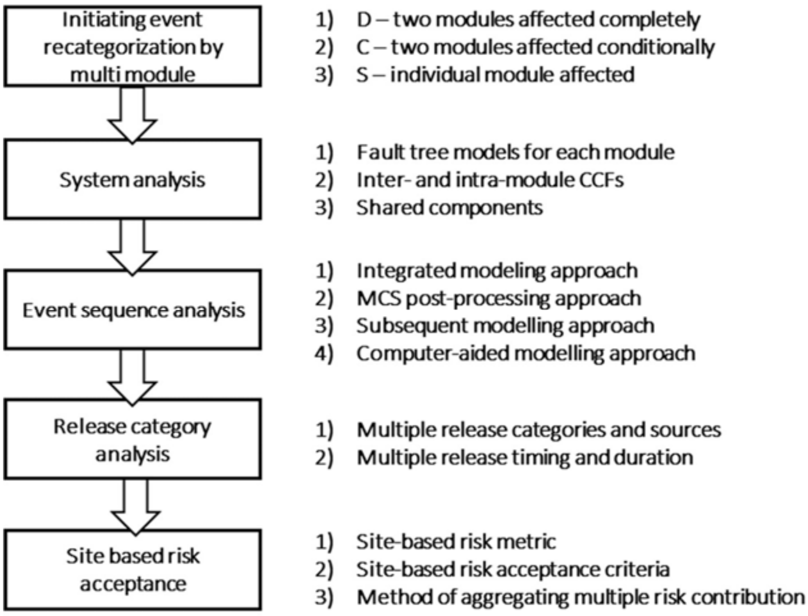


FIG. 36. Overall modelling framework of HTR–PM MUPSA.

3.3.3.1. Integrated modelling approach

Integrated modelling approach is the first approach we have proposed and piloted. This approach is intended to model the response evolution process of all the modules in one event tree model. Function Events representing the accident mitigation responses from each of the NSSS modules are modelled sequentially in the event tree as headings. Each of the function events is still binary–state and linked to the fault tree representing the corresponding mitigation function/system failure in the candidate reactor. Same as what we do in single module PSA, dependencies such as shared SSCs and CCFs are automatically solved by the fault tree linking. However, such an integrated modelling approach would undoubtedly result in a huge amount of event sequences; thus, a phased evolution method is recommended allowing for the whole set of event sequence evolutions to be modelled by a set of smaller event trees.

Three key steps to implement the phased evolution method are: (1) all the failure events involved during the event tree evolution process are recommended to be categorized into groups and these failure events groups to be prioritized accordingly; (2) each phase develops the event tree sequences based on the current failure events group only; (3) when the sequences under current phase are generated, an analysis is introduced to conclude the end states of these sequences and group these sequences into several intermediate end states accordingly. These are then treated as the initiating events of event trees for the next phase. Such phased method could help reduce the number of sequences to a manageable level. This approach requires the used PSA software used have the capability to propagate complex logic information between event trees. The input of the next phase event trees includes the whole set of logic structure information from the previous phase event tree sequences. RiskSpectrum PSA software provides such a feature to support this modelling approach.

3.3.3.2. Subsequent modelling approach

From the HTR–PM case study of integrated modelling approach, we find that although the multiunit event trees are new, the sequence evolution knowledge behind are actually not new, they are still based on the existing understanding we have established for the candidate single reactor. If we could maximize the usage of existing event trees, the whole work would be improved significantly. This finding leads to the proposal of subsequent modelling approach. Subsequent modelling approach intends to use the existing event trees established for the single module HTR–PM PSAs. Taking the existing event tree of module #1 as the prior, we connect the event tree of module #2 as the event tree structure template to the end of each event tree branch of module #1. By taking account of the conditions already introduced by event tree branches of module #1, the subsequent event tree structures of module #2 usually can be simplified a lot. Due to the fact that modification actions on event tree structures of module #2 are mainly to group, delete or conclude the branches from the template event tree, this step is somewhat a relatively easy task.

3.3.3.3. Computer-aided modelling approach

Computer–aided modelling approach is proposed to improve the effectiveness of subsequent modelling. Since the next development goal for HTR–PM type of NPP is intended to introduce more modules into one NPP unit, i.e. HTR–PM600. HTR–PM600 is designed as a 600MWe HTGR NPP having 6 NSSS modules and one shared steam turbine. It is probable to build two HTR–PM600 units on one site for the purpose of better economy and operational management. Hence, it is expected that 6 modules and even 12 modules will be considered in MUPSA in the near future. Currently, subsequent modelling is implemented manually.

3.3.4. Finland/VTT

In the Nordic SITRON project [7], the analyses were performed using SUPSA models. In this approach, SUPSAs need to be complete, and single unit risk metrics need to be calculated correctly covering contributions of significant multiunit scenarios, e.g. related to use of shared systems. Particularly, the consequences of multiunit events need to be modelled correctly in single unit models.

The analysis process can be presented as six steps presented in Fig. 37. In the process, relevant POS combinations, multiunit initiating events and multiunit dependencies are selected. Then probabilities related to relevant multiunit dependencies are estimated, and finally risk metrics are calculated.

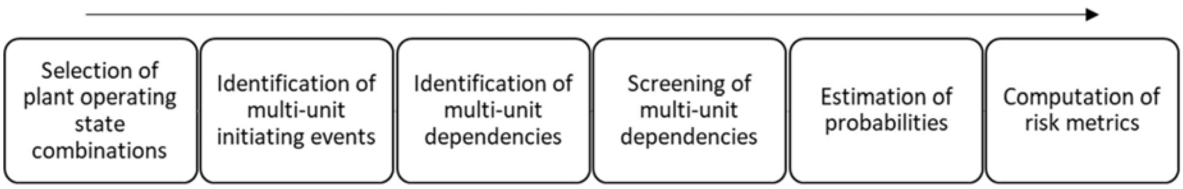


FIG. 37. Analysis process.

For the selection of POS combinations, POSs that are sufficiently similar, e.g. based on the configuration of residual heat removal systems, can be merged. POS combinations that are very short can be screened out. If the CDF related to a POS is very small, the POS can be screened out. After the merging and screening of POSs, site level POS combinations are created. Identified dependencies are analysed qualitatively and classified into categories 'very important', 'important', 'less important' and 'insignificant.' Some guidance for the classification can be found in [7]. The dependencies not classified as 'insignificant' can also be screened quantitatively. The quantitative screening is based on basic events related to the dependencies in the single unit models. The maximum contributions of the dependencies are estimated based on the Fussell–Vesely values of the basic events in the single unit models. If the maximum contribution is smaller than 1×10^{-8}/year, the dependency can be screened out [7].

In the SITRON project two alternative quantification approaches were used:

a) MCS list approach: MCS of individual units are combined considering the multiunit dependencies to create MCS for multiunit core damage. The frequencies of these new MCS are based on the estimated probabilities for multiunit dependencies. Risk metrics are calculated based on the MCS. This approach requires a tool to combine and analyse MCS;
b) Multiunit event combinations approach: relevant multiunit scenarios are identified. A scenario includes a multiunit initiating event and a set of multiunit events related to other dependencies. The scenarios can, for example, be presented in event trees that include a multiunit initiating event as the initiator and multiunit dependencies as event layers. For each multiunit scenario, the frequency and the CCDP in each unit are calculated. The CCDP values are calculated using SUPSA models. The multiunit CDF is calculated for each multiunit scenario by multiplying the frequency of the scenario with the CCDP values. The total multiunit CDF is calculated by summing the multiunit CDF values of individual scenarios.

3.3.5. Ghana/GAEC

An integrated Level 1 PSA approach to estimate SCDF was adopted. This approach involves mainly the superimposition of SUPSA models with unit–unit dependencies to reflect the multiunit case. The following are details of the unit–unit dependencies that were considered in this study:

— Common cause dependencies ($A^{(1)}$and $A^{(2)}$): these type of dependencies accounts for failures of a component (A), which is found in both units. The CCF may result from the common characteristics of $A^{(1)}$and $A^{(2)}$ including, same design from the same manufacturer as well as same installation and maintenance procedures;

— Causal dependencies between different events ($I^{(2)} \rightarrow C^{(1)}$, ($C^{(1)}|I^{(2)}$)): event in unit 2 ($I^{(2)}$) leads to a condition in unit 1 ($C^{(1)}$);
— Causal dependencies between a component and an initiating event ($D^{(2)} \rightarrow I^{(1)}$, ($I^{(1)}|D^{(2)}$)): the degradation of component D in unit 2 leads to an initiating event in unit 1 ($I^{(1)}$);
— Dependencies between identical initiating events caused by an external coupling condition such as an earthquake or LOOP.

For the demonstration of the methodology, a case study of a system was selected from the CAFTA user's manual tutorial. The function of the system is to supply water to a safety system for more than three hours. It was assumed for the purposes of this work that this system represents an auxiliary feedwater system of a typical PWR. For the multiunit scenario, two units of the same design were located on the same site. Fault trees were developed for the two auxiliary feedwater system using CAFTA version 6. The following modifications were made to the original single unit fault tree of the auxiliary feedwater system, which reflects the methodology of superimposing the static PSA model with unit–unit dependencies for the multiunit case with details as follows [18]:

— Basic event was added to each fault tree: one represents a condition in unit 1 that results from initiating event in unit 2 while the other represents a condition in unit 2 because of an event in unit 1. The probabilities assigned to these basic events were obtained from, which are licensee event reports submitted to the USNRC by operating plants in the United States;
— Dependencies between initiating events caused by a LOOP that affects both units is accounted for by the initiating event used in the event tree with probability (P(I|F)=1.1×10^{-2}/site.year).

The CCF for the multiunit was modelled for two groups of components in each system as follows: check valves had a common cause component group of size 4 (2 in each unit) and Motorized operated valves (instead of manual valves) had a common cause component group of size 4 (2 in each unit). The alpha factor model with staggered testing was selected and applied. Alpha factors used were obtained from the USNRC CCF parameter estimates from 1997 to 2015 [18]. Although the data used from literature is for single units, authors have developed a method for estimating posterior estimates of alpha factors using the Bayesian method applicable to multiunit and these results would be utilized in future studies.

3.3.6. Hungary/NUBIKI

Combined POSs of the four reactors and the adjacent SFP are characterized using the distinct POSs as defined for the unit–level PSA models. The approaches found viable to assessing the site–level risk are evaluated. Besides the use of common PSA methods, the analysis included some developmental work for risk quantification software.

3.3.6.1. Plant operational states

There are 25 POSs in the reactor PSA model for a single unit of the Paks NPP. These 25 states include one full power and 24 low power and shutdown states representing the refuelling outages. The operational states of the SFP are decomposed into six categories in the PSA based on the level of decay heat, the number and storage configuration of fuel assemblies, and water inventory of the pool (normal operational level and refuelling level).

In the analysis of an initiating event that impacts multiple units or release sources, the operational state of the four reactors and the four SFPs at the time of the event, need to be considered. The combined states of the different release sources are called overall POSs. To define overall POSs the operational cycles of the four reactors and the four SFPs have been evaluated for a 10–year period between The plant has recently introduced a 15–month operational cycle instead of the earlier 12–month cycle. Three types of refuelling outages are used in the new cycle: short, medium and long. The periodicity of the cycle for a plant unit is 10 years. The evaluation has led to the definition of 115 distinct overall POSs. Each state is characterized by a unique and physically viable combination of operational states for four reactors and four SFPs. The duration of these POSs is normalized so that they sum up to a year, i.e. 8,760 hr.

The nature of the overall POSs for 15 month (10,950 hr) operational cycle following the long outage of unit 1 are shown in Fig. 38 for different states of the reactors and SFPs. A sub–model within the multiunit risk model has to be developed for each overall NPP state in thus appropriately representing the distinguishing characteristics of a state. There may be ways to reduce the number of overall NPP states based on comparative analyses of the states to find bounding plant states for some groups of states (e.g., merging states with similar operational features, bounding low frequency states with less favourable states, etc.). If risk due to configuration of a certain low power and shutdown POS at unit 1 and full power operation at unit 2 is similar to the risk due to the configuration of the same low power and shutdown POS at unit 2 and full power operation at unit 1, the modelling of low power and shutdown states of one unit may be sufficient (i.e., modelling all operational states of unit 1 and only full power operation of all the other units).

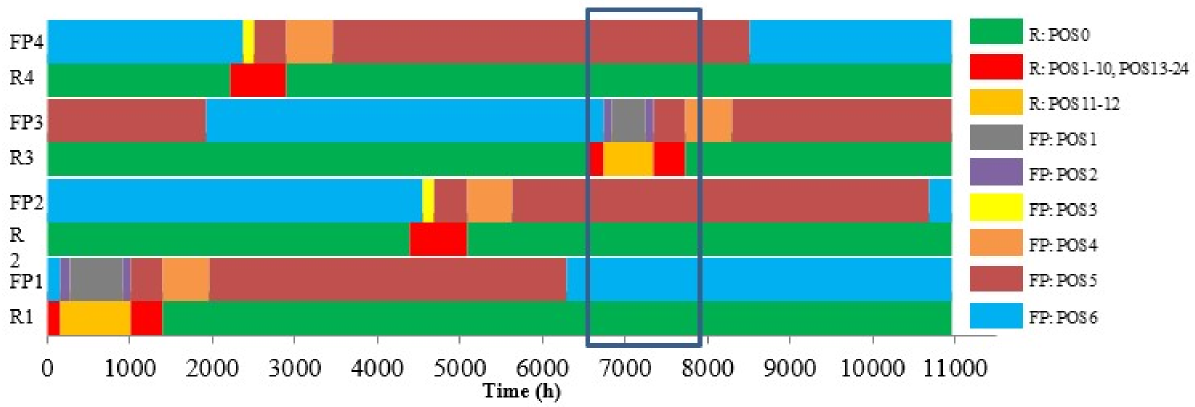


FIG. 38. PSA based POSs for a 15 month operating cycle for four reactors and four SFPs at NPP Paks [25].

3.3.6.2. PSA model development approach

Regarding all screened in multiunit events, the unit specific PSA models have been improved so that they can be appropriate for elaborating the multiunit model by integrating the event trees and fault trees of unit specific models. These model improvements have been completed for full power as well as for low power and shutdown states. The following main tasks were in the focus of the model upgrade:

— System of model identifiers has been revised and modified by giving a unit specific identifier to all unit specific model elements, and to others twin unit specific or plant specific identifiers, as appropriate;
— Fault trees of shared systems have been standardized;
— Modelling maintenance related unavailability of dedicated plant equipment as a true event in the relevant POSs as opposed to using average unavailability as in SUPSA;
— Size of the integrated multiunit model has been rationalized.

After performing unit specific model improvements, an initial MUPSA model was developed based on linking the modified unit specific PSA models to enable quantification of plant risk for each POS.

Three basic options were studied and evaluated for modelling and quantifying site level risk:

— Option 1: event tree linking approach;
— Option 2: event tree conversion to fault trees;
— Option 3: MCS conjunction.

Option 1 is the interconnection of the unit–level accident sequences for each initiating event that can induce transients in more than one unit or release source. The interconnection can be developed by building a single large event tree that includes all the combined event trees of the four units, or by connecting a continuing event tree for a unit to each event sequence (to success as well as to failure sequences) of another unit. Option 2 is an alternative way of an event tree linking conversion of all core damage sequences of the event trees for the relevant initiating events at a unit, into fault trees. This can be achieved by building a fault tree representation of each core damage sequence and connecting these fault trees under an OR–gate. This is modelled by a fault tree conversion of core damage sequences in the fault trees linked to the headers of the event tree. This solution does not result in a large event tree or many event trees; it can be completed manually by using traditional PSA software. However, the complexity of fault trees significantly increases. Option 3 is a combination and succeeding Boolean reduction and quantification of unit level MCS created for a given end state (core damage or fuel damage) for an initiating event that induces transients at multiple units.

The findings of a small scale pilot assessment were used to outline the advantages and the disadvantages of the different modelling options. By comparing these advantages and disadvantages, use of the event tree linking approach with (and in some cases without) fault tree conversion of accident sequences was chosen for the purposes of the full scale Level 1 MUPSA of NPP Paks.

3.3.6.3. Shared resources

Because of the Paks NPP design, there are available resources common to two twin units or even to all four units. For example, the demineralized water system is shared by two units. Open loop cooling by steam dump to the atmosphere is required for successful secondary side heat removal in some accident conditions, and demineralized water to be injected into the steam generators. According to the Technical Specifications of the NPP, when the inventory of demineralized water tanks decreases below the prescribed limit, the twin unit has to be shut down causing a reactor trip transient at the twin unit. This combined scenario is not modelled in the SUPSA but needs to be considered in the multiunit model. Total of 16 categories of shared technical resources are defined for four Paks units, including shared systems, shared structures and shared plant areas outside the building enclosures. Among these shared resources, the followings have been studied in detail:

— ESWS;
— Demineralized water system;
— Auxiliary emergency feedwater system;
— Condenser cooling water system;
— Steam system for house load use;
— Electrical power supply system and diesel generators;
— Fire water system.

The needs (e.g. water volume or flow rate) of multiple plant units using a designated shared resource were assessed taking into account all possible multiunit scenarios and all overall POSs. Also, the capacity of each shared resource was examined (e.g. flow rate of a pump). Based on the needs of the plant and the capacity of the system trains, the success criteria (e.g. required number of systems trains to fulfil a safety function in question) were determined for each shared system in each accident scenario.

3.3.7. India/AERB

To develop a MUPSA, it is important to understand the interconnection between the units. These interconnection leads to unfavourable interaction and dependencies between units. These interconnects may be present at event initiation level, prevention level and mitigation level or in any combination of these and are due to the physical coupling like electrical, fluid systems, ventilation ducts, cables trays, locations, procedures, similar design and design basis, CCFs, common location and environment, recovery actions, etc. To assess the risk and risk contributors at multiunit sites, accidents sequences that have potential to affect the multiple units needs to be accessed. This requires:

— Assessment of multiunit interactions and dependencies;
— Preparation of comprehensive lists of multiunit initiating events;
— Modelling multi–nit accident scenarios.

Figure 39 depicts the overall guiding principle for MUPSA. Figure 40 shows the overall approach followed by India for conducting the Level 1 MUPSA.

The Level 1 full scope PSA was reviewed, with all initiating events examined from a multiunit perspective to assess the likelihood of an event in a unit having influence on another unit and potentially impact the normal operation of another unit through either spatial interaction, cascading, or propagation, or the event has potential to affect the multiunit simultaneously. Further, these initiating events are classified broadly into events that are restricted to single unit or have influence on another unit. These initiating events are also reviewed by plant operators to search for additional multiunit initiating events. An initial list of multiunit initiating events is prepared for assessment of multiunit accident scenario. There may be additional initiating events depending upon the influence of progression of accident progression in a unit on another unit. Fault trees are modified to incorporate multiunit CCFs, success criteria of common systems with respect to multiunit considerations. Inter and intra unit CCF quantification needs supporting data.

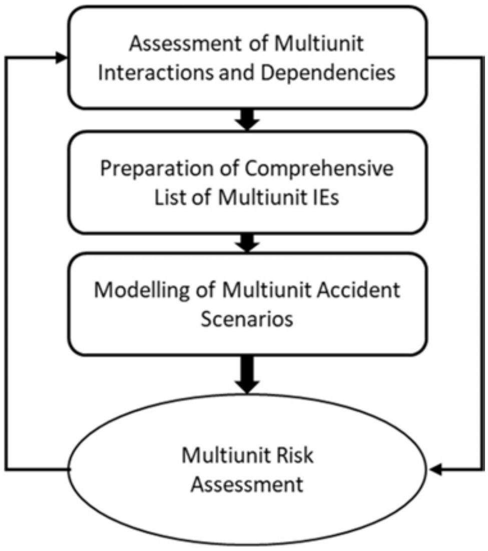


FIG. 39. MUPSA guiding principle.

For multiunit core damage modelling, both multiunit event tree and multiunit fault tree approach are explored. In multiunit event tree approach, amalgamation of event sequences of multiple units in single event tree or the end state of event trees of a unit can be linked to event tree of other unit for assessment of multiple core damages. Multiunit fault tree approach is by converting the core damage event tree sequences for a given initiating event in each unit into sequence fault trees and eventually a new event trees/fault trees are built logically connecting sequence fault trees according to number of core damages.

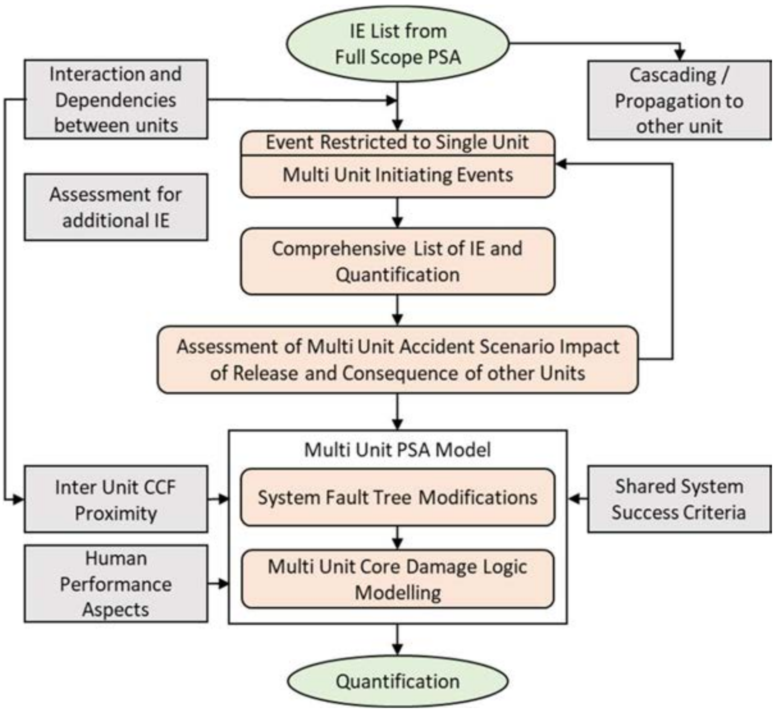


FIG. 40. Level 1 MUPSA framework.

3.3.8. India/BARC

The stages of analysis involved in development for multiunit site PSA study is shown in Fig. 41. The analysis majorly consists of Level 1 and Level 2 PSA study for reactor core and spent fuel facility.

Since there are two risk metrics devised in this benchmark study, i.e., SCDF and SiRF, the level of analysis considered for each metric is different.

For site CDF, Level 1 PSA models for both units are utilized, considering the initiating events from Category 1 & 2. For SiRF, Level 2 PSA of reactor units and PSA of SFF are used. However, in SiRF, initiating events from all categories have been analysed. An integrated PSA model approach is used to estimate SCDF and SiRF. Event trees are developed for each identified initiating event and safety functions in event trees are modelled using fault trees. Small event tree and large fault tree methodology defines the major framework for this analysis. Fault tree analysis is extensively employed for system modelling while accident sequence propagation is modelled using the event tree approach. The event tree development in MUPSA is like that of SUPSA, with the inclusion of the various functional events from multiple units. In the present study the event trees are developed from multiunit context in both Level 1 and Level 2 PSA. Some of the multiunit initiating events that are considered in the analysis from Level 1 PSA point of view are as follows:

— Reactor 1 structural failure;
— Reactor 2 structural failure;
— Pump house structural failure leading to SWS failure of both reactors;
— Turbine building structural failure leading to MSLBORB of both reactors;
— LOOP affecting both reactors.

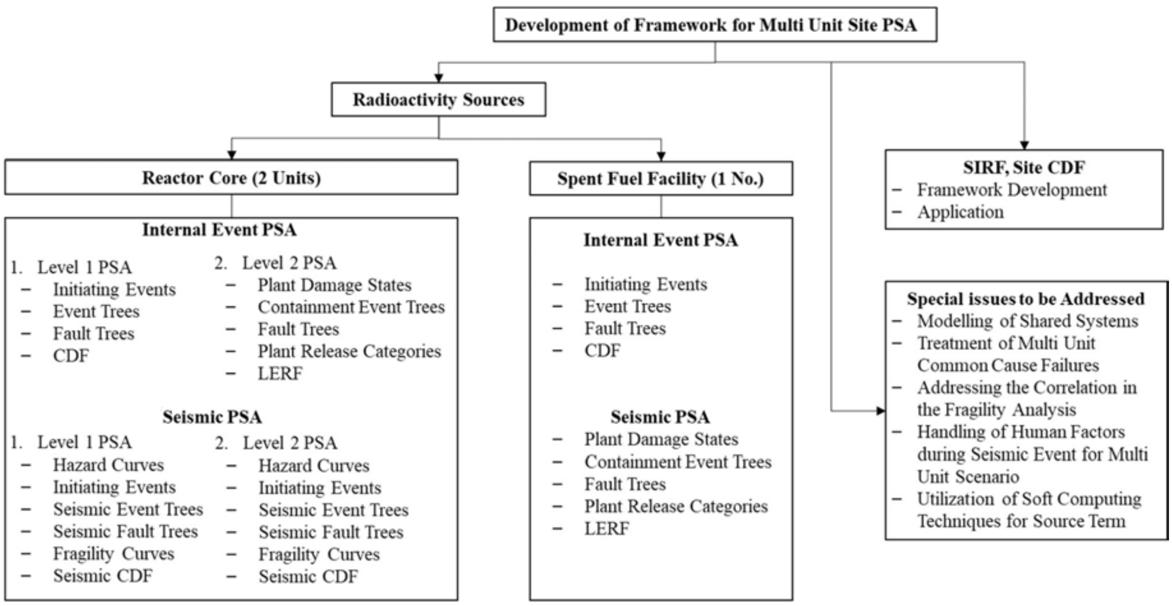


FIG. 41. Flow diagram of various stages of analysis involved in MUPSA study.

The event trees representing the above initiating events are developed as primary, secondary and tertiary event trees. Figure 42 shows the primary event tree developed from Level 1 PSA point of view, considering above initiating events as functional events.

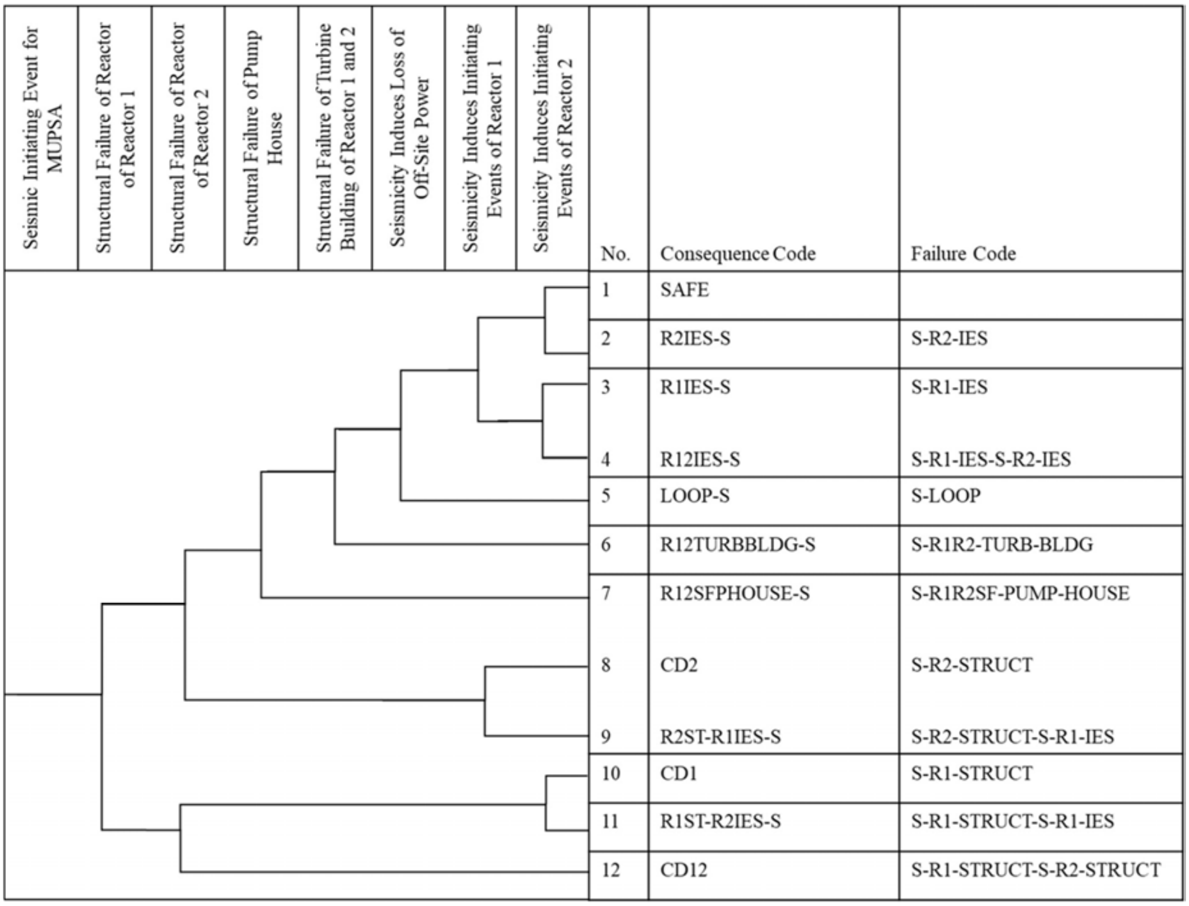


FIG. 42. Seismic primary event tree developed for MUPSA study.

Apart from this, deployment of state space approach using Markov diagram has been a special feature of this study. Many complex scenarios like incorporation of CCFs, preventive maintenance and repairs have been analysed using Markov models. MCS are found for each of the event tree. Various consequences are considered depending on the sources of radioactive releases. In Level 1 PSA studies, consequence categorization has been devised based on thermal hydraulics studies, i.e.:

— Core damage state (peak clad temperature beyond 1200˚C);
— Core degradation state (peak clad temperature beyond 800˚C, and within 1200˚C);
— Deviation from safe state (peak clad temperature beyond 400˚C, and below 800˚C);
— Success state (peak clad temperature is less than 400˚C).

Similarly, in the case of SFF, the following criteria have been used for fuel damage categorization:

— Safe state (pool temperature < 40°C));

— Deviation from safe state (40°C < pool temperature < 60°C);
— High pool temperature (> 60°C and < 100°C);
— Fuel damage (pool temperature =100°C, water not replenished).

For SiRF a generic and conceptual model has been developed that considers the location of NPP units and can include SFF. This model estimates the site LERF from individual unit LERF, release percentage from the sequences contributing to LERF and spatial location of the units. In general, there may be many NPP units at a site of different design. Performing MUPSA for a given site under consideration involves integration of various analysis for the multiple units in that site.

The following analyses are envisaged in the MUPSA study:

- PSA model development
  - Seismic Level 1 PSA for reactor core
  - Seismic Level 2 PSA for reactor core
  - Seismic PSA for spent fuel facility
- Thermal hydraulics analysis
  - Core thermal hydraulic analysis
  - Spent fuel facility thermal hydraulic analysis
- Containment analysis
  - Thermal hydraulic analysis
  - Fission product transport analysis
- Seismic analysis
  - Dynamic response analysis
  - Fragility analysis

To perform this analysis, various software tools have been utilized. The details of the tools are listed in Table 20.

Table 20. VARIOUS SOFTWARE TOOLS USED IN THE MUPSA STUDY

S. No.	Analysis Type	Software Tool
1.	Seismic Level 1 PSA model for reactor core	Risk Spectrum
2.	Seismic Level 2 PSA model for reactor core	Risk Spectrum
3.	Seismic PSA for spent fuel facility	Risk Spectrum
4.	Core thermal hydraulic analysis	RELAP5.0
5.	Spent fuel facility thermal hydraulic analysis	RELAP5.0
6.	Containment thermal hydraulic analysis	CONTRAN
7.	Fission product retention analysis in containment	CONTRAN
8.	Containment fission product transport analysis and source term estimation	CONTRAN
9.	Seismic dynamic response analysis	ATENA, MIDAS
10.	Seismic fragility analysis	SFRAG, Risk Spectrum

3.3.9. Pakistan/PAEC

The model is developed based on existing SUPSA model. The methodology adopted in this study for MUPSA Level 1 is described in subsequent sections.

3.3.9.1. Step 1: identification and selection of initiating events

Initiating event: seismic event at Chashma site is considered in the benchmark amongst other external hazards (e.g., flood, high wind) considering the Chashma site specific features. Tsunami hazard is not applicable at Chashma site as it is far away from the nearest sea or. Furthermore, a detailed flood study of Chashma site reveals that all safety related systems will remain dry in case of the worst flooding scenario due to breaking of all upstream dams (existing and planned) and superposition of probable maximum precipitation concurrent with dam breaks. Therefore, external flood is less important for Chashma site. Similarly, no high wind is observed or recorded in Chashma area and thus is an insignificant phenomenon. SUPSA shows that LOOP is the top contributing event resulting in core damage. Therefore, LOOP due to seismically induced collapse of switchyard buildings is selected for Chashma site as multiunit initiating event for the present study.

Shared connections (systems and equipment): there are some shared systems at Chashma site. All four units share the 132 kV offsite power supply system. Further details are as follows:

- Shared systems between C–1 and C–2:
  - 220 kv and 132 kv switchyard buildings;
  - Circulating cooling water pumping station system building;
  - Raw water purification system building;
  - Intake structures;
  - Drainage structures.

  Similarly, C–3 and C–4 NPPs are constructed simultaneously with provision of shared systems and buildings. But C–3 and C–4 NPPs do not share any systems and buildings with C–1 and C–2
- Shared systems between C–3 and C–4:
  - AAC power supply;
  - 220 kV and 132 kV switchyard buildings;
  - Circulating cooling water pumping station system building;
  - Raw water purification system building;
  - Fire protection water supply system;
  - Intake structures;
  - Drainage structures.

Switchyard buildings are shared among C–1 & C2 and C–3 & C–4 NPPs. Further, Chashma site is connected with four (04) independent off–site power sources. The three (03) off–site power sources are at 220 kV (Bannu grid, D.I.Khan grid, Ludewala grid) and one at 132 kV (Wan Bhachran grid). In this study, LOOP due to seismically induced collapse of shared switchyard buildings is assumed which results in unavailability of 220 kV and 132 kV offsite power lines. Also, credit of recovery is not considered in the analysis. The frequency of collapse of the switchyard buildings is 8.5×10^{-5}/yr.

Identical components: components/systems identical in all the units eligible to be evaluated for relevancy in CCF are:

- Main steam safety valves (spring loaded);
- Atmospheric power operated relief valves;

— Auxiliary feed water system pumps and valves.

Main steam system is divided into two main steam lines which are connected to corresponding steam generator outlet nozzle. Each steam line contains four (04) main steam safety valves and two (02) atmospheric power operated relief valves which are located outside the containment. The four (04) main steam safety valves are conventional spring loaded safety valves and operate automatically when pressure reaches set point value. Similarly, two (02) atmospheric power operated relief valves operates automatically when pressure reaches set point value. Auxiliary feedwater system serves as a backup of main feedwater system at times when main feedwater is not available. The function of auxiliary feedwater system is to provide adequate cooling water to steam generators during unavailability of main feedwater system under postulated incident, or accident conditions. It also maintains plant at hot shut condition for sufficient time and cool down reactor to conditions required for Residual Heat Removal System operation. In this process, heat of RCS is transferred to the secondary system via steam generators. auxiliary feedwater system is designed as safety class 3, seismic category SSE and qualification requirement QA2. It consists of two trains (each train contains motor driven pump and diesel driven pump) capable of supplying auxiliary feedwater to corresponding steam generator from either of three water sources (emergency feed water tank, conventional island demineralized water tank, firewater pool). Each motor driven or diesel driven pump is designed to provide sufficient flow to steam generator to remove decay heat of the reactor.

Proximity dependencies: nature of selected multiunit initiating event i.e., LOOP due to seismically induced collapse of switchyard buildings (multiunit LOOP) is such that it will not spread from any NPP to other NPP. Therefore, no proximity dependencies are found in the present study.

Human and organizational dependencies: operational and maintenance teams are independent for all four units operating at Chashma Site and thus human dependency may be neglected in the analysis. Further, each plant has a Plant Manager, and all four plant managers are under one General Manager. The organizational dependency is not considered in the current study for the sake of simplicity.

3.3.9.2. Step 2: dependency analysis

The dependencies identified in Step 1 are ranked qualitatively for screening purposes in Table 21.

TABLE 21. RANKING OF DEPENDENCIES

Dependency	Importance	Remarks
Shared Systems & Equipment		
Off Site Power & Switchyard	Very Important	Ordinary power supply. A failure may result in a multiunit initiating event (multiunit LOOP)
Identical Components		
Main Steam Safety Valves	Very Important	
Atmospheric Power Operated Relief Valves	Very Important	
Auxiliary Feedwater System Pumps and Valves	Very Important	
Proximity, Human & Organizational		
N/A	Less Important	No potential dependencies identified

3.3.9.3. Step 3: CCF modelling

CCF has been modelled based on identical SSCs for inter–unit and intra–unit model of single unit. The number of combinations of more than four components increases exponentially for modelling of CCF. Therefore, for the sake of simplicity, only complete CCF basic event of all components failing is added as new CCF group to the existing single unit model, and its probability is assumed to be 0.1 multiplied by the complete CCF probability used in single unit model for components more than four (04). The inter–unit components (vales, pumps) prone to CCF are greater than four (04). Therefore, proposed simplified methodology for inter–unit CCFG is adopted for all components.

3.3.9.4. Step 4: extension of SUPSA model

The existing SUPSA model is extended to include result of dependency analysis. Therefore, event trees and fault trees of existing single unit models are extended for computation of multiunit CDF. In this regard, different approaches for event tree development are considered in the study. The simplified event tree approach is adopted in contrast to huge event trees. In this approach, dependencies are modelled explicitly.

The details of functional events are as follows:

a) SR SGA INTER–INTRA CCF & SR SGB INTER–INTRA CCF: These functional events are used to model inter–unit CCF of main steam safety valves and atmospheric power operated relief valves in addition to existing single unit model. The function of these valves is to release the energy to final heat sink i.e., atmosphere, so as to prevent over pressurization in secondary side and from decay heat removal mechanism with feed water system;
b) Auxiliary feedwater system INTER–INTRA CCF: This functional event is used to model inter–unit CCF of auxiliary feed water systems pumps (motor driven and diesel driven) and valves in addition to existing single unit model. The auxiliary system fulfils the requirements of steam generator inventory makeup after a transient event. It provides secondary side heat sink in combination with steam generator steam removal system to remove heat generated in RCS.

3.3.9.5. Step 5: computation of risk metrics

The risk metrics considered in the study are single unit CDF, multiunit CDF and SCDF for MUPSA Level 1. While CDF is used in computation of risk metrics for MUPSA, obtained from SUPSA Level 1 study, result of core damage in all four reactor cores concurrently due to multiunit LOOP is considered for multiunit CDF. Both, CDF and multiunit CDF are used in the computation of single unit CDF and SCDF. Seismic fragility and risk assessment of the selected shared buildings for multiunit LOOP event is conducted in the current study using the EPRI procedures, IAEA guidelines [2, 3] and various other relevant codes and standards. The methodology/modelling technique for evaluating fragility and risk of shared buildings is described in this section.

3.3.9.6. Seismic fragility and risk assessment of shared switchyard structures

Shared buildings details: among four Switchyard Buildings. (i.e., two GIS 220 and two GIS 132 buildings), one set of GIS 220 and GIS 132 buildings are shared between C1–C2 and the other set of GIS

220 and GIS 132 buildings are shared between C3–C4 NPP units. These buildings are Reinforced Concrete frame structures and are non–nuclear safety class buildings. The structural system of these buildings is Intermediate Moment Resisting Frame system. These building are designed according to Chinese National Standard GB (50010–2002) for seismic intensity level of 7 (approximately equal 0.1g). These buildings are neither designed for SSE level nor OBE earthquake level.

Structural modelling and finite element analysis: structural models are prepared in the SAP2000 software by using the as–built architectural and structural drawings. Beams and columns are modelled using the line elements whereas roof slabs are modelled using shell elements. To account for the inelasticity in the nonlinear static analysis (pushover analysis), plastic hinges are modelled in the buildings. Different finite element analyses are conducted in this study which are modal analysis, linear static analysis, linear dynamic analysis (response spectrum analysis), non–linear static analysis.

Load cases and load combinations: dead (D), live (L), including crane loading, and earthquake (E) loadings are applied to the structural models for evaluating the structural response. For the earthquake loading, USNRC RG 1.60 spectra anchored to the review level earthquake of 0.25g (i.e. SSE level for Chashma site) is used. The spectra are 7% damped as recommended by IAEA guide for assessment of existing structures. Different load combinations corresponding to the normal operating load and normal operating load plus earthquake are used in the finite element analysis. These load combinations are:

— First 1.0 D+1.0 L (Service Load i.e. Normal operating load for shared GIS Buildings);
— Second 1.0 D+1.0 L ± 1.0 EQX (Service Load+ Earthquake Load in X–direction);
— Third 1.0 D+1.0 L ± 1.0 EQY (Service Load+ Earthquake Load in Y–direction).

Capacity and demand evaluation: for evaluating the yield moment capacity (M_y), flexural capacity (M_p) and other plasticity parameters of the beams and columns cross–sections, moment–curvature curves are developed for each cross section. Moreover, for columns P–M2–M3 interaction curves are developed to get the appropriate moment capacity. Beams of all the four buildings are divided into groups for comparison of capacity and demand. Different beams groups have varying sizes and reinforcements. From capacity and demand comparison results it is clear that seismic forces demand on beams along the earthquake direction is generally higher and exceeding capacity in top and bottom directions in most of the beam groups. For different columns in each building, the seismic demand exceeds capacity in different critical earthquake direction (*X* or *Y*). The capacity vs. demand outcomes are used in evaluating strength factor of safety 'F_s' for fragility analysis.

Failure mode identification: failure mode identification is important for fragility analysis. The comparison between nominal shear capacity (V_n) and flexural–shear demand (V_p) is used to identify the failure mode of structural components. All beams and columns of GIS buildings have $V_n>V_p$, which means all beams and columns have flexural (ductile) failure mode and the members will remain elastic in shear. Other criteria to identify the flexural failure mode of columns are used, which are $Av/bws \leq 0.0002$ (Av is transverse steel ratio, *bw* is width of section and s is ties spacing), spacing to depth ratio less than 0.5. Moreover, specifically for columns if $1.05V_p<V_n\leq 1.4V_p$ or $V_n>1.4V_p$ the columns will potentially fail in flexure. The identification of failure mode helps in defining the type of plastic hinges (e.g. flexural, shear) for the inelastic analysis.

Fragility model and parameters: double lognormal model is selected for developing family of fragility curves for the case buildings. These fragility functions are given as follows:

$$f = \Phi\left(\frac{ln\frac{a}{\hat{a}}+\Phi^{-1}(Q)\cdot\beta_u}{\beta_r}\right) \tag{8}$$

where, Q is the confidence level, and:

$$f = \Phi\left(\frac{ln\frac{a}{\hat{a}}}{\beta_c}\right) \tag{9}$$

where:
a : ground motion parameter;
$\hat{a}$: deterministic value representing the median capacity;
β_u and β_r : logarithmic standard deviation for uncertainty and randomness, respectively;
$\hat{a}$, β_u, β_r : fragility parameters and needs to be calculated/defined for development of the fragility curves;
$\hat{a} = \hat{f}.a_{RLE}$, where $\hat{f}$ is median factor of safety which is intermediate random variable to estimate fragility parameters and can be evaluated for civil structure by using the following;
$\hat{f} = \hat{f}_S.\hat{f}_\mu.\hat{f}_{RS}$, where strength factor = $\hat{f}_S$, inelasticity absorption factor = $\hat{f}_\mu$, response conservatism factor = $\hat{f}_{RS}\cdot a_{RLE}$ is RLE which is 0.25g for the case structures equal to SSE.

After defining the median values of strength factor $\hat{f}_S$, inelasticity absorption factor $\hat{f}_\mu$, response conservatism factor $\hat{f}_{RS}$ and their corresponding logarithmic standard deviation values associated with randomness and uncertainty variables, the fragility parameters $\hat{a}, \beta_u, \beta_r$ for generating the family of fragility curves with different confidence levels can be evaluated. The other fragility parameters $\beta_r$and β_u can be evaluated using:

$$\beta_u = (\beta_{S,u}^2 + \beta_{\mu,u}^2 + \beta_{RS,u}^2)^{1/2} \tag{10}$$

For mean fragility curve the composite logarithmic standard deviation is calculated using:

$$\beta_C = \sqrt{\beta_r^2 + \beta_u^2} \tag{11}$$

Fragility curves are further developed by using the fragility functions using the calculated / defined parameters. The $\hat{f}$, $\hat{a}$ values calculated for the shared GIS Buildings are given in Table 22. Calculated β_r, β_u and $\beta_c$values for the shared GIS buildings are given in Table 23.

TABLE 22. FRAGILITY PARAMETERS FOR DEVELOPING FRAGILITY CURVES OF SHARED GIS BUILDINGS

Buildings	$\hat{f}$	$\hat{a}$
GIS 220 Building C1–C2	1.524	0.381
GIS 132 Building C1–C2	2.171	0.542
GIS 220 Building C3–C4	2,774	0.693
GIS 132 Building C3–C4	2.564	0.641

TABLE 23. FRAGILITY PARAMETERS (LOGRITHIMIC STANDARD DEVIATION FOR UNCERTAINTY AND RANDOMNESS) USED FOR DEVELOPING FRAGILITY CURVES OF SHARED GIS BUILDINGS

Buildings	$\boldsymbol{\beta_r}$	$\boldsymbol{\beta_u}$	$\boldsymbol{\beta_c}$
GIS 220 Building C1–C2	0.369	0.457	0.587
GIS 132 Building C1–C2	0.369	0.457	0.587
GIS 220 Building C3–C4	0.369	0.457	0.587
GIS 132 Building C3–C4	0.369	0.457	0.587

The calculation/definition of the median values of strength factor $\hat{f}_S$, inelasticity absorption factor $\hat{f}_\mu$, response conservatism factor $\hat{f}_{RS}$ is discussed in the next section.

3.3.9.7. Median factor of safety calculation for different shared GIS buildings

Median strength factor $\hat{f}_S$: definition of a failure could be considered either from strength or from deformation. In the current study, strength based approach is adopted. Derivation of fragility from strength consideration involves with element level failure; an element reaching limit state of strength (or stress). Using the strength approach structural fragility is governed by the capacity of the weakest element. Fragility derivation from element strength (or stress) consideration would result in a conservative estimate as compared to the deformation approach. Median strength factor of the case structures is calculated using:

$$\hat{f}_S = \bar{F}_1 = \frac{S-R_N}{R_T-R_N} \tag{12}$$

where:

S: capacity of the element for a given failure mode;

R_N: response of the element for the specified failure mode against normal operating load (concurrent non–seismic load like dead load, operating temperature load, etc.);

R_T: response of the element for the specified failure mode against total load on the structure (sum of seismic load and normal operating load; it basically represents the ratio of capacity to demand. To account for uncertainty and randomness associated with median strength factor, the logarithmic standard deviation $\beta_{S,U}$ and $\beta_{S,R}$ value range proposed for civil structures and are adopted in this study.

Inelastic energy absorption factor $\hat{f}_\mu$: accounts for the fact that an earthquake represents a limited energy source, and many structures or equipment are capable of absorbing substantial amounts of energy beyond yield without loss–of–function. This factor basically corresponds to the level of conservatism in assessing the capacity; it depends primarily on the energy absorption capacity of SSC beyond elastic limit. Inelastic energy absorption factors are a function of system ductility μ (the ratio of maximum displacement to displacement at yield). A recommended value is 1.0 for $\hat{f}_\mu$, if the failure modes are brittle. The following relations are suggested to define $\hat{f}_\mu$:

— If the dominant frequency range is between 2–8 Hz:

$$\hat{f}_\mu = \sqrt{2\mu - 1} \tag{13}$$

— If the dominant frequency range is less than 2 Hz:

$$\hat{f}_\mu = \mu \tag{14}$$

— If the dominant frequency range is above 33 Hz:

$$\hat{f}_\mu = \mu \tag{15}$$

3.3.9.8. Methodology for evaluating median in–elastic energy absorption factor for shared GIS buildings

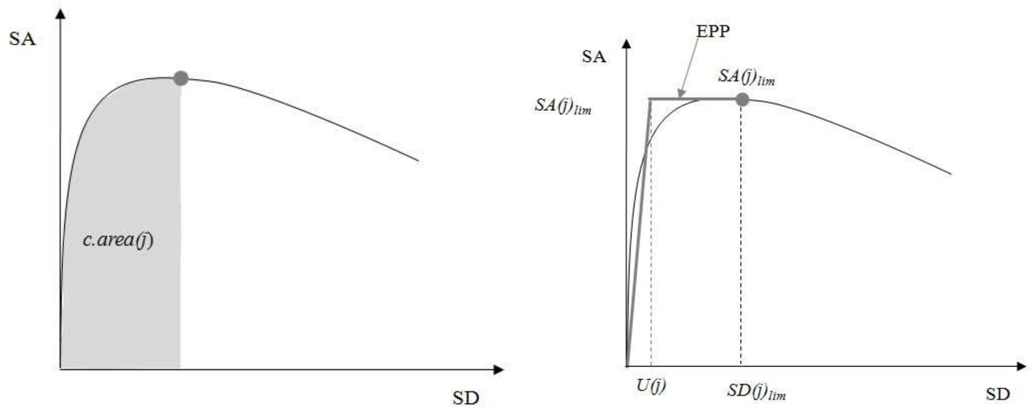
To evaluate the median inelastic energy absorption factor of the shared GIS Buildings, capacity spectrum is generated using pushover analysis and an ultimate displacement limit corresponding to collapse limit. Plastics hinges are defined in beams and columns to capture the inelastic behaviour at different limit states. Elastic perfectly plastic (EPP) system corresponding to ultimate displacement limit is used and using equal energy principle ductility is evaluated.

Figure 43 (a) shows the cumulative area under the capacity curve at *SD*(*j*) corresponding to the capacity point corresponding to defined limit. EPP corresponding to this point is shown in Fig. 43 (b). The equal area rule is applied to evaluate the yield displacement *U*(*j*) using the following relation where *j* represents the building population:

$$U(j) = -\frac{2(C.area(j)-(SD(j)_{lim}SA(j)_{lim}))}{SA(j)_{lim}} \tag{16}$$

Drift limit state for collapse of reinforced concrete structures: different codes and researchers have recommended different Interstory drift ratio (IDR) for the collapse damage state of ductile reinforced concrete frame structures. Structural Engineers Association of California Vision 2000 has recommended an IDR value of ≥2.5% whereas Federal Emergency Management Agency 356 recommends IDR ≥4 for collapse for reinforced concrete frame structures. In the current study the structures are pushed to 4% to achieve the flexural failure mode (plastic hinge generation in push over analysis).

Time dependent properties of concrete: in this study, time dependent properties of concrete properties are considered for C1–C2 shared GIS building which is around 19 years old and is important for time limiting aging analysis of structure. These properties are used while evaluating $\hat{f}_\mu$. In SAP2000, CEB–fib 90 model are used for evaluation of shrinkage strain using the parameters mean 28 days compressive strength, relative humidity, age of concrete at the beginning of shrinkage, cement type and shape of specimen.



(a) *(b)*

FIG. 43. Evaluation of yield displacement for EPP system: (a) Cumulative area at a particular spectral displacement, (b) Implementation of equal energy rule for yield displacement evaluation.

Median response conservatism factor for case structures '$\hat{f}_{RS}$': this factor represents the conservatism associated with calculating demand. Civil engineering structures, which generally support, house and protect the equipment and other SSC, are the primary structure and median value of response conservatism factor $\hat{f}_{RS}$ is calculated using:

$$\bar{F}_R = \bar{F}_{RS} = \hat{f}_{RS} = \bar{F}_{SA}\bar{F}_{SS}\bar{F}_{\delta}\bar{F}_M\bar{F}_{MC}\bar{F}_{EC}\bar{F}_{SD} \tag{17}$$

where:
$\bar{F}_{SA}$ = Factor for ground motion and associated response spectra for a given PGA;
$\bar{F}_{SS}$ = Soil Structure Interaction factor;
$\bar{F}_{\delta}$ = Factor for energy dissipation. i.e. damping;
$\bar{F}_M$ = Structural modelling factor;
$\bar{F}_{MC}$ = Factor for combination of modes and earthquake analysis results;
$\bar{F}_{EC}$ = Factor for combination of earthquake components;
$\bar{F}_{SD}$ = Factor to reflect reduction of seismic input with depth.

The generic values of the intermediate factors in above Equation and their logarithmic standard deviation associated with randomness and uncertainty are used in the current study.

Fragility curves development: by using the calculated fragility parameters, family of fragility curves are developed for all four shared GIS Buildings from the defined fragility model.

High confidence of low probability of failure: seismic capacity of SSC is represented by high confidence of low probability of failure. It represents the PGA, as the hazard parameter, corresponding to 5% conditional probability of failure on the 95% confidence fragility curve or alternatively can be defined as PGA on the mean fragility curve corresponding to 1% conditional probability of failure. The high confidence of low probability of failure capacity is a conservative representation of capacity and corresponds to the earthquake level at which it is extremely unlikely that the loss of shutdown capability or unacceptable performance will occur. The values are calculated from the developed family of fragility curves for the shared GIS Buildings at Chashma site.

Probabilistic seismic hazard assessment of the Chashma site: PSHA of Chashma site is based on available geological and seismological database. Methodology adopted for PSHA of CHASHMA site is in accordance with guidelines and procedures presented in IAEA Specific Safety Guide SSG–9, 2010 Seismic Hazards in Site Evaluation for Nuclear Installations [23]. PSHA is computation of probabilities of occurrence per unit time of certain levels of ground shaking caused by earthquakes. The results of this analysis are commonly presented in the form of Seismic hazard curve, which shows annual probability of exceedance versus ground motion amplitude. The methodology/steps adopted for conducting the PSHA study of the Chashma site is shown in the flowchart in Fig. 44.

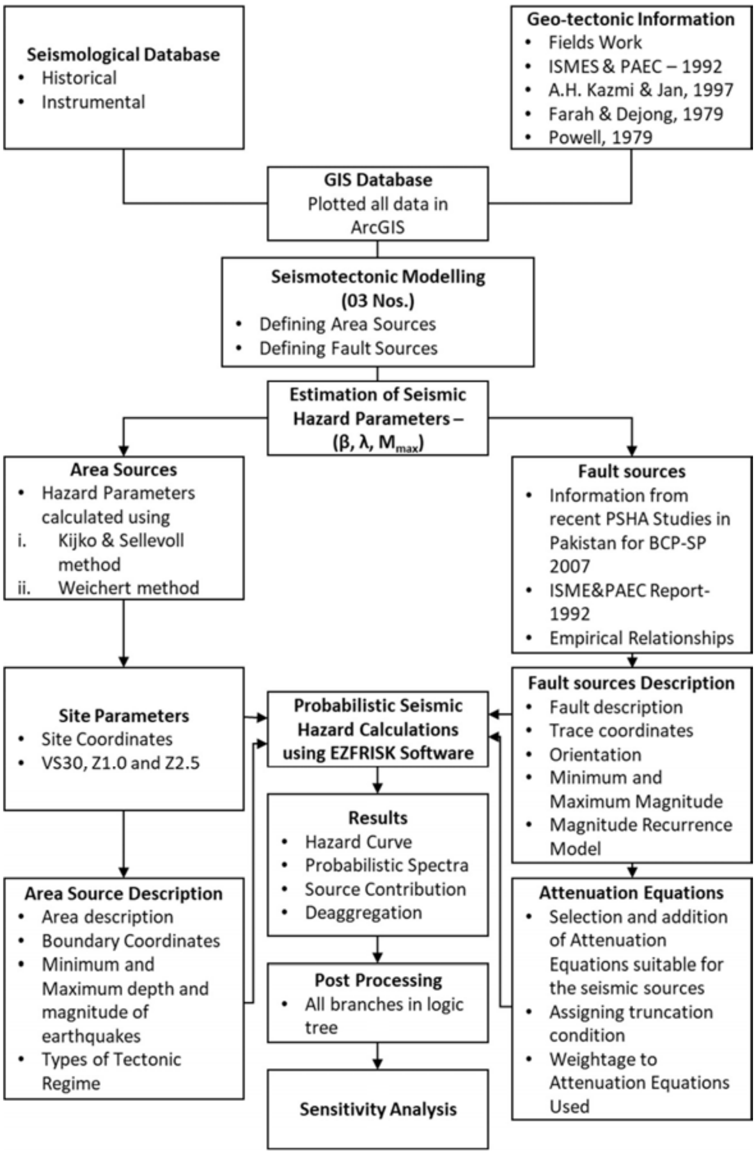


FIG. 44. Flow chart for different steps of PSHA.

Calculation of the initiating event frequency: mean hazard curve is used along with the mean fragility curve of all buildings to calculate the Initiating Event Frequency through convolution integration. This procedure is shown in Fig. 45. To calculate the annual frequency of unacceptable performance, the same PGA intervals produced in the seismic hazard curve are used for the unacceptable performance fragility curve. The midpoint PGA, a_i, of each interval corresponds to a probability of unacceptable performance, F_i. The sum of the products of F_i and ΔH_i for all PGA intervals is the total annual frequency of unacceptable performance, AF, as given with:

$$AF = \sum_{i=1}^{n} \Delta H_i(a) F(a_i) \tag{18}$$

where, ΔH_i represents the annual frequency of exceedance at a given value of ground motion, a_i (seismic hazard curve). F_i represents the probability of unacceptable performance at a given value of ground motion, a_i. In Fig. 45, the a_{s11} to a_{s1n} are the mean PGA values of different PGA ranges. Annual frequency of unacceptable performance is evaluated for a selected hazard intervals/ranges (Hazard Bins) based on specific criteria. For the current study, the total annual frequency of unacceptable performance is calculated from 0.01g to 0.5g (one bin with mean of 0.25g) and for multiple seismic hazard bins within this range.

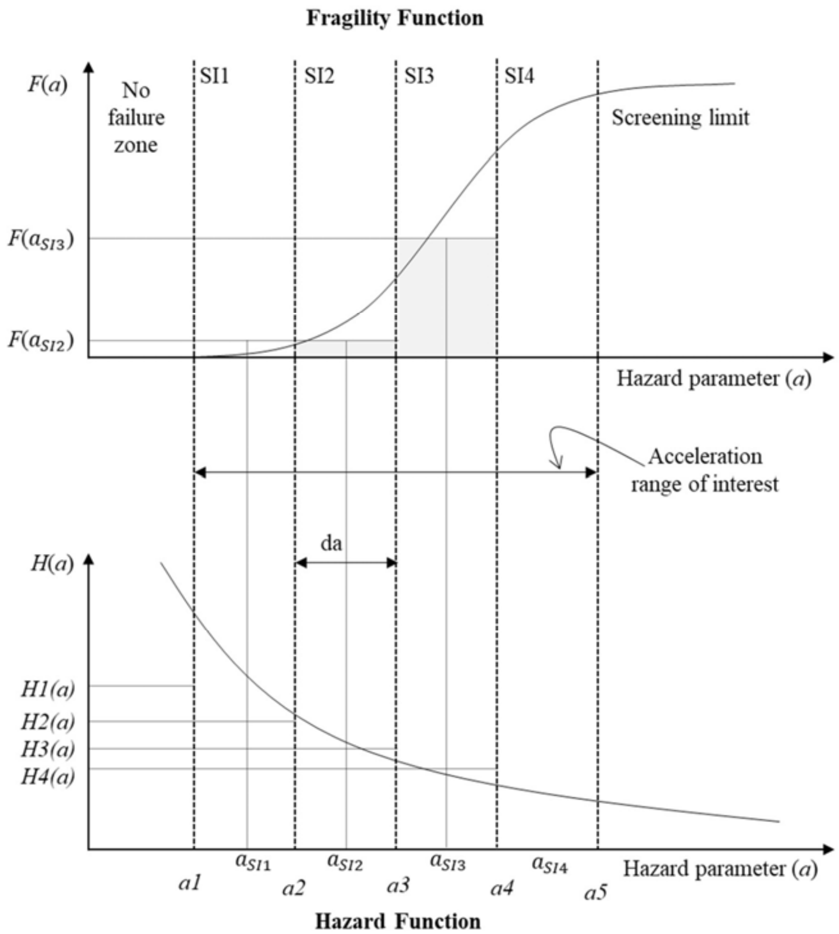


FIG. 45. Representation of calculation method for annual frequency of unacceptable performance by convolution integration using fragility and hazard curves.

Software: EZ–FRISK v 7.62 software package is used to perform the PSHA of the CHASNUPP site. The software has three main capabilities;

- Seismic hazard analysis;
- Spectral matching;
- Site response analysis.

With the EZ–FRISK module the earthquake hazard at a site under certain assumptions as specified by the user can be calculated. These assumptions include the earthquakes' location, characteristics, and the associated ground motions. Seismic hazard analysis is performed as a single site analysis. With the EZ–FRISK module, both probabilistic and deterministic seismic hazard calculations, can be carried out. The results of the probabilistic calculations include the annual frequencies of exceedance of various ground motion levels at the site of interest. These deterministic calculations estimate ground motions (for the mean and specified fractals of the ground motion dispersion) corresponding to the largest magnitude occurring on each seismic source at its closest approach to the site of interest. These results are applicable to various types of structural analyses. Seismic hazard analysis with EZ–FRISK relies on databases of ground motion equations and seismic sources. EZ–FRISK provides the users with tools to create and maintain databases, and to download extensive and up–to–date databases from Risk Engineering's web server for the user's licensed regions. The input parameters required for EZ–FRISK are as follows:

- Site parameters;
- Area source parameters;
- Fault sources parameters;
- Attenuation equations.

The structural analysis SAP2000 is a software package based on the finite element method. In addition, it has the capability to support designing and optimizing the building structures including the modal analysis, static and dynamic analysis, linear and nonlinear analysis. The analytical modelling is the member type model which means that beams or columns are modelled using single elements. The inelasticity formed in these elements is assumed to be concentrated at the ends of the element, which is a response of building elements to earthquakes. The hysteretic response of the concentrated plasticity at the ends of a member can be described by a moment curvature relationship. With the SAP2000 for each material one or more stress–strain curves that are used to generate nonlinear hinge properties in frame elements can be specified. Different curves can be used for different parts of a frame cross sections. For example, in a reinforced concrete material, SAP2000 can specify stress strain curves for confined reinforced concrete, unconfined reinforced concrete, longitudinal reinforcing steel, and hoop confinement reinforcing steel. For steel and other metal materials, SAP2000 typically only specify one stress–strain curve. Different time dependent concrete properties can be assigned to model creep and shrinkage effects for time limiting aging analysis. The SAP2000 element library contains a variety of cross sections, including rectangular sections as used for modelling the beams and columns of the reinforced concrete buildings. Also, the cross sections used for steel building are chosen from the built–in sections included in the steel sections library. Frame line elements and shell elements are used to model beam, columns and slabs, respectively in the case structures. SAP2000 is utilized in this study for static, dynamic and nonlinear static pushover analysis. For dynamic analysis, response spectrum method is used by defining the appropriate spectrum to obtain the seismic response of structure in different directions. Modal super position method is used for the response spectrum analysis.

Different summation rules such as square root of sum of squares or complete quadratic combination can be used for combination of modal responses in the dynamic analysis. For the pushover analysis SAP2000 provides the following:

- Nonlinear static analysis procedures to handle the sharp drop–off in load carrying capacity of frame hinges;
- Displacement control nonlinear static analysis procedures, so that the structure can be pushed to a desired target displacement;
- Display capabilities in the graphical user interface to generate and plot pushover curves, including demand and capacity curves in spectral ordinates. Also, to plot and get information about the state of every hinge formed at each step in the pushover analysis.

3.3.10. Republic of Korea/KHNP

As we need to consider nine units to develop MUPSA models, it is impossible to consider all combinations of operating status and it is not expected to get any significant insights from considering all combinations. So, the concept of site operating status was applied to simplify MUPSA models. Based on the operating experiences of O/H and long term O/H schedules for Kori/Saeul sites, we decided to consider five kinds of site operating status to develop MUPSA models.

- Site operating status 1: all nine units on full power operation;
- Site operating status 2: eight units on full power operation;
- Site operating status 3: seven units on full power operation;
- Site operating status 4: eight units on full power operation & one unit in O/H;
- Site operating status 5: seven units on full power operation & two units in O/H.

In site operating status 2, one unit out of nine units is not considered because there is no fuel in the reactor vessel during the specific period in O/H. With the same concept, only seven units out of nine are considered when developing MUPSA models. As for the modelling structures, we modified the structures of the SUPSA models from a format of event tree and fault tree to a format of single top fault trees. In addition, the names of gates and basic events in the fault tree are modified to identify their units, and shared resources such as off site power, the alternative alternate current diesel generator (AAC–DG) are modified to reflect sharing characteristics. The logical fault tree to consider multiunit scenario combinations are developed; Figs. 46 and 47 show the modelling structures. There are a few shared SSCs between twin units such as:

- Switch yard & off site power (shared);
- Intake & discharge structures of sea water (shared);
- Alternate AC diesel generator (shared);
- Instrument air (inter–unit connection).

Based on engineering judgement, we considered LOOPs simultaneously occur in all units and recovery actions for off site power are totally dependent for all units. And, conservatively, we did not consider instrument air supply from adjacent unit. As for the intake & discharge structures, we considered multiunit loss of circulating water due to marine lives as a multiunit initiator. AAC–DG is shared between two units or among four units, which could be used for only one unit. So, we decided the order to connect AAC–DG to a unit and considered this in the models. For example, AAC–DG can be available for the first unit, and it can be used in the second unit only if SBO occurs in the first unit.

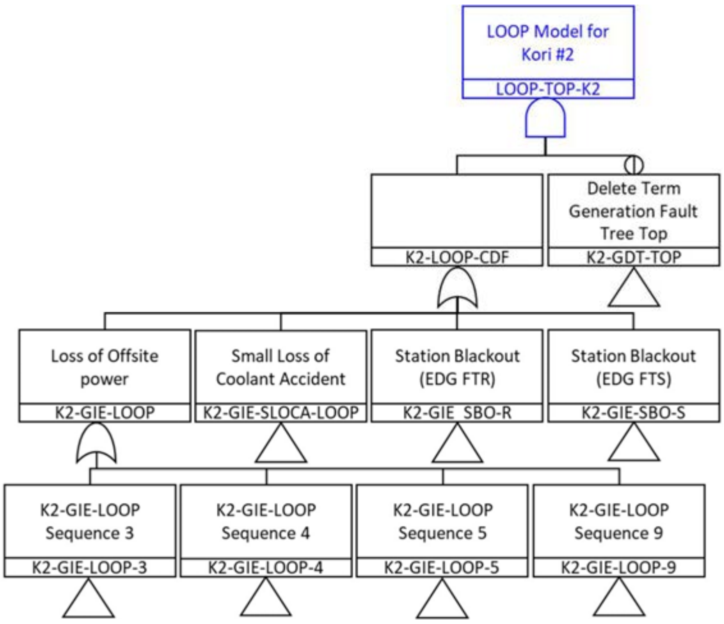


FIG. 46. Modified SUPSA models [25].

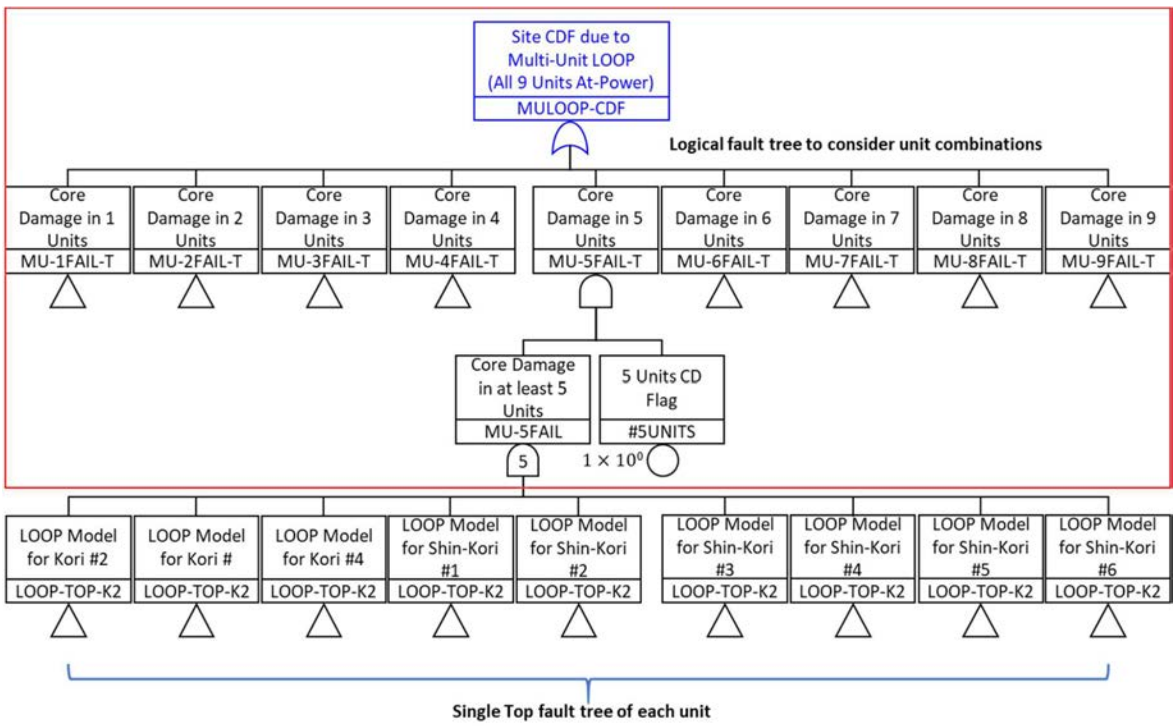


FIG. 47. Integrate MUPSA models considering nine units [25].

3.3.11. Republic of Korea/Hanyang University

The SRA framework shown in Fig. 48, is used in the benchmark to assess multiunit issues and site risks. Based on the framework, the operation records and data of Korean NPPs were surveyed to develop the model reflecting the POS such as full power operation and shutdown operation. For Kori 2, Shin Kori 1, and Shin Kori 2, a one top model for each unit was created considering the fraction of full power operation period and Shutdown operation period. Among Shutdown operation, it was assumed that there would be no core damage during the period in which nuclear fuel was withdrawn. The models for the rest of reactors are built assumed to be at full power operation.

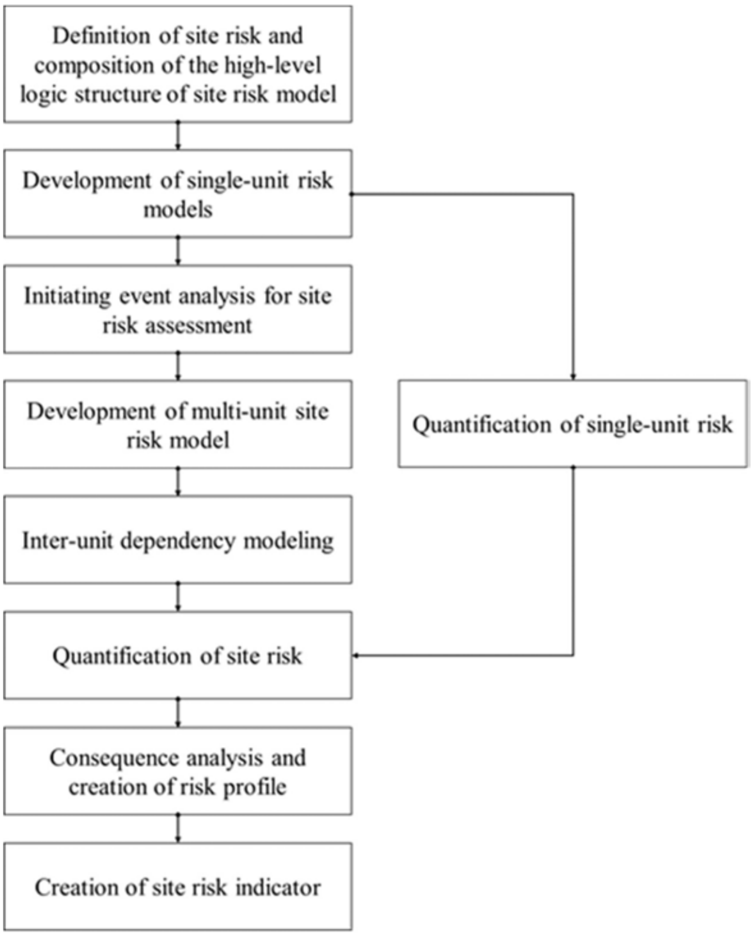


FIG. 48. Site risk assessment framework.

With the SUPSA models constructed as per Figs. 49 and 50, an SRA model was built using the truth–table method through AIMS–PSA and FTREX program. This model was developed including single unit initiator and common cause initiator and the inter–unit dependency between multiple units. For seismic event, a separate model will be built as the logic composition in the model would be different from the internal event. The inter–unit dependency model considers the shared components, inter–unit CCF and human failure events (HFEs).

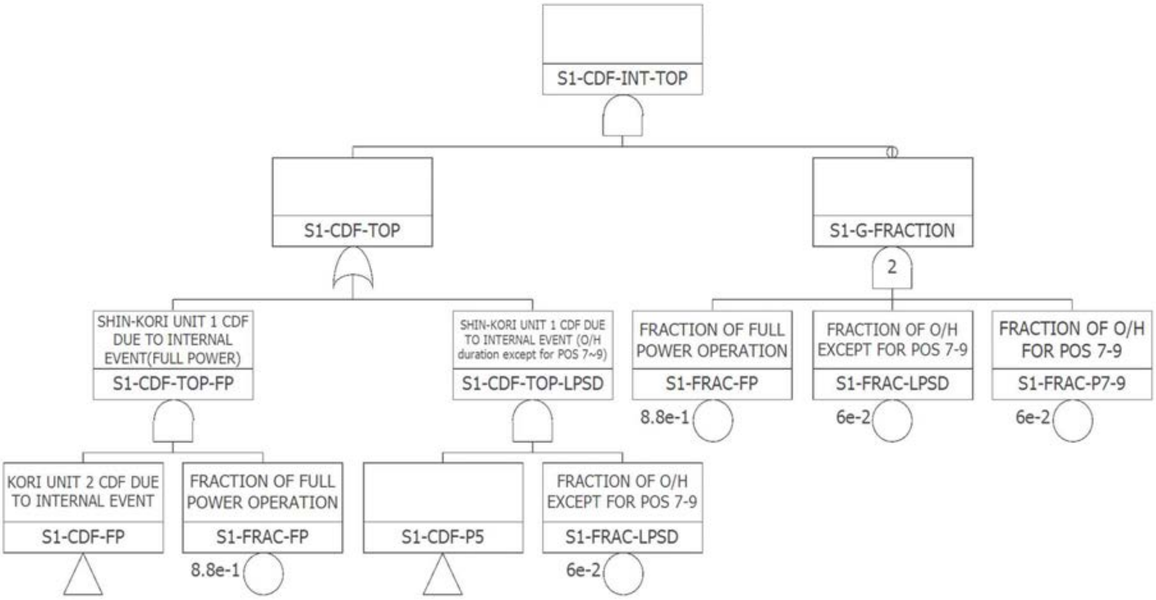


FIG. 49. Single unit one top model.

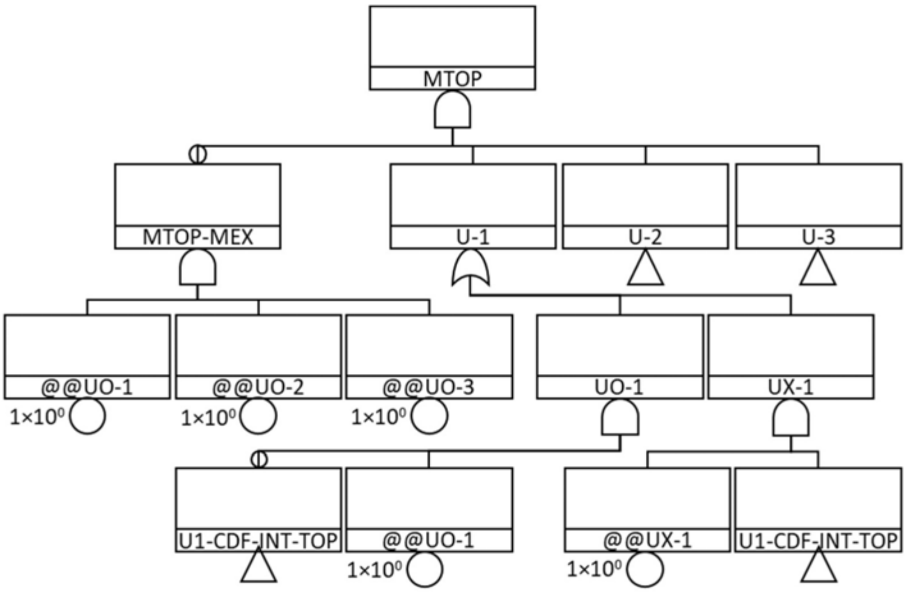


FIG. 50. Concept of truth–table method (3 unit).

The current state of shared component between units is same as KHNP presented. First, even though there are some cases that instrument air system is interconnected between units, it is neglected for a conservative assumption. Intake structure sharing is considered in multiunit loss of condenser vacuum and switchyard sharing is reflected in off site power recovery assuming they are fully correlated between related units. Lastly, shared AAC–DG usage is modelled giving the same credit on the units under the same accident sequence. Therefore, considering the accident sequences and operation state combination in the case of multiunit LOOP, the average fraction of AAC–DG connection to each unit in certain accident scenario is reflected. As none of the data is available for inter unit CCF analysis, the inter unit CCF probabilities were calculated with the developed approach for the inter–unit dependency estimation, if inter–unit CCF event would occur between the CCFs of every related component in a unit. To consider inter–unit dependency in human error, the dependency analysis tree was developed reflecting the characteristics of multiunit accident. The FTREX would be applied for quantification of internal event SRA, providing MCS and calculating CDF. In the case of seismic event, FTREX and binary digital diagram quantification utilizing MCS would be implemented because of non–rare events.

3.3.12. Romania/CNCAN

The MUPSA study is performed using a scoping analysis to define the main aspects of further detailed evaluation of the developing a MUPSA model starting from SUPSA Level 2, as part of an extended sensitive analysis. The methodology starts and uses the existing experience in considering SUPSA Level 2, which embed the SUPSA Level 1 model. The model is built for the multiple sources of the Case A and is using a fault tree format to integrate all the sources. For each source the model develops a master fault tree, as in the IAEA fault tree approach recommended in documents under publication. A representation of the fault tree built to integrate the radiological impact from various sources is in Fig. 51 for single unit and multiple unit models.

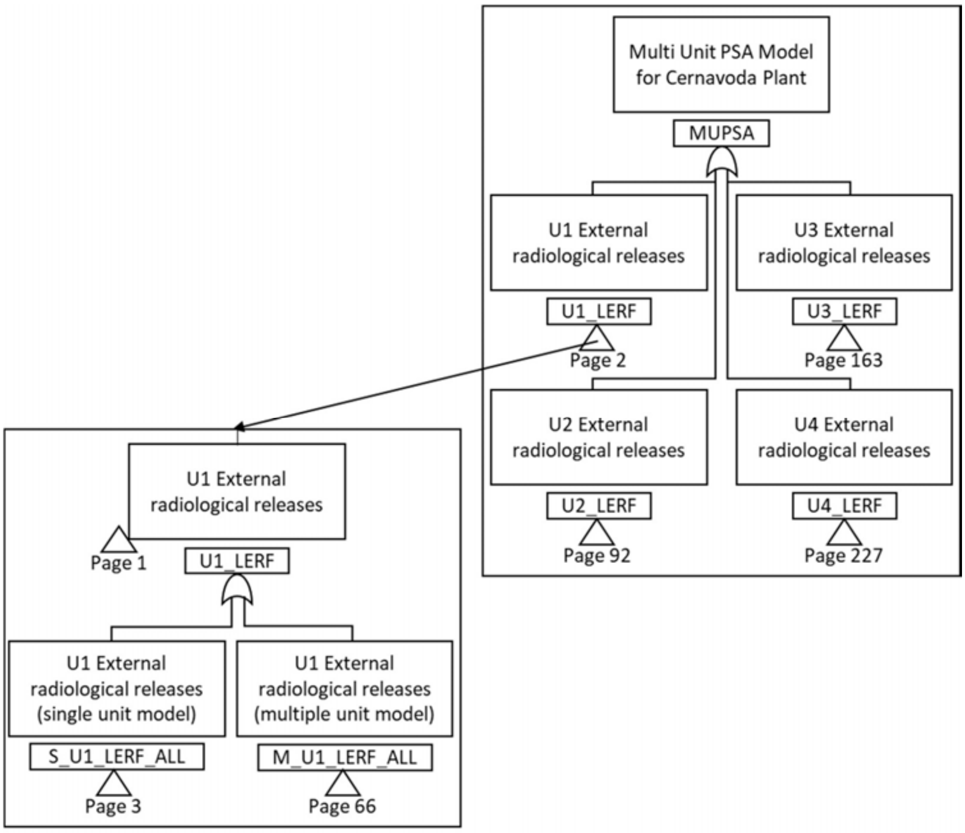


FIG. 51. Step of connecting radioactive releases from various sources.

The main techniques used for building MUPSA model from SUPSA consider the fact that SUPSA are built from the very beginning and/or experience a transformation able to identify the main dominant sequences contributing to risk metrics, defining in such a manner the barriers for various initiating event. These barriers are similar for plant reaction to a multiunit initiating event and are an important assumption for MUPSA. The MUPSA model uses the coding system of SUPSA, adding the mark M for specific multiple impacts (for initiating event and basic events) and the correlations introduced for the POSs and ageing level. Dominant sequences for a given initiating event (loss of class IV, MSLB, general transient) describe the plant reaction and barriers that will be challenged in a similar multiunit initiating event. The model illustrated in Table 24 is implemented in CAFTA and RiskSpectrum as master fault tree, as represented Fig. 51. It is assumed that the barriers identified for SUPSA are the same to similar challenges in MUPSA. as represented in in Fig. 52. Therefore, the master fault tree of the Single unit Case A start point modelling defines the main barriers to the challenges of the plant for the use in using them in the same format in MUPSA model.

TABLE 24. DOMINANT SINGLE UNIT SEQUENCES – BARRIERS IN SUPSA TO BE USED IN MUPSA

TLCLIV sequences/ Function events	**LOOP_BR**	**ESC_HE**	**SOLID_HE**	**FIRE_HE**	**CLASS III_MTCE**	**MKUP_CAL_HE**	**EQ_DOOR**	**SEAL_CC F**	**EQ_LEA K**
1	X	X	X	X			X		X
2	X	X	X	X	X			X	X
3	X		X		X	X	X	X	X

MSLB sequences/ Function events	**EPS_HE**	**ESC_HE**	**FIRE_HE**	**FIRE_TRUCK**	**MKUP_CAL_HE**	**EQ_DOOR**	**SEAL_CCF**	**EQ_LEAK**
1	X	X	X			X		
2	X				X		X	X
3	X					X		X
4	X		X	X			X	X
5	X					X	X	

General transient sequences/ Function events	**LOCLIV_CONSEQ**	**CLASSIIIDG_CCF**	**EPS_HE**	**SEAL_CCF**	**EQ_DOOR**	**SEAL_CCF**	**EQ_LEAK**
1	X	X	X		X		X
2	X	X	X	X	X	X	

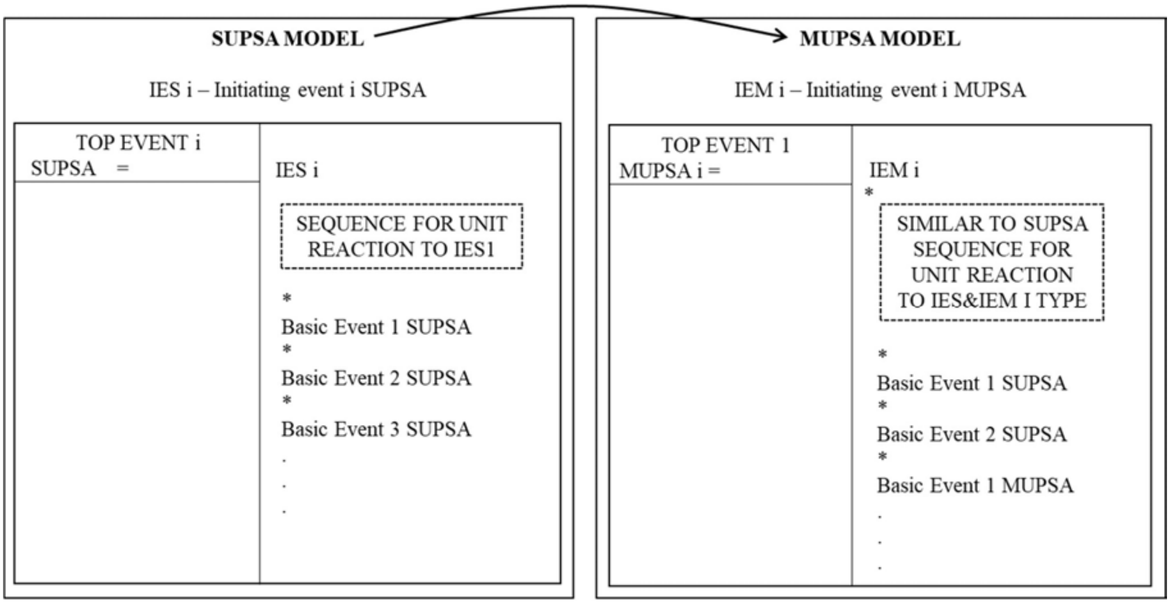


FIG. 52. Transformation of SUPSA model in MUPSA model.

3.3.13. Russian Federation/JSC A

3.3.13.1. Analytical tool

The RiskSpectrum 1.3.2 software is used as the main analytical tool for development of single and multi source PSA models and quantification of all risk metrics. The software was selected for the following reasons:

— The original SUPSA models were developed using RiskSpectrum 1.3.0 version. The latest version RiskSpectrum 1.3.2 is fully compatible with this version. The software allows development of comprehensive Level 1 and Level 2 PSA models. This software obtained verification certificate in Russian Federation;
— The latest version RiskSpectrum 1.3.2 has additional feature extremely useful for MUPSAs purposes. This feature is illustrated on the Fig. 53 and is simply allows to transfer MCS obtained after quantification of core damage (00–CD–UNIT1) from the model for one unit to the events trees of the model for another unit as an input to initiating events heading.

3.3.13.2. Modelling technique

The modelling technique for Level 1 PSA is completely based on the methodology applied for Level 1 PSA for single unit/source. The only, but major difference with Level 1 SUPSA is that input to event trees for the second unit are the MCS obtained in quantification of the associated event trees for the first unit. This technique is illustrated in Fig. 54, where event tree for loss of off site power for 2nd unit model is shown. One can see that event tree for loss of of–site power for the second unit has an input in the form of MCS from consequence analyses case from the quantification of the event tree for LOOP for the 1st unit. The boundary conditions set MUPSA enables the house event MUPSA. This house event is used to exchange

CCF models and human errors probabilities aimed to account for multiunit effects. These aspects are discussed further in this report and multiunit CCF inclusion in the PSA model is shown in Fig. 54. It can be seen that when house event MUPSA is set to TRUE additional multiunit CCFs of multiunit diesel generators are connected to the fault tree of diesel generator failure

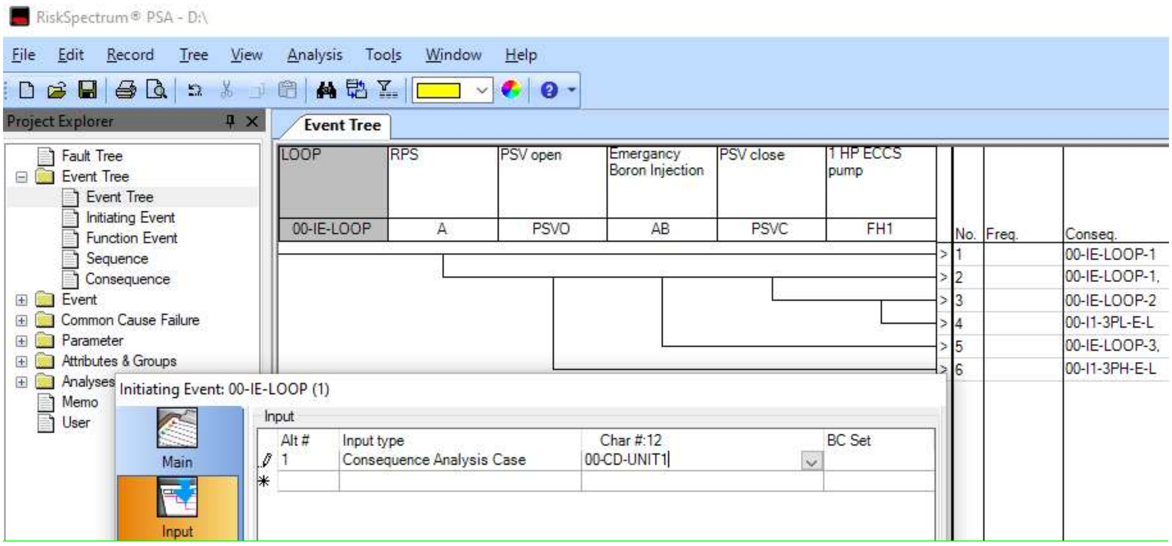


FIG. 53. Illustration of the RiskSpectrum 1.3.2 feature.

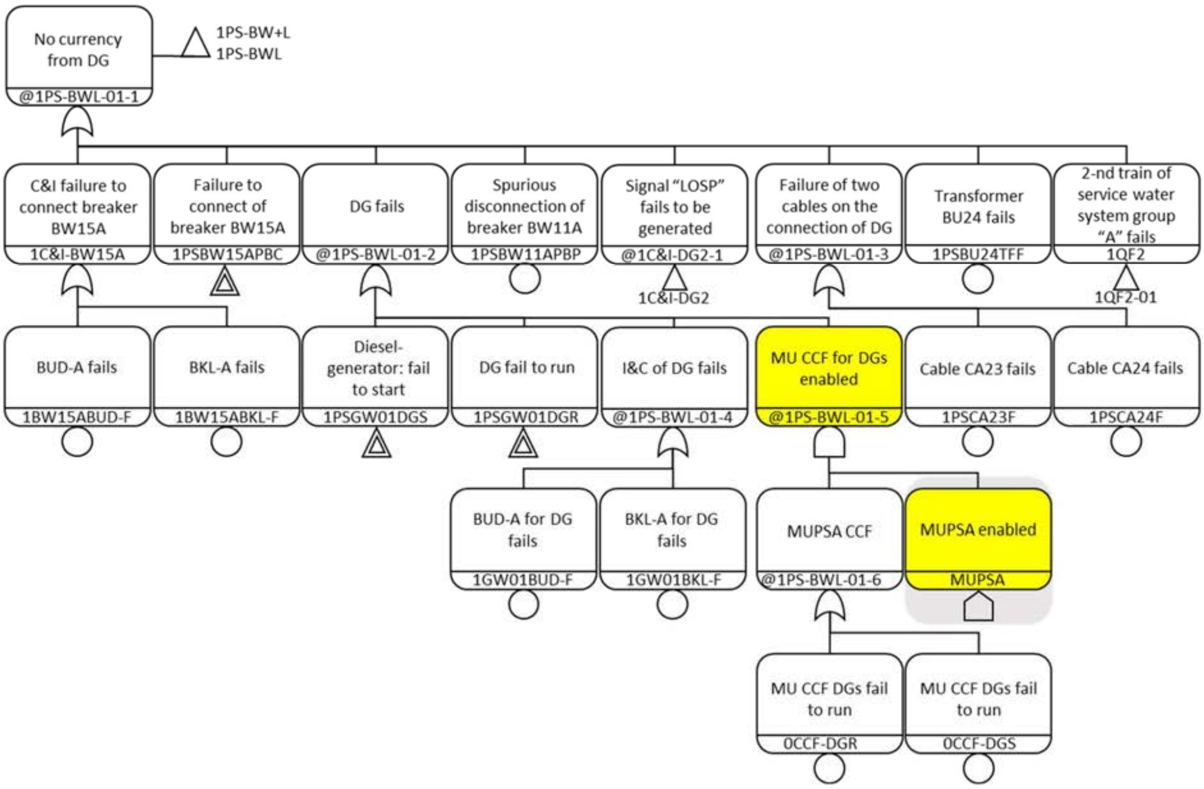


FIG. 54. Multiunit CCF in the PSA model (MU stands for multiunit; DG for diesel generator).

3.3.14. Tunisia/STEG

3.3.14.1. Analytical software/tool

SAPHIRE8, developed by U.S. Nuclear Regulatory Commission, is used to develop a Tunisian PSA model allowing to model NPP's response to initiating events, quantify associated core damage frequencies and identify important contributors to core damage by enabling the user to build event trees and fault trees, and thus to obtain MCS and to quantify the model.

3.3.14.2. Methodology/modelling techniques

The methodology described here includes the following main tasks:

— Plant familiarization and information collection;
— Initiating event analysis;
— Accident sequence analysis;
— Dependent failure analysis;
— System analysis;
— Data analysis;
— CCF analysis;
— HRA;
— Analysis and interpretation of the results (sensitivity studies, importance and uncertainty analysis).

A set of initiating events and hazards (internal or external) to be considered in a PSA is selected based on a progressive and iterative process that includes the following steps:

— STEP1: Selection of initiating events for the internal events PSA (one NPP, all reactor states);
— STEP2: Selection of hazards scenarios for internal / external hazards PSAs (with no correlations between hazards, limited to one NPP, all reactor states):
— STEP3: Selection of combinations of hazards (with all correlations between internal / external hazards / initiating event, limited to one NPP, all reactor states)
— STEP4: Selection of combinations of hazards for a site extended PSA (with all correlations between internal / external hazards / initiating event, all NPPs on a site, all reactor states);

In this study the core damage states are classified into four states:

— First CDS (CD1): state when transient happens without reactor scram or when reactor core is uncovered due to large LOCA; all fuel assemblies in the reactor will be damaged;
— Second CDS (CD2): it is assumed that transient occurs with scram and no means of decay heat removal is available; the fuel in the reactor may be partially damaged due to a lack of primary cooling or natural circulation cooling;
— Third CDS (CD3): state in which unstoppable LOCA happens and where the reactor is successfully scrammed;
— Fourth CDS (CD4): state defined as fuel degradation due to blockage of one fuel assembly

3.3.15. Ukraine/Energorisk

The development of MUPSA model at SAPHIRE–8 was performed as follows:

a) Updating of existent PSA models for each ZNPP unit; since the PSA models for separate ZNPP units use different component reliability and CCF data bases, the one harmonized and actual statistical data (reliability parameters, initiating event frequency) for all PSAs is adopted;
b) Integration of six separate models into one model;
c) Development of MUPSA event trees/functional fault trees for selected initiating event;
d) Preliminary quantification, model enhancement and final quantification.

Multiunit effects are modelled with linked event tree approach, where successful and unsuccessful accident sequences for considered initiating event are linked for several units in such way that the final states of one event tree for the first considered unit becomes input (or transfer) for accident sequences for other units. Linking scheme is presented in Fig. 55 (OK000 means no core damage at all units; CD001–CD010–CD100 designate core damage at one unit; CD011–CD110–CD101 designate core damage at two units; CD111 – core damage at three units).

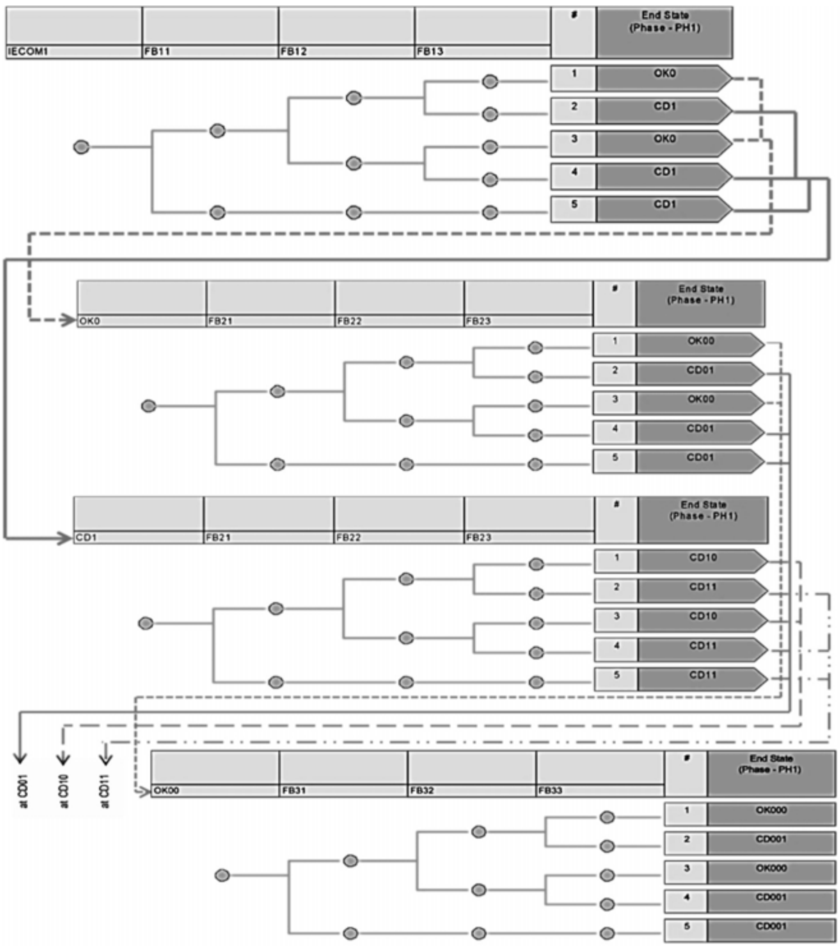


FIG. 55. Illustration of linked event tree approach (example for three units).

3.3.16. Ukraine/ SSTC NRS

3.3.16.1. Verification of the SUPSA models

The CDF was calculated in relation to internal initiating events at reactor full power operation to verify relevant PSA models. The results of model verification confirmed convergence of calculated CDF regarding internal initiating events at reactor full power operation with results obtained earlier.

3.3.16.2. Combination of PSA models

The existing PSA model regarding internal initiating events at reactor full power operation of unit 2 was corrected for the purposes of further combination of the models of units 1 and 2 and estimation of multiunit CDF.

Codes of basic events, house events, top events and logical operators of fault trees (system fault trees, functional fault trees) of unit 2 PSA model were extended with unit identifier. As a result, codes of the specified events received the following view U2–Event_Code.

The PSA models were combined through the integration of basic events and fault trees of the improved PSA model regarding internal initiating events at reactor full power operation of unit 2 into PSA model regarding internal initiating events at reactor full power operation of unit 1. The integration was implemented using built–in means of SAPHIRE 8 computer code applying 'Integrate Project' function.

3.3.16.3. Consideration of interconnections between units 1 and 2 in PSA model

The scope of this task includes the analysis of existing system models (fault trees) with a view to considering the intersystem relationships of units 1 and 2. The analysis shows the need for revision of existing system models (fault trees) to consider interconnections between the units 1 and 2, which are modelled in functional fault trees of T8 initiating event selected for the analysis.

Table 25 summarizes the description of interconnections between units, components and how these are accounted for in the PSA models. The interconnections are related to RNPP 1 and RNPP 2.

The column 'Accounting in individual PSA models' of Table 25 presents the description of the performed improvement of system models. System fault trees, which simulate interconnections between units, were extended with service basic events to exclude mutually exclusive events from the calculation.

These basic events are included in the rules for processing MCS with the aim of subsequent exclusion of illogical MCS containing mutually exclusive events from CDF calculations.

TABLE 25. ACCOUNTING FOR INTERCONNECTIONS OF RNPP 1 AND RNPP 2

No. of units with interconne-ctions	System name, equipment	Description of interconnections between units	Common components	Accounting in individual PSA models	Accounting in combined PSA model
Components common for units 1 and 2	Auxiliary emergency feedwater system, Auxiliary emergency feedwater pump (AEFWP)	System is common for units 1 and 2 consist of: — 2 condensate storage tanks common for two units, — 6 AEFWP, three AEFWP for each unit, — piping and valves. Demineralized water from CST is supplied to the pumps by two collectors DN 350. It is supplied by piping DN 150 to AEFWP suction side. Two AEFWPs are connected each to its own collector. Third AEFWP is actuated upon seismic signal and connected to both channels. Demineralized water is supplied by piping DN 100 to semicollectors of steam generator source point: from AEFWP–2 to the semicollector of steam generator–1,3,5 source point; from 1AEFWP–3 to semicollector of steam generator–2,4,6 source point; from 1AEFWP–1 to both semicollectors.	2 common CST; common piping; common valves with manual drive; system location	RNPP 1,2 models: CST failures are presented by basic event EFWTNK–BZK1–2–L ‘CST–1(2) leak’ in fault tree DWS–100, EFW–100, EFW–200, EFW–300. Failures of valves DW–6, DW–7, DW–8, DW–9 are not modelled, since valves are not equipped with manual drives and are in open position, which eliminates false closing of valves due to the failure of control system components.	CST failures are presented by one basic event EFWTNK–BZK1–2–L ‘CST–1(2) leak’ to eliminate duplication of events.
Common for units 1 and 2	ESWS	Essential cooling system is recirculated with spray ponds and is not connected to other water supply systems. Onsite pump station is common to 2 power units. System within industrial site consists of three independent channels. Each channel includes: — water intake chamber on UPS–1; — one section of the spray pond designed to cool water after loads; — emergency service water storage tank (BTV–1,2,3) designed to fill a channel with water and protect it from emptying and unacceptable interruption in water supply to essential loads; — 4 pumps in each separate compartment of the unit pump station: two pumps per one power unit (operating and redundant) designed to supply water to loads: 2NTO–1 (2) (essential service water pump) – 1st channel, 2NTO–3 (4) – 2nd channel, 2NTO–5 (6) – 3rd channel; — piping and valves, including one supply piping with diameter=800 mm, one drain piping with diameter=900 mm, one piping with diameter=1000 mm for water supply from spray pond to water intake chamber of the pump station. Pump stations of each channel of 2 power units are in separate compartments in unit pump station No. 1. Water cooled in the spray pond goes by gravity through water conduits to the compartments where essential pump groups are installed. The pumps separate chambers 10 m deep on the suction side into which water goes from the spray pond. Water from each pumping installation is supplied by a separate DN 800 pipe to the turbine building, where it is distributed from the main piping by separate piping to power unit 1, 2, and emergency service water storage tanks.	All system components common to units 1 and 2	RNPP–1 model: in ESWS fault trees the failures of pumps 2NTO–1,2, 2NTO–3,4, 2NTO–5,6 are considered. CCFs of 2NTO–1,2,3,4,5,6 are considered. Failures of unit 2 ESWS valves are considered. RNPP–2 model: In ESWS fault trees the failures of pumps 1NTO–1,2, 1NTO–3,4, 1NTO–5,6 are considered. CCFs of 1NTO–1,2,3,4,5,6 are considered. Failures of unit 1 ESWS valves are considered. RNPP–1,2 models: Failures of ESWS system pumps are presented in individual fault trees ESWS–UNIT1–1NTO–1,2,3,4,5,6 and ESWS–UNIT2–2NTO–1,2,3,4,5,6.	Improvement of system models (fault trees) is not required.

TABLE 25 ACCOUNTING FOR INTERCONNECTIONS OF RNPP 1 AND RNPP 2 (CONT.)

No. of units with interconnections	System name, equipment	Description of interconnections between units	Common components	Accounting in individual PSA models	Accounting in combined PSA model
Common for units 1 and 2	Circulation water system, Cooling towers and cooling tower station pumps	The circulating water system is two–channel and includes the following equipment: — circulation pumps (for unit 1 – TsN–1A, 1B, 2A, 2B, and for unit 2 – TsN–3A, 3B, 4A, 4B); — two cooling towers; — cooling tower station pumps (1NG–1A,1B, 2NG–2A,2B); — cooling pump for oil tanks and circulation pump bearings (for power unit 1 – NMO–1 (NMO–2), and for power unit 2 – 2NMO–3 (2NMO–4)); — head and drain circulation water conduits; — valves, — rotating screens of circulation water conduits (1VVS–1(2,3,4)A,B for power unit 1 and 2VVS–3A(3B, 4A, 4B for power unit 2).	cooling towers; cooling tower station pumps; location of circulation pumps of two units in UPS–1	Components, which are common, not modelled in RNPP 1 and RNPP 2 models.	Improvement of system models (fault trees) not required.
Connection between No.1 and 2	Uninterruptible power supply system, Power supply backup both from 1 TR IHBT (in–house backup transformer) and 6TR IHBT	The following was provided within the implementation of C(I)SIP 35206: — possibility for separate operation of 6kV RSh–6kV–A (B) bus bars of backup power supply at unit 1 (from 1TR) and at unit 2 (from 6TR); — flexible and maintenance–friendly 6kV backup power supply circuit, which allows power supply of backup power bus bars of power unit 2 from 1 TR IHBT by switching on Vs–RShA–6kVA, B sectional switches and vice versa, from 6 TR IHBT: provide 6kV backup power supply at power unit 1; — possibility of long term parallel operation of 1TR and 6TR, which allows switching in the 6kV backup power supply circuits of power units 1 and 2 without power interruption; — operation algorithms for 6kV section automatic load transfer (ALT) circuits of power units 1, 2 1RA, 1RPB, 2RA, 2RB, 3RA, 3RB, 4RA, 4RB provide reliable power backup from both 1TR IHBT and 6TR IHBT; — – meeting the main requirements of the conclusions and recommendations of the reports on testing in–house electric motors of power unit 1, 2 in group self–starting mode in terms of the need to install a second backup 6TR power source. Thus, in case of simultaneous loss of power at power units 1 and 2, self–actuation of 6 kV and 0.4 kV in–house mechanisms from 1TR IHBT at power unit 1 and from 6TR transformer at power unit 2 is provided.	Possibility of power supply to backup power bus bars of unit 2 from 1 TR IHBT, and vice versa, from 6 TR IHBT: provide 6kV backup power supply at power unit 1	*RNPP 1 model:* Possibility of power supply from unit 2 considered by simplified fault tree HLPSS–KRU–6–1RA, HLPSS–KRU–6–1RB,HLPSS–KRU–6–1RB–M, HLPSS–KRU–6–2RA, HLPSS–KRU–6–2RB. Failures of 1TR, 6TR IHBT, switches and human errors on arrangement of power supply from unit 2 considered. *RNPP 2 model*: Possibility of power supply from unit 1 considered by simplified fault tree HLPSS–KRU–6–1RA, HLPSS–KRU–6–1RB, HLPSS–KRU–6–1RB–M, HLPSS–KRU–6–2RA, HLPSS–KRU–6–2RB. Failures of 1TR, 6TR IHBT, switches and human errors on the power supply from unit 1 considered. Operational designations of transformers, switchgear 6 and 0.4 kV components of unit 2 do not correspond to principal scheme of in–house power supply (1RA, 1RB, 2RA, 2RB instead of 3RA, 3RB, 4RA, 4RB)	Interconnections between units 1 and 2 for power supply systems were considered in the unified model. RNPP–2 model has corrected identifications of components of the power supply system. Fault trees HLPSS–KRU–6–1RA, HLPSS–KRU–6–1RB, HLPSS–KRU–6–1RB–M, HLPSS–KRU–6–2RA, HLPSS–KRU–6–2RB are improved in RNPP–2 model for compliance of system models with principal schemes of power supply.

TABLE 25. ACCOUNTING FOR INTERCONNECTIONS OF RNPP 1 AND RNPP 2 (CONT.)

No. of units with interconnections	System name, equipment	Description of interconnections between units	Common components	Accounting in individual PSA models	Accounting in combined PSA model
No.1 – No.2	In–house normal power supply, Backup transformers	Normal operation in–house power supply system starts with the devices connecting the generator–transformer to 330/110 switchyard and ends at switch terminals of process equipment loads. The following in–house power networks exist at RNPP: — 6 kV network to energize electric motors with a power of 200 kW and above and 6/0.4 kV in–house stepdown transformers; — 0.4/0.23 kV network to energize electric motors with a power to 160 kW, welding, lighting; — 0.4/0.23 kV network for group 2 uninterruptible power supply loads; — 0.4/0.23 kV network for group 1 uninterruptible power supply loads; — 220 V direct current network. In–house loads of reactor and turbine installations of each power unit are energized from two operating transformers with a power of 25 MV□A, with a voltage of 15.75/6.3–6.3 kV. Transformers are connected by taps between step–up transformers and VGM 20 91/11200 generator switches (rated voltage 20 kV, breaking current 90 kA). Group 3 in–house loads of each power unit are connected to four 6 kV sections and to six 0.4 kV sections, of which 2 are unit sections, 2 sections are of pressurizer heaters, 2 sections are plant. Power supply of 0.4 kV section is performed from dry 6/0.4 kV 1000 kV A transformers. 6/0.4 kV backup transformers of each power unit are connected to 6 kV normal power supply sections of adjacent power unit (unit 1 backup transformer is connected to unit 2 6 kV section), which ensures redundancy if operating 6 kV in–house power sources of this unit are damaged.	– 2 plant sections; – 6/0.4 kV redundant transformers of each unit are connected to 6 kV normal power supply sections of an adjacent unit.	RNPP 1 model: Possibility of power supply from unit 2 to 0.4 kV buses of unit 1 is considered in fault tree HLPSS–RESPS–RSH–04, HLPSS–RESPS–RSH–04–M. RNPP 2 model: Possibility of power supply from unit 1 to 0.4 kV buses of unit 2 is considered in fault tree HLPSS–RESPS–RSH–04, HLPSS–RESPS–RSH–04–M.	Improvement of system models (fault trees) is not required.

3.3.16.4. Analysis of accident sequences (development of event trees)

To calculate multiunit CDF in the occurrence of initiating event T8, the existing T8 event tree was improved taking into account the progression of accident scenarios at units 1 and 2. Besides, the event tree of the event ‘Uncontrolled cooling through BRU–A(K)’ was extended, which is transfer with respect to T8 event tree.

The accident sequences leading to severe fuel damage at units 1 and 2 were assigned the end state of CD12. The event tree was improved in the combined PSA model as shown in Fig. 56.

Table 26 contains a description of the accident sequences of the improved T8 event tree, which are characterized by core damage of RNPP–1 and RNPP–2.

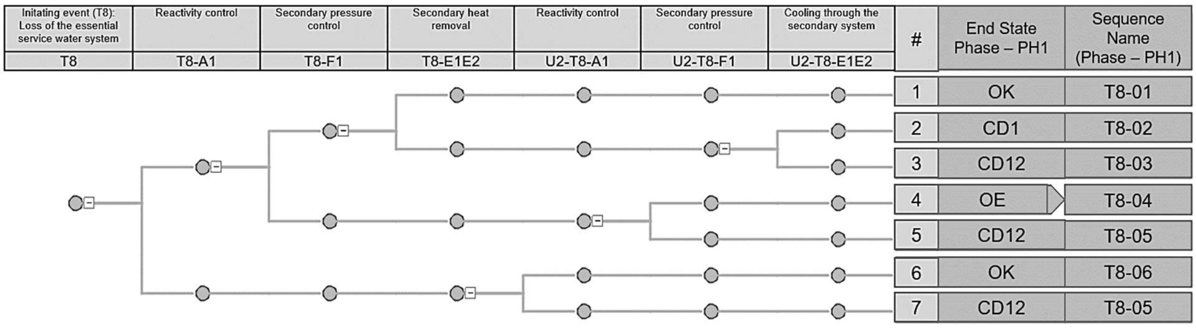


FIG. 56. Event tree for T8 initiating event: loss of the ESWS.

TABLE 26. DESCRIPTION OF ACCIDENT SEQUENCES OF T8 EVENT TREE

Accident sequence code	Accident sequence description
T8–03	Accident sequence is characterized by simultaneous failures of the emergency feedwater system, auxiliary feedwater system and auxiliary emergency feedwater system. Lack of units 1 and 2 steam generator feedwater leads to fuel damage at high pressure in the primary system.
T8–05	Accident sequence is characterized by the non–fulfilment of the pressure control function in the secondary system of units 1 and 2. Failure to close after opening at least of one steam generator steam relief systems leads to uncontrolled reactor cooling.
T8–07	Accident sequence is characterized by the failure of reactor scram. In this case, the requirement for the introduction of boron into the primary system becomes critical. Non–fulfilment of the function to create a shutdown concentration due to a failure of the ESWS leads to core damage at high pressure in the primary system.

As it was previously specified, frequency of the initiating event T8 was calculated by the event tree method for the case of failure of three channels of ESWS considering failures of service water pumps of unit 1 (1NTO–1,2,3,4,5,6) and unit 2 (2NTO–1,2,3,4,5,6). Provided this, the previously obtained frequency of occurrence of the initiating event T8 (equal to $9{,}530\times10^{-7}$) can be assigned to the initiating event of T8 event tree and used for the purposes of evaluating the calculation of multiunit CDF.

3.3.17. UAE/Khalifa University

The SBO event is created when an impacted unit has a LOOP and both of its onsite dedicated EDGs fail to start and run, while an extended SBO for the impacted unit develops when the shared AAC–DG between units as a coping diesel generator for the SBO fails to start and run (or it could be unavailable because it is occupied by another unit that is also under the SBO). After LOOP and SBO, it is expected that one of the turbine–driven auxiliary feed water pump starts and deliver auxiliary feedwater to one steam generator (secondary side heat removal–shutdown heat removal). If the reactor coolant pump is intact, the turbine–driven auxiliary feed water pump successful operation with essential components depends on the availability and support of the DC battery; the lifetime of the battery is ~2 h without battery load shedding and 16 h with load shedding. Depending on the success of load shedding operation, within 2 h or 16 h, the operator is required to remove the non–essential loads from DC battery load to extend its lifetime. To ensure the continuity of the secondary heat removal, the AC power needs to be recovered. In this pilot PRA model, there is no restoration of offsite power after 1 hr. It is considered that the alternative options for AC recovery are EDG crosstie from another unit (if adopted) or other AC sources such as mobile diesel generator. For the EDG crosstie, the operator needs to align EDG from other units utilizing the shared AAC bus to supply AC power to the essential loads and keep the NPP in the safe state; the failure to restore AC power will lead to reactor core damage. The electrical crosstie is a dynamic action and involves operators' actions from two MCRs and two locals' operators with proper procedure and communication. The crosstie steps are expected to be indicated in the relevant plant procedure for the extended SBO and involve the participation of two units (unit with extended SBO and an EDG donor unit) or with more units in case of multi-unit LOOP and SBO. If the crosstie option is adopted for the extended SBO, the traditional way for estimating the extended SBO is to add the crosstie failure to the SBO fault tree and find out the end result of extended SBO frequency, as shown in Fig. 57.

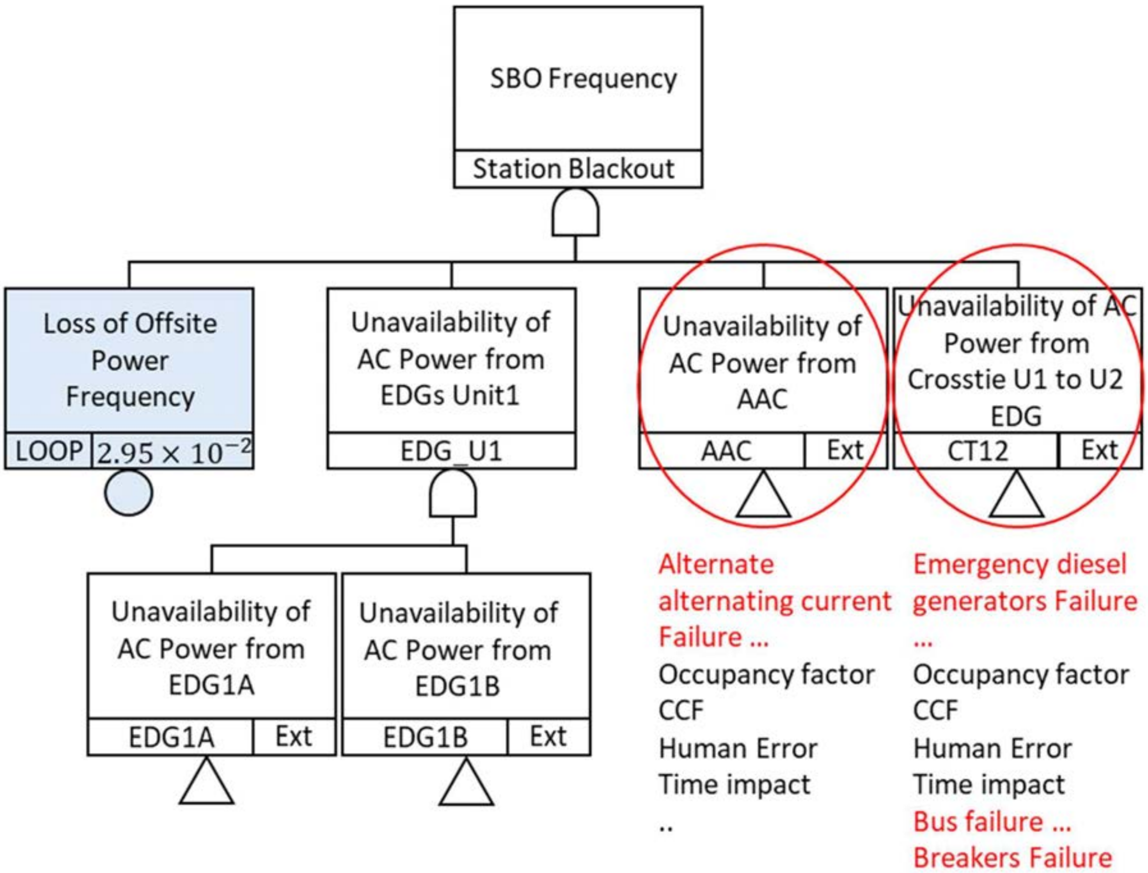


FIG. 57. Station blackout.

The inter–unit impact between involved units needs to be considered such as CCFs (alpha factor of CCF group–HRA) impact and unavailability of shared AC power sources that could be used by other units when it is needed for the impacted unit in case of multiple LOOP and multiple SBO occurred on one site. Re–evaluations of SBO of multiunit event is described in Fig. 58 and Fig. 59 in terms of event tree and fault tree.

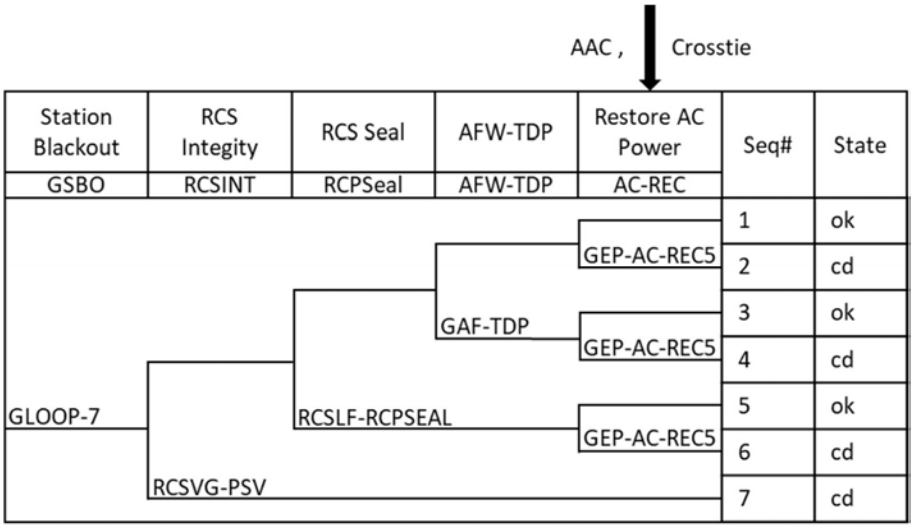


FIG. 58. Re–evaluation of station blackout of multiunit event.

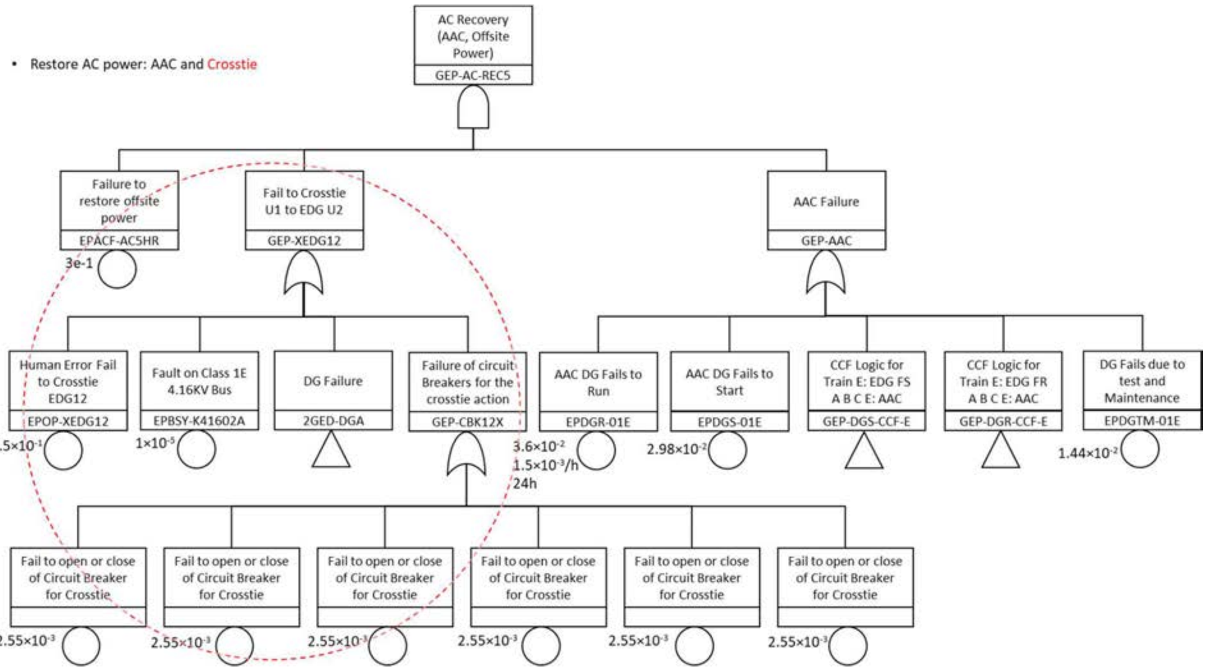


FIG. 59. Re–evaluation of station blackout of multiunit event.

3.4. LEVEL 1 TECHNICAL CONSIDERATIONS

Level 1 PSA for multiunit generally have the same features as SUPSA. However, Ref [20] suggests set of attributes that are specific only for PSA that involve several units in the consideration. These attributes relate to practically all PSA tasks, described in SSG–3 [2]. The most important attributes listed in [20] suggest the following:

a) While selecting POSs in MUPSA it is essential to consider the most likely combinations at all units involved in the assessment. It is suggested that these combinations include at least the one with all reactors in operation at full power and one for each reactor unit in a low power and shutdown POS;
b) When performing multiunit fire or flood PSA multiunit impacts of internal fire and flooding on SSCs and plant initiating events needs to be considered and analyzed;
c) In multiunit PSA i.e. is essential to identify and include in the list of initiating events all initiating events that may impact two or more reactor units concurrently;
d) Accident sequence analyses performed in MUPSA need to consider the various possible responses to initiating events of two or more reactor units and adverse impacts of accident at one unit on other units. This includes consideration of multiunit dependencies associated with the occurrence of an accident on one reactor unit and its impact on the control and accident management in another reactor unit;
e) Single integrated multiunit PSA model is recommended; it accounts for the effect of multiple units being impacted at one time for a single reactor unit PSA and for a multiunit PSA. This model is recommended for quantification as it accounts for the effect of multiple units being impacted at one time for a single reactor unit PSA and for a multiunit PSA (for example, SCDF from accidents involving core/fuel damage on multiple units, whose total frequency is MCDF).

These attributes quite clear and do not require special methods for implementation. However, there are several attributes suggested in [20] that are not evident and require more explanation. These attributes are discussed below in more details.

Common cause consideration in MUPSA: in SUPSA common cause groups are typically developed for identical equipment performing redundant functions. The number of components in the group typically reasonable and it is possible to define justified parameters for the models that are used for CCF quantification. The situation is rather different when CCF groups for MUPSA are defined.

Reference [20] states that for multiunit PSAs the common cause groups are defined such that distinguishes common cause events confined to components within reactor units and those involving combinations of components in different reactor units. For multiunit PSAs special common cause models may be needed to resolve common cause basic events within and between or among multiple reactor units. The reason for this attribute is based on the experience of data analyses for CCFs that shows instances of multiple failures that occur on similar components on different reactor units. If CCF occur in combination with an initiating event that affects several units, it can be large contributor to multiunit risk. Examples of such initiating event are LOOP and CCF failure of all EDGs on more than one units at the site. There could be reasons that support attributes in [20], e.g., common diesel generator design and manufacturer; however, there can also be reasons for dividing EDGs in separate groups if maintenance personnel at different units is different and use different maintenance procedures. In this case at least low coupling factor can be assigned in CCF model for one unit and CCF models for CCF in more than one unit. Moreover, if EDGs at different units are of different design, manufacturer and have different maintenance practice these all is a good reason for dividing them in separate CCF groups.

Correlated hazards are those hazards that either result from the case where the occurrence of the:

— Initial hazard creates conditions resulting in a second hazard caused to occur closely in time; or

— Hazard has multiple manifestations such that secondary effect often accompanies the primary effect.

Hazards that are independent but can occur simultaneously also need to be addressed in MUPSA. An example of an induced hazard is the Fukushima Daiichi NPP accident, where seismic event was followed by a tsunami. Similarly, hurricanes are always accompanied by precipitation and this correlation is considered. Examples of independent hazards that have a high potential to occur simultaneously are any hazards that can occur during a long hot summer period, prolonged external flood, etc. While SSG–3 [2] does not provide recommendations on how correlated hazards are considered in Level–1 PSA and therefore Member States have relatively low experience in such assessment; Ref. [20] suggest that multiunit PSAs need to thoroughly consider correlated hazards if they do have the potential to cause a multiunit initiating event. These correlated hazards are subjected to detailed realistic analysis in multiunit PSA. In the context of MUPSA this assessment is more important than in SUPSA, because even with low frequency (that makes them less important for SUPSA) they have high potential to affect several units at the site and provide high contribution to multiunit risk. Reference [20] focuses on the hazard specific human interactions that have to be defined. For hazards that involve multiunit initiating events and/or accident sequences, interactions associated with the need to manage multiple reactor units need to be considered. The reason is that human interactions, depending on the nature of the hazard, are based on specific operating instructions for the hazard, such as fire response procedures, and therefore may have additional dependencies to be addressed. Also, it is suggested that the definition of each HFE accounts for the scenario specific factors that include the impact of multiunit accidents. Multiunit aspects need to be addressed also while assessing performance shaping factors, where their complexity can be higher if an accident involves several units. These considerations lead to the need to reassess the probabilities of all post initiator human errors modelled in the Level 1 PSA for single units for the specific hazard conditions and multiunit considerations.

3.4.1. Common cause failures

3.4.1.1. Argentina/CNEA

Regarding CCF, they were not modelled in this research project. However, some insights can be made. The practice in Level 1 PSA is to identify (CCFG for the same type of components in a system. For SMR it can be observed that each unit would have the same systems and probably components from the same supplier. Moreover, other coupling factors could be identified. As conclusion, it could be needed to identify CCFG inter units. Then, regarding CCF modelling approach, it is needed to review tools to model CCF in an implicit way. This means that the calculation code can make simplifications or present incongruent MCS.

3.4.1.2. Canada/COG

The Canadian utilities' PSA methodology describes the following dependencies:

— Functional dependencies, that is, equipment failures due to failure of shared sub–systems or common support systems;
— Physical interaction dependencies, for example, a hostile environment which impacts multiple components in a common location;
— Residual CCFs, that is, dependent cause equipment failures not caused by first two items;
— Pre–event human interaction dependencies, that is, undetected prior human errors such as

equipment being left in an incorrect state after maintenance;
- Post–event human interaction dependencies, that is, dependencies between human interactions that are performed close in time and location after the onset of an initiating event.

The modelling of residual CCF in the Canadian PSAs includes contributions from identical components common to all units, e.g. CCF of the Emergency Power Generators at a four unit NPP and contributions from identical components within a single unit, e.g. CCF of the shutdown cooling pumps within a single unit. In the seismic PSA for the multiunit NPPs, identical components on the same unit or on adjacent units are assumed to be fully correlated.

3.4.1.3. China/INET

Both inter–system and intra–system CCFs are considered for the identical components in this study, revealing the following issues:

- In case of MUPSA, the number of identical components increases by 2 times or more. Current CCF parameter databases cannot support the modelling of common cause component groups consisting of so many components. As it is reported that Idaho National Lab is working to scale the CCF group sizes up to 16, this issue is expected to be alleviated.
- Some PSA software has the limitation to model a CCF group with large size. Taking RiskSpectrum PSA software as an example, CCF parameters that the software can accept are beta, gamma and delta. This means that if the CCF group size is larger than 5, a considerable part of the possible combinations cannot be distinguished and are treated as a lumped CCF basic event. However it is not a serious issue because it can be solved by using special model skills, i.e. modelling the CCF basic event directly instead of using the automatic generation function provided by the software.

It seems that with the enlargement of CCF group size, the single reactor risk decreases. It can be explained from the mathematical viewpoint that some of the new CCF combinations don't contribute to either of the reactors. But this observation still needs further interpretation from the CCF mechanism. Therefore, not only CCF modelling methodology but also CCF data collecting/analysing processes need to be revisited.

3.4.1.4. Finland/VTT

The CCFs between all identical components are considered possible. The analysis can focus on the combinations of intra–unit CCFs and single failures that are important in SUPSAs. To estimate inter–unit CCF probabilities, the best option is to use multiunit data. A multiunit impact matrix approach is outlined in [7]. However, no sufficient multiunit data is available currently. Therefore, use of single unit data is recommended. In this approach, it is conservatively assumed that inter–unit CCFs are as likely as intra–unit CCFs. For example, if both of two units include four components, an inter–unit CCF group of eight components is used. CCF data for group sizes up to 8 can be found from some sources, [24]. Based on generic data, the conditional probability of complete inter–unit CCF between all components given complete intra–unit CCF was ~0.2 in case of four components in both of two units.

3.4.1.5. Ghana/GAEC

A methodology for quantifying CCF for MUPSA based on parameter mapping and the Bayesian approach was proposed and demonstrated. Further investigations were conducted to estimate the effect of prior distributions on posterior estimates of alpha factors. The main methodology is presented in Fig. 60. The method commences with data acquisition from CCF databases including the International common cause data exchange and CCF database owned by the USNRC. Information acquired from these databases include alpha factors, CCF events and various component unavailability. Mapping equations were then utilized to map parameters either up or down depending on the component groups for which data is available. Prior distributions were assigned to fundamental parameters usually of the largest component group size for which data is available. The Jeffrey's prior, which is a specialized form of the Dirichlet distribution was suggested in the literature as an appropriate choice for this purpose. Likelihood functions were then defined based on the multinomial distribution for component groups with data available. Bayesian updating was carried out using an appropriate method. In this study, the OpenBUGS program based on the Markov Chain Monte Carlo technique was used. After successfully updating the model to achieve convergence, a posterior distribution was obtained with estimates including updated alpha factors, probability density functions and component unavailability.

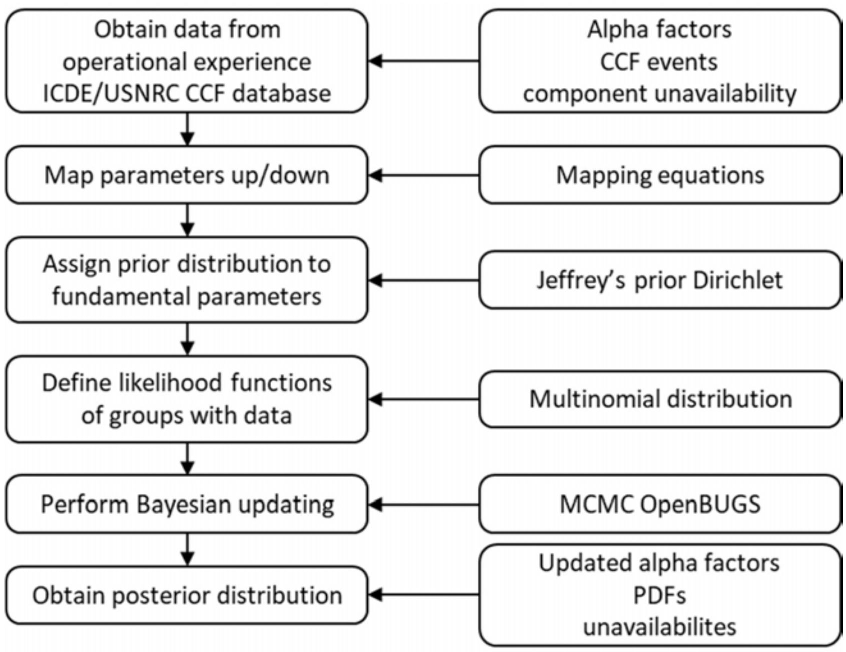


FIG. 60. Summary of methodology for parameter mapping and Bayesian updating (MCMC stands for Markov Chain Monte Carlo).

3.4.1.6. Hungary/NUBIKI

The CCFs as residual failure events not modelled explicitly in PSA can theoretically be extended to CCFs of components belonging to different units. However, the parametric models used generally to describe and quantify CCFs do not seem readily applicable to multiunit CCF events. To this end it is noted that even inter–system CCFs or CCFs of a large number of components are rarely addressed in contemporary PSA. The feasibility study did not endeavour to propose a method to overcome this shortcoming. It is also highlighted that, as generally, intra–unit CCF groups incorporate failures of redundant components, that cannot be the case for components in inter–unit CCF groups. Nevertheless, sensitivity analysis was performed to assess the potential effects of inter–unit CCFs by establishing inter–unit CCF groups to some selected components (e.g. to diesel generators of the same type), and by assigning a beta–factor by one order of magnitude lower than the corresponding one used in modelling intra–unit CCFs.

3.4.1.7. India/AERB

Explicit SSC dependency is modelled through logical fault tree connections. Dependent failures like common manufacture, design, procedures, crews, etc., use parametric models. Generic alpha factor models are used for SUPSA, while from a multiunit perspective, new CCF groups consisting of similar components in different units are considered (parameters need support data from plant experience).

3.4.1.8. India/BARC

The CCFs are considered in system modelling. Both inter and intra unit CCF are modelled using beta factor model. The various identical components in both the units are grouped under several common cause groups. The grouping of the identical components and the value of the beta factor is based on the nature and severity of the hazard. The components which share the same operating environment or failure of components that can induce failure of the other nearby components are grouped together under CCF. In seismic analysis, functional as well as spatial coupling are considered.

3.4.1.9. Pakistan/PAEC

The CCF has been modelled on the basis of identical SSCs for both inter–unit and intra–unit. For intra–unit, CCF model of single unit is used. In the SUPSA model, MGL model is used for CCFGs. It is highlighted here that number of combinations of more than four (04) components increases exponentially for modelling of CCF. Therefore, for the sake of simplicity, only complete CCF basic event of all components failing is added as new CCF group to the existing single unit model, and its probability is assumed to be 0.1 multiplied by the complete CCF probability used in single unit model for components more than four (04). The inter–unit components (vales, pumps) prone to CCF are greater than four (04). Therefore, proposed simplified methodology for inter–unit CCFG is adopted for all components. CCF for shared civil structures has been modelled based on identical structural components in all four shared GIS structures. The identical beams and columns in each shared structure having same design are divided into groups. Different factors important for calculation of median factor of safety in fragility analysis are calculated based on their median value of the weakest element to get a conservative estimate from the statistical analysis of scatter. These values are further utilized for fragility curves development of shared structures for use in MUPSA–1.

3.4.1.10. Republic of Korea/KHNP

As for inter–unit CCF, some important components are selected, such as EDG, battery, auxiliary feedwater motor driven pump, etc. based on Fussell–Vesely importance measures of SUPSA results. To simplify the models, we considered one basic event of inter unit CCF between twin units, another basic event among units of the same reactor types, and the other basic event for all units as shown in Fig. 61. For the preliminary stage, it is assumed the inter–unit CCF multipliers are based on the intra–unit n/n CCF multiplier.

The weighting factors of 0.5 for twin units, 0.25 for the same reactor type units, and 0.1 among all units are applied. In case that the impact of inter–unit CCF to multiunit CDF is estimated significant due to conservative assumptions, the detailed approach is considered. The similarities of coupling factors in hardware/operational/environmental respects, which could cause inter–unit CCF, can be estimated between or among the units. Considering both, the value of (0.5)n and similarity factors could be more realistic alternative when estimating the inter–unit CCF multipliers, where n is the intra–unit common cause component group size [22].

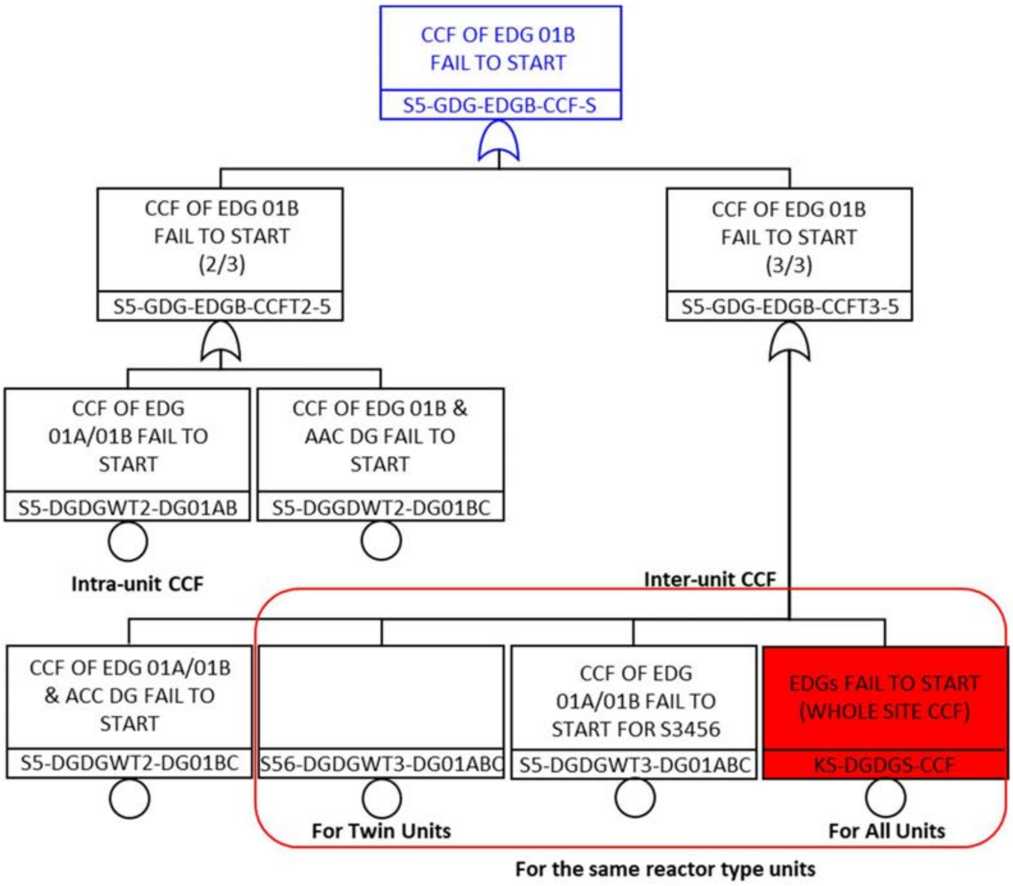


FIG. 61. Simplified modelling approach on inter–unit CCF.

3.4.1.11. Republic of Korea/Hanyang University

To adopt CCF probabilities of SUPSA model, inter–unit dependency analysis methodology was developed implementing Swain's dependency model. Even though inter–unit CCF can be analysed with the same approach applied for single unit CCF, the common cause component group would be too large, resulting in tremendous number of CCF combinations. Also, the data acquired for the SUPSA would be appropriate for estimating the single unit CCF, not for inter–unit CCF. Therefore, inter–unit CCF analysis methodology was developed. The inter–unit CCF probabilities were calculated assuming that inter–unit CCF would occur between the CCF of every related component in a unit. As shown in Fig. 62, the factors are calculated based on the probability of single unit CCF, where, Q_{AB} means CCF probability of SUPSA model for unit 1, Q_{ABCD} inter–unit CCF probability considering inter–unit CCF factor (ρ) and Q'_{AB} means a remaining CCF probability in unit 1 excluding Q_{ABCD}.

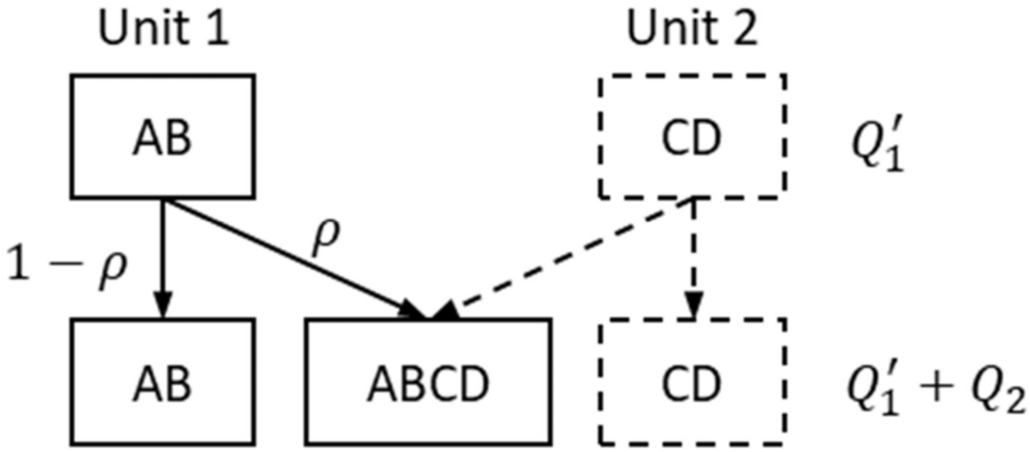


FIG. 62. Concept of the inter–unit CCF modelling.

To determine the factor for inter unit CCF, the decision tree for inter–unit dependency level was developed considering design difference, accordance of maintenance and operation, environmental factor, and the number of components. Using these dependency level, the conditional probability estimated based on Swain's dependency model is shown in Table 27 and Fig. 63. Since analysing all of component's inter–unit CCF combination would be hard, the components analysed are selected from the importance analysis that is conducted on each SUPSA model.

The preliminary SRA model applying the models for inter–unit dependency except inter–unit CCF is quantified, resulting in MCS occurring core damage at more than two units. Among CCF events diagnosed as important, the inter–unit combination of CCF events identified in the MCS are selected for the analysis.

TABLE 27. CONDITIONAL PROBABILITY ACCORDING TO INTER–UNIT DEPENDENCY LEVEL

Inter unit dependency level	**Conditional probability**
Zero dependency	0
Low dependency	$P = \frac{1 + 19 \times Q_B}{20}$
Medium dependency	$P = \frac{1 + 19 \times Q_B}{20}$
High dependency	$P = \frac{1 + 19 \times Q_B}{20}$

*Q_B: Single unit CCF probability of representative unit

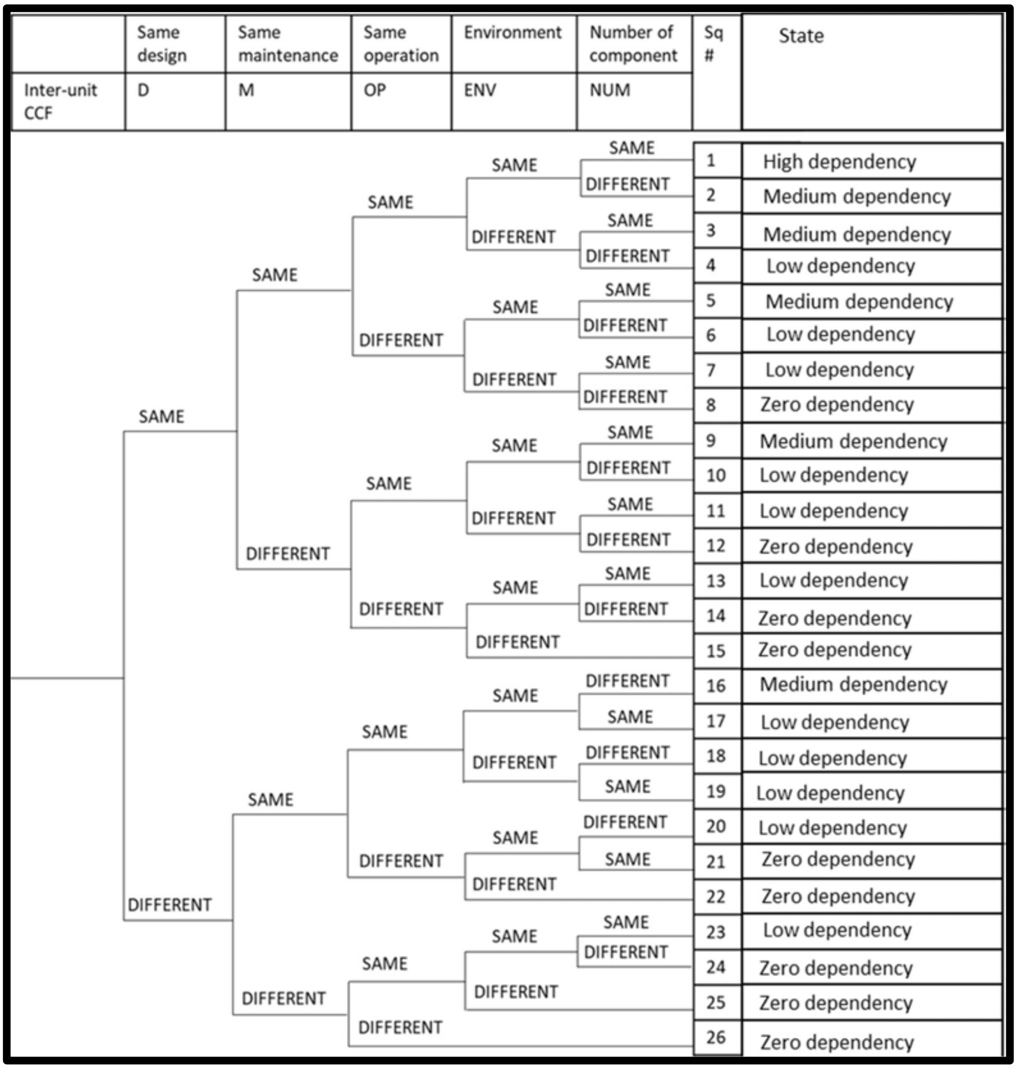


FIG. 63. Decision tree of the inter–unit dependency level.

3.4.1.12. Romania/CNCAN

The Level 1 PSA for units 1 and 2 results in identifying CCFG for the same type of components in a system. MUPSA study is performed using a scoping analysis to define the main aspects related to CCFG embedded in Level 2 and are based on SAMG latest results for the plant and the emergency procedures. Details on the evaluation and of the results are in the chapters related to Level 2

3.4.1.13. Russian Federation/JSC A

The CCF is the most valuable contributor to multiunit CDF. The following issues has to be resolved to determine how to consider CCF for identical equipment at different units of multiunits site:

— CCF at different units are possible not only because of identical design, but also because common maintenance procedures, procurements and safety culture are identical for all units at the Balakovo

site;
- Combining all identical components in one CCF group (using alpha factor model) can lead to excessively conservative results, considering that units started operation at different time. It means that identical equipment is produced with certain time difference that may have impact of CCF coupling factors.

In the Balakovo NPP MUPSA beta factor model may be used; in this approach all CCF groups developed for SUPSA remain, but new CCF basic events activated only for MUPSA quantification are added. This allows to evaluate CDF for both single and multiple units. Considered CCF groups and parameters adopted in MUPSA for Balakovo NPP are shown in Table 28.

TABLE 28. CCF GROUPS AND PARAMETERS FOR MUPSA FOR BALAKOVO NPP UNITS

CCF–group	Description	Parameter (beta–factor)
0CCF–DGR	Failures of diesel generator to run	1.00×10^{-03}
0CCF–DGS	Failures of diesel generator to start	1.00×10^{-03}
0CCF–PSV–C	Failures of PSV to close	2.00×10^{-04}
0CCF–PSV–O	Failures of PSV to open	2.00×10^{-03}
0CCF–QF1–PMR	Failures of service water pumps to run	2.00×10^{-03}
0CCF–QF1–PMSS	Failures of service water pumps to restart	3.00×10^{-03}
0CCF–QF2–PMR	Failures of service water pumps to run	2.00×10^{-03}
0CCF–QF2–PMS	Failures of service water pumps to start	2.00×10^{-03}
0CCF–RL–PMR	Failures of pumps to run	1.00×10^{-03}
0CCF–RL–PMS	Failures of pumps to start	2.00×10^{-03}
0CCF–TQHP–PMR	Failures of pumps to run	1.00×10^{-03}
0CCF–TQHP–PMS	Failures of pumps to start	5.00×10^{-03}
0CCF–TQLP–PMR	Failures of pumps to run	1.00×10^{-03}
0CCF–TQLP–PMS	Failures of pumps to start	5.00×10^{-03}
0CCF–TX–PMR	Failures of pumps to run	1.00×10^{-03}
0CCF–TX–PMS	Failures of pumps to start	2.00×10^{-03}
0CCF–SDS–A	Failure of shutdown system–A to start and operate	5.00×10^{-03}
0CCF–TL10, etc.	Failure of fans to run	1.00×10^{-03}
0CCF–UA	Failure of funs to run	1.00×10^{-03}
Emergency feedwater pumps	Failures of pumps to run	1.00×10^{-03}
	Failures of pumps to start	2.00×10^{-03}

3.4.1.14. Tunisia/STEG

The component groups are identified to incorporate the impact of CCF into systems' reliability models based on the following attributes:

- Component type;
- Component use and function;
- Component failure modes;
- Housing systems' interfaces;
- Component environmental conditions;

— Component maintenance characteristics.

For an appropriate combination of components that share one or more of the above features, the combination is analysed for a potential CCF. The maximum of six members inside one group are considered. This number originates from the design specifications of the system; e.g. each of the coincidence, initiation and actuation circuits is composed of six relays which could malfunction due to a CCF. To handle different levels in redundancy, especially in shutdown systems, the alpha factor model is considered to model the CCFs. Depending on the number of members inside each group and the type of components, the most suitable alpha factors set was retrieved from the U.S. Nuclear Regulatory Commission, CCF Parameter Estimations; where not applicable, the general alpha factors are retrieved from Idaho National Engineering and Environmental Laboratory document: *Common Cause Failure Analysis for Reactor Protection System Reliability Studies* [24] which depends only on the number of components inside a group.

3.4.1.15. Ukraine/Energorisk

Inter–unit candidates for CCFs were selected after evaluation of MUPSA MCS list. For further consideration were selected only dominant MCS with similar equipment from different units. For testing of MUPSA modelling technique and taking into account lack of reliability data for large CCF groups inter–unit CCF for only two units were modelled. Since inter–unit CCF groups contain up to 4 elements (i.e., auxiliary feedwater pumps and valves), contemporary CCF modelling approach and reliability data was used. For inter–unit CCF alfa–model was used instead of beta–factor. Al possible combinations – 2 out of 4, 3 out of 4, 4 out of 4 were explicitly modelled. It is highlighted lack of data and techniques for modelling of CCF combinations for CCF groups with more than six components.

3.4.1.16. UAE/Khalifa University

Similarities of redundant safety systems are contributing to the repetition of system failure if a similar system is failed by a common cause and challenging multiple redundant systems. The CCF is defined as "failure of two or more components during a short period of time as a result of a single shared cause" (ASME/ANS RA–Sb–2013, 2013) and treated as a type of dependency. In a single unit PRA, the CCF is considered as a source of risk for components that are similar within the redundant trains in an intersystem. The CCF for same components is applied in one system and not between different systems in a unit or between similar systems in multi units. The components that are shared by multi units are affected by alike maintenance and test procedures, by similar personnel, as well as by alike environmental conditions, and / or same manufacturing; these similarities may present some additional risks. The inter–unit CCF risks may underestimate the calculated risk of dependency between the units. In the EDG crosstie implementation, there are many common components in the connected AC system. The AAC–DG bus is used as a sharing line to feed the impacted unit from other unit's EDG power supply (non–impacted unit). The common components, such as EDGs from unit 1, EDGs from unit 2, shared AAC–DG, circuit breakers, and others need to be considered when sharing of multi unit features are in between the units. The CCFs of the alpha factor are considered in the crosstie option. The different cases involving AC sources of diesel generators are discussed for a single unit PRA with the impact of the multi unit features. The EDGs include dedicated unit's EDGs, shared AAC–DG and crosstie EDG from other units. The CCFs group is based on the number of diesel generators. The example of the inter–unit CCFs is the modelling by alpha factor as a part of the crosstie fault tree at the Khalifa university PRA model. As shown in Fig. 64. the common cause component group of six EDGs is applied in the fault tree of the EDG crosstie between the unit 1 up to unit 4.

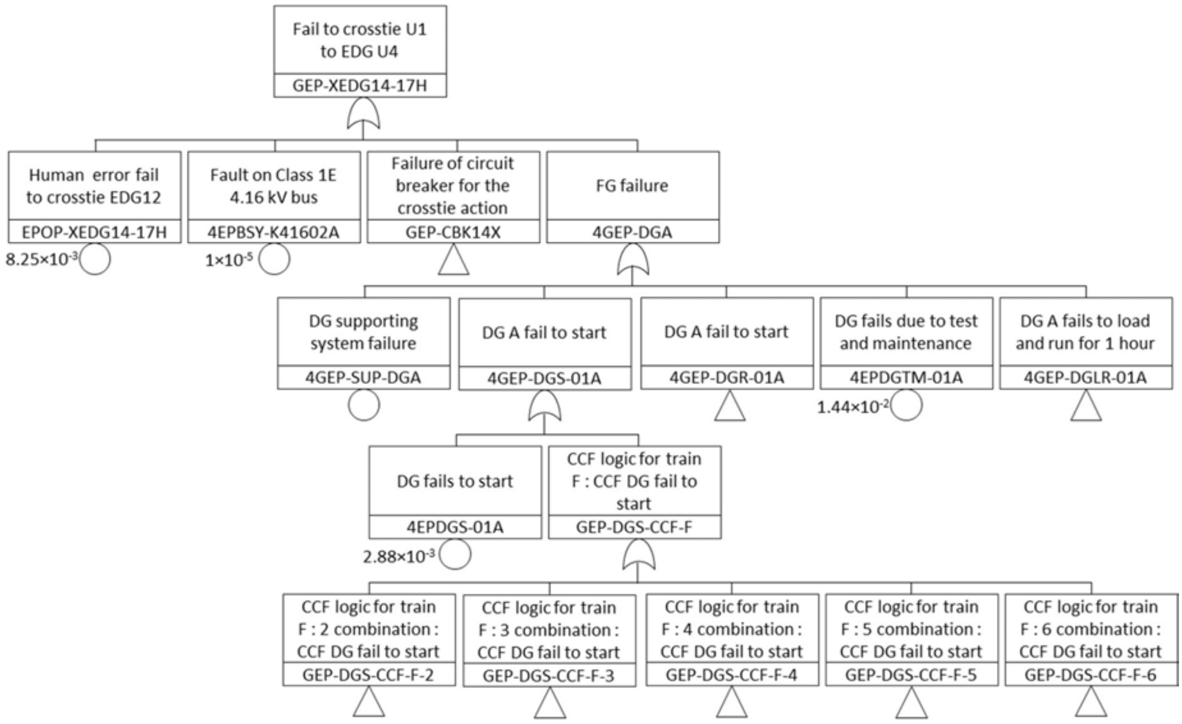


FIG. 64. Fault tree of the EDG crosstie between unit 1 up to unit 4.

3.4.2. Hazard correlations

3.4.2.1. Canada/COG

The Canadian industry practice on modelling the degree to which SSCs need to be treated as having correlated seismic response is, for most SSCs, to consider either fully correlated or fully un–correlated (independent) response. Accordingly, the PSA–based model for PNGS follows the practice to consider the seismically–induced failures as fully correlated for equipment of the same design and located in the areas of the plant with similar seismic response. This is a conservative approach used due to the lack of data for developing justifiable correlation factors that could de–couple the seismic response. Dissimilar SSCs and those with different design, location, elevation, installation (anchorage), orientation, functionality and other attributes affecting their seismic response may be treated as uncorrelated where sufficient justification is provided. To account for dependencies between identical equipment in the same system (i.e., caused by response correlations), one seismic fragility curve and basic event is often used to represent all of the trains (i.e., if one train fails, they all do) containing the correlated equipment (so–called fragility groups which represent failure of groups of SSCs). Structural failures of buildings are conservatively assumed to fail all equipment inside the building.

3.4.2.2. *China/INET*

Preliminary analysis of seismic risk with respect to HTR–PM has been explored. Obviously, earthquakes may cause accidents at the two reactors at the same time. However in the course of our research, we don't think we need to develop new technologies. The way to model the dependency caused by earthquake is generally same as that in single reactor. The hardest part is how to decide the degree of correlated seismic response quantitatively.

3.4.2.3. *Hungary/NUBIKI*

The external events may lead to multiple yet simultaneous failure events at the NPP that can be correlated to certain extent; they may occur within the boundaries of a single unit as well as at multiple units. For example, based on the assumptions and results of fragility analysis in the Paks seismic PSA, it was found that seismic accident sequences of the reactor and the SFP overlap for a unit, and that a number of fragility groups used in the seismic SUPSA need to be extended to contain relevant systems, structures and components of four units. A complete correlation of many inter–unit seismic failures suggests that the cumulative frequency of multi unit seismic induced core damage sequences is close to the single unit CDF. Common to four units, is the damage of a large turbine building complex. In fragility analysis this building complex is considered as a single component, for the probability of total building collapse to be assessed. Accordingly, a seismic failure event that is common to all four units is considered in multiunit seismic PSA; consequential component and system failures need to be identified at each unit to assess the impact of turbine hall failure. If the rigorous and seemingly overly conservative assumption of full correlation is simplified, substantial additional analyses and modifications need to be made to fragility analyses. Comparable considerations hold for other external events, although the level of correlation among multiple failures depends on the type of the external event and the associated loads, on the types and modes of induced component and system failures. Nevertheless, it was decided to perform sensitivity analysis to assess the potential effects of dependence among similar fragility groups.

3.4.2.4. *India/AERB*

The SSC response to extreme external hazard is completely correlated and almost independent for lower values of hazards. Conservatively, it is assumed completely correlated for external hazards if the SSC are located at same elevation or similar locations.

3.4.2.5. *India/BARC*

In the present study apart from the internal initiating events only one external hazard is considered that is seismic hazard. The seismic hazard can initiate several internal initiating events simultaneously in both reactor units as well as SFP. In the preliminary analysis while evaluating the fragility of SSC the conservative approach that is the identical components with same geometry, same orientation located in same elevation are considered as fully correlated and if one component fails all the components are assumed failed during the seismic event. However, depending on the contribution to the overall risk the fragility of the components are further evaluated by applying various dependency and correlation modelling approaches such as multiple integration method, Mankamo model and Reed McCann procedure in conjunction with various numerical methods such as Monte Carlo Simulation, Latin Hypercube Sampling and discrete probability distribution methods depending on the applicability of the methodology and component correlations. In this detailed analysis in dealing with the seismic hazard both the inter as well as intra unit correlations existing among the various identical and unidentical components located at same as well as different elevations have been considered and are modelled using the concepts of multidimensional multiunit correlations. In general, the correlation can exist among the components both from seismic input ground motion as well as response of the components. The schematic representation of the correlations is shown in Fig. 65. Where, ρ_{12i}, ρ_{12L} are intra unit (within same unit) correlation coefficient between two components at same elevation i in unit 1 and unit 2 respectively. ρ_{1ki}, ρ_{2ki} are intra unit correlation coefficient between two components at different elevations k and i in unit 1 and unit 2 respectively. ρ_{12i}, ρ_{12L} are inter unit (between unit 1 and unit 2) correlation coefficient between two components at elevation i and L, respectively.

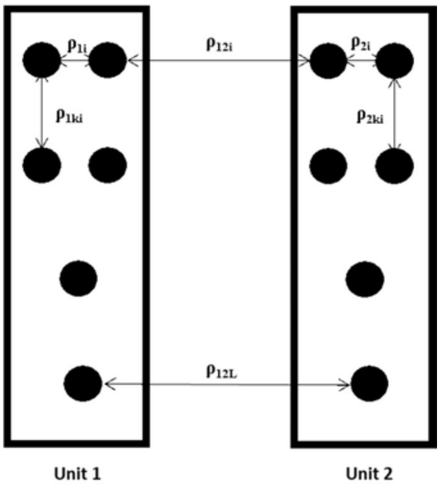


FIG. 65. Schematic representation of multidimensional and multiunit correlations.

3.4.2.6. Republic of Korea/KHNP

As for hazard correlation, KHNP assumed that the off–site power systems of all units are completely dependent against external hazards such as typhoon, heavy snow, and seismic hazard.

As for seismic correlations on other components, we considered only two correlations based on SUPSA for seismic events; one is fully correlated (1.0), the other is zero correlation (0.0). For the same components of the same systems between twin units, we considered those components fully correlated. For other cases, we considered zero correlation.

3.4.2.7. Republic of Korea/Hanyang University

Only seismic hazard is considered currently in the multiunit risk study for Kori site. Seismic correlation of equipment between units is evaluated. However, correlation of intra–unit within a twin unit is assumed fully dependent.

3.4.2.8. Russian Federation/JSC A

Correlation of hazards by itself does not require special attention in MUPSA comparing to SUPSA, unless the units at the site can experience different impact from the same hazard. For example, if one unit at the site has higher elevation, but low wind fragility it may experience structures damage due to high wind and another unit with lower elevation can experience damage from external flooding. If the units are identical in terms of their resistance to external hazards then an external hazards impacts simultaneously both units and the same damage inside units is assumed and modelled, whether from single or correlated hazards. However, there is one hazard that requires special consideration – seismic hazard. Seismic hazard can affect different units located at the site in very different manner. If distance between units is large or even orientation of components is different the same hazard can damage structures and components at one unit but not on the other. Therefore, seismic correlations represent a special subject for the assessment. One of the possible approaches was suggested in the Annex 2 to the IAEA Safety Series Report # 96. However, seismic hazards were out of the scope of current CRP for Russian Federation and will be considered when common understanding on the approach for inter–units seismic correlations is achieved in Member States.

The general approach for consideration of any type of correlations and dependencies accepted in MUPSA for Balakovo NPP based on the information provided above can be summarized as follow:

a) Generally, when internal event occurred at one unit the only dependent LOOP that affect another unit is considered in MUPSA. No other dependent failures or human errors are considered;
b) Only internal events considered in MUPSA for both units are:
    - Administrative shutdown caused by failure of one SWS train. For this event it is conservatively assumed that failure of SWS train requires maintenance work that affects common parts of the system for all units, and therefore the second unit will be shutdowned. In this case both units will experience shutdown with one safety train unavailable at each unit;
    - LOOP was considered as internal event and was modelled for both units as it affects them identically and at the same time.
c) For all cased listed above the following assumptions are made:
    - If external power is lost and not restored for one unit it cannot be restored for another unit;
    - It is not possible to use mobile devices at another unit. It is assumed that they are already occupied for the first unit;
    - Human error at the second unit has no dependency with human errors at the first unit.
    - CCFs of similar equipment at both units are considered.
d) Only event caused by external hazards that can affect simultaneously equipment of both units are considered. This is very specific for Balakovo NPP as no external events have been identified for this plant that has measurable impact on plant safety. In particular the magnitude of the external hazards that can directly lead to core damage at single units is extremely high and has negligible frequency.

The only exception is the seismic hazard. Seismic hazard of high magnitude can damage equipment at all units simultaneously. However, this hazard was not considered in the CRP as currently there is no common approach for seismic fragility correlations for structures and equipment available.

3.4.2.9. Tunisia/STEG

The assessment of a multiunit site against multiple concurrent and correlated hazards has generally not received much attention in previous publications on PSA methodology and was not subject of extensive studies by the nuclear industry or regulatory bodies. Correlated hazards and events are handled by defining major criteria like significance (CA, SE) or frequency (DE, RO). Seismic hazard is one hazard that requires special consideration. Multiunit hazard correlations are expected to affect the final results. The level of correlation is impacted by a number of factors, including the separation between the units, homogeneity of seismic hazard, orientation of the component, and component design. The sensitivity analysis is aimed to check the need of refinement of the fully correlated assumption if the results are impacted by modelling of partial correlations. This sensitivity analysis needs to determine what impact may occur if the correlation refinement is performed. The sensitivity can include a range of correlation, including assuming zero correlation as well as use of a 50% correlation factor. Additionally, this can include implementing these assumed correlations for all fragilities, as well as a smaller subset of risk–significant fragilities.

3.4.2.10. Ukraine/SSTC NRS

This aspect was not included in SSTC NRS analysis for Level 1 MUPSA. Also, consideration of spatial effects (floods, steaming, fire, damage of building structures, etc.), which are a consequence of the development of an accident process on one of the power units and could potentially cause the occurrence of an initiating event at an adjacent unit, are not within the analysis scope.

3.4.2.11. UAE/Khalifa University

At unit 1 with an extended SBO event, the AC power is needed, while the crosstie can be started for the impacted unit. The EDGs in other units may be available or not that could prevent the success of the crosstie initiation. To SBO frequency assessment for unit depends on the LOOP frequency impacting that unit and the failure probabilities of its EDGs and shared AAC–DG. In case of a multiunit LOOP the SBO needs to be re–estimated; this is because the AAC–DG may be in use by another unit who is also experiencing effects of a multiunit LOOP, with loss of all its EDG. LOOP data of LOOP and the conditional probability of all units at a multiunit site with LOOP are used.

Occupancy factor for AAC–DG at multiunit site: an unavailability of the shared AAC–DG due to the occupancy from other units experiencing multiunit LOOP can lead to a failure of EDG crosstie. This is the case of failure of two EDGs from unit 2, and failure of two EDGs can be obtained from the cut sets of the fault tree. The occupancy factor represents the unavailability of AAC–DG (addition to the failure of AAC–DG) and therefore:

- — If the AAC is shared between 2 units, the occupancy factor will be (failure of two EDGs);
- — If the AAC is shared between 3 units, the occupancy factor will be (2 × failure of two EDGs);
- — If the AAC is shared between 4 units, the occupancy factor will be (3 × failure of two EDGs).

Occupancy factor for EDG crosstie from another unit: other unit's EDG may be unavailable to support unit 1 for the crosstie option, if one of the other unit's EDGs is failed. In this case the crosstie is failed and two EDGs from other's units need to be available to crosstie to unit 1 (this is a proposed condition that could be changed). Probability of failure of one EDG at one unit that represents the failure of donner unit EDG can be obtained by the fault tree cut sets as a failure of one EDG. The EDG crosstie is credited from other unit, the occupancy factor will be the failure of one EDG. The occupancy factor of the EDG crosstie is modelled as a failure of one EDG from the donor unit. Both occupancy factors of AAC–DG and crosstie EDGs from other units are estimated and combined into the PRA model within the relevant accident sequences. As shown in Fig.66 the failures of EDGs in the multiunit site are represented as a cascading of the failures from unit 2 to unit 3 to unit 4.

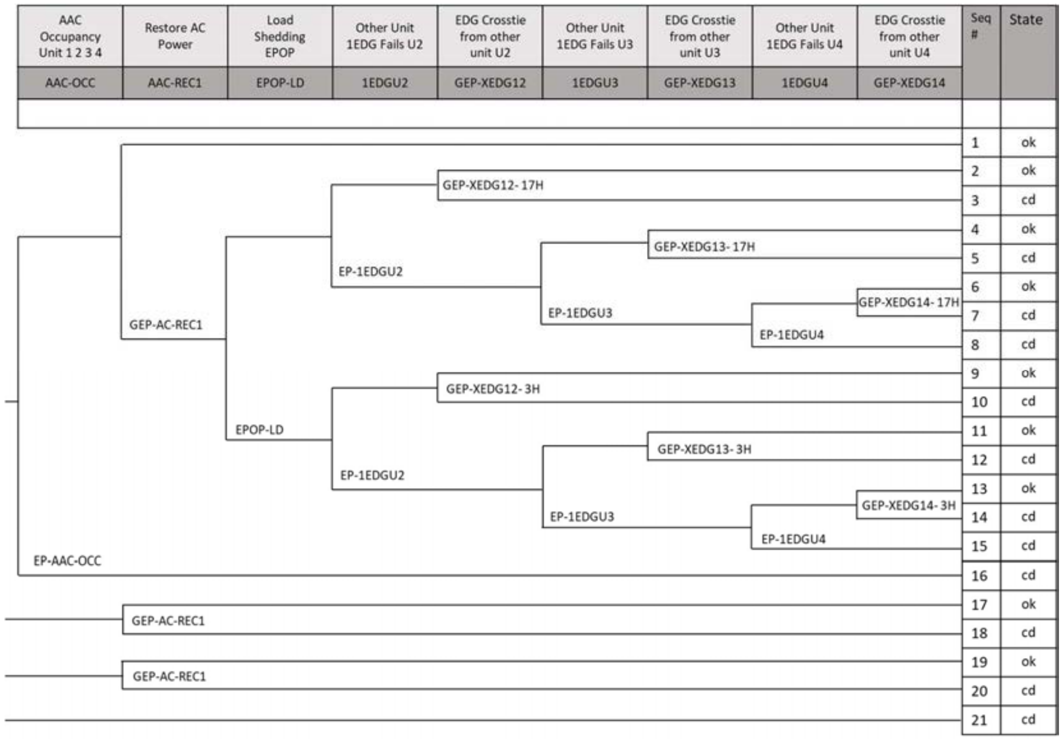


FIG. 66. Cascading approach in occupancy factors from U2, U3 and U4 in MUSBO for unit–1.

3.4.3. Human reliability analysis

3.4.3.1. Argentina/CNEA

The SMRs are expected to be highly automated plants. Moreover, SMRs designs have as feature the use of passive safety systems. An analysis of a complete strategy for controlling initiating events, including the extension of mission time for event sequences, can allow identifying requirements of human actions. In other words, despite of high level of automation and design passive features, some human actions will be required in the medium term. It highlights that passive feature has as advantages a greater availability of time for human actions. As a generalization for SMRs, it is expected that operators in MCR have to monitor and control more than one reactor at the same time. Also, the units can share systems. In the multiunit initiating event scenario where these human actions have to be fulfilled are complex. They imply to develop human actions not only at control rooms but also at field, considering environment severe conditions, more than one unit demands, complex planning and coordination tasks, among other characteristics. Another aspect to consider is that changes in Human Systems Interfaces are being implemented in new designs. It is considered that first generation HRA methods are not appropriated for capturing scenario characterization and the interaction with cognitive functions. Then, a method like CREAM [26] could be adjusted for needed HRA in MUPSA framework. In the framework of the case study, it has developed a preliminary HRA, limited to post–initiators human actions, to highlight methodological aspects regarding HRA in MUPSA framework. In spite of some HFE have been quantified, they have not considered in event trees quantification. It is observed that the CREAM method like others, requires a detail scenario characterization, and operational in emergency information, which is obtained by walk down site. Then, for this case study, hypothesis and assumptions can be used to quantify HFE. As conclusion, regarding the MUPSA model, it highlights those human actions are critical for management of share equipment (DEC systems) among units before fuel damage and during severe accident management. Then, in the framework for SMR multiunit site, HRA is an issue to be analysed.

3.4.3.2. Canada/COG

The PSA methodology already includes modelling of multiunit events. Therefore, the HRA methodology already addresses issues related to staff availability considerations and shared control rooms. Habitability challenges are accounted for in the quantification of the human error probability (HEP). Both pre–initiating event and post–initiating event activities, including those modelled in support system fault trees, are addressed. Firstly, a preliminary, conservative HEP is calculated. The error events are generic meaning that they are applicable to the description of the human interaction in a wide variety of systems. If the preliminary HEP is potentially a significant contributor to risk, a final value is calculated from a detailed model developed by the HRA Specialist using an established methodology such as the Technique for Human Error Rate Prediction (THERP). Post–Fukushima Daiichi NPP accident, an HRA methodology for deployment of emergency mitigation equipment was developed by all Canadian utilities. The SAMGs are modelled only as a sensitivity case. The SAMG needs to consider the possibility of the accident occurring concurrently on more than one unit. The accident unit may be in a different condition from the other units; the SAMG strategies will need to be coordinated to ensure that no negative impact appears between the activities performed at different units.

3.4.3.3. China/INET

The dependency between HFEs in multi reactor accidents was identified and correlated by setting the dependency probabilities of all HFEs equal to 1. This operation can highlight the MCS containing more than one HFE. These HFEs are re–examined from the perspective of multi reactor scenarios. Using the SPAR–H method [12], typical influencing factors, such as insufficient human resources, unfamiliar accident scenario, chaotic working environment, can be associated with performance shaping factors that SPRA–H has adopted, but with the scores reassigned.

3.4.3.4. Finland/VTT

It is anticipated that HEPs can be larger in some multiunit scenarios than in single unit scenarios. There can also be identical events at different units. In that case, the level of dependency needs to be analysed. In the Nordic SITRON project [7], two complementary HRA approaches were proposed:

— Penalty factor method: In a multiunit scenario, an additional multiplier is applied to a single unit

HEP. This multiplier represents the more difficult context of the multiunit scenario compared to a single unit scenario. It is determined based on shared diagnosis personnel between units, shared execution personnel, execution location and shared recovery personnel;

— Dependency approach: Dependency approach aims to consider whether there are shared resources between the unit–related operated actions. A dependency category is determined: zero (no common actors), low (shared recovery), medium (shared diagnosis), high (shared personnel for execution) or full (common action). Probabilities for different dependency categories are proposed in [7].

3.4.3.5. Hungary/NUBIKI

Type A and type B human errors were taken as unchanged from the unit level PSAs as multiunit effects are not considered to play a role in these types of potential human errors. As modelled in the Level 1 PSA for Paks, responses of the plant personnel to NPP accidents are governed by the symptom oriented emergency operating procedures. The responses at each unit are decided and taken by the operators separately. This is mostly true for multiunit accidents too, but it is the responsibility of the shift supervisor to decide on the use of shared resources, and also to help the operators as needed. This decision and the associated human related dependence were included in the MUPSA by considering full correlation of corrective actions and decision–making by the shift supervisor. The PSA model includes some recovery actions that are relevant and common to more than one unit (e.g., recovery from LOOP, establishing plant level island mode operation). These actions were modelled with common model parts, i.e., using the same basic events or fault trees. If core damage occurs at one of the units due to any single unit initiating event, the neighbouring unit or even all other units have to be shut down. In comparison to usual manual shutdown operations the probability of operator errors at those units that need to be shut down may increase significantly under these conditions because of the radiological conditions on the site and the increased stress level in the first place. To assess the effect thereof, sensitivity analysis was performed by changing all operator errors in the PSA model of the unit(s) in need of forced shutdown by one order of magnitude higher than the nominal value as well as by setting all of these HFEs to true.

3.4.3.6. India/AERB

Human factors applicable to severe accident scenario were reviewed for applicability to multiunit PSA and the SPAR–H method [12] was used for estimation of HEPs.

3.4.3.7. India/BARC

Considering the provision of many passive safety features in the AHWR design, very minimal human interactions are anticipated and human error contribution towards core damage for Category 1, 2, 3 and 4 may be small. However, in the present study, for the case of internal initiating events, wherever the human interactions are anticipated, HRA is modelled with human cognitive reliability, Accident Sequence Evaluation Programme and SPAR–H models for calculating Human error in diagnosis and action taken. In the case of seismic event, HEP is modelled considering the severity of the seismic input ground motion such as PGA. In this modelling, HEP increases with increase in PGA value. However, the base case HEP is kept same as internal initiating events.

3.4.3.8. Pakistan/PAEC

From civil structures point of view, under normal circumstances buildings can collapse due to human error within design or construction of the buildings. HRA in structural engineering may be done for the analysis and design process using suitable models. In current study, it is found that high confidence of low probability of failure seismic capacity of GIS shared structures is generally higher than its design PGA values at the review level earthquake higher than design basis, which shows design is adequate as per original design parameters with no human error. Moreover, these structures are constructed under strict QA/QC programme and have sustained many frequently occurring earthquakes therefore independent study on HRA for case structures is not conducted in this study. From MUPSA–1 model point of view as per CRP 20919, no potential dependency found since the operational and maintenance teams are independent for all the units operating at Chashma Site and thus HRA is not explicitly modelled.

3.4.3.9. Republic of Korea/KHNP

The MCRs of all units in Kori/Saeul sites were independently installed and have been operated by independent operators in each unit. So, we have not considered any inter–unit dependencies in operator actions between units. However, we considered off–site power recovery actions totally dependent for all units.

3.4.3.10. Republic of Korea/Hanyang University

The overall approach to HRA is to model the possible human actions in the overall PSA logic. The modelled human actions are then analysed using the SPAR–H method [12] both qualitatively and quantitatively by considering the unique features and timeline of the multiunit events. The quantitative results are fed into the overall MUPSA logic. The SPAR–H method is chosen because of its flexibility for the reflection of multiunit considerations such as adjusting the performance shaping factors levels. A new dependency analysis tree is developed for qualitative analysis with new rules for each branch point to reflect the unique features of the multi unit scenario. The method applied for quantification was based on the SPAR–H method (although other methods may be adopted) but the rules for determining conditional HEPs and joint HEPs were modified. The results so far, show that the proposed methodology can effectively estimate the dependencies in multiunit HFEs, although the accuracy of these results is highly dependent on the correct estimation of the timelines of the modelled events. For analysing HFEs in seismic scenarios, the EPRI approach suggested in 2016 has been applied. The feasibility of HFEs was analysed based on the perspectives suggested in the EPRI approach and the screening tree was applied to assign initial HEPs. When a detailed analysis was necessary, the further analysis using the SPAR–H was performed to produce an HEP.

3.4.3.11. Romania/CNCAN

The Level 1 PSA for units 1 and 2 results in identifying HRA issues and models are included in Level 2 MUPSA model., as presented in the corresponding chapters on Level 2. The model is based on SAMG latest results for the plant and the emergency procedures.

3.4.3.12. Russian Federation/JSC A

Pre–accident human errors dependencies: pre–accident human errors dependencies at different units can be neglected due to the reason that they can be done by the completely different maintenance teams. Note, that at VVER–1000/320 plants each unit has its own maintenance team, and all procedures are developed for each unit individually.

Post–accident human errors dependencies: post–accident human errors dependencies were considered in the assessment separately for actions aimed to mitigate the consequences of the accident and recovery action. In addition separate consideration was applied for actions performed during the accident occurring for the same reason simultaneously at different units and actions performed at another unit after severe accident already occurred at one unit.

Post–accident human errors for actions performed in the accident occurring at the same time at different units prior to core damage: the most common situation that has to be considered in MUPSA is when the accident occurs for the same reason at several units at the site. The best example is LOOP, when all units loose power simultaneously. In this case operators at each unit have to manage the accident independently and it believed that there is no negative or positive impact on human errors that can be done at one unit on human errors at another unit. No single example of dependency between this type of human errors on different units was identified in MUPSA for Balakovo NPP.

Post–accident human errors for actions performed in the accident occurring at the same time at different units after core damage occurred at one unit: situation can be quite different for the case when the events occurring at one unit has led to core damage, but another unit is still in the mitigation process. This situation can happens when due to different component failures or human errors at one unit the core damage was not prevented, but at another unit mitigation is successful. In this case certain impact on the probability of operator errors at the second unit can be envisioned due to:

— Higher stress level after information on the negative accident progression at one unit will be made available for operators at another unit;
— Mobile devices common for several units may not be available for the second unit as they are already utilized for the first unit that experienced core damage;
— Radiation impact on operators of another unit may restrict their flexibility to perform certain actions (for example local actions performed manually in the areas where radiation from another unit can propagate).

The first impact theoretically can be accounted for by increasing stress level for the actions to be performed at another unit. The second and third effect can be simply accounted for by non–crediting mobile equipment and local actions in the potentially high radiation areas at one of the units under investigation. However, both situations require certain modelling technique not currently clear. For example, it is not clear how time dependencies can be considered in the PSA static logic. Currently not all aspects of this type of dependency were addressed in MUPSA for Balakovo NPP and this subject require further elaboration. The only effect that was addressed is the use of mobile equipment: the usage of mobile equipment was disabled in the MUPSA model for the second unit of Balakovo NPP.

Post–accident recovery actions performed in the accident occurring at the same time at different units prior to core damage: only recovery action considered in the MUPSA model is the restoration of off–site power. However, this recovery action was not analysed using HRA methods, but the recovery probability was calculated based on the statistical data on grid recovery in Russia. Therefore, this type of dependences was not changed in MUPSA.

Applicability of HRA methods to multiunit context: no specific features of the HRA methods which are specifically required for MUPSA have been identified. The THERP [27] and SPAR–H [12] methods have been used in both single and MUPSA models.

Timing considerations, impact from affected unit (e.g. radiation impact): issue of proper consideration of timing of the accident and impact of the radiation on the human actions seems to be quite complicated and not only for HRA view. The sequence (timing) of events is difficult or may not be even possible at all to model in PSA static logic.

3.4.3.13. Tunisia/STEG

In Tunisian approach and according to the classification per the IAEA Safety Series No. 50–P–10 an expanded study of the HRA is developed to study possible human actions after various initiating events. The three human actions classified as types A and B (including the operation and training modes), and type C human are considered. The HRA for the Tunisian NPP contains a detailed analysis for all probable human actions of Types A, B and C. The quantification process used to calculate Types A and B human errors probabilities is based on THERP as follows:

- Type A human errors: the probabilities are calculated if these errors lead to unavailability of any system or component during the NPP operation;
- Type B human errors that could lead to an initiating event: the probabilities are calculated for all of them. Given that there is a possibility of a recovery of the error, for some Type B human errors, the recovery mechanism is assumed;
- Type C human errors: the analysis is developed based on the HRA SPAR–H method [12]. The probabilities are calculated for the postulated control room scenario (operator actions) after all initiating events.

Although the Type A human errors do not cause an initiating event, it is important to study them; this is because some errors could lead to unavailability of important systems during accident conditions, thus making the accident progression more severe.

3.4.3.14. UAE/Khalifa University

The PRA is considered as an essential input for regulatory decision making. Therefore, it is essential to have confidence in the PRA results, including associated HRA from their scope and quality. In other words, the impact of human actions needs to be reflected in the risk assessment, quantified and modelled properly in the PRA. The HRA is a systematic method by which the estimated HRA and HEP are quantified for the credited mitigation actions. The quantification of the HEP requires an HRA event tree to be developed, with identified HFEs for the corresponding tasks. The quantification continues with qualitative evaluation of those factors that are influencing human actions. The identified HFEs are to represent what is the contribution of human errors resulted from human actions and leading to failure of a function, system, or component with the impact of the dependency and any recovery actions. Development of HRA requires inputs from the NPP PRA and design, operations, procedures, thermal hydraulics, ergonomics (in–out control room), and cognitive sciences. Many HRA methods are adopted for human error quantification able to implement different approaches based on published HEPs data. The EPRI approach (EPRI/NRC, 2012), representing a combination of two methods, the cause based decision tree methodology for the cognitive part and the THERP for the execution part of task based actions, is used to estimate the HEPs. These methods can bring an adequate resolution to meet the internal events PRA modelling of human failures (EPRI/NRC, 2012). The operator errors are divided into two categories: (a) cognitive part that is related to detection, diagnosis, and decision making and (b) the execution part related to manipulation or implementation failure, an error of omission and error of commission. The combination of cognitive and execution parts estimate the total HEP as follows:

$$HEP = P_C + P_e \tag{19}$$

Effect of DC battery load shedding on HRA: for the impacted unit with an extended SBO, the extension of DC battery life is important factor required to accomplish critical safety functions needed for core heat removal by the secondary system. The implementation of the safety functions is related to essential components, instrumentation, and controls powered by DC battery for a limited period before the AC power or crosstie are restored. In case of LOOP, the EDGs from the impacted unit would start automatically; if both EDGs fail, the operator will actuate the shared AAC–DG; if it also fails, the operator will start DC battery load shedding. The load shedding is achieved by removal of non–essential loads to extend DC battery life from 2 h to 16 h until AC is recovered or the EDG crosstie from another unit implemented. The operator has only a certain available time to carry the EDG crosstie operation; thus, the success of battery load shedding has important impact to that action. Given the load shedding and EDG crosstie are human actions, they both are associated with the probability of failure. Therefore, the HRA needs to be conducted to estimate the HEPs for both situations of the success of load shedding (16 h available to crosstie) and failure of load shedding (2 h available to crosstie). The results of HRA analysis for both cases are discussed in the following section. The HEP of human action of AC recovery from offsite power within 1 h has the same value as used in the KAERI pilot PRA model. The HEPs estimation is carried out using the EPRI approach (cause based decision tree methodology + THERP methods) for different human actions in relation to SBO and extended SBO as following:

- Restore AC power from AAC (Operator Fails to Actuate and Provide Power to Class 1E 4.16 KV Switchgear);
- LOAD shedding (operator fails to shed non–essential load after SBO);
- Crosstie – (operator fails to align EDG crosstie within 3 hr (2 + 1));
- Crosstie – (operator fails to align EDG crosstie within 17 hr (16 + 1)).

3.5. RESULTS OVERVIEW

Table 29 provides relevant information on the multiunit sites and the Level 1 MUPSA scope, while Table 30 is an overall summary of the CRP results of the Level 1 MUPSA analyses from all participants. The following sections discuss each participant's results and their concluding findings.

TABLE 29. SUMMARY OF NPP SITES AND MUPSA LEVEL 1 ANALYSIS SCOPE

MS/Org.	Plant & Site Information			Initiating Event		
	NPP Site / Reactor Type	#NPP Units / Total site power (MWth)	SFF (SFP/ SFB)	Hazard type	Events	Frequency (1/yr)
ARG/ CNEA	Hypothetical site / CAREM type SMR	2 / 200		External hazard	LOOP	2.5×10^{-2}
CAN/ COG	Pickering (PNGS) / CANDU	6 / 10 460		Internal events, internal fires & floods, seismic, high wind	LOOP, LOCA, SSLB, fire, flood, seismic and high wind	1×10^{-2} ~ 1×10^{-5}
CPR/ INET	Shidao Bay NPP / HTGR-PM SMR modules with shared turbine	2 / 500		Internal Seismic	LOOP	1×10^{-2}
FIN/ VTT	Forsmark (Sweden) / BWR	2 / 6200		External hazard	LOOP	N/A
	Ringhals (Sweden) / PWR	2 / 5550		External hazard	LOOP	N/A
GHA/ GNEC	Hypothetical coastal site / same design PWR	2 / ~7200	2 SFPs	Defined internal	LOOP	1.1×10^{-2}
HUN/ NUBIKI	PAKS / VVER–440	4 / 5940		ALL	LOOP, internal flooding, and fire, seismic, wind, snow, ice, river contamination.	10^{-5} ~ 10^{-2}
IND/ AERB	Kakrapar / PHWR	2 / 1490		Internal events and External hazards	LOOP Seismic induced LOOP LOUHS	
IND/ BARC	Tarapur hypothetical coastal site / AHWR	2 / 1840	1 SFP	Internal Events External Hazards (Seismic)	LOOP, LOCA, SWS	10^{-4} ~ 10^{-2}
PAK/ PAEC	Chashma NPP (CHANUPP) / PWR	4 / 4100		External hazard (seismic event)	LOOP due to seismically induced collapse of shared switchyard buildings	8.5×10^{-5}
ROK/ KAERI	Kori site / Westinghouse PWR (3), OPR1000 (2)	5 / 10 500		External hazard: typhoon, snow, lightening, marine life	LOOP, general transient, loss of circulating water	4.4×10^{-3} ~ 2.2×10^{-2}
	Saeul Site / APR1400	4 / 15 930		External hazard: seismic and tsunami		seismic / tsunami hazard analysis curves
RUS/ JSC A	Balakovo NPP / VVER–1000/320 (2/4 included)	2 / ~6000		Internal event	Internal LOOP of both units, Loss of SWS, LOOP induced by grid disconnection of one operating unit	10^{-6} ~ 6.8×10^{-2}
TUN/ STEG	Proposed Skhira Site / ACP 100 SMR	3 / ~1000		Internal event	LOOP	2.0×10^{-2}
UKR/ Energo	Zaporizhzhya (ZNPP) / VVER–1000/320	6 / 18 000	6 SFPs	Internal events	Loss of essential service water system	2.41×10^{-6}
UKR/ SSTC	Rivne (RNPP) / VVER440/B–213 (2) VVER1000/B–320 (2)	4 / ~9000		Internal events, external hazards except seismic	Multiunit LOOP	9.04×10^{-7}
UAE/ KU	Barakah NPP / APR1400	4 / 15 930		Internal events	LOOP SBO	

TABLE 30. SUMMARY OF MUPSA LEVEL 1 ANALYSIS RESULTS

MS/Org.	MUPSA Model Approach	SUPSA Result (Single unit–CDF)	Dependencies	Screening Criteria	MUPSA Risk metric and frequency (1/yr)	Safety Goal(s)
ARG/ CNEA	Integrated event tree (reactors and SFP), simplified fault trees.	After 24h: 5.6×10^{-11} After 48h: 2.3×10^{-7}	Shared SSCs	Not applied.	multiunit CDF: After 24 h: 8.4×10^{-19} After 48 h: 2.1×10^{-8}	Not analyzed.
CAN/COG	Integrated event trees, fault trees with simplified models for other units, with MCS manipulation	N/A	Shared SSCs, CCF, HRA and fragility	Human interaction	SCDF: 0.1×10^{-5} to 1.8×10^{-5}	$10^{-}4$/yr/reactor & hazard
CPR/ INET	Event tree and fault tree	N/A	Shared SSCs, CCF, HRA and fragility	Quantitative	multiunit CDF/SCDF ≈ 0.05	Accumulated frequency of exceeding 50mSv
FIN/VTT - Forsmark	Single unit MCS combined	N/A	Identical components, shared SSCs, human dependencies	Dependency of maximum contribution <1x10^{-8}/year screened out	multiunit CDF/SCDF ≈ 0.1	Not analyzed.
FIN/VTT - Ringhals	Event tree, CCDP for scenarios calculated using single unit models	N/A	Identical components, shared SSCs, human dependencies	Dependency of maximum contribution <10^{-8}/year screened out	multiunit CDF/SCDF ≈ 0.02	Not analyzed.
GHA/ GAEC	Integrated Fault & Event Trees with unit–unit dependencies	6.5×10^{-5}	SSCs, Inter unit CCFs		SCDF: 3.9×10^{-4} /site–yr	SCDF<$N\times10^{-4}$ /site yr
HUN/NUBIKI	Event tree linking approach by consequence event trees and ET conversion to FTs	1×10^{-6} (LOOP) 1×10^{-7} (Loss of power due to onsite causes) 5.0×10^{-6} cascading	Shared SSCs; Inter unit CCFs, Type C human errors -shift supervisor Fragility-sensitivity studies	Inter–unit CCF = 10*intra–unit; HRA: shift supervisor: full correlation; independent crews in independent MCRs	multiunit CDF(LOOP) in 2 units: 5×10^{-8}/yr 3 units: 4×10^{-10}/yr 4 units: 2×10^{-11}/yr multiunit CDF(Loss of power due to onsite causes) in 2 units: 1×10^{-11}/yr Cascading effects: 2 units 1.9×10^{-10}/yr–1.5×10^{-7}/yr	N/A SUPSA CDF<10^{-4}/yr
IND/ AERB	MU event trees, CCF model and fault trees		Common SSCs and HRA	Qualitative and quantitative (risk importance)	multiunit CDF	
IND/ BARC	Fault tree Event tree Markov models Fragility analysis PSHA		SSCs, fragility, inter unit CCF modeled with β factor method,	Engineering analysis	Single unit CDF is 47% of SCDF	–
PAK/ PAEC	Integrated event trees, fault trees with simplified models for other units, with MCS manipulation	C1: 3.0×10^{-4} C2: 8.7×10^{-5} C3&4: 7.4×10^{-6} (Plant level)	SSCs CCFs Fragility	Functional and Physical dependencies analysis	CDF: 1.8×10^{-4} 1/yr Multiunit CDF:8.0x10^{-8} 1/site–yr single unit CDF:7.3×10^{-4} 1/site–yr SCDF: 7.3×10^{-4} 1/site–yr	

TABLE 30. SUMMARY OF MUPSA LEVEL 1 ANALYSIS RESULTS (CONT.)

MS/Org.	**MUPSA Model Approach**	**SUPSA Result (Single unit–CDF)**	**Dependencies**	**Screening Criteria**	**MUPSA Risk metric and frequency (1/yr)**	**Safety Goal(s)**
ROK/ KAERI	Single Top Fault Tree		Shared SSCs, inter–unit CCF, seismic correlation	Conservative engineering judgment, operating experience, references to SUPSAs	Multiunit CDF to SCDF: 2.0% (SCDF ~10^{-5} /site–yr) Multiunit CDF to SCDF: 0.2% (SCDF ~10^{-8} /site–yr) Multiunit CDF to SCDF: 0.7% (SCDF ~10^{-7} /site–yr) Multiunit CDF to SCDF: 49.2% (SCDF ~10^{-5} /site–yr) Multiunit CDF to SCDF: 2.0% (SCDF ~10^{-8} /site–yr)	
RUS/ JSC A	Small event tree with large fault trees	Ranges from 2.3×10^{-8} to 4.5×10^{-6}	*Depending on analyzed scenario.*	–	Multiunit CDF ranges from: 5.9×10^{-11} to 4.1×10^{-8} (details and uncertainty ranges given in Table 37)	10^{-5} 1/year
TUN/STEG	Explicit event tree	3.85×10^{-6} for CD1, 4.43×10^{-7} for CD2, 5.00×10^{-6} for CD3 1.00×10^{-2} for CD4 For single unit	Diesel Generator System shared among unit 1, unit 2 and unit3 External Water Supply System for DEC	Functional and Physical Dependences Analyses	multiunit FDF	Individual radiological risk
UKR/ Energorisk	Small fault trees, large event trees OR Large fault tree,, Small event tree		Identical components, proximity	Dependency of maximum contribution <10^{-12}/year screened out	Reactor cores: Multiunit CDF: 8.03×10^{-8} SFP: Multiunit FDF: 7.05×10^{-9}	10^{-4} N/A
UKR/ SSTC NRS	Event Tree, Fault Tree; MCS analysis		Shared SSCs and support systems, identical components, proximity dependencies	Criteria based on Fussell–Vesely and RAW	Multiunit CDF: ??? For SUPSA 10^{-4}	
UAE/KU	Event tree and fault tree. Accident sequence development.		Shared AAC s, , electrical crosstie (EDG) identical components, proximity, HR will be modelled		Single unit CDF with multiunit features	

3.5.1. Summary analysis

3.5.1.1. Argentina/CNEA

The Level 1 MUPSA risk metric multiunit FDF has been calculated for case study, considering LOOP as IEM. The obtained multiunit FDF is 2.1×10^{-8}/yr, which considers fuel damage in more than one radioactive source in Stage 1 or Stage 2. As reference, for LOOP multiunit initiating event the core damage frequency (CDF) calculated for single unit (unit 1 or unit 2) is 2.3×10^{-7}/yr, which includes fuel damage in Stage 1 or Stage 2.

Regarding multiunit FDF, it highlights that the higher contributions are due fuel damage in at least one unit during Stage 2. As result of the analyses, it was observed that the sequences with higher importance in the multiunit FDF are those related with shared systems during Stage 2.

— Multiunit FDF considering fuel damage in unit 1 and 2 only during Stage 1, with or within fuel damage in SFP: 8.4×10^{-19}/yr;
— Multiunit FDF considering fuel damage in at least one unit during Stage 2, with or within fuel damage in SFP: 2.1×10^{-8}/yr.

Equivalent results are obtained for CDF (single units). It was also observed that including SFP in the analysis is relevant. As conclusion, based on the passive characteristics of the systems operating during the grace period (Stage 1) and the defined strategy to control initiating events for this case study, for SMR MUPSA would be necessary to extend the mission time beyond the first Stage (grace period stage). Besides, the multiunit FDF is increased considering extended mission time. It is important to mention that for the case study conservative screening values were used for Stage 2 systems fault trees quantification. They are calculated using the best estimate approach.

3.5.1.2. Canada/COG

Caution needs to be exercised with any form of numerical risk aggregation for a multiunit NPP. With this consideration, the SCDF for a 4 unit CANDU plant (Pickering B) for a given hazard was calculated as follows:

Site SCDF or LRF = 4 × single unit SCDF/LRF + 2 × two unit SCDF/LRF + 1 × four unit SCDF/LRF. where,

— Single unit SCDF or LRF is a subset of the per–unit SCDF/LRF that includes initiating events for which only a single unit is affected (i.e., reference unit only);
— Two unit SCDF or LRF is a subset that includes accident sequences where two units are simultaneously affected, i.e., the reference unit + one other unit;
— Four unit SCDF or LRF is a subset that includes initiating events that affect all units simultaneously; the three unit sequences are very few; lumped with four–unit cases.

Based on the above aggregation approach, SCDF for various hazards for the 4 unit Pickering B site was calculated and the results are as follows:

— At–power internal events: 3.2×10^{-6}/yr;
— At–power fire: 1.7×10^{-6}/yr;
— At–power flood: 7×10^{-7}/yr;
— At–power seismic: 2×10^{-7}/yr;
— At–power high wind: 3.8×10^{-6}/yr;
— Outage internal events: 2.4×10^{-6}/yr.

The LRF results for the 4 unit Pickering B site are as follows:

— At–power internal events: 6.0×10^{-7}/yr;
— At–power fire: 8×10^{-7}/yr;
— At–power flood: 5×10^{-7}/yr;
— At–power seismic: 2×10^{-7}/yr;
— At–power high wind: 2.1×10^{-6}/yr;
— Outage internal events: negligible.

Analysis of loss of heat sink at the IFBs indicated that the time to reach boiling was greater than 72 hr. Consequence assessments, both deterministic and probabilistic led to the overall conclusion that the risk associated with the IFBs is negligibly low LRF is of the order of 10^{-9}/yr. For the UFDS, since there is no additional containment for the dry storage containers, a direct containment bypass or failure is always assumed in case of failure of a UFDS. Thus, to release ^{137}Cs, which is the radionuclide of concern for the LRF in a PSA, the fuel would need to be melted. There were no hazards identified that could result in melting of the used fuel. From PSA perspective, therefore, risk from the dry used fuel storage can be neglected.

3.5.1.3. China/INET

The INET's research interests are the RCs with possible consequence exceeding 50mSv, i.e. SU LARGE and MU LARGE. None of these accident sequences with a large release category is identified with frequency above 1×10^{-8}/site year. Hence, it can be concluded that the risk of HTR–PM NPP under the situation of LOOP is low. The results based on the pilot study with LOOP are shown in Table 31. Therefore, it leads to the following conclusions:

— Multi module risk holds a certain proportion (5.64%) to the whole site risk;
— When simply doubling the single module risk, i.e. LARGE frequency obtained from the single module based PSA, the frequency part of the site risk is somewhat overestimated, however the consequence part is underestimated;
— Inter module CCF acts as a dominant dependency, contributing to most of the multi module large MCS.

Despite of the low risk figure, the relative contribution of multi module characteristics is of interest. It can be derived from the quantification that single module LARGE frequencies contribute 94.36% to the total site large release frequency, and multi module one takes the remainder (5.64%).

With respect to the multi module LARGE frequency, the statistics shows that:

— Contribution from the MCS containing 1 inter module CCFE = 97.86%;
— Contribution from the MCS containing 2 inter module CCFEs = 1.42%;
— Contribution from the MCS containing Bes shared between both modules = 2.88%.

Therefore, it leads to following conclusions:

— Multi module risk holds a certain proportion (5.64%) to the whole site risk;
— When simply doubling the single module risk, i.e. LARGE frequency obtained from the single module based PSA, the frequency part of the site risk is somewhat overestimated, however the consequence part is underestimated;
— Inter module CCF acts as a dominant dependency, contributing to most of the multi module large MCS.

TABLE 31. RESULTS BASED ON THE PILOT STUDY

Release Category		Description	Frequency (1/yr)	Individual Effective Dose (mSv)
1	OK12	Both reactors are successfully protected from the initiating event LOOP	4×10^{-2}	
2	F–0EXP	Filtered release into the environment under 0EXP cases	4.25×10^{-8}	0.23
3	F–1EXP	Filtered release into the environment under 1EXP cases	ε[8]	4.31
4	NF	Non–filtered release into the environment	1.87×10^{-8}	10.88
5	2EXP	Safety membrane is subject to multiple challenges	ε	13.86
6	SU LARGE	Bounding end states for extreme combination of postulated failures in one unit	ε	>50
7	MU LARGE	Bounding end states for extreme combination of postulated failures in more than one unit	ε	>50

3.5.1.4. Finland/VTT

The conditional probability of a core damage in one unit given a core damage in the other unit was around 0.1 in the Forsmark pilot study and 0.02 in the Ringhals pilot study. The majority of the multiunit risk was related to the loss of power supply to core cooling systems. House turbine, EDGs, gas turbines and mobile diesel generators were among the most important components from the multiunit core damage risk point of view, and recovery of offsite power had also some significance. Ringhals units include steam driven feedwater pumps to backup diesel generators, which decreases the multiunit risk compared to Forsmark. The SCDF is almost the sum of single unit CDF. Risk importance results are very similar at the site level as in the SUPSAs.

[8] Estimated result lower than 1×10^{-8} is marked as ε in the table.

3.5.1.5. Ghana/GAEC

The following key results are obtained after quantifying the fault trees and event tree with CAFTA–6:

- Frequency of 2 unit core damage without the incorporation of unit unit dependencies (common cause (both intra–unit and inter unit CCF) and causal) = $[6.52 \times 10^{-5}] \times 2 = 1.30 \times 10^{-4}/yr$;
- Frequency of 2 unit core damage with common cause (inter–unit CCF) and causal dependencies between units = $3.92 \times 10^{-4}/yr$;
- Contribution from the MCS containing CCF = 92.47%;
- Contribution from MCS containing basic events shared between both units = 39.56%
- Contribution to site core damage from multiunit core damage = 0.93%;
- Contribution to site core damage from single unit core damage = 99.07%.

Table 32 shows MUPSA summary results from literature in comparison to this study. The model used in this study is simplified in comparison to others; the methodology can be applied to include more NPP systems in a full scale study. Summaries of CCF quantifications is shown in Figs. 67 and 68.

TABLE 32. SUMMARY OF MUPSA STUDIES

	Site risk metric	**Model**	**No. of initiating events**	**Single unit contribution**	**Multiunit contribution**
This work	SCDF	Simplified 2–unit PWR	1MU	99.07%	0.93%
Le Duy et al.	SCDF	Simplified 2–unit NPP	1MU; 1SU	99.5%	0.45%
Seabrook	SCDF	Realistic 2–unit LWR	4MU; All SU	92.59%	7.44%
Modarres	SCDF	Conceptual 2–unit NPP	1MU; 2SU	83.74%	16.26%
Zhang et al.	SCDF	Realistic 2–unit HTGR	1MU	94.36%	5.64%

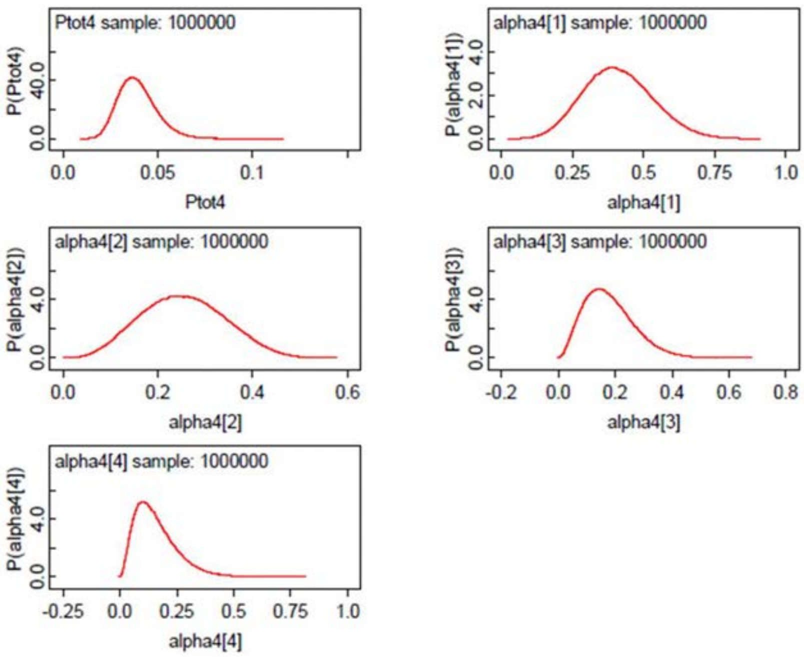


FIG. 67. Posterior distribution of alpha factors in 4 component group for emergency diesel generator fail to load and run failure mode.

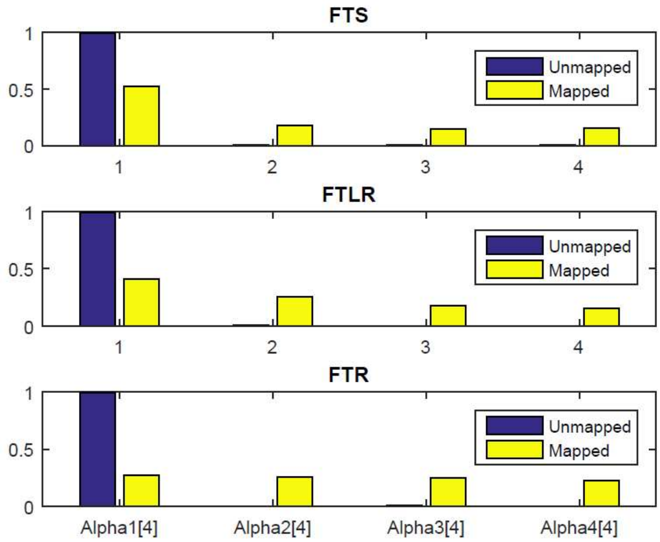


FIG. 68. Mapped and unmapped alpha factors for 4–component group and 3 failure modes of EDGs.

The use of alpha factor method of estimating CCF has certain advantages including the possibility to differentiate partial failure of component groups, which is not possible with the use of the direct method of calculating CCF estimates. Additional insight gained from the results is that the distribution of prior alpha factors is more skewed compared to the posterior estimates, which are approximately bell shaped. Furthermore, the Jeffery's prior distribution was observed to produce longer confidence intervals of alpha factors compared to the normal Dirichlet prior. Finally, the study confirms the suitability of Mapping up or down of alpha factors for application to MUPSA with good approximations of actual failure probabilities of components.

3.5.1.6. Hungary/NUBIKI

For benchmark purposes, risk has been preliminarily quantified for a limited number of multiunit initiating events. Model development and risk quantification of internal and external hazards is still ongoing and is planned to be finished during the coming years of the CRP. The frequencies of those event scenarios were quantified when a single unit internal initiating event evolves to core damage at power operation at one unit (unit 1), and core damage occurs at the neighbouring unit (unit 2) during the subsequent forced shutdown. The results of the sensitivity analysis performed by changing all type C operator errors related to unit 2 by one order of magnitude higher than the nominal value as well as by setting all of these HFEs to true, are presented hereby (the values reflect the frequency of accident sequences leading to core damage at both units):

— Probabilities of type C human errors unchanged: 1.9×10^{-10}/yr;
— (Nominal values)×10 assigned to type C human errors at unit 2: 1.5×10^{-8}/yr;
— All operator actions fail at unit 2: 1.5×10^{-7}/yr.

LOOP induced single as well as multiunit risk results are presented in Table 33.

TABLE 33. MULTIUNIT RISK RESULTS FOR LOOP INITIATING EVENT

No. of simultaneous CDs	CDF (inter–unit CCF neglected)	CDF (inter–unit CCF considered)
1	1.0×10^{-6}/yr	1.0×10^{-6}/yr
2	7.0×10^{-9}/yr	5.0×10^{-8}/yr
3	4.0×10^{-11}/yr	4.0×10^{-10}/yr
4	3.0×10^{-13}/yr	2.0×10^{-11}/yr

Core damage at twin units induced by the initiating event ‘loss of power due to onsite causes’ was also assessed. The frequency of the twin–unit initiating event, i.e. simultaneous loss of power due to onsite causes, is 1.7×10^{-3}/yr, however, the frequency of core damage at both units is as low as 1.0×10^{-11}/yr, even if inter–unit CCFs are considered.

3.5.1.7. India/AERB

The single unit CDF is 7.1×10^{-8} for LOOP event. The LOOP event is directly affecting both units concurrently, hence analysed for multiple unit also. The fire water system is the only common system but can cater to the demand of both the units simultaneously. Fire water system has three diesel driven fire water pumps along with an electrically driven fire water pump. Due to the LOOP, the frequency of event involving core damage of both the unit is 9.1×10^{-12}. The main contributors are dependent human error in crash cool down of reactors and subsequent injection of fire water into steam generators, and the CCF of fire water pumps.

3.5.1.8. India/BARC

As part of the Level 1 PSA work has been carried out both for internal as well as external event (seismic) for both reactor cores. Level 1 internal event PSA of advanced reactor has been relooked and finalized. Level 1 external event (seismic) PSA has been completed for the advanced reactors. Primary and secondary seismic event trees have been developed and dominant accident sequences have been identified. Seismic hazard curves have been developed for the site under consideration. Seismic fragility curves have been developed at the component level and later propagated to system level by using seismic fault trees. Seismic CDF has been estimated by convoluting seismic hazard curves and seismic fragility curves of the dominating accident sequences. The summary of the results is shown in Table 34.

TABLE 34. RESULTS FOR SITE CDF

S. No.	Consequence	Description	% Contribution
1.	CD1	Core Damage in Reactor 1	47% of Site CDF
2.	CD2	Core Damage in Reactor 2	47% of Site CDF
3.	CD12	Core Damage in both Reactor 1 & 2	6% of Site CDF

Following are the specific conclusions from the SCDF point of view.

- LOOP is the main contributor towards core damage frequency
- Contribution for combined core damage frequency (CD12) is mainly due to the common initiating event (LOOP, pump house structural failure, turbine building structural failure).
- Major contribution towards SCDF is coming from earthquake PGA level more than 0.5g.

3.5.1.9. *Pakistan/PAEC*

The single unit CDF for the selected scenario i.e. LOOP based on SUPSA Level 1 study is 1.83×10^{-4}/y. The analysis of current study shows that multiunit CDF is about 8.01×10^{-8} per site per year due to multiunit LOOP. The single unit CDF for the selected scenario is about 7.3×10^{-4} per site per year. It shows that there is a sharp decline in the CDF when considering multiple units concurrently. To validate the obtained results, literature from various research papers was studied and it was observed that contribution of multiple units concurrently in core damage at site level is minimal. Furthermore, SCDF is calculated using CDF and multiunit CDF which is about 7.31×10^{-4} per site per year. The contribution of single unit CDF and multiunit CDF in SCDF is about 99% and 1% respectively. It clearly indicates that contribution of multiunit CDF in SCDF is negligible in comparison with single unit CDF. This study also shows that multiple of single unit specific CDF which is 7.32×10^{-4}/y ($4\times1.83\times10^{-4}$/y) in case of Chashma site is comparable to SCDF (7.31×10^{-4} per site per year).

3.5.1.10. *Republic of Korea/KHNP*

As reviewing all possible initiating events, which could have an impact on multiunit risk, we developed Level 1 MUPSA models for multiunit LOOP, multiunit general transient, multiunit loss of circulating water, multiunit loss of essential service water caused by seismic induced tsunami event and seismic event. As for the seismic induced tsunami event and the seismic event, we performed hazard analysis and applied probabilistic hazard curves to estimating the initiating event frequencies. For other initiating events, we estimated multiunit initiating event frequencies by using Jeffrey’s non informative Bayesian update method based on operating experiences. The evaluation of the MUPSA models, which consider nine units, quantified multiunit CDF and SCDF, and, as shown in Table 35, identified that multiunit general transient, multiunit circulating water and seismic–induced tsunami event has no significant impact on multiunit risk.

TABLE 35. RESULTS OF LEVEL 1 MUPSA

	IEM Frequency (per site year)		**SCDF (per site year)**	**Contribution of multiunit CDF to SCDF**
	Site wide initiating event	**Plant wide initiating event**		
Multiunit LOOP	2.18×10^{-2}	1.31×10^{-2}	Order of 10^{-5}	2.0%
Multiunit General transient	1.31×10^{-2}	4.79×10^{-2}	Order of 10^{-8}	0.2%
Multiunit LOCW	4.36×10^{-3}	4.79×10^{-2}	Order of 10^{-7}	0.7%
Tsunami Event (Multiunit LOESW)	Seismic induced tsunami hazard curve		Order of 10^{-8}	2.0%
Seismic Event	Seismic hazard curve		Order of 10^{-5}	49.2%

3.5.1.11. *Republic of Korea/Hanyang University*

The preliminary SRA model for internal events considering inter–unit dependency was built and quantified (truncation limit= 1.0×10^{-13}). As shown in Table 36, most of the SCDF was induced by the core damage accident scenarios in one unit and multiunit CDF takes ~0.77% of SCDF.

TABLE 36. PRELIMINARY QUANTIFICATION OF 9–UNIT SRA FOR INTERNAL EVENT

No. of damaged unit	Initiating Event				Site (%)
	Single unit initiator (%)	CCI LOOP (%)	CCI loss of condenser vacuum (%)	CCI General transient (%)	
1 unit	71.1	27.7	0.2	0.2	99.2
2 units	–	0.77	<0.1	<0.1	0.77
3 units	–	<0.1	–	0.0	<0.1
4 units	–	<0.1	–	0.0	<0.1
5 or more units	–	0.0	–	0.0	0.0
Total	71.1	28.4	0.2	0.2	100.0

Most of multiunit CDF was induced by CCI–LOOP and the rest of CCIs were insignificant. The case that more than 5 units got core damage could not be found because of truncation limit. The SRA model for seismic event is under development.

3.5.1.12. Russian Federation/JSC A

Table 37 presents the results of preliminary assessment, limited to the modelling of initiating events listed in the table, along with 95% uncertainty limits on multiunit CDF. Frequencies of some of these events were not always obtained based on detailed assessment but assigned based on expert judgment. This was done intentionally to obtain information needed to facilitate further activity in more efficient manner.

Table 37. SUMMARY OF BALAKOVO NPP MUPSA LEVEL 1 ANALYSIS RESULTS

MUPSA scenario model and input	Single unit CDF (1/yr))	Dependencies	MUCFD and uncertainty (1/yr)
LOOP–MUPSA Input: Consequence analyses case 1–LOOP	5.5×10^{-7}	Inter–unit CCFs, HRA (non–crediting LOOP recovery and use of mobile equipment for 2nd unit)	**4.1×10^{-8}** 5%: 3.9×10^{-9} 95%: 1.2×10^{-7}
SWS2–MUPSA Input: Consequence analyses case 1–AS–01	1.47×10^{-7}	Same as above, plus one SWS train is disabled for both units.	**5.0×10^{-9}** 5%: 1.4×10^{-10} 95%: 1.5×10^{-8}
00DWT–MUPSA Input: consequence analyses case 1–19–GT–DWT	2.6×10^{-7}	Multiunit CCFs for similar equipment. DWTs unavailable for both units	**4.7×10^{-9}** 5%: 4.4×10^{-10} 95%: 2.6×10^{-8}
00SWS2–0MUPSA Input: consequence analyses case 1!QF2	4.44×10^{-8}	Multiunit CCFs for similar equipment. Two SWS trains are disabled for both units	**9.5×10^{-10}** 5%: 3.5×10^{-11} 95%: 4.3×10^{-9}
00SWS2–2MUPSA Input: consequence analyses case 1!QF2	4.44×10^{-8}	Multiunit CCFs for similar equipment. Two SWS trains are disabled for both units.	**9.5×10^{-9}** 5%: 3.5×10^{-10} 95%: 4.3×10^{-8}
00SWS3–3MUPSA Input: consequence analyses case 1!QF3	2.3×10^{-8}	Multiunit CCFs for similar equipment. Three SWS trains are disabled for both units	**2.3×10^{-8}** 5%: 6.1×10^{-10} 95%: 9.1×10^{-8}
00LHRH –INDUCED Input: consequence analyses case 1–12–LHRS	1.4×10^{-6}	Multiunit CCFs for similar equipment. Loss of normal heat removal for both units	**6.2×10^{-8}** 5%: 3.1×10^{-8} 95%: 1.4×10^{-7}
00LOOP–INDUCED Input: consequence analyses case 1!QF2	4.48×10^{-6}	Multiunit CCFs for similar equipment and HRA dependencies. LOOP conditions for second unit	**5.9×10^{-11}** 5%: 2.3×10^{-12} 95%: 4.1×10^{-10}

The largest contribution to multiunit CDF is provided by two internal events LOOP and administrative shutdown due to SWS train failure This is due to high frequencies of the events and high inter–units CCF values for SWS pumps and diesel generators were accepted in the assessment. All other internal events considered in the analyses provides negligible contribution to multiunit CDF, primarily because their contribution to CDF for single unit was very low and there is no single CCF that prevent accident mitigation at both units. There are several initiating events, caused by external hazards that also provides considerable contribution to multiunit CDF, however, this contribution is mainly driven by the frequency of the initiating event, which was assessed in a very conservative manner.

3.5.1.13. Tunisia/STEG

The accident sequences leading to core damage are quantified to evaluate the CDF and main contributors to core damage. The CDF of each of the four end states which represents the summation of the frequencies of all the event tree sequences leading to that state is estimated. For the analysis of the fault tree and accident sequence a truncation value of 1×10^{-15} is used. The point estimate values for the CDF are as follows:

- 3.85×10^{-6} for CD1;
- 4.43×10^{-7} for CD2;
- 5.00×10^{-6} for CD3;
- 1.00×10^{-2} for the CD4.

3.5.1.14. Ukraine/ SSTC NRS

The quantitative assessment was performed by means of the probabilistic code SAPHIRE 8 through the calculation of frequency of occurrence of accident scenarios leading to the end state CD12, which is characterized by severe fuel damage at RNPP 1 and RNPP 2. The degree of separation of MCS is equal to 1×10^{-22}. According to calculation results, multiunit CDF for initiating event T8 ‘Loss of the ESWS’ is 8.88×10^{-16} 1/year. According to results of quantitative assessment of Level 1 PSA regarding internal initiating events at reactor full power operation, CDF for initiating event T8 calculated for each of adjacent units is 2.95×10^{-11}.

Table 38 presents the comparison of results of the quantitative assessment of CDF for initiating event T8 ‘Loss of the ESWS’ for single and multiunit.

TABLE 38. COMPARISON OF RESULTS OF CDF QUANTITATIVE ASSESSMENT

Initiating event	**RNPP 1 CDF, 1/year**	**RNPP 2 CDF, 1/year**	**Multiunit CDF of RNPP 1 and RNPP 2, 1/year**
Internal initiating event T8 ‘loss of the ESWS’	2.95×10^{-11}	2.95×10^{-11}	8.88×10^{-16}

3.5.1.15. Ukraine/Energoris

Calculations were performed with a cut off degree of minimal sections equal to 1×10^{-14}. Calculations with cut off degrees of 1×10^{-15} and smaller show a slight increase in risk metrics (< 0.001%), but they require large computational resources. Results are presented in Tables 39 and 40.

TABLE 39. CORE DAMAGE FREQUENCIES FOR ZNPP SITE

End state code	**Reactor**	**CDF, 1/year**	**multiunit CDF, 1/year**	**CCDP**	**Multiunit CCDP**
CD01000	ZNPP Unit 2	1.61×10^{-08}	7.95×10^{-08}	6.68×10^{-03}	3.34×10^{-02}
CD00001	ZNPP Unit 5	1.60×10^{-08}		6.68×10^{-03}	
CD00010	ZNPP Unit 4	1.58×10^{-08}		6.68×10^{-03}	
CD00100	ZNPP Unit 3	1.58×10^{-08}		6.68×10^{-03}	
CD10000	ZNPP Unit 1	1.58×10^{-08}		6.68×10^{-03}	
CD00011	ZNPP Units №4 and №5	7.81×10^{-11}	7.81×10^{-10}	4.37×10^{-05}	4.36×10^{-04}
CD00101	ZNPP Units №3 and №5	7.81×10^{-11}		4.36×10^{-05}	
CD00110	ZNPP Units №3 and №4	7.81×10^{-11}		4.31×10^{-05}	
CD01001	ZNPP Units №2 and №5	7.81×10^{-11}		4.43×10^{-05}	
CD01010	ZNPP Units №2 and №4	7.81×10^{-11}		4.38×10^{-05}	
CD01100	ZNPP Units №2 and №3	7.81×10^{-11}		4.38×10^{-05}	
CD10001	ZNPP Units №1 and №5	7.81×10^{-11}		4.36×10^{-05}	
CD10010	ZNPP Units №1 and №4	7.81×10^{-11}		4.30×10^{-05}	
CD10100	ZNPP Units №1 and №3	7.81×10^{-11}		4.30×10^{-05}	
CD11000	ZNPP Units №1 and №2	7.81×10^{-11}		4.37×10^{-05}	
CD00111	ZNPP Units №3,4 and №5	$< 1.00\times10^{-14}$	$< 1.00\times10^{-14}$	8.00×10^{-06}	8.00×10^{-05}
CD01011	ZNPP Units №2,4 and №5	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD01101	ZNPP Units №2,3 and №5	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD01110	ZNPP Units №2,3 and №4	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD10011	ZNPP Units №1,4 and №5	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD10101	ZNPP Units №1,3 and №5	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD10110	ZNPP Units №1,3 and №4	$< 1.00\times10^{-14}$		7.99×10^{-06}	
CD11001	ZNPP Units №1,2 and №5	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD11100	ZNPP Units №1,2 and №3	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD11010	ZNPP Units №1,2 and №4	$< 1.00\times10^{-14}$		8.00×10^{-06}	
CD01111	ZNPP Units №2,3,4 and №5	$< 1.00\times10^{-14}$	$< 1.00\times10^{-14}$	3.83×10^{-08}	1.92×10^{-07}
CD10111	ZNPP Units №1,3,4 and №5	$< 1.00\times10^{-14}$		3.83×10^{-08}	
CD11011	ZNPP Units №1,2,4 and №5	$< 1.00\times10^{-14}$		3.83×10^{-08}	
CD11101	ZNPP Units №1,2,3 and №5	$< 1.00\times10^{-14}$		3.83×10^{-08}	
CD11110	ZNPP Units №1,2,3 and №4	$< 1.00\times10^{-14}$		3.83×10^{-08}	
CD11111	ZNPP Units №1,2,3,4 and №5	$< 1.00\times10^{-14}$	$< 1.00\times10^{-14}$	2.55×10^{-12}	2.55×10^{-12}

TABLE 40. FUEL DAMAGE FREQUENCIES FOR ZNPP SITE SFP

SFP	**FDF, 1/year**	**multiunit FDF, 1/year**	**CFDP**	**MUCFDP**
ZNPP Unit 6	2.17×10^{-09}	7.05×10^{-09}	9.01×10^{-04}	2.94×10^{-03}
ZNPP Unit 4	2.17×10^{-09}		9.01×10^{-04}	
ZNPP Unit 3	2.17×10^{-09}		9.01×10^{-04}	
ZNPP Unit 2	1.82×10^{-10}		7.80×10^{-05}	
ZNPP Unit 1	1.82×10^{-10}		7.77×10^{-05}	
ZNPP Unit 5	1.79×10^{-10}		7.69×10^{-05}	
ZNPP Units№5 and №6	$< 1.00\times10^{-14}$	$< 1.00\times10^{-14}$	6.92×10^{-08}	3.08×10^{-06}
ZNPP Units№2 and №6	$< 1.00\times10^{-14}$		7.02×10^{-08}	
ZNPP Units№4 and №5	$< 1.00\times10^{-14}$		6.93×10^{-08}	
ZNPP Units№2 and №5	$< 1.00\times10^{-14}$		5.87×10^{-09}	
ZNPP Units№3 and №6	$< 1.00\times10^{-14}$		8.12×10^{-07}	
ZNPP Units№3 and №5	$< 1.00\times10^{-14}$		6.93×10^{-08}	
ZNPP Units№2 and №4	$< 1.00\times10^{-14}$		7.03×10^{-08}	

TABLE 40. FUEL DAMAGE FREQUENCIES FOR ZNPP SITE SFP (CONT.)

SFP	FDF, 1/year	multiunit FDF, 1/year	CFDP	MUCFDP
ZNPP Units№3 and №4	$< 1.00\times10^{-14}$		8.13×10^{-07}	
ZNPP Units№2 and №3	$< 1.00\times10^{-14}$		7.03×10^{-08}	
ZNPP Units№1 and №6	$< 1.00\times10^{-14}$		6.99×10^{-08}	
ZNPP Units№1 and №5	$< 1.00\times10^{-14}$		5.85×10^{-09}	
ZNPP Units№1 and №4	$< 1.00\times10^{-14}$		7.00×10^{-08}	
ZNPP Units№1 and №3	$< 1.00\times10^{-14}$		7.00×10^{-08}	
ZNPP Units№1 and №2	$< 1.00\times10^{-14}$		5.94×10^{-09}	
ZNPP Units№4 and №6	$< 1.00\times10^{-14}$		8.12×10^{-07}	
ZNPP Units №4, №5 and №6	$< 1.00\times10^{-14}$	$< 1.00\times10^{-14}$	1.10×10^{-11}	6.00×10^{-10}
ZNPP Units №3, №5 and №6	$< 1.00\times10^{-14}$		1.10×10^{-11}	
ZNPP Units №3, №4 and №6	$< 1.00\times10^{-14}$		5.01×10^{-10}	
ZNPP Units №3, №4 and №5	$< 1.00\times10^{-14}$		1.10×10^{-11}	
ZNPP Units №2, №3 and №4	$< 1.00\times10^{-14}$		1.10×10^{-11}	
ZNPP Units №2, №4 and №6	$< 1.00\times10^{-14}$		1.10×10^{-11}	
ZNPP Units №2, №3 and №6	$< 1.00\times10^{-14}$		1.10×10^{-11}	
ZNPP Units №1, №4 and №6	$< 1.00\times10^{-14}$		1.10×10^{-11}	
ZNPP Units №1, №3 and №6	$< 1.00\times10^{-14}$		1.10×10^{-11}	
ZNPP Units №1, №3 and №4	$< 1.00\times10^{-14}$		1.10×10^{-11}	
Other combinations (4 out of 6, 5 out of 6, 6 out of 6)		$< 1.00\times10^{-14}$		

3.5.1.16. UAE/Khalifa University

Results of the developed Khalifa University PRA model for a single unit: the CDF of a single unit per Khalifa University PRA pilot model is 1.36×10^{-5}/y, while the CDF related to SBO with LOOP has a value of 8.50×10^{-6}/y. It is found that the SBO and LOOP are the highest contributors to the CDF representing ~62.42% of the total CDF. The CDF and SBO risks obtained in the model, are highly increased because of adding all failure modes and unavailability due to test and maintenance of involved AC power sources after LOOP (two EDGs and one AAC–DG). The consideration of CCFs of alpha factors between three involved CCF groups of diesel generators contributes additionally to high CDF. The evaluation of SBO risk in a single and multiunit site with possible risk reduction options applied to pilot model to evaluate the SBO risk for each of the involved risk reduction options that could contribute to the reduction of total CDF risk and increase NPP safety.

Results of the addition of third and fourth EDGs to the Khalifa University PRA model: after the addition of the third and fourth EDGs to PRA model, the CDF is highly reduced; this is because of the risk of the highest contributor of the SBO by increasing the onsite AC power availability (redundancy factor). The CDF associated with SBO is reduced by 74.75% by adding the third EDG and by 86.54% by adding the fourth EDG, while the total CDF is reduced by ~46.58% and 53.95%, respectively. Further reduction of CDF associated with SBO and total CDF by the addition of 4th EDG are 12% and 7%, respectively.

Results of the SUPRA with impact of multiunit features (2 units/site, 3 units/site, and 4 units/site): Khalifa University PRA model is developed to estimate a realistic CDF of a single unit with multiunit features by incorporating data of all failure's mode of EDG and AAC, and LOOP and multiunit LOOP frequency. The model also uses by the alpha factor method the CCF's factors of CCF groups of involved diesel generators.

Case 0 represent a reference base in the Khalifa University PRA model with indicated features related to LOOP and SBO. The reduction defined as ΔSBO % and ΔCDF % is calculated as a percentage of the difference between the risk of (LOOP+SBO)/y and CDF/y, respectively. It is considered that unit 1 is the unit under the study and the unit 2, unit 3 and unit 4, are the donner units of EGD crosstie if that option is adopted. In Case 1, the results of the applied EDG crosstie show a reduction of CDF related to (LOOP + SBO) and total CDF of unit 1 by 69.71% and 43.44%, respectively. This reduction is caused by an increase of redundant onsiteAC power of additional EDG crosstie from unit 2. The shared AAC–DG between the units has an occupancy factor. If the other unit experience SBO, it will be used by the unit having SBO (not unit 1). In case of crosstie option, two EDGs in the other units are to be available to the unit under the extended SBO. In case 2, a failure of one of the two EDGs (from unit 2) is considered in the occupancy factor of EDG from another unit. In comparison to case 1, a consideration of occupancy factors of AAC and crosstie EDG (case2) reduces the availability of AAC–DG and EDG crosstie to unit 1 and SBO and CDF risk are then increased by ~5.2% and 3.2%, respectively. With battery load shedding the available time for the operator increases, and it therefore contributes to the reduction of SBO risk. For case 3 with battery load shedding and without the occupancy factors, the SBO and CDF have additional reduction by 4.9% and 3.1%, respectively in comparison to case 1. For case 4, with the occupancy factors for AAC–DG and U2 EDG crosstie applied with battery load shedding, the final reduction of CDF associated with (LOOP + SBO) and total CDF are 69.16% and 43.11%, respectively in reference to case 0. A reduction of CDF induced by LOOP + SBO and reduction of total CDF by 69.16% and 43.11%, respectively are obtained if the crosstie is adopted from unit 2 only (case 5). A more reduction is obtained by ~6.9% and 4.3%, respectively with additional EDG crosstie from unit 3 (case 6). In case unit 4 also contributes one of its EDG crossties to unit 1, an additional reduction of CDF induced by LOOP + SBO and total CDF are 1.8% and 1.2%, respectively (case 7).

Selected results are graphically presented in Fig. 69. and summarized in Table 41.

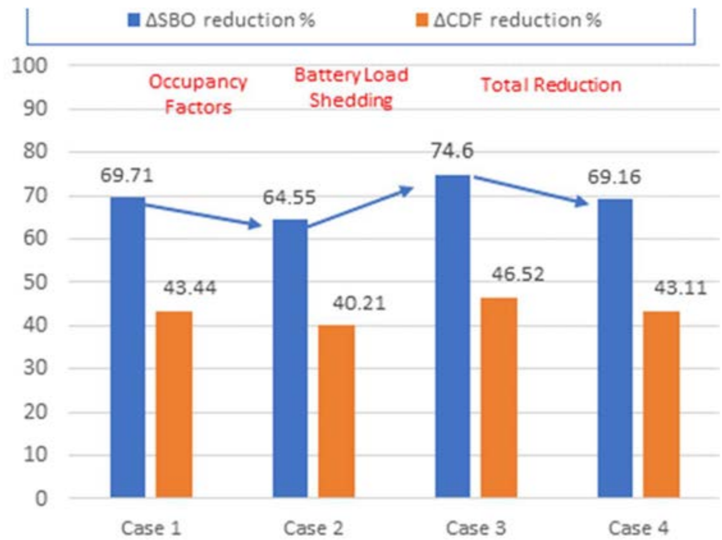


FIG. 69. PRA pilot model of Khalifa University: result's comparison for unit 1 with EDG crosstie from unit 2 (ΔSBO and ΔCDF).

TABLE 41. PRA RESULTS FOR UNIT 1 WITH MULTIUNIT FEATURES (SU–: SINGLE UNIT; MU–: MULTIUNIT)

Case No	Common features	Added features for LOOP and SBO	CDF Induced by LOOP+SBO / Year		Total CDF / Year	ΔSBO % Reduction	ΔCDF % Reduction
0 (Ref)	No EDG crosstie. Recent data of EDG and AAC–DG failures and LOOP. 3 CCF groups of diesel generators of unit 1 (Two EDGs+ AAC)	With human error for restoration of AC power from AAC–DG. No battery load shedding. No Multi initiating events (MU–LOOP)	8.5×10^{-6} reference SBO		1.4×10^{-5} reference CDF	n/a	n/a
Case 1	EDG crosstie unit 1 and unit 2. Recent data of EDG AAC failures and LOOP	No occupancy factor of AAC–DG and EDG unit 2 crosstie. No battery load shedding	SU–LOOP +SU–SBO MU–LOOP +MU–SBO **Total**	2.2×10^{-6} 4.1×10^{-7} $\mathbf{2.6\times10^{-6}}$	7.7×10^{-06}	69.7	43.4
Case 2	EDG crosstie from unit 2 (EDG failures+ EDG supporting system failures + breakers failure). Multiunit initiating events (MU–LOOP and MU–SBO)	With occupancy factor of AAC–DG. With occupancy factor for EDG crosstie from unit 2. No battery load shedding	SU–LOOP +SU–SBO MU–LOOP +MU–SBO **Total**	2.5×10^{-6} 5.3×10^{-7} $\mathbf{3.0\times10^{-6}}$	8.1×10^{-06}	64.6	40.2
Case 3	EDG Crosstie from Unit–2. Human error for restoration of AC power from AAC–DG. Human error for EDG Crosstie	With battery Load shedding. No occupancy factor of AAC–DG and unit 2 EDG Crosstie	SU–LOOP +SU–SBO MU–LOOP +MU–SBO **Total**	1.8×10^{-6} 3.4×10^{-7} $\mathbf{2.2\times10^{-6}}$	7.3×10^{-06}	74.6	46.5
Case 4	EDG Crosstie from Unit–2. CCF groups (alpha factor) with: 4 CCF group (Three EDGs+ AAC–DG)	With occupancy factor of AAC–DG and EDG crosstie from unit 2. With battery load shedding	SU–LOOP +SU–SBO MU–LOOP +MU–SBO **Total**	2.2×10^{-6} 4.7×10^{-7} $\mathbf{2.6\times10^{-6}}$	7.7×10^{-06}	69.2	43.1
Case 5	Unit–1 – EDG Crosstie for multiunit site (2 units)	Occupancy factor of AAC–DG from U2. Occupancy factor for EDG Crosstie from U2. Battery load shedding	SU–LOOP +SU–SBO MU–LOOP +MU–SBO **Total**	2.2×10^{-06} 4.7×10^{-07} $\mathbf{2.6\times10^{-06}}$	7.7×10^{-06}	69.2	43.1
Case 6	Unit–1 – EDG Crosstie for multiunit site (3 units)	Occupancy factor of AAC–DG from U2 and U3. Occupancy factor for EDG Crosstie from U2 and U3 Battery load shedding	SU–LOOP +SU–SBO MU–LOOP +MU–SBO **Total**	1.60×10^{-06} 4.4×10^{-07} $\mathbf{2.0\times10^{-06}}$	7.2×10^{-06}	76.1	47.4
Case 7	Unit–1 – EDG Crosstie for multiunit site (4 units)	Occupancy factor of AAC–DG from U2, U3 and U4. Occupancy factor for EDG Crosstie from U2, U3 and U4. Battery load shedding	SU–LOOP +SU–SBO MU–LOOP +MU–SBO **Total**	1.4×10^{-06} 4.7×10^{-07} $\mathbf{1.9\times10^{-06}}$	7.0×10^{-06}	77.9	48.6

3.5.2. Lessons learned and insights

3.5.2.1. Argentina/CNEA

Regarding Level 1 MUPSA development, according to scope of this study, the followed conclusions have been obtained:

- For SMR Level 1 MUPSA would be necessary to extend the mission time beyond the Stage 1. Regarding design features of SMRs, passive safety systems allows achieving a safe state. After that, active systems are required to achieve a final safe state. Then, in MUPSA framework, the extension of mission time allows considering other functional dependencies that can affect multi radioactive sources, considering the strategy to control initiating events;
- Dependencies analyses among radiological sources, in particular during Stage 2, is a key task for MUPSA development;
- Level 1 MUPSA event trees can be large. It demands simplifying hypothesis and grouping of systems in headers as event tree modelling approach;
- CCFs: it is required to model inter–units CCF. In the framework of SMR, it is credible a scenario where more than one unit will be constructed at the same time. In that sense, components will be purchased to same suppliers and installed by same teams;
- HRA: SMR implies new concepts of operations, where operators can monitor and control more than one unit at the same time. Moreover, changes in Human Systems Interfaces are being implemented in new designs. Consequently, HRA methods need to be adapted, in special with the objective of calculating probability of HFE;
- In the event tree modelling approach, another important aspect to evaluate is the objective of the event tree: for licensing or for support to operations. For licensing, the objective could be to evaluate a risk metric. For giving support to operations or training, the objectives could be to show dependences among systems. These objectives can affect the headers and how they are modelled;
- In Level 1 MUPSA, deterministic analysis for single unit can be used. There is not interaction inter units and/or SFP from this point of view;
- For calculating proposed site risk metric (section 4.2.1), Level 1 event sequences derives from a multiunit initiating event that lead to fuel damage only in one radiological source have to be identified.

3.5.2.2. Canada/COG

Caution needs to be exercised with any form of numerical risk aggregation for a multiunit NPP. With this consideration, the SCDF for a 4 unit CANDU plant (Pickering B) for a given hazard was calculated as a combination of single unit, two unit, and four unit SCDF and LRF results. Based on the above aggregation approach, SCDF for various hazards for the 4 unit Pickering B site was calculated and the results are as follows:

- At–power internal events = 3.2×10^{-6}/yr
- At–power fire = 1.7×10^{-6}/yr
- At–power flood = 7×10^{-7}/yr
- At–power seismic = 2×10^{-7}/yr

— At–power high wind = 3.8×10^{-6}/yr
— Outage internal events = 2.4×10^{-6}/yr

The LRF results for the four unit Pickering B site are as follows:

— At–power internal events = 6.0×10–7/yr
— At–power fire = 8×10^{-7}/yr
— At–power flood = 5×10^{-7}/yr
— At–power seismic = 2×10^{-7}/yr
— At–power high wind = 2.1×10^{-6}/yr
— Outage internal events = negligible

Analysis of loss of heat sink at the IFBs indicated that the time to reach boiling was greater than 72 hr. Consequence assessments, both deterministic and probabilistic led to the overall conclusion that the risk associated with the IFBs is negligibly low LRF is of the order of 10^{-9}/yr. For the UFDS, since there is no additional containment for the dry storage containers, a 'direct containment bypass or failure' is always assumed in case of failure of a UFDS. Thus, to release ^{137}Cs, which is the radionuclide of concern for the LRF in a PSA, the fuel would need to be melted.

There were no hazards identified that could result in melting of the used fuel. From PSA perspective, therefore, risk from the dry used fuel storage can be neglected.

3.5.2.3. China/INET

Some insights which can be obtained from INET's study are:

a) Multi module releases are not random combinations of single module releases:

   — Common initiator and CCFs dominate the releases involving multiple modules;
   — Multi module and single module releases are physically similar in terms of source term category.

b) Multi module risks contribute a small portion of the total risk (e.g. 5%) with respect to the frequency part of risk:

   — Multi module releases contribute a step wise effect to the consequence part of risk.

3.5.2.4. Finland/VTT

In the pilot studies of the SITRON project, the multiunit CDF values were nonnegligible. However, the SCDF was almost the sum of single unit CDFs, and risk importance results are very similar at the site level as in the SUPSAs. This means that from SCDF point of view the analysis did not produce much new information. The multiunit CDF was dominated by inter–unit CCFs, whereas human dependencies were not very important. Without inter–unit CCFs, the multiunit CDF would be negligible. Conservative inter–unit CCF probabilities were used due to lack of multiunit data. Hence, the analysis could be made significantly more accurate by collecting sufficient multiunit data. Computation of risk metrics is relatively straightforward given relevant input data. The only significant POS combination was that both units were at power operation.

3.5.2.5. Ghana/GAEC

The results indicate that a significant contribution to site risk is attributable to core damage of single units, which had been already accounted for in the PSA models of the individual units. Further insights gained from the results is the confirmation that CCF contribution to MCS is significant. This emphasizes the fact that MUPSA studies need to account for CCF in analysis models.

3.5.2.6. Hungary/NUBIKI

Based on the assessment of those event scenarios when a single unit internal initiating event evolves to core damage at power operation at one unit (unit 1), and core damage occurs at the neighbouring unit (unit 2) during the subsequent forced shutdown, it can be concluded that the consequences of a large release (i.e. harsh radiological conditions) can have a significant effect on the operators of the neighbouring units. Also, human errors have a significant contribution to CDF at forced shutdown. The sensitivity analysis was based on simplified assumptions and was meant to demonstrate the potential effects of radiological conditions on a high level. A more detailed, event sequence based analysis may be required to refine the results and gain more insights. Based on the LOOP induced single as well as multiunit risk results it can be concluded that LOOP induced twin unit CDF is moderately important; however, the frequency of core damage at 3 or 4 units simultaneously due to LOOP seems negligible. This may be originated from the fact that diesel generators installed at units 1 and 2 differ considerably from the diesel generators belonging to units 3 and 4; hence no inter–unit CCF was assumed among these groups of 6 diesels. It is also noted that the effect of accounting for some designated inter–unit CCFs is significant. Regarding the assessment of core damage at twin units induced by the initiating event ‘loss of power due to onsite causes’, the twin unit initiating event can be regarded as a multiunit event with relatively considerable frequency. However, it can be concluded that risk originated from simultaneous loss of power due to onsite causes can be regarded as negligible.

3.5.2.7. India/AERB

Some of the salient observations and lessons learnt from MUPSA development and during the implementation of the methodology for the benchmark study are as follows:

— Scope of MUPSA needs to be clearly defined. It is important to clearly define the operating states addressed, whether non–reactor facilities that may be colocated are included, etc. in the scope of MUPSA.
— A systematic approach is required to limit the number of IEMs as the number of event trees can be very high that can result in enormous number of accident sequences, especially when there are large number of units at a site. Further, the consequences, radioactivity releases, time and duration could have a wide range. Such large PSA models are very difficult to handle.
— Identification of potential initiating events that can affect more than one unit simultaneously
— Estimating the multiunit initiating event frequency is vital in MUPSA.
— As in most of the cases, SUPSA is completed prior to MUPSA, a clear understanding is required to either include or exclude single unit accident sequences in the risk metrics defined for MUPSA.

— Modelling of inter and intra–unit CCFs includes appropriate weightage factors taking into account the vintage or age of identical components, suppliers, installation team, working environment, etc.
— HRA needs to consider the size of crew for more than one reactor at a site and the human system interfaces implemented, especially in the new reactors. Performance shaping factors used for SUPSA may not be directly adopted for MUPSA.
— The new design features incorporated in various units at a multiunit site needs to be given due credit for a realistic estimate of the risk metrics.
— Similarly, for the dependence of SSCs in multiple units.

3.5.2.8. India/BARC

Following are some of the observations made during Level 1 MUPSA analysis:

— SUPSA results cannot be directly summed up for getting the MUPSA results. This will lead to double counting of the events.
— Separate PSA models with different identifiers to be created for individual units. Attention is given for coding the components/systems which are common to multiple units by adopting same identification code.
— A separate list of MUPSA initiating events to be created and corresponding event trees to be developed. Apart from this a separate MUPSA model to be created. In the present study, simplified event tree approach while integrating multiple units and sources has been adopted.
— Attention is given for the treatment of CCFs, shared systems and HRA.
— Separate HRA models to account for sharing the resources, man power and modelling HRA under extreme events (seismic) to be developed. Dependency of HEP with respect to earthquake intensity needs to be evaluated properly. If existing models are being used modification of existing performing shaping factors is recommended.
— Care to be taken while modelling dependency and correlations between inter and intra unit, identical and non identical components. Considering complete dependency between identical components and zero dependency between the non identical components may not be realistic. Hence, the conservative assumption of failure of similar SSCs at the same elevation due to seismic event needs to be addressed by proper consideration of dependency and correlations.

3.5.2.9. Pakistan/PAEC

The most conservative value of annual frequency of unacceptable performance of all building structures corresponding to mean fragility curve is adopted from the seismic risk metrics of shared buildings for input in MUPSA–1 model.

3.5.2.10. Republic of Korea/KHNP

From Level 1 MUPSA, KHNP has identified the following lessons learned and insights:

— Impact of internal events, internal fire and internal flooding on multiunit risk are recognized negligible. It is because of the design characteristics, which do not share safety related SSCs between units.

— SCDF for multiunit LOOP (multiunit LOOP) is relatively higher than that for other IEMs. The SCDF is sensitive to the depth of details of POS models of the unit in low power and shutdown operating mode, especially of the old unit.
— Multiunit CDF for multiunit LOOP was estimated at the level of 10^{-7} per site per year, and about 97% of multiunit CDF is from any unit combination of two out of nine units.
— Multiunit CDF for multiunit LOOP is sensitive to off site power recovery dependency and the inter–unit CCF.
— SCDF and multiunit CDF for multiunit loss of circulating water, general transients and tsunami induced loss of essential service water were identified negligible comparing to those for multiunit LOOP and seismic event. Accordingly, we could determine not to perform further assessment such as Level 2 MUPSA.
— Multiunit CDF for seismic events is estimated in a range of 10^{-6} per site per year, and the portion of multiunit CDF to SCDF takes up about 49%.
— SCDF and multiunit CDF for seismic events is not sensitive to the depth of details of POS models and inter–unit CCF.
— Seismic correlation factor has little impact on multiunit CDF for the seismic hazard group corresponding to PGA from 0.1g to 0.2g, and over 0.5g.
— Based on the result of Level 1 MUPSA, KHNP decided to develop Level 2 MUPSA models only for multiunit LOOP and seismic event.

3.5.2.11. Republic of Korea/Hanyang University

To get insight from the result of SRA and characteristic of site risk, the risk index was developed based on the result that have been considered in PSA. In this analysis, the site risk indexes were developed and estimated based on the SRA result of internal event as shown in Tables 42 and 43, and based on definition in Fig. 32, the preliminary value of the multiunit accident importance shows 0.77% which indicates multiunit CDF of 0.77% among SCDF and it can be interpreted for the influence of internal event to multiunit accident to be insignificant. Multiunit accident importance by common cause initiator showed 2.67% indicating that multiunit CDF of 2.67% of the SCDF induced by common cause initiator and core damage in 1 unit would be probable to occur though simultaneous initiating event occurs.

TABLE 42. PROPOSED SITE RISK INDEX

Site risk index	**Definition**	**Practical usage**
Multiunit accident importance in Site Risk	Fraction of multiunit risk among site risk	– Comprehension of the multiunit risk contribution – Understanding multiunit risk regulation necessity
Multiunit accident importance by common cause initiator in site risk	Fraction of multiunit risk among CCI’s risk	– Comprehension of multiunit risk contribution among CCI’s risk

TABLE 43. EVALUATED SITE RISK INDEX BASED ON LEVEL 1 SRA FOR INTERNAL EVENT

Site Risk Index	**Value (%)**
Multiunit accident importance	0.77
Multiunit accident importance by common cause initiator	2.67

3.5.2.12. Romania/CNCAN

The Level 1 elements for MUPSA were included in the Level 2 model as presented in the corresponding chapters.

3.5.2.13. Russian Federation/JSC A

From the results obtained the following insights can be drawn:

— Multiunit CDF is relatively high for Balakovo NPP (1.51×10^{-7}/y), but still two orders of magnitude lower that CDF for single unit (4.4×10^{-5}/y);
— Only few internal initiating events contribute to multiunit CDF for power operation mode;
— Seismic hazards are to be included in the scope of the assessment. However, it requires elaboration if the common approach for seismic fragility correlations;
— Multiunit CDF is highly driven by inter units CCFs for similar equipment. Elaboration of common approach for assessment of these CCFs is also required.
— Multiunit CDF is highly dependent on external hazards disabling common equipment at the site. More thorough assessment of such hazards is needed.

3.5.2.14. Tunisia/STEG

From the preliminary results, it was noticed that the partial CDF that may result from the fuel blockage of one fuel assembly (CD4) is not negligible and need to be taken into account. All internal events can be neglected from many assessment because of their negligible impact. Seismic fragility curves have been developed at the component level and later propagated to system level by using seismic fault trees. Seismic CDF has been estimated by convoluting seismic hazard curves and seismic fragility curves of the dominating accident sequences.

3.5.2.15. Ukraine/Energoris

From the results obtained the following insights can be drawn:

— Only few initiators can be considered as potentially significant for MUPSA. Seismic hazards are to be included in the scope of the MUPSA;
— Development of integrated MUPSA model requires simplification assumptions;
— Multiunit CDF is ≤ 1% of total CDF for sites with low inter–unit dependencies;
— Multiunit CDF for damage combinations is linearly distributed due to very low system dependencies between ZNPP units;
— Truncation value need to be significantly decreased, comparing to SUPSA;
— Multiunit CDF is driven by inter units CCFs for similar equipment and external hazards disabling common equipment at the site. Large uncertainties in inter–unit CCF/large SSC groups requires detailed elaboration of methods/approaches for assessment of inter–unit CCF parameters.

3.5.2.16. Ukraine/ SSTC NRS

The dominant MCS of MUPSA for initiating event 'Loss of the ESWS' are presented in Table 44. The most significant is the MCS, which is characterized by a failure of control rods to enter the core of RNPP 1 and RNPP 2 in the occurrence of initiating event 'Loss of the ESWS'.

TABLE 44. DOMINANT MINIMUM CUT SETS (MCS)

No.	Probability / frequency	Contribution to CDF, %	Event code	Event description
1	9.158×10^{-16}	99,9	SFCOM–01	Failure of control rods to enter
			U2–SFCOM–01	Failure of rods
2	9.441×10^{-20}	0.01	SFCOM–01	Failure of control rods to enter
			U2–EPSACB–B140–S–S	Short circuit on panel B–140 (1 hour)
			U2–RPS–OA–SCRAM–DC	Human error on scram
3	9.441×10^{-20}	0.01	EPSACB–B140–S–S	Short circuit on panel B–140 (1 hour)
			RPS–OA–SCRAM–DC	Human error on scram
			U2–SFCOM–01	Failure of rods
4	5.111×10^{-20}	< 0.01	SFCOM–01	Failure of control rods to enter
			U2–C–EPSTKE–2XN1A–1–SFC	CCF 3/3 TKEO–21,22,23N1A–1 within 1 hour
			U2–RPS–OA–SCRAM–DC	Human error on scram
5	5.111×10^{-20}	< 0.01	C–EPSTKE–1XN1A–1–S–F–C	CCF 3/3 TKEO–11,12,13H1A–1 during one hour
			RPS–OA–SCRAM–DC	Human error on scram
			U2–SFCOM–01	Failure of rods
6	1.018×10^{-20}	< 0.01	SFCOM–01	Failure of control rods to enter
			U2–C–RPSRYL–GCNX–X–F	CCF 4/6 main circulation pump s power relays
			U2–RPS–OA–SCRAM–DC	Human error on scram
7	1.018×10^{-20}	< 0.01	C–RPSRYL–GCNX–X–F	CCF 6/6 of reactor coolant pump 1(2–6) power relay
			RPS–OA–SCRAM–DC	Human error on scram
			U2–SFCOM–01	Failure of rods
8	3.394×10^{-21}	< 0.01	SFCOM–01	Failure of control rods to enter
			U2–C–EPSTRF–2X05TN–F–S–C	CCF 3/3 transformers 2105(2205,2305)TN within 1 hour
			U2–RPS–OA–SCRAM–DC	Human error on scram
9	3.394×10^{-21}	< 0.01	C–EPSTRF–1X05TN–F–S–C	CCF of 3/3 transformers 1105(1205,1305)TH (1 hour)
			RPS–OA–SCRAM–DC	Human error on scram
			U2–SFCOM–01	Failure of rods
10	3.150×10^{-21}	< 0.01	SFCOM–01	Failure of control rods to enter
			U2–C–EPSBAT–AB–2XP–F–C	CCF 3/3 AB–21(22,23)P (1 hour)
			U2–RPS–OA–SCRAM–DC	Human error on scram
11	3.150×10^{-21}	< 0.01	C–EPSBAT–AB–1XP–F–C	CCF of 3/3 for functioning during 1 hour AB–11(12,13)p
			RPS–OA–SCRAM–DC	Human error on scram
			U2–SFCOM–01	Failure of rods
12	2.743×10^{-21}	< 0.01	SFCOM–01	Failure of control rods to enter
			U2–C–EPSINV–PTS–2X–AFSC	CCF 3/3 PTS–21(22,23)A
			U2–RPS–OA–SCRAM–DC	Human error on scram
13	2.743×10^{-21}	< 0.01	C–EPSINV–PTS–1X–A–F–S–C	CCF 3/3 for functioning of PTS–11(12,13)A

TABLE 44. DOMINANT MINIMUM CUT SETS (MCS) (CONT.)

No.	Probability / frequency	Contribution to CDF, %	Event code	Event description
			RPS–OA–SCRAM–DC	Human error on scram
			U2–SFCOM–01	Failure of rods
14	1.953×10^{-22}	< 0.01	SFCOM–01	Failure of control rods to enter
			U2–C–RPSSNC–PTAZ–X–F–A	CCF 2/2 SCRAM subsystems / complement 1(2)
			U2–RPS–OA–SCRAM–DC	Human error on scram
15	1.953×10^{-22}	< 0.01	C–RPSSNC–PTAZ–X–F–A	CCF 2/2 of scram software, set 1(2)
			RPS–OA–SCRAM–DC	Human error on scram
			U2–SFCOM–01	Failure of rods
16	1.206×10^{-22}	< 0.01	SFCOM–01	Failure of control rods to enter
			U2–RPS–OA–SCRAM–DC	Human error on scram
			U2–RPSTRN–SET–1–TM	Testing or maintenance of scram subsystem (complement 1)
			U2–RPSTRN–SET–2–TM	Testing or maintenance of scram subsystem (complement 2)
17	1.206×10^{-22}	< 0.01	RPS–OA–SCRAM–DC	Human error on scram
			RPSTRN–SET–1–TM	Testing or maintenance of scram subsystem (set 1)
			RPSTRN–SET–2–TM	Testing or maintenance of scram subsystem. (set 2)
			U2–SFCOM–01	Failure of rods

3.5.2.17. UAE/Khalifa University

Based on conventional safety regulations, each reactor unit is supposed to be independent with dedicated safety features. Therefore, most of risk analysis is accomplished for a single reactor unit. As already understood from Fukushima Daiichi NPP accident that involved more than one unit, a conventional PRA based on a single unit assessment is not sufficient to characterize the risk at multiunit site. The electric crosstie option for the multiunit site for SBO was suggested to reduce the risk for core damage.

The PRA evaluation in the multiunit site includes the aspects of sharing and dependencies such as CCFs, related human actions, multiunit initiators and occupancy factors of shared components (AAC–DG) and electrical crosstied EDG. From the single unit PRA model, the reduction of CDF and associated SBO risk may be achieved by increasing a factor of redundancy for onsite AC power (increasing number of EDG). Two cases, adding new EDG and borrowing an EDG from the non–impacted unit are assessed providing the following results: (a) adding the third EDG reduced CDF of unit 1 by 46.58% and CDF associated with SBO by 74.75%, (b) adopting the electrical crosstie from unit 2 reduced total CDF by 40.21% and CDF associated with SBO by 64.55%. In case the EDG crosstie is adopted from unit 2, unit 3 and unit4 with battery load shedding, the resulting reduction of a total CDF is 43.11%, 47.43%, and 48.60% respectively. From the perspective of PRA analysis and risk management regarding SBO, it can be concluded that adopting an electrical crosstie reduces the total CDF, however the total benefit of EDG crosstie is limited by CCF and factors of occupancy of AAC–DG and crosstie EDG. Although sharing safety components between the units in multiunit site is not recommended by conventional safety regulation, it is worth considering as it reduces CDF in the case of SBO.

4. LEVEL 2 MUPSA

The basic MUPSA tasks are analogous to the SUPSA and are shown as task (or step) 1–8 in Fig. 70. However, there are tasks that have an indirect impact on the MUPSA Level 2 analysis, as well as tasks that require novel approaches and diverse methodologies from the established practice in SUPSA. Figure 71 provides the organization of these tasks for this publication. This section provides information on the following categories of MUPSA Level 2 tasks:

— Tasks within Level 1 and important for Level 2 (Task 1–3, their inputs and outputs);
— Tasks, which provide direct input to Level 2 (Task 4b);
— Tasks of Level 2 (Task 4b–8).

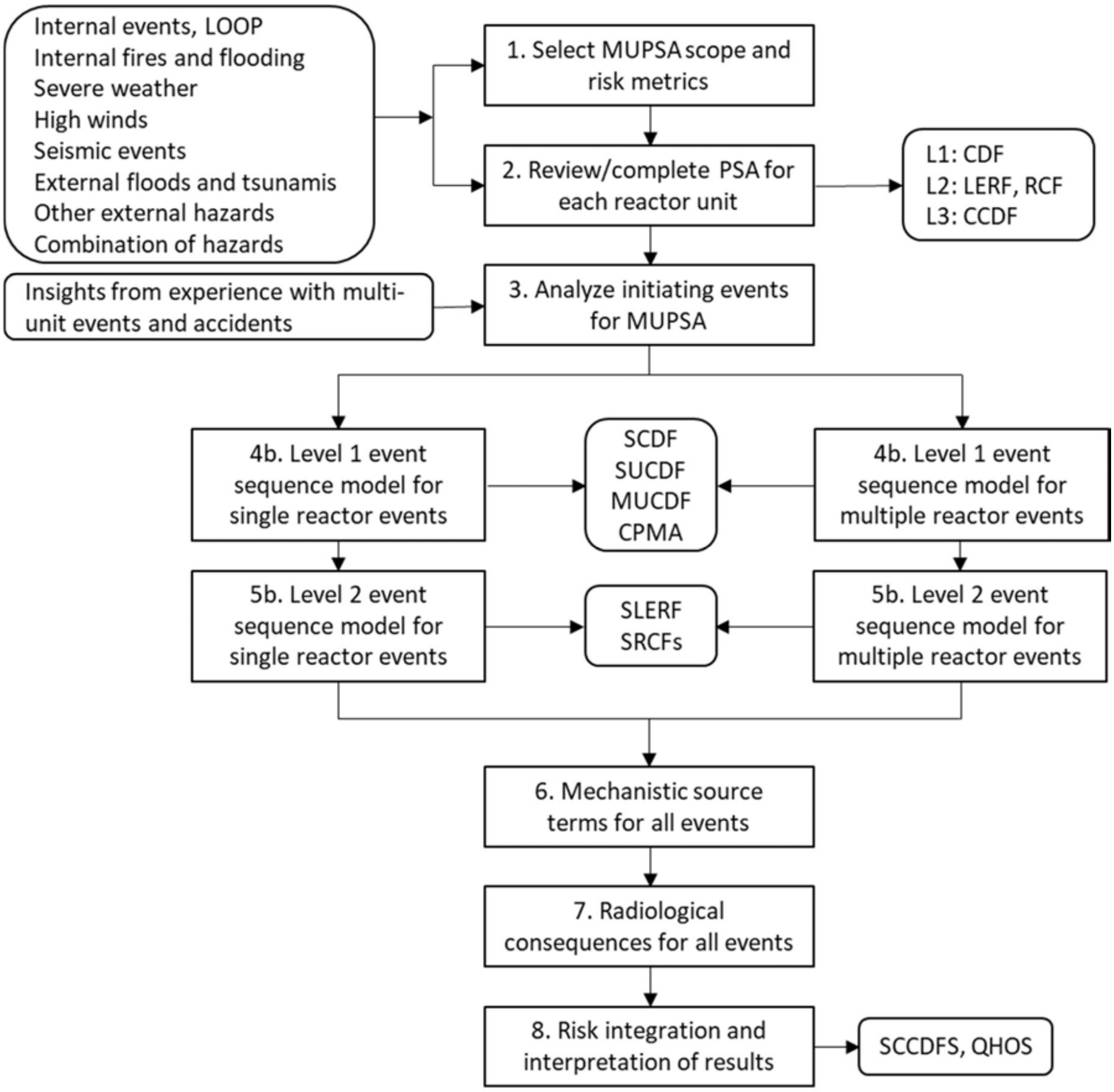


FIG. 70. MUPSA Level 2 tasks (after [20]).

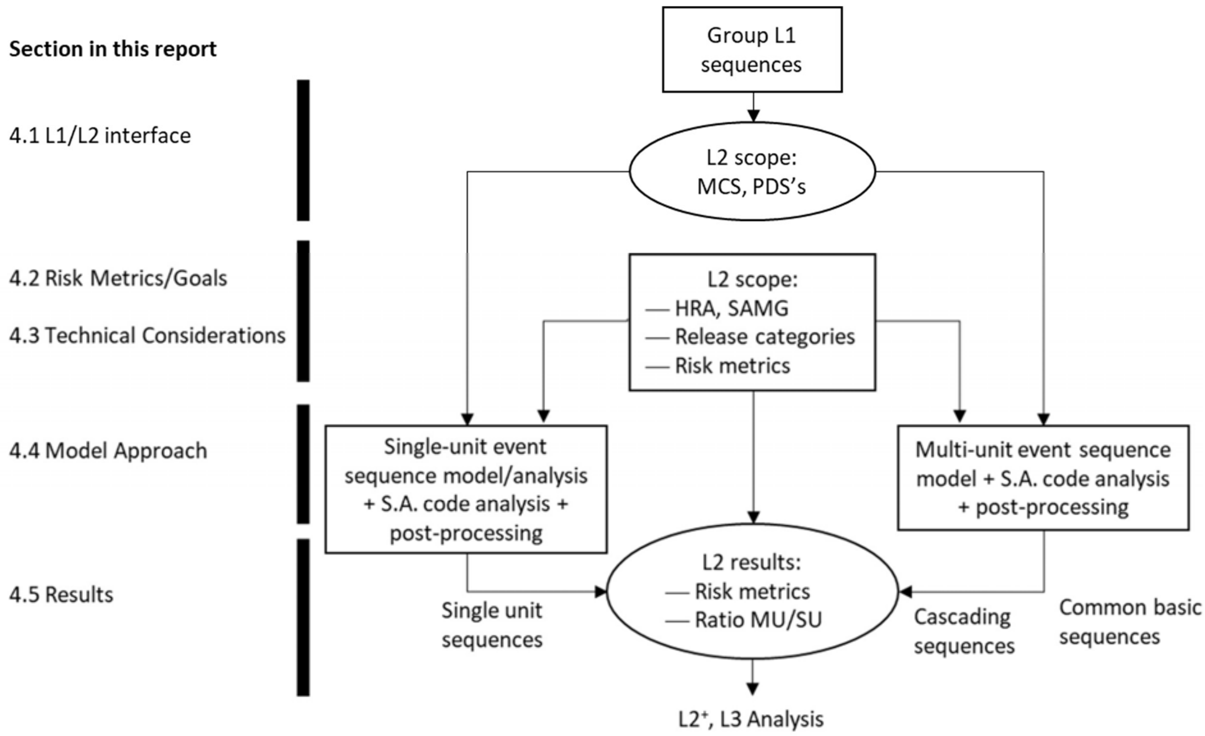


FIG. 71. MUPSA Level 2 tasks requiring novel approaches.

It can be noted that currently the scope of MUPSA, in particular the Level 2 tasks, are highly dependent on the national safety goals adopted, and they include interface with other safety analyses and documentation for a site. The performance of those tasks is also iterative, and the tasks may be performed several times. A more detailed screening of the degree of challenge and/or novelty in those tasks is presented in Table 45, along with the content taken from SRS–96.

All MUPSA projects are guided by general standards available for SUPSA. However, the task details for MUPSA are under review and testing in various case studies and not yet standardized. Each project is also guided by its goals, national regulations and existing recommended practices. As a result, the insights from the national experience on the technical considerations of the project are very important for the PSA and safety community.

Present MUPSA studies have a high degree of research character, mainly for tasks which need clarification and testing, as compared to the existence of established guides and standards for SUPSA. The specific approaches of MUPSA Level 2 are very important for their use in the decision–making process and technical basis for emergency plans and SAMGs, because it is tightly connected with severe accident codes runs.

TABLE 45. TASK (STEP) DESCRIPTIONS IN [20] AND THEIR LEVEL OF CHALLENGE IN MUPSA

Section in SRS-96 Chapter 4[20] [20] and title			Challenge in MUPSA (high/medium/low)	Task description relevant to Level 2 MUPSA [20]
4.1 Summary of Steps	4.1.1	Step 1: Selecting PSA scope and risk metrics	Option 1	Perform a limited scope Level 2 PSA that is sufficient to estimate the site risk metrics of SCDF (defined as the CDF to one or more reactor units on the site) and site LERF (SLERF) (defined as the frequency of a large early release from one or more reactors or radionuclide sources on the site).
			Option 2	Perform Level 3 PSA to provide additional complete set of risk metrics, such as CCDF for public health effects and property damage, and individual risks for quantitative health objective type of risk metrics. This option means added complete treatment of radionuclide sources such as spent fuel storage.
	4.1.2	Step 2: Reviewing and completing the single reactor PSA for each reactor and facility		PSA is completed to the scope selected in Step 1 for each reactor using established PSA methods. If a PSA already exists for one or more reactor units, it is only necessary to extend the scope, as needed, to achieve the scope selected in Step 1.
	4.1.3	Step 3: Analysing initiating events for MUPSA		Analyse the selection of initiating events to resolve which apply to individual reactor units and which impact two or more reactor units on the site concurrently. Resolve the initiating event causes, including internal and external events/hazards. This may require rescreening the initial list of events considered in the single reactor PSA and some events may need to be subdivided to resolve the multiunit CCIEs.
	4.1.4	Step 4: Level 1 event sequence model	4a	The single unit event sequence model in Step 4a is largely based on what was already developed in the SUPSA in Step 2, but it may need to be altered to interface with a more refined definition and selection of initiating events. As with SUPSAs, plant walkdowns are necessary to identify the potential for accidents involving two or more units.
			4b	Develop a new model to identify event sequences involving core damage to two or more units, resulting from a multiunit CCI or from the cascading effects of a single unit accident propagating to another unit.
	4.1.5	Step 5: Level 2 event sequence model	5a 5b	The models are based on what was already done in Step 2 for the individual reactor units if Step 2 had been developed to Level 2. Otherwise, if Step 2 was for a Level 1 PSA, it would be expanded in Step 5a to address Level 2 scenarios involving single reactor units. The event sequences for the scenarios with core damage to two or more units are developed and quantified. a) Level 1 / Level 2 interface treatment b) Level 2 event sequence model for single and multi unit accidents c) Level 2 event sequence quantification (SRC, SELRF, analysis of sensitivities, uncertainties and significant risk contributors

TABLE 45. TASK (STEP) DESCRIPTIONS IN [20] AND THEIR LEVEL OF CHALLENGE IN MUPSA (CONT.)

Section in SRS-96 Chapter 4 [20]and title		Challenge in MUPSA (high/medium/low)	Task description relevant to Level 2 MUPSA [20]
4.1.6	Step 6: Mechanistic source terms for all events		The purpose of this step is to develop the radioactive release source terms for all the event sequences and release categories of Step 5. The step is completed for the Level 3 risk metric option. It can be noted that the single reactor core damage events were already addressed in Step 2. When the single reactor PSA is expanded to a Level 2 PSA, the single unit initiating events and accident sequences are fully developed to support the Level 2 PSA in Steps 3, 4a and 5a, which establishes the scope of single reactor accidents for which mechanistic source terms are needed. To support the MUPSA, it is necessary to address the unique accident sequences associated with multiunit source terms (already defined in Step 5b).
			a) Single unit accidents
			b) Multiunit accidents
			c) Non core sources
			d) Uncertainties
4.1.7	Step 7: Radiological consequences for all events		The purpose of this step is to develop the radiological consequences for all of the release categories and source terms of Steps 5 and 6. Similar to Step 6, the step is completed for the Level 3 risk metric option. If the single reactor PSA developed in Step 2 was a Level 3 PSA, all that is now required is to analyse the multiunit core damage sequences for the necessary source term information. a) Level 3 analysis of all site release categories (Early/later cancer fatalities, release times, conditional risk curves) b) Level 3 analysis of external hazards with hazard specific evacuation models
4.1.8	Step 8: Risk integration and interpretation of results		The results for the event sequence frequencies and consequences are combined into Level 3 risk metrics, such as SCCDF curves for public health and safety impact, property damage and economic impacts. The integrated risk results are compared to the selected risk significance criteria and safety goals. Risk insights are then developed with regard to plant vulnerabilities and site and design specific factors that give rise to risk management opportunities.
	— Risk integration		a) Aggregation of complementary cumulative distribution functions
			b) Safety goal evaluation
			c) Sensitivity and uncertainty evaluation
	— Interpretation and documentation		d) Identification of risk insights towards site safety improvements
			e) Evaluation of DiD
			f) Documentation

TABLE 45. TASK (STEP) DESCRIPTIONS IN [20] AND THEIR LEVEL OF CHALLENGE IN MUPSA (CONT.)

Section in SRS-96 Chapter 4 [20] and title			Challenge in MUPSA (high/medium/low)	Task description relevant to Level 2 MUPSA [20]
4.2 Selection of initial conditions for sequence development				For example, estimates for three unit NPP for the following site configurations, are: — All three units operating at–power; — Two units at–power and one in shutdown (three combinations); — One unit at–power and two units in shutdown (three combinations); — All three units in shutdown; — Variations of the above with SFP status. If the units are identical, different combinations do not have to be modelled separately. Both at–power PSA and low power and shutdown PSA are needed for each unit on the site.
4.3 Multiunit site risk metrics				SCDF: Frequency per site–year of core damage to one or more reactor units; SLERF: Frequency per site–year of a large early release from one or more reactors or onsitefacilities; SRCF: Frequency per site–year of each distinct release category for a Level 2 MUPSA. These release categories include those already defined in a Level 2 SUPSA for each unit and for releases from a single reactor unit, as well as categories for accidents involving multiple reactor units or facilities; Multiunit CDF: Frequency per site–year of an accident involving core damage to two or more reactor units.
4.4 Selection of risk significance criteria				SRS–96
4.5 Summary of PSA models for risk metrics	4.5.1	Sites with identical reactor units		The multiunit site metrics SCDF and CPMA are dependent on the number of reactor units at the site. It becomes more complex for sites with more reactor units.
	4.5.2	Sites with non–identical units		For sites with large number of units, it is impractical to model each possible combination of multiunit accident cases separately. Hence, assumptions would need to be made to simplify it to something manageable.
4.6 Treatment of multiple hazards				Combined effects of correlated internal and external hazards are considered, such as: — Seismically induced tsunamis and dam failures (upstream and downstream); — Seismically induced fires, floods and high energy pipe breaks; — Combined effects of wind hazards and flooding from severe storms;
4.7 Ensuring technical adequacy				Strategies to ensure the technical adequacy of MUPSA are like those that for SUPSAs. A key challenge for technical adequacy is that there is very little experience of performing MUPSAs. Most of the available guides and standards for performing PSAs and conducting peer reviews are based on the single reactor PSA model.
4.8 Terminology for multiunit PSA				New site risk metrics, such as SCDF, SLERF, SRCF and SCCDF, which parallel the SUPSA risk metrics, CDF, LERF, release category frequency (RCF) and CCDF, but yet have different meanings.

4.1. LEVEL 1 AND LEVEL 2 INTERFACE

4.1.1. General aspects of Level 1 and Level 2 interface for single unit PSA

Level 1 and Level 2 interface is the Level 2 PSA task aimed to transfer information from Level 1 PSA and to form initial conditions for Level 2 PSA performance[3]. Level 1 PSA identifies a large number of accident sequences that lead to core or fuel damage. It is impractical to treat each accident sequence independently in the Level 2 PSA when assessing accident progression, containment response and radionuclide release. Accident sequences are grouped together into PDSs in such a manner that all accidents within a given PDS can be treated in the same way in the Level 2 PSA.

PDSs represent groups of accident sequences with similar accident progression and, more important, which generate similar loads on the containment, thereby resulting in a similar severe accident progression that will influence the chronology of the accident, the containment response or the release of radioactive material to the environment. The attributes of the PDSs provide boundary conditions for severe accident analysis.

All severe accidents can be classified into two following main classes:

- Class in which radioactive material is released from the RCS to the containment; or
- Class in which the containment is either bypassed or is ineffective.

Therefore, initial status of the containment (e.g. intact and isolated, intact and not isolated, failed or bypassed) is the main attribute of PDSs. For PDSs where the containment is bypassed additional information is essential:

- Type of initiating event;
- Attributes that affect the timing of release;
- Attributes associated with attenuation of concentrations of radioactive material.

Typically, for PDSs with initial containment bypass, containment analyses are not needed and only source terms analyses are performed; however, even though assessment of mitigation features aimed to reduce releases (e.g. scrubbing of the releases), delay release or stop release (e.g. isolation of the pathway) can be performed.

For PDSs with intact containment, a containment analysis can be performed. Attributes that are considered in SUPSA for accidents with initially intact containment are well defined in Table 3 of [3]. These attributes define features of the plant at onset of core/fuel damage such as:

- Type of initiating event that cause core/fuel damage;
- Pressure in the reactor;
- Status of systems that can impact accident progression after core/fuel damage;
- Status of containment systems that are aimed to control pressure in the containment, conditions for hydrogen detonation/deflagration, source terms released into the environment;
- Status of systems aimed to maintain in–vessel retention (e.g. ex–vessel cooling systems).

If the consideration of all factors and parameters that affect the Level 2 PSA results in too large a number of potential PDSs, then they can be reduced to a manageable number. Two methods are recommended as per [3]: (a) combine similar PDSs and perform bounding analysis to select a representative sequence that characterizes the PDS for Level 2 PSA, and (b) use a frequency cut off as to screen out less important PDSs. In practice, both approaches are used in parallel. For example, PDSs are grouped in one final PDS even when consequences of scenarios associated with them differ if:

— Scenario associated with the representative PDS has the most severe consequences in terms of potential for containment damage and high source terms;
— Frequencies of other PDSs are much lower than frequency of representative PDS.

Important attributes that are included in PDS analyses, or distinct plants damage states that associate with external hazards, are discussed in [3]. For example, the potential to induce containment failure in a seismic event can be included in PDSs attributes, or seismically induced containment failure can be assimilated into attributes responsible for containment isolation failure.

4.1.2. Aspects of Level 1 and Level 2 interface for MUPSA (fuel in the reactor core)

The list of final PDSs defined for single unit typically contains up to 100 PDSs. When multiunit Level 2 PSA is performed it is impossible to consider all combinations of PDSs even for the site with 2 units (it is clear that for sites with more than 2 units the task is even more fantastic). Therefore, an approach for limiting the number of multiunit PDSs are developed. Multiunit Level 2 PSA considers accidents at several units (two or more) at the site that results in core damage and releases close in time. This means that the multiunit CDF is quantified for initiating events that occurred also close in time. This fact has significant impact on the attributes that are included in PDSs assessment in multiunit Level 2 PSA.

The term ‘close in time’ can be understood differently in member States and highly depends on national safety goals/targets/criteria established in regulations. For instance, in some Member States safety goals are defined for LER and early is limited to 24 hr and ‘close in time’ means that releases at several units occur within 24 hr and thus can be only generated by accidents affecting units simultaneously. In Member States where safety goals are defined for large releases and time is not limited (or has much longer duration, e.g. 10 days in Russian Federation) the ‘close in time’ means that releases at other units can occur much later than at the first damaged unit. In such case the damage at other units can be caused by different initiating events or even can be the results of core damage at the first unit. As an example, the accident caused by external flood due to slow water build up in spring snowmelt can affect one unit and much later other units, elevated higher. This flood event may not be considered in Member States where only LER is assessed but is important in Member States where the safety goals are established in terms of large releases. Also, effects of fuel damage at one unit on other units may not be accounted for if releases occur after 24 hr for the first group of Member States but need to be considered for the second. Further discussion assumes that large release frequency is in the focus of the assessment.

From Table 46 one can see that the four attributes that remain for non–seismic hazards are (1): pressure in the reactor, (2), status of active primary injection systems, (3) status of power supply system, and (4) spray system status. All other attributes listed in Table 46 are considered for the single unit that is damaged first, but for other units their status can be assigned as ‘Success.’

TABLE 46. CHANGES IN PDs ATTRIBUTES IN LEVEL 2 MUPSA

Attribute	In Level 2 SUPSA	Change in the attribute in Level 2 MUPSA	Comment
Containment integrity at onset of core damage	Steam generators tubes rupture	Can be excluded from multiunit Level 2 PDS analyses as frequency of close in time Steam generators tubes ruptures and consequential core damage at several units is negligible.	Seismic hazard has very limited potential to cause steam generators tubes ruptures.
	Loss of coolant accident in interfacing systems	Requires special consideration and can be excluded from multiunit Level 2 PDS analyses if it is shown that event cannot be caused by spurious opening of valves. Spurious operation can be caused by fires in specific plant locations.	When the event can be caused by spurious opening of valves the attribute still can be eliminated if it is shown that the same fire cannot affect several units.
	Containment isolation failure	Can be excluded from multiunit Level 2 PDS analyses as frequency of independent containment isolation failure at several units is negligible. The only LOSP events may impact containment isolation failure probability; however, typically LOSP has negligible impact on the system as it isolation systems always has back–up from non–interruptible power sources and in many pants has 'fail–safe' design.	When containment isolation systems does not have 'fail–safe' design attribute can remain for seismic hazards only if it not possible to neglect seismic damage of all power sources.
Type of initiating event that cause core/fuel damage	LOCA	All LOCAs except for Small and Very Small LOCAs can be excluded from multiunit Level 2 PDS analyses as frequency of close in time events and consequential core damage at several units is negligible. Small and Very Small LOCAs can remain in the list of attributes, but only for seismic hazards.	Seismic hazard can cause Small and very small LOCAs. LOCAs induced by pressurizer Safety Valves opening may also be considered for seismic hazards.
	Stream/feedwater line breaks	All steam/feedwater lines breaks can be excluded from multiunit Level 2 PDS analyses as frequency of close in time events and consequential core damage at several units is negligible. Small steam/feedwater lines breaks can remain in the list of attributes, but only seismic for hazards.	Seismic hazard can cause small secondary side breaks. Steam line leaks induced by steam generator safety valves opening may also be considered for seismic hazards.
	Initiating events, caused by operating system failures	All initiating events, caused by operating systems failures can be excluded from multiunit Level 2 PDS analyses as frequency of close in time events and consequential core damage at several units is negligible, except for seismic hazards.	
Pressure in the reactor	No changes	No changes.	No changes.
Status of systems that can impact accident progression after core/fuel damage	Active primary injection systems	No changes.	Status of primary injection systems may be important for multiunit Level 2 PSA as inter–units CCFs are the main reason for multiunit failures in these systems.
	Hydro accumulators	This attribute may be excluded from PSAs analyses as independent failures in passive systems have negligible probabilities and these failures are less dependent on seismic hazards.	Passive systems are robust for seismic hazards. However, justification is needed to support their exclusion.
	Power supply systems.	No changes.	Status of power supply is important for multiunit Level 2 PSA as external grid failure and inter–units CCFs are the main reason for multiunit failures in these systems.

TABLE 46. CHANGES IN PDs ATTRIBUTES IN LEVEL 2 MUPSA (CONT.)

Attribute	In Level 2 SUPSA	Change in the attribute in Level 2 MUPSA	Comment
Status of containment systems that are aimed to control pressure in the containment	Spray system	No changes.	Status of spray systems may be important for Level 2 MUPSA as inter–units CCFs are the main reason for multiunit failures in this system.
	CVS	This attribute can be excluded from PDSs analyses for all non–seismic PDS as independent failures of containment cooling systems have negligible probability.	CVSs typically vulnerable to seismic hazards.
	External and internal systems for containment cooling	Attributes related to these systems can be excluded from PDSs analyses for all non–seismic PDS as their independent failures have negligible probability.	External and internal containment cooling systems typically vulnerable to seismic hazards.
Status of containment systems to control hydrogen detonation/deflagration	Hydrogen recombines and igniters	Attributes related to hydrogen recombines and igniters can be excluded from PDSs analyses for all non–seismic PDS as their independent failures have negligible probability.	Hydrogen recombines and igniters remain attributes in PDSs analyses for seismic hazards if their robustness for seismic hazards is not justified.
	Core catcher	May be excluded from PSAs analyses as independent failures of core catchers have negligible probabilities and these failures are independent from seismic hazards.	Core catchers are robust for seismic hazards.
Status of systems for in–vessel retention	Ex–vessel cooling systems	Attributes related to ex–vessel cooling systems can be excluded from PDSs analyses for all non–seismic PDS as their independent failures have negligible probability.	Ex–vessel cooling systems remain an attribute in PDSs analyses for seismic hazards if their robustness is not justified.

Possible meaning for the states of attributes listed above are shown in Table 47. This table provides examples for the case of a two unit site; however, a similar approach can be extended to multiunit sites. The 15 atm pressure reference is based on low pressure ECCS actuation pressure.

It can be seen that instead of 2×4×2×4= 64 combinations of attributes for single units PDSs 4×7×4×9= 1,008 are used in multiunit PDSs analyses. This number is significant, but it can be managed with current PSA software. However, further reduction in PDS can be achieved with approaches that are the same or like those used for single units PDS analyses.

The first approach that is based on grouping of similar PDSs remains the same as in single unit PDS analyses but may involve additional considerations. One of them is that, when plants are identical or similar in terms of power, plant systems, and reactor design, there is no need to distinguish status of attributes at a particular plant. In this case Table 46 can be simplified as shown in Table 48. Instead of 1,008 combinations only 3×5×3×6= 270 can be used in multiunit PDS analyses.

TABLE 47. SUGGESTED VALUES FOR PDS ATTRIBUTES IN LEVEL 2 MUPSA (FULL SET)

Attribute (example)	**Meanings at 1^{st} unit**	**Meanings at 2^{nd} unit**	**Meaning in Level 2 MUPSA**
Pressure in the reactor	P1L<15 atm → (L1) P1H>15 atm →H1)	P2L<15 atm → (L2) P2H>15 atm → (H2)	L1& L2 →LL L1& H2 →LH H1& L2 →HL H1& H2 →HH
Status of active primary injection systems	High pressure injection available (J1) Only low–pressure injection available (I1) Low pressure injection and recirculation available (R1) Low pressure injection, and recirculation not available (F1)	High pressure injection available (J2) Only low–pressure pressure injection available (I2) Low pressure injection and recirculation available (R2) Low pressure injection, and recirculation not available (F2)	J1&J2 →JJ I1&I2 →II I1&R2 →IR I1&F2 →IF I2&R1 →RI I2&F1 →FI F1&F2 →FF
Status of power supply system	All safety buses are lost (O1) Power available on at least 1 safety bus (A1)	All safety buses are lost (O2) Power available on at least 1 safety bus (A2)	O1&O2 →OO O1&A2 →OA A1&O2 →AO A1&A2 →AA
Spray system status	Spray system available only in the injection phase, Spray system not available in the recirculation phase (I1) Spray system available in both injection and recirculation phase (R1) Spray system unavailable (F1)	Spray system available only in the injection phase, Spray system not available in the recirculation phase (I2) Spray system available in both injection and recirculation phase (R2) Spray system unavailable (F2)	I1&I2 →II I1&R2 →IR I1&F2 →IF R1&I2 →RI R1&R2 →RR R1&F2 →RF F1&I2 →FI F1&R2→FR F1&F2 →FF

TABLE 48. SUGGESTED VALUES FOR PDS ATTRIBUTES IN LEVEL 2 MUPSA (REDUCED SET)

Attribute (example)	**Meanings 1^{st} unit**	**Meanings at 2^{nd} unit**	**Meaning in Level 2 MUPSA**
Pressure in the reactor	P1L<15 atm → (L1) P1H>15 atm →H1)	P2L<15 atm → (L2) P2H>15 atm → (H2)	L1& L2 →LL L1& H2 (H1& L2) →LH H1& H2 HH
Status of active primary injection systems	High pressure injection available (J1) Only low–pressure pressure injection available (I1) Low pressure injection and recirculation available (R1) Low pressure injection, and recirculation not available (F1)	High pressure injection available (J2) Only low–pressure pressure injection available (I2) Low pressure injection and recirculation available (R2) Low pressure injection, and recirculation not available (F2)	J1&J2 → JJ I1&I2 →II I1&R2 (I2&R1) →IR I1&F2 (I2&F1) →IF F1&F2→FF
Status of power supply system	All safety buses are lost (O1) Power available on at least 1 safety bus (A1)	All safety buses are lost (O2) Power available on at least 1 safety bus (A2)	O1&O2 →OO O1&A2 (A1&O2) →OA A1&A2 →AA
Spray system status	Spray system available only in the injection phase, Spray system not available in the recirculation phase (I1) Spray system available in both injection and recirculation phase (R1) Spray system unavailable (F1)	Spray system available only in the injection phase, Spray system not available in the recirculation phase (I2) Spray system available in both injection and recirculation phase (R2) Spray system unavailable (F2)	I1&I2 →II I1&R2 → (R1&I2) IR I1&F2 F1&I2 →IF R1&R2 →RR R1&F2 → (F1&R2) RF F1&F2→>FF

The second approach (screening by frequency) has the highest priority as it is expected that frequency cut off screening of numerous PDSs can be done. This approach uses the same technique as in Level 2 SUPSA, after PDS quantification those that are below a predefined threshold are excluded from the analyses. Transparent results of PDS analyses are obtained if the coding scheme, similar to that in Level 2 SUPSA plant damage states analyses, is used. An example that is based on attributes listed in Tables 46–48 for non–seismic Level 2 MUPSA is as follows:

- 1st and 2nd symbols – pressure (LL,LH, HL, HH);
- 3rd and 4th symbols – ECCS status (JJ, II, IR, IF, RI, FI, FF);
- 5th and 6th symbols – spray system status (II, IR, IF, RI, RR, RF, FI, FR, FF);
- 7th and 8th symbols – containment status (II,IF,FI,FF);
- 9th and 10th symbols – status of power supply system (OO,OA,AO, AA);
- Example: LLIIIRIAIFIIOO:
  - LL– low pressure in both reactors;
  - II – high pressure injection available at both plants;
  - IR spray system available in injection mode at 1st unit and in recirculation at the second;
  - II containment isolated at both units;
  - OO – loss of all emergency power buses at both units.

The example of a bridge tree that can be used to develop PDSs using RiskSpectrum software is shown in Fig. 72. It can be noted that when Level 2 MUPSA is developed for seismic hazards all simplifications discussed above are not applicable. However, currently there is no clear guidance on how Level 1 and Level 2 MUPSA need to be done for seismic hazards.

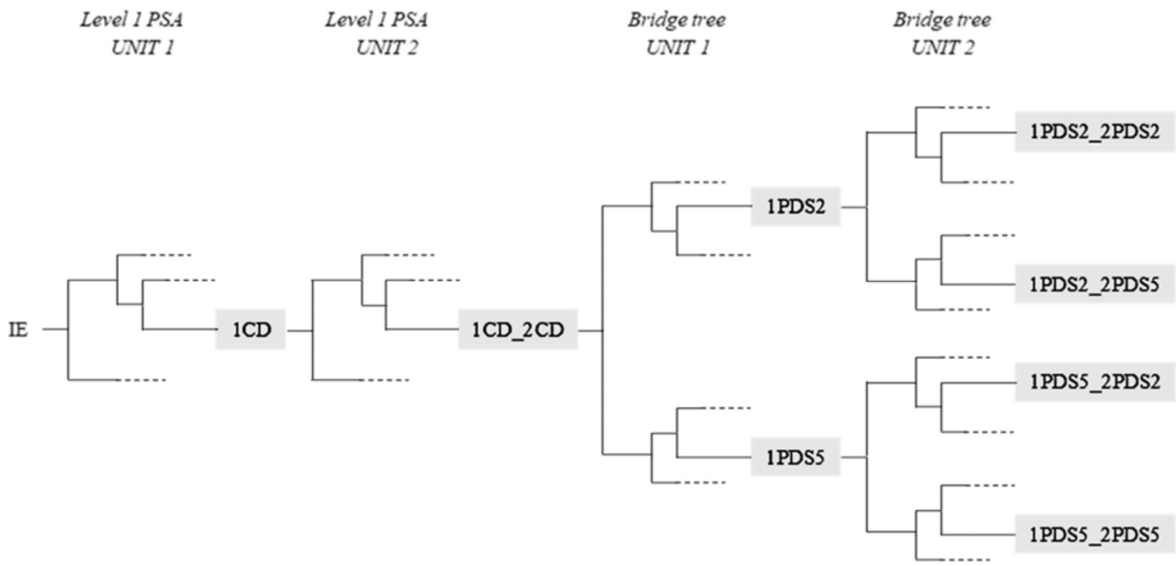


FIG. 72. Bridge tree for multiunit PDS.

4.1.3. Aspects of Level 1 and Level 2 interface for MUPSA (fuel in the reactor core and in refuelling pool)

It is noticed that Level 1 PSA in [2] is limited to consideration of fuel in SFP only for shutdown modes when fuel is removed from the reactor (see para 1.13 [2] Also, SSG–3 in para 2.3 recommends considering the impact of radioactive releases from other radioactive material on the site while assessing the total risk from the plant to members of the public near the site. It does not provide recommendation to perform Level 1 PSA for spend fuel pool for other than refuelling states conditions, and therefore this also limits consideration of fuel in spend fuel pool for Level 2 and Level 3 PSAs. Member States experience in performing spend fuel pool PSAs generally confirms that fuel melt frequency in SFP is rather low, comparing with core damage frequency. However, these estimations are typically based on the 24 h mission time assumption. Amount of water in SFP is large and heat release rate from the fuel is low (at least for full power operation) and therefore most of accident sequences in spend fuel pool typically has no impact on safety within 24 hr. Therefore, only sequences with SFP leaks have certain small contribution to fuel damage risk.

However, if consideration is extended beyond 24 hr this contribution may be much higher. Taking into account that cooling of fuel in the reactor and SFP is provided by the systems that depends on the same support systems it is highly likely that this damage may occur in single accident scenario caused by the same initiating event. This aspect was not discussed neither in [2] nor in safety guide on Level 2 PSA [3].

Another aspect of this limitation is that probability of containment damage in severe accidents depends on hydrogen generation rate and static loads due to pressure in the containment. For those reactors where spend fuel pool and reactor share the same containment building the damage of the fuel in the reactor accompanied with damage of fuel in spend fuel pool may lead to higher hydrogen generation and faster pressure increase in the containment at latest stages of the accident. This in turn may lead to higher loads on the containment and higher probability of containment damage.

Concerning Level 1 PSA consideration of simultaneous damage of fuel in the reactor and spend fuel pool typically does not provide additional insight in overall plant risk. If core damage and fuel damage in spend fuel pool occurs due to same initiating event or failure of the same support systems, the overall risk of fuel damage will be mainly driven by core damage risk. However, in Level 2 (and Level 3) PSA situation is different. Fuel damage in the reactor and spend fuel pool will lead to large releases of radioactive materials in the containment and much severe releases to the environment in case of containment damage comparing with releases with damage of fuel in single source. Therefore, Level 2 PSA has to consider both sources of radioactivity for all operational states, focusing on accident sequences where damage occur due to the same reasons.

This aspect is also important for multiunit Level 2 PSA, where one unit may have damage of fuel in the reactor only, another – in the reactor and SFP. Even when radioactive releases from single unit and single source of radioactivity are below safety goals, consideration of damage of several units and several sources may lead to violation of safety goals and requires technical or organizational measures to prevent such releases.

4.2. RISK METRICS AND SAFETY GOALS

The risk metrics depend on the adopted national regulatory framework for PSA and quantitative health objectives and on the decision taken for the manner MUPSA is to be performed; MUPSA may be performed in two options as per IAEA SRS-96 [20]:

In Option 1 of PSA Level 2, i.e. without doing Level 3 analysis:

- RCF;
- SLERF;
- Others as required.

In Option 2 of PSA Level 2, i.e. by doing Level 3 analysis

- Site quantitative health objectives for individual risk; (Level 3);
- RCF;
- SLERF;
- Others as required.

The overall Level 2 risk is composed of three component groups in a MUPSA:

1) Sequences involving accidents at single reactor unit that correspond to LERF sequences in the single reactor unit PSAs for that site;
2) Sequences involving multiple reactor unit accidents that correspond to LERF sequences in the single reactor PSAs with increased source terms reflecting multiple releases;
3) Sequences involving multiple reactor unit accidents that involve combinations of non–LERF sequences in the single reactor PSAs but are not sufficient to produce early health effects because of an increased total source term.

The SLERF is the frequency per site year of accidents with a large early release, either from a single unit or from the combination of releases from multiple units. There are some specific features to be considered for the MUPSA Level 2 risk metrics:

- Quantitative objectives for multiunit plants are based on either Level 3 PSA or other studies;
- These objectives may be direct measurements or surrogates to multiunit quantitative safety goals or health objectives, such as the traditional CDF and LRF for MUPSA applications;
- MUPSA metrics involve aggregating risks from multiple radiological hazard sources and initiating events, which is a decision at national level.

Insights from supporting information to MUPSA Level 2 (Level 3 MUPSAs, EP, etc.) and supporting radiological consequence analyses are very limited at this point for MUPSA but are necessary to establish suitable site level metrics. In this section, some of these metrics presently proposed for MUPSAs are discussed as follows:

Level 2 multiunit risk metrics: the frequency of radionuclide release from the site forms the basis for the multiunit Level 2 metric. The definition of what release is large, specially for multiple discrete releases or nearly concurrent releases from the sites is very important. However, it is still under review and presently is decided in each case based on national regulations or requirements. There is a diversity of categories (or grouping) of releases and their frequency per year associated with the identified multiunit accident sequences in terms of the nature, timing, and magnitude of the release and there is a need to support them for MUPSA by evaluations (in mechanistic codes for instance).

In the evaluation of the releases the important factors include:

— Response of the containment structure, timing and mode of containment failure;
— Timing, magnitude, and composition of released radionuclides;
— Thermal energy of release;
— Deposition and removal of radionuclides.

Consistent with the multiunit CDF definitions, two LRF multiunit measures are currently proposed and were included in the MUPSA risk metrics:

— Frequency of all possible scenarios (or group of scenarios), each leading to a large release from one or more radiological source terms on a site per year, referred to as the site large release frequency (SLRF);
— Frequency of a specific release category from core damage of one or more units on a site per year or due to damages of other radiological sources per year, referred to as the SRCF.

Level 3 Multiunit Risk Metrics (Argentina/CNEA): in many cases MUPSA results in site consequences from one or more nearly concurrent or sequential releases that form the basis for defining the Level 3 risk metric. Argentina/CNEA has developed a proposed site MUPSA risk metric safety goal that correspond to partial Level 3.

The consequences are often in the form of:

— Prompt fatality;
— Long–term health effects or fatalities;
— Economic losses.

Radiological consequence results are also useful in determining whether safety targets and goals are attained. The risk aggregation over all site–level initiating events needs to be consistent with the risk metrics adopted for MUPSA.

Possible definition of Level 2 MUPSA risk metrics (Hungary): The Level 1 PSA for NPP Paks includes the quantification of LERF in the reactor PSA and in the SFP PSA separately for each of the four units. In principle, the frequency of single and multiple large or early release sequences have to be known and aggregated correctly to quantify risk at site level. Since the assessment covered two units, the following risk metrics were applied that have been adopted from similar definitions used in Level 1 PSA:

— LERF on at least unit 1 (or unit 2);

- Single unit LERF (single unit LERF): frequency of large or early release only on unit 1 (or unit 2);
- Multiunit LERF (multiunit LERF): frequency of large or early release on both units (1 and 2);
- SLERF frequency of large or early release at the site.

LERF was calculated by assessing the unit specific PSA model and multiunit LERF was quantified based on the MUPSA model. Single unit LERF and SLERF are derived metrics that were calculated on the basis of the following relations:

Single unit LERF = LERF – (Multiunit LERF)

SLERF = (Single unit LERF1) + (Single unit LERF2) + (Multiunit LERF) = LERF1 + LERF2 – (Multiunit LERF)

Moreover, the ratios of LERF, (Single unit LERF) and (Multiunit LERF) to SLERF (i.e. LERF / SLERF; (Multiunit LERF) / SLERF; (Single unit LERF) / SLERF), as well as LERF/CDF, (Multiunit LERF) / (Multiunit CDF) and SLERF / SCDF were also calculated, since these metrics are seen as important to yielding valuable risk insights. There are no multiunit or multi source safety goals available in Hungary at present, so the results were not compared to any predefined goals or criteria. At present no country has defined safety goals for Level 2 multiunit analysis. Multiunit sites use single unit safety goals.

4.2.1. Example of site and MUPSA risk metric and safety goals proposal based on IRR from Argentina/CNEA

In this section is resumed the proposal of IRR as risk metric and safety goal for the site, considering the contribution of the MUPSA.

In first time, the conceptual base of IRR as risk metric is explained. Considering the conceptual base of the safety goal that is used for a single unit, a safety goal for the site has been derived. This proposal has demanded the development of a risk metric for the site. The IRR as site risk metric has to consider the risk that is calculated from single units and the risk that comes from initiating events that affect multiple units. From this metric, a proposed IRR risk metric for MUPSA is derived. Methodological aspects for IRR calculation due to releases from two or more radioactive sources are explained.

The safety goal is derived considering the risk value calculated from the dose limit established to normal radioactive exposures and imposing that the potential risk (due to accidental sequences) need to be lower. The safety goal for MUPSA is derived as a contribution to the total site risk.

4.2.1.1. Conceptual base of IRR as single units risk metric

The IRR is based on the Argentine Acceptability Criterion for licensing NPP and research reactors, defined by the Regulatory Body in Argentina (Autoridad Regulatoria Nuclear – ARN). The IRR is defined as the probability of intersection of two events: exposure to ionizing radiation and the fatality due to this exposure, as $IRR = P(E \cap F)$, where IRR is the Individual radiological risk (risk metric), E is the event of exposure to ionizing radiation and $F$is the event of fatality due to exposure. Considering the dependence between the events of exposure and fatality, this can be rewritten as $IRR = P(E) \times P\left(\frac{F}{E}\right)$, where $P(E)$ is the probability of event of exposure to ionizing radiation, $P(F/E)$: is the Probability of event of fatality due to exposure.

On one hand, the probability of exposure (*P(E)*) is calculated through Level 1 and Level 2 PSA development and a Partial Level 3 PSA to model transport and dispersion of the release to the environment, considering the site characteristics and individual exposure paths among other aspects to quantify the dose in the members of the public. On the other hand, the term *P*(*F/E*) can be evaluated as a function of dose as IRR = *P*(*E*)×*f*(*d*), where$f(d)$ is the probability of fatality as result of dose.

The function $f(d)$ considers stochastic and deterministic health effects and is defined by the regulatory authority of Argentina as follows:

$$f(d) = \begin{cases} 1 \times 10^{-5} & d \leq 0.0002\,Sv \\ 0.05\,d & 0.0002\,Sv \leq d \leq 1Sv \\ 0.05\,d^{1.67} & 1\,Sv \leq d < 6Sv \\ 1 & d \geq 6Sv \end{cases} \tag{20}$$

where *d* is a dose in Sv.

The definition of the IRR as risk metric developed by the Regulatory Body in Argentina, is based on Farmer proposal. This criterion evaluates the consequence in Curies. When developing partial Level 3 PSA, required to calculate the IRR, the unknown is the moment of the accident occurrence, and therefore the meteorological conditions at that time, which in turns produces a given value of dose in the members of the public in the NPP surroundings. In the present evaluation, it is assumed a uniform probability distribution for the moment of the accident along the year. Each meteorological condition is considered with its probability of occurrence, in a year period, and its consequent doses in the members of the public at the NPP surroundings are evaluated. Then, the associated risks (IRR) in each position are calculated.

For a given release category, all the IRR, due to the different meteorological conditions, in each position are added. The maximum value in the domain is selected. The total NPP risk to the public is the sum of maximum IRR of all release category and can be compared with the safety goal. The sums over the risks can be done as they are disjoint events. If the accident occurs in each day, it will not occur in another (different meteorological conditions), and if a given release category occurs, another will not occur.

4.2.1.2. Definition or individual radiological risk as a risk metric for the site

As a general expression for site risk metric proposal, the site total risk can be evaluated as the sum of the risks related to single initiating events that could affect only one unit or radioactive source and the risk due to events that can affect multiunit. It can be expressed as it is indicated in:

$$IRR_{Total-Sit}^{MAX} = \sum_{k=1}^{m} IRR_{PSA_k}^{MAX} + IRR_{MUPSA}^{MAX} \tag{21}$$

where:
$IRR_{Total-Site}^{MAX}$: maximum IRR, calculated for all the site
$IRR_{PSA_k}^{MAX}$: maximum IRR, calculated for each kth radioactive source due to single initiating events

IRR_{MUPSA}^{MAX}: maximum IRR, calculated for the site, due to IEMs
kth: radioactive source

For calculation of the $IRR_{Total-Sit}^{MAX}$ the following points have to be taken into account:

- Initiating event that can affect more than one unit is to be considered only into MUPSA analysis, and they have to be excluded from PSA of units or radioactive sources. Because, the dependences among units are not considered in the sequences of single PSA;
- In Level 1 MUPSA development, sequences that imply core damage of at least one unit or radioactive source have to be considered. It highlights that in other MUPSA development, these sequences are excluded from MUPSA, but they are required for proposed site risk metric assessment.

Then, considering the release categories in the IRR calculation, can be expressed as:

$$IRR_{j,\,Total-Sit} \;=\; \sum_{k=1}^{m}\sum_{n=1}^{p} IRR_{j,PSA_k}^{RC^n} + \sum_{k=1}^{m}\sum_{x=1}^{t} IRR_{j,\,MU-U_k}^{RC^x} + \sum_{y=I}^{z} IRR_{j,\,MUPSA}^{MURC^y} \quad (22)$$

where:
$IRR_{j,\,Total-Sit}$: IRR calculated at j^{th} position, for all the site.
$IRR_{j,PSA_k}^{RC^n}$: IRR calculated at j^{th} position, due to the nth release category that comes from each k^{th} radioactive source. It applies to internal initiating events that result in fuel damage in single radioactive source.
$IRR_{j,\,MU-U_k}^{RC^x}$: IRR calculated at j^{th} position, due to the xth release category that comes from each k^{th} radioactive source. It applies to a IEM that derives in fuel damage in only one radioactive source.
$IRR_{j,\mathrm{MUPSA}}^{MURC^y}$: IRR calculated at j^{th} position, calculated for the y^{th} multiunit release category (multiunit release category). It applies to a multiunit initiating event that derives in fuel damage in more than one radioactive source.

The first term represents the risk regarding to single units (or single radioactive source), at the j^{th} position. Then, the k^{th} and n^{th} sub index represent the radioactive source and the release category respectively. This first term ($\sum_{k=1}^{m}\sum_{n=1}^{p} IRR_{j,PSA_k}^{RC^n}$) is calculated from single PSA (Level 1, Level 2 and Partial Level 3 PSA).

The second and third terms are calculated through MUPSA development. The second term represents the risk derived from sequences originated by multiunit initiating event that implies damage only in one radioactive source, at j^{th} position. The x^{th} sub index represent the release category from single unit. The last term represents the risk associated to y^{th} multiunit release category, the releases from more than one radioactive source. To calculate each term of Eq. (22), the release category or multiunit release category can occur at any instant within the temporary domain, then, whatever meteorological condition can occur. The probability of intersection of event exposure to a release category and fatality due the exposure in the j^{th} location, considering the different meteorological conditions is shown in:

$$IRR = P(E_{ji} \cap F_{ji}) \quad (23)$$

where:
E_{ji}: event of exposure to ionizing radiation, at j^{th} position, due to *ith* meteorological condition;

F_{ji} : event of fatality due to exposure, at j^{th} position, due to *ith* meteorological condition.

Then, the IRR in the j^{th} location can be expressed as the sum of IRR due to every meteorological condition, as is shown in:

$$IRR = P(E_{ji} \cap F_{ji}) \tag{24}$$

where:
$IRR_j^{RC^n}$: IRR due to n^{th} release category, at j^{th} position;
$IRR_{ji}^{RC^n}$: IRR due to n^{th} release category, at j^{th} position, due to i^{th} meteorological condition.

The sums over the risks can be done as they are disjoint events. If the accident occurs in a given moment it will not occur in another. It implies different meteorological conditions and therefore different doses. Moreover, the location with the highest risk can be identified. It is considered that protecting the member of the public located there, taking into account the safety goal, everyone is protected:

$$IRR_{PSA_k}^{MAX} = MAX(IRR_j^{RC^n}) \tag{25}$$

where $IRR_{PSA_k}^{MAX}$: Maximum IRR calculated for each k^{th} radioactive source. Then, according to Eq. (24), Eq. (21) can be expressed as it is shown in:

$$IRR_{Total-Site}^{MAX} = \sum_{k=1}^{m}\sum_{n=1}^{p} MAX(IRR_{j,PSA_k}^{RC^n}) + \sum_{k=1}^{m}\sum_{x=1}^{t} MAX(IRR_{j,MU-U_k}^{RC^x}) + \sum_{y=1}^{z} MAX(IRR_{j,MUPSA}^{MURC^y}) \tag{26}$$

The sums over the risks can be done as they are disjoint events. If a given release category occurs, another will not happen. Considering the identified terms in Eq. (26), Eq. (21) can be rewritten in the following way:

$$IRR_{Total-Site}^{MAX} = \sum_{k=1}^{m} IRR_{PSA_k}^{MAX} + \sum_{k=1}^{m} IRR_{MU-U_k}^{MAX} + IRR_{MUPSA}^{MAX} \tag{27}$$

where $IRR_{MU-U_k}^{MAX}$: maximum IRR calculated for each *kth* radioactive source. A multiunit initiating event occurs, but only one radioactive source is affected).

The contributions to the second and third term of Eq. (27) are represented in Fig. 73. It considers a multiunit initiating event like LOOP that can affects generic units A and B. Then, one of the units or both can derive in fuel damage. Finally, $IRR_{Total-Sit}^{MAX}$ can be calculated.

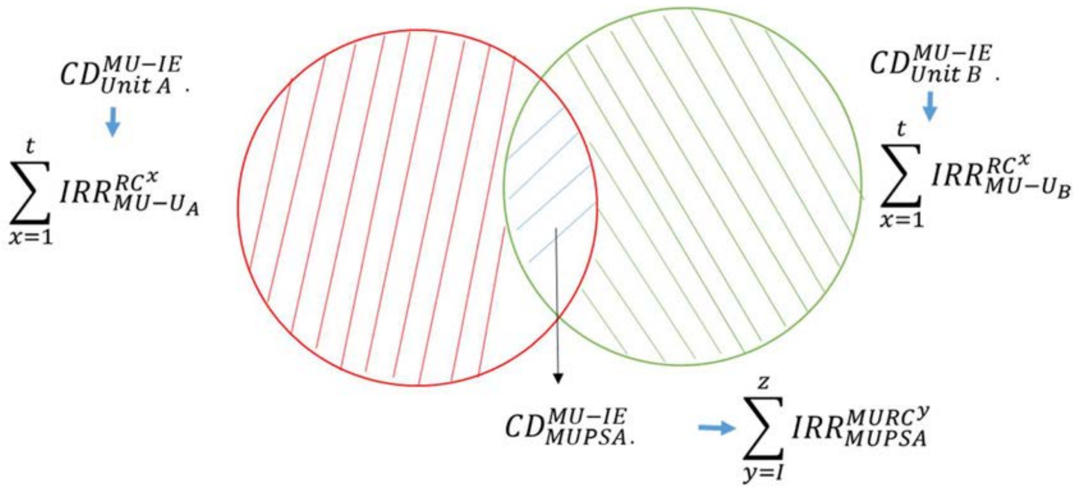


FIG. 73: Representative graph of contributors to 〖IRR〗_(Total-Site)^MAX due to multiunit initiating event.

4.2.1.3. *Definition or safety goal for IRR site and MUPSA risk metric*

The Regulatory Body in Argentina has established an Acceptability Criterion for IRR risk metric applied to single NPP or Research Reactors. The objective of the legal safety goal, to apply to single units, is to limit the IRR for members of the public due to potential accidental exposures during a given period, surrounding to the nuclear plant, to risk values lower than risk accepted for normal radiological practices. For obtaining a limit value, the following assumptions are made:

— Taking into account the ICRP criteria, the Regulatory Body in Argentina has established as limit for a normal radiological practice for members of the public an annual dose of 1 mSv;
— Dose is located in the stochastic region;
— *P(E)=1*, normal radiological practice.

Then:

$$IRR^{MAX}_{Total-Sit} = 1 \times 0.05Sv^{-1} \times 0.001Sv = 5 \times 10^{-5} \quad (28)$$

Instead of the factor applied by the Regulatory Body for uncertainties in the case of IRR for a NPP or Research Reactors, it is proposed include the uncertainties in the PSA models. The uncertainties in PSA give treatment not only regarding the values of the parameter used in probabilistic models but also regarding PSA scope, technical adequacy and epistemic uncertainties in the deterministic and probabilistic models. Nowadays, calculation codes allow the uncertainty treatment at least in the parameter values, using for example Monte Carlo simulations. For instance, the proposal safety goal need to be compared with the 95% of the IRR obtained for the site. The proposed safety goal implies that single PSA and MUPSA need to be developed to evaluate its accomplishment. As a consequence, partial safety goals, like the safety goal for MUPSA, then need to be analyzed.

4.3. MODELLING APPROACH

The technical considerations are specific to each study. However, there are some practices for building the models in MUPSA. They are included in documents under development at national and international (IAEA) levels and they fall under two approaches:

- To build master fault trees for the entire model;
- To build event trees and assure their interface with the plant reaction (function events) by considering the correlation (switches) for the elements that do not have to be accounted for in any type of multiunit initiating event. This approach also implies either to evaluate all the possible combinations for failures for the units or to consider one combination as conservative.

Combinations of the two methods is also possible and the individual reports reflect the fact that this methodological aspect of MUPSA is yet under review and testing. Therefore, the information on particular experience is very important.

4.3.1. Argentina/CNEA

4.3.1.1. Level 2 MUPSA modelling approach

In the next paragraphs, the developed Level 2 MUPSA for case study is explained. Similar to Level 1 MUPSA, the Level 2 analysis is developed following several steps based on event tree technique. A simplified and integral Level 2 event tree with units 1, 2 and the SFP for each site damage state is the followed approach. The steps are described as follows:

Step 1 – Objective: objective of Level 2 MUPSA is to obtain multiunit release category, characterized by deterministic attributes and its frequencies. Moreover, taking into account the proposed risk metric, the release categories derived from Level 1 sequences that implies fuel damage in only one radioactive source also need to be obtained.

Step 2 – Scope, hypothesis and general assumptions: given the proposed case study and regarding Level 2 MUPSA for units 1 and 2, the following assumptions are made:

- Two possible phases are considered during the severe accident progression: in–vessel and ex–vessel. For in–vessel phase, the RPVECS is considered to avert an ex–vessel progression;
- CVS and PARS are analyzed to set the release pathway. Two possible pathways are evaluated: catastrophic failure of containment building or pulsed release by controlled venting;
- Regarding the site, it is considered that the RPVECS is shared between units 1 and 2. Thus, if the severe accident occurs in both units at the same stage, it is considered that this system can be used to mitigate the accident in only one of the units at a time. On the contrary, if the severe accident occurs in different stages for each unit, given the time availability between Stage 1 and 2, it is considered that a failure of this system during Stage 1 could be recovered to be demanded in Stage 2;
- For Level 2 consequences and multiunit release category definition, it is not considered the stage where the core damage has occurred. It is supported by doses calculation in the context of the current

Partial Level 3 MUPSA scope.

Regarding the SFP, it is considered that:

- Spray system will limit the fuel elements damage. The analysis of the failure of that system is left out of scope;
- It is assumed the failure of the confinement function;

Simplified event trees are modelled to facilitate the calculations to obtain the frequencies associated to the multiunit release category and proceed with the focus on the risk metric proposal and calculation. The headers are modelled as basic events or simplified fault trees. Screen values are used to carry out quantification of the model.

Step 3 – Model development:

- **Mitigation strategy for severe accident**

  Given a core damage scenario, the adopted strategy is to maintain the corium inside the RPV. This is possible due to the ratio between the low plenum area and the core power in integral type PWRs. Thus, after core damage occurrence, the RPV will be cooled by means of the RPVECS to maintain the corium inside. Moreover, the hydrogen concentration present inside containment will be controlled with the PARS. Containment pressure will be controlled with the CVS. The combined performance of the PARS and the CVS will attempt to maintain the integrity of the containment building. Regarding the SFP building, the the SFP spray cooling system will attempt to limit the fuel elements damage, given a scenario of fuel elements uncovering due to coolant boiling when the cooling system for SFP and SFP injection DEC system are assumed to fail.

- **Dependencies analyzes**

  Dependencies analyses are a main element to be evaluated in MUPSA. For this case of study, it is analyzed the dependencies among radioactive sources. In this case, only the RPVECS is shared between both units. EWSS is shared by the units and SFP.

- **Headers identification**

  According to previous steps and assumptions, headers for the event tree development are identified. Some systems are grouped in the same header to reduce event tree complexity.

- **Event tree development**

  For each site damage state described in Level 1, an event tree is constructed considering the headers of unit 1, 2 and SFP in an integral event tree. RiskSpectrum PSA 1.3.2 is used as calculation code. Level 1 and Level 2 MUPSA are integrated in the same RiskSpectrum PSA project. To mitigate the severe accident, the following actions are set for each unit. In the first place, the RPVECS will be required to prevent ex–vessel progression.

If the core damage in units 1 and 2 occurred in the same stage, and if this system succeeds for unit 1, then it will not be possible to provide external cooling of RPV in unit 2. On the contrary, if the CD had occurred during different stages for the units (e.g. during Stage 1 for unit 1 and Stage 2 for unit 2), then it is considered that there is enough time to perform recovery actions in case of failure during Stage 1, and to require the system during Stage 2.

The success or not of RPVECS determines the types of RCs from the core to the containment. Regardless RPVECS success or failure, CVS and PARS will be required to maintain containments integrity. These systems success or failure will determine the release pathway:

— Pulsed: if both systems succeed;
— Catastrophic containment failure: if any of these systems fail.

With respect to the SFP, the spray cooling system is required to limit the damage of the fuel elements. Taking into consideration previous statements, four release categories are defined for each unit and one for the SFP. For each sequence of each MUPSA Level 2 event tree, a multiunit release category is built with the units or SFP respective release categories. Finally, the different multiunit release category groups are defined. In this stage the same release category for each unit are used, but without distinguishing the stage of the occurrence of the core damage in each unit, to reduce the number of categories for this case of study. The frequency of each multiunit release category group is the sum of the frequencies of the multiunit release categories allocated to them. As an example, a developed event tree is shown in Fig. 74 for the case derived from site damage state where the fuel damage in both units was at the same stage, and no fuel damage occurred in the SFP. Due to previously explained design criteria for RPVECS, its respective header for unit 2 it is not part of this event tree. If the system is required by unit 1 and succeeds then it is not available for unit 2, and if it fails it cannot be recovered to be available for unit 2.

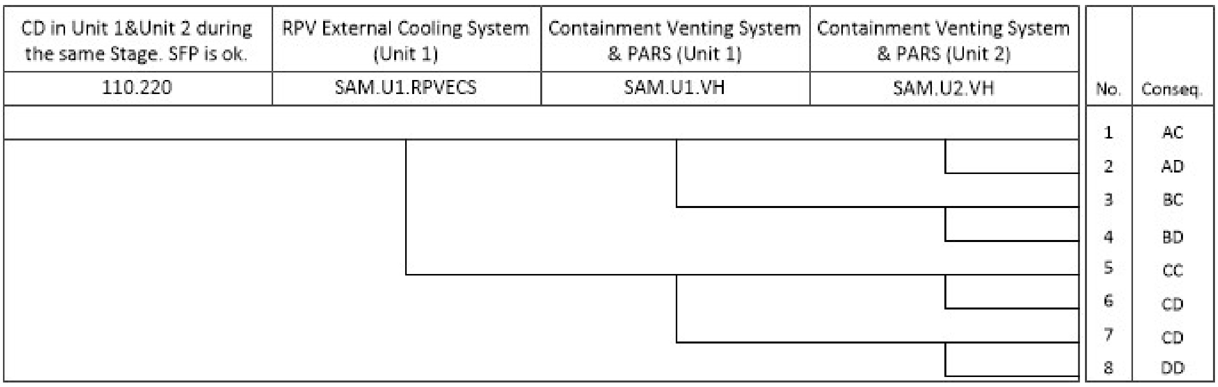


FIG. 74. PSA Level 2 event tree for case with fuel damage in units 1 and 2 during the same stage.

Step 4 – Deterministic analysis

For this case of study, the deterministic characterization of release categories that are release from units and SFP are extrapolated from bibliography, with additional simplifications. Different release categories are defined in Tables 49 and 50, considering the successful or failure of RPVECS and CVS and PARS, and the damage of SFP. The associated release categories for each unit and SFP are characterized, considering the SOARCA study [27] and additional hypotheses.

TABLE 49. DETERMINISTIC CHARACTERIZATION OF RELEASE CATEGORIES FROM UNIT 1 AND UNIT 2

RPVECS/ CVS+PAR	Release Category	Description	Hypotheses for release category definition
Yes/ Yes	A	RPV intact. In–vessel progression. Successful venting from the chamber of the suppression pool and H_2 control that leads to a pulsed release.	The release fractions for all the release category are obtained from SOARCA study. Pool retention coefficients are applied considering the venting from the suppression pool chamber. The release energy is calculated based on the energy accumulated in each unit due to the decay power and the venting time. The release through the venting begins 20 h after reactor SCRAM. It is assumed that it is required because the success of the RPVECS generates an additional source of steam. Each venting last one hour, with an interval of eight hours between the onsets of each venting. Given the 48 h analysis frame, four venting are considered. Height of release, as the venting is through the stack: 40 m are considered.
Yes/ No	B	RPV intact. In–vessel progression. Failure of venting and H_2 control that leads to a catastrophic failure of containment building.	The release fractions for all the release category are obtained from SOARCA study. The release energy for is taken from SOARCA study that corresponds with a catastrophic containment failure. The release begins 42 h after reactor SCRAM, when a catastrophic containment failure is postulated. Then, a continuous release is assumed, and modelled as one release per hour. Height of release, 20 m are considered.
No/ Yes	C	RPV damaged. Ex–vessel progression. Successful venting from the chamber of the suppression pool and H2 control that leads to a pulsed release.	The release fractions for all the release category are obtained from SOARCA study. Pool retention coefficients are applied considering the venting from the suppression pool chamber. The release energy is calculated based on the energy accumulated in each unit due to the decay power and the venting time. The release through the venting begins 24 hr after SCRAM, a few hours later than release category A, as there is no steam production because of the RPVECS failure. Each venting last one hour, with an interval of 8 hr between the onsets of each venting. Given 48 hr analysis frame, four venting are considered. Height of release, as the venting is through the stack, 40 m are considered.
No/ No	D	RPV damaged. Ex–vessel progression. Failure of venting and H2 control that leads to a catastrophic failure of containment building.	The release fractions for all the release category are obtained from SOARCA study. The release energy for is taken from SOARCA study that corresponds with a catastrophic containment failure. The release begins 36 hr after SCRAM, because of the venting failure, when a catastrophic containment failure is postulated. Then, a continuous release is assumed, and modelled as one release per hour. Height of release, 20 m are considered.

TABLE 50. DETERMINISTIC CHARACTERIZATION OF RELEASE CATEGORY FROM SFP

SRS	Release Category	Description	Hypotheses for release category definition
Yes	X	Fuel elements limited damage	a) The release begins 60 h after SCRAM. b) Accident time: 5 h. c) The release is decrescent exponential function. d) Inventory: half core with 30 days of decay; half core with 365 days of decay. e) Release fractions from [29]

Step 5 – Results analysis: after the development of the Level 2 event trees and considering the combination of release categories of each unit and SFP, 28 multiunit release categories are obtained. Each multiunit release category has associated RCs (identified as A, B, C and D) that comes from each unit and/or release category (X) from SFP with respective frequency of occurrence. These frequencies and release category deterministic characterization are the required input for Level 2– Partial Level 3 MUPSA interface.

4.3.1.2. Level 2 – partial level 3 MUPSA interface

The objective of Level 2 – Partial Level 3 MUPSA Interface is to define multiunit release category groups, considering same RC. As results, 18 multiunit release categories are defined. The multiunit release categories group frequencies are required as input to Partial Level 3 MUPSA. The multiunit release category deterministic attributes are the same that were characterized in Level 2 MUPSA.

4.3.1.3. Level 3 – partial level 3 modelling approach

Step 1 – Objective: objective of Partial Level 3 MUPSA is to calculate the proposed site risk metric (IRR), due to LOOP.

Step 2 – Scope, hypothesis and general assumptions: considering the proposed risk metric, the site total risk is evaluated as the sum of the risks related to single units and the risk due to events that can affect multiunit. The hypothesis and assumptions used for Partial Level 3 MUPSA are the following:

- Population distribution: uniform distribution is used for the entire domain;
- Dose factors FGR13: those included in WinMACCS code;
- Population Shielding: extracted from the sample case included in the WinMACCS code, for normal condition;
- Meteorological Information: one year is considered;
- Dispersion parameters: the same that were used in Surry NPP site [27];
- Release paths to the atmosphere: according to the Level 2 PSA, are considered two release paths: the units stack and the containment failure hole;
- Domain discretization (in km): 0.95, 1.05, 2.95, 3.05, 4.95, 5.05, 6.95, 7.05, 8.95, 9.05, 10.95, 11.05, 12.95, 13.05, 14.95, 15.05, 16.95, 17.05, 18.95, 19.05, 20.95.

Step 3 – Model development: proposed site and MUPSA risk metric, IRR, required on one hand, the probability of exposure and on the other hand, the effective doses. The doses are calculated with

WinMACCS, code version 3.6.0. It is important to mention that this version does not calculate multiunit consequences. For this reason, the initial inventory used for dose calculations is the sum of the inventory of the units and/or SFP. Although the latest version of WinMACCS has the possibility of calculating doses generated by multiunit releases, the new version does not generate the binary output file needed for the IRR calculation. The probability of exposure is calculated considering multiunit release category group frequencies and probabilities of incurred doses due to meteorological conditions. An ad–hoc Python program was developed.

Step 4 – Results analysis: the $IRR^{MAX}_{Total-Sit\ ,LOOP}$ due to LOOP initiating event has been calculated: 1.3×10^{-9}. Figure 75 shows the contributions to $IRR^{MAX}_{Total-Site,LOOP}$, if only one radioactive source has fuel being damaged or if more than one radioactive source is damaged, IRR^{MAX}_{MUPSA}; the MUPSA contribution to risk metric is only 2%.

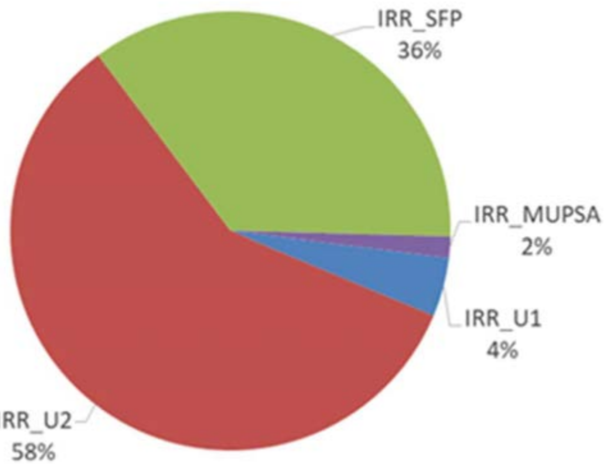


FIG. 75. Case study: contributions to $IRR^{MAX}_{Total-Site}$ due LOOP: only one radioactive source has fuel damaged (IRR^{MAX}_{MU-U1}, IRR^{MAX}_{MU-U} , IRR^{MAX}_{SFP}) or more than one radioactive source have fuel damaged (IRR^{MAX}_{MUPSA}).

The difference of IRR^{MAX}_{MU-U1} and IRR^{MAX}_{MU-U2} is due modelling approach and due to the hypothesis posed for the case study. From Level 1 MUPSA, the shared systems are assigned in first place to unit 1. Then, the contribution of unit 1 to multiunit FDF is lower than unit 2. Regarding the safety goal accomplishment, it can be partially evaluated because only an initiating event has been analysed. Thus. LOOP multiunit initiating event for case study represents less than 1% of the safety goal.

Figure 76 shows the contribution of release category A, B, C and D to IRR^{MAX}_{MU-U1}, for unit 1, (for unit 2 the percentage contribution of each release category is the same). Considering that the IRR is the probability of the intersection of the exposure and fatality (consequence) events, the release category D implies the highest doses, but the frequency is the lowest. The release category D contributes with 13% to IRR^{MAX}_{MU-U1}. In comparison, RC_B implies lower doses, but this release category has a higher frequency; its contribution to IRR^{MAX}_{MU-U1} is higher than release category D.

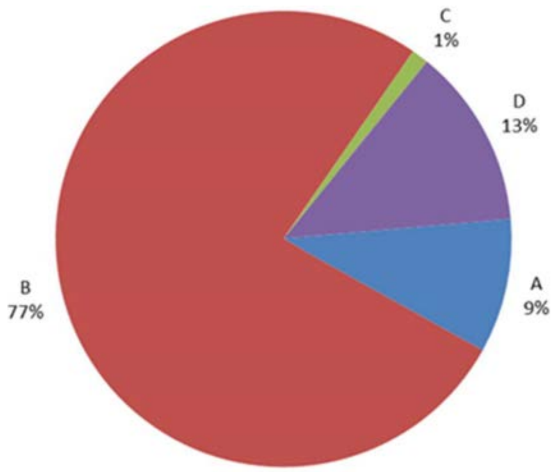


FIG. 76. Contribution of release category A, B, C and D to IRR_{MU-U1}^{MAX} (unit 1 or unit 2).

Figure 77 shows the contributions of each multiunit release category group to IRR_{MUPSA}^{MAX}, considering fuel damage in at least two radioactive sources (one or both units and/or SFP). The contributions higher or equal to 1% are shown. The highest contribution is the IRR due to multiunit release category AX. The occurrence frequency of multiunit release category AX is the highest. It means that multiunit release category that can represent lower doses to member of public can imply higher risks because they are more frequent. Same as was analyzed previously for release category, the multiunit release category group BDX implies the highest doses, but the frequency is the lowest. Then, the contribution to IRR_{MUPSA}^{MAX} is only 1%. The IRR is an integral risk metric that considers not only the consequences but also frequencies regarding multiunit release category.

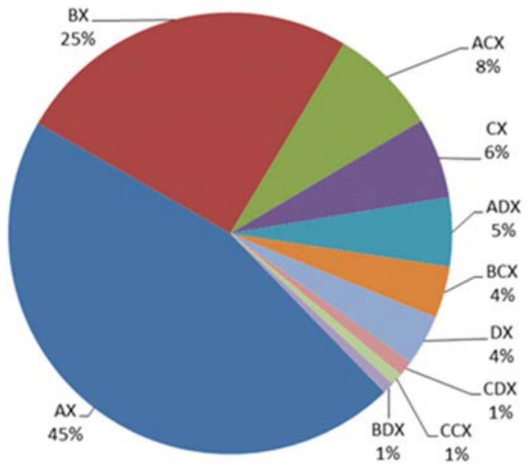


FIG. 77. Contribution of each multiunit release category group to IRR_{MUPSA}^{MAX} (multiunit release category groups that has a contribution equal or higher than 1% are graphited).

4.3.2. Canada/COG

4.3.2.1. Containment system fault trees

The process for developing the containment system fault tree is similar to the process of developing systems fault trees in Level 1 PSA. However, during the development of the containment systems fault trees in regards to severe accidents, it may be observed that the role of the system may be different based on the physical characteristics of the severe accident sequences. For example, some characteristics such as mission time, human interaction and success criteria may impact the modelling of the containment system fault trees. Furthermore, systems such as end shield and shield tank cooling system is not credited in preventing accident progression to severe core damage. However, if available, it may be capable of removing decay heat to prevent severe accidents from progressing beyond the calandria vessel. On another note, consequential failure modes that reflect severe accident progression do not need to be included in the system fault tree due to its dependence on the sequence.

4.3.2.2. Containment event trees

The CET are logic models that address the uncertainties to predict potential impact of the accident progression and associated physical phenomena on the containment response. Additionally, they are used to address the consequential challenges to containment and containment systems as the severe accident progresses. The CET top events are built from questions that interrogate the state of the plant at a given stage of the accident. (e.g. ‘Is containment integrity maintained’). The CETs are generally developed with the lower branch representing ‘Success’ and the upper branch representing ‘Failure’. The CET branch points represents major events in accident progression and the potential for fission product release to the environment. The same question may appear more than once in the event tree since the event tree also considers the evolution of the progression with time. Therefore, the focus of the CET is to estimate the probabilities of various ways that containment failure may occur leading to a release to the environment.

Relative to LWR, the modelling of the accident progression for CANDU reactors can be quite complex prior to corium relocating into containment. Once fuel channel integrity has been compromised, a number of potential core retention boundaries exist. Core debris can potentially be retained in the calandria vessel or shield tank, given that adequate heat sinks are present to reject the decay heat. These heat sinks can be established via the continued operation of existing systems or via operator interventions according to emergency operating procedures or SAMG. Once the corium has relocated into containment, the progress of a severe accident in a CANDU reactor is similar to the accident progression in a LWR. In this instance, the logic of a CANDU CET is developed in two (2) stages. In the first stage, branch point questions track the state of core degradation and identify those accident sequences where corium will relocate into containment. In the second stage, branch point questions relate to the challenges to containment integrity to assess the potential for releases. To support the CET assumptions, severe accident progression and consequence analysis need to be conducted. This can be done with the support of software such as MAAP–CANDU. Insight can be taken from thermal hydraulic analysis, structural or other deterministic safety analyses.

4.3.2.3. *Branch point probabilities*

Branch point probabilities represent the relative likelihood of the alternative possible outcomes of a given physical interaction. Branch point probabilities can be quantified using four approaches:

— A combination of probability distribution functions obtained from variation in analysis parameters or mathematical relationships;
— Decomposition event tree or logic diagram;
— Engineering judgment;
— Expert elicitation.

The method appropriate for use will depend on the nature of the event and the kind of information available.

4.3.2.4. *Grouping of CET end states*

The CET end states are grouped (or binned) to collect accident sequences with similar release characteristics in terms of potential offsite radiological impacts. This approach facilitate comparison with the Level 2 Safety Goals as it allows to identify sequence of events for which release from the plant exceeds the release threshold. It also provides a suitable basis for the Level 3 analysis for dose economic consequences and related risk measures. The Canadian regulator does not require Canadian utilities to produce a Level 3 analysis. The consequences of a CET end state is expressed in terms of magnitude and timing of ^{131}I and ^{137}Cs release to assigned them to the appropriate release category. The release categories and the sequences assigned to them form the basis of the Level 1 / Level 2 integration.

4.3.2.5. *Human reliability analysis*

The HRA will follow the same approach as in the Level 1 PSA. However, in the Level 2 PSA, the analyst need to credit operator actions that may take place after severe core damage has occurred. For example, an action covered by an existing procedure that remains valid under severe accident conditions e.g. actions expected to be conducted under emergency operating procedures or SAMG need to be quantified and included in the Level 2 PSA. Furthermore, consideration for radiological habitability issues need to be considered. Factors such as high radiation level at the location where the action is performed, along with the time taken to accomplish this action, need to be considered in the evaluation of the operator action.

4.3.2.6. *Release categories*

Release categories are typically defined based on release magnitude and timing with no distinction made between event sequences involving single and multiple units or between pathways such as containment bypass. There is also no distinction if the source comes from a non–reactor source such as the IFB. Additional release categories could be created to highlight those characteristics.

4.3.3. China/INET

Due to the reason that HTR–PM doesn't apply CDF and LERF as the risk metric, it adopts an integrated modelling framework, starting from initiating events and ending at the release categories as well as the dose estimates. That is, we complete Level 1, Level 2 and part of Level 3 in one step, i.e. Level 2+. In brief, the overall modelling approach is event tree and fault tree linking. Shared SSCs are explicitly modelled in the event trees and fault trees. Both inter–reactor and intra–reactor CCFs are considered. Release categories are concluded at the end of event tree branches. Each release category is then analysed to determine its source term and dose estimates. Event tree and fault tree models are developed by using RiskSpectrum PSA software. Dose consequence assessment is done by the software ARCAT which is developed by INET for the purpose of multiple source releases. The technical issues related to modelling are basically the same as those of Level 1.

4.3.4. Hungary/NUBIKI

The Level 1 MUPSA model as well as the severe accident analyses performed by using the MAAP–VVER code for a single unit served as the main input to the Level 2 assessment. The LOOP related unit specific bridge trees and CETs with the corresponding fault trees were integrated into the Level 1 MUPSA model to enable the quantification of the twin unit risk for the LOOP event. Some preparatory steps were seen necessary in the unit specific Level 2 PSA models to enable the model integration, namely:

— The system of model identifiers has been revised and modified by giving a unit specific identifier to all unit specific model elements, and twin–unit specific or plant specific identifiers to other elements, as appropriate;
— Fault trees of shared systems have been standardized.

In the Level 2 MUPSA model first the unit 1 specific bridge tree was interconnected to the dual core damage end states (core damage occurs on both units) of the Level 1 multiunit combined event tree. Based on the quantification of different end states of the bridge tree, those end states (PDSs) that cannot lead to large or early release (even simultaneously with a similar release on the other unit) or have insignificant contribution to the aggregated PDSs frequencies were screened out from further detailed assessment. Besides, all the large or early release sequences of the CETs for the relevant screened–in PDSs scenarios at unit 1 were converted into fault trees. This was done by building a fault tree representation of each large or early release sequence and connecting these fault trees under an OR–gate. The complement of successful response (no LER occurs) at a unit is modelled by a fault tree conversion of large or early release sequences in the fault trees and the top gate was linked to an additional, final header of the unit specific bridge tree. In this manner, the original bridge tree was extended by a header that enables to assess the LERF originated from PDSs with significant contribution to the overall PDSs frequency. Subsequently, the unit 2 specific bridge tree, extended by an additional header representing the fault tree conversion of the CET of unit 2, was interconnected to the large or early release end states of the aforementioned event tree. Consequently, by means of these 3 interconnected event trees the frequency could be calculated that corresponds to large or early release on both units.

After developing the integrated Level 2 MUPSA model, the dependency among the CET headers were studied and modelled as seen necessary. The following types of dependencies were identified and quantified:

— Shared systems: systems and supporting systems that are common to both units (e.g. common reactor hall and its ventilation system, common severe accident diesel generators, common transportation vehicle for the unit specific mobile diesel generators for severe accident

measurements and interventions) were modelled by the same basic events;
- Sequential recovery actions: when the same personnel is responsible for the recovery of a certain SSC if failed at both units (e.g. emergency diesel generators), hence the actions can only be performed sequentially, and the time to recovery is relatively low, then the failure probability of recovery on both units were quantified by Markov chains;
- Actions of the Technical Support Centre (TSC): the circumstances of the different actions were studied including the underlying performance shaping factors in different multiunit scenarios and the dependence in the HEP of the TSC was quantified by expert judgement, considering (in comparison to single unit scenarios) aggravating effects (i.e. the workload on the TSC doubles in case of a multiunit accident, whilst only one additional person joins the group) too.

4.3.5. Ghana/GAEC

The following release categories and risk metrics relevant to Level 2 MUPSA are defined as follows:

- Acceptable release is a release of less than 0.1% of the core inventory of ^{134}Cs or ^{137}Cs;
- Unacceptable release is a release above 0.1% of the core inventory of ^{134}Cs or ^{137}Cs;
- Large release is a release of more than 10^{14} Bq of ^{137}Csfrom the site to the environment;
- LER is the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the implementation of off site emergency response and protective actions such that there is potential for early health effects (ASME/ANS PRA standard);
- SLERF is the frequency of a large early release from an accident involving one or more reactor units simultaneously per site calendar year;
- SRCF is the frequency per site and calendar year of each distinct release category for Level 2 MUPSA.

The following safety goal applicable for Level 2 MUPSA in Ghana is proposed: large offsite release safety goal as the aggregate of large LRFs of all event sequences that can lead to a total release from the site to the environment of more than of ^{137}Cs is less than per site year. Initiating events analysed for SUPSA need to be reviewed for the multiunit analyses. The review considers events that may affect only single units as well as those that have the likelihood of affecting multiunit on the site. Generally, initiating events can be categorized as follows:

- Initiating events that affect single units independently and separately such as a primary circuit pipe break;
- Initiating events that affect multiunit, such as seismic events and other external events, and
- Initiating events that occur in single units but may impact multiple units depending on the cause, severity or conditions of the plant at the time of the event; an example is a LOOP event.

It has been established that all multiunit sites are subject to LOOP events that may either affect single unit or multiunit depending on the event. There is greater possibility of a LOOP affecting multiple units on a site where there exist shared SSCs among the units. In this study, accident sequences analysed were assumed to be initiated by a LOOP event that affected multiunit and resulted in SBO. The modelling approach for conducting Level 2 MUPSA is shown in Fig. 78 with the steps as follows:

1) Develop Level 2 SUPSA and calculate containment failure probabilities
2) Map PDS event tree to source term categories using mapping fractions from Level 2 SUPSA results.
3) Estimate MUPSA Level 2 risk metrics depending on selected scope (large release frequencies and late large release frequencies).

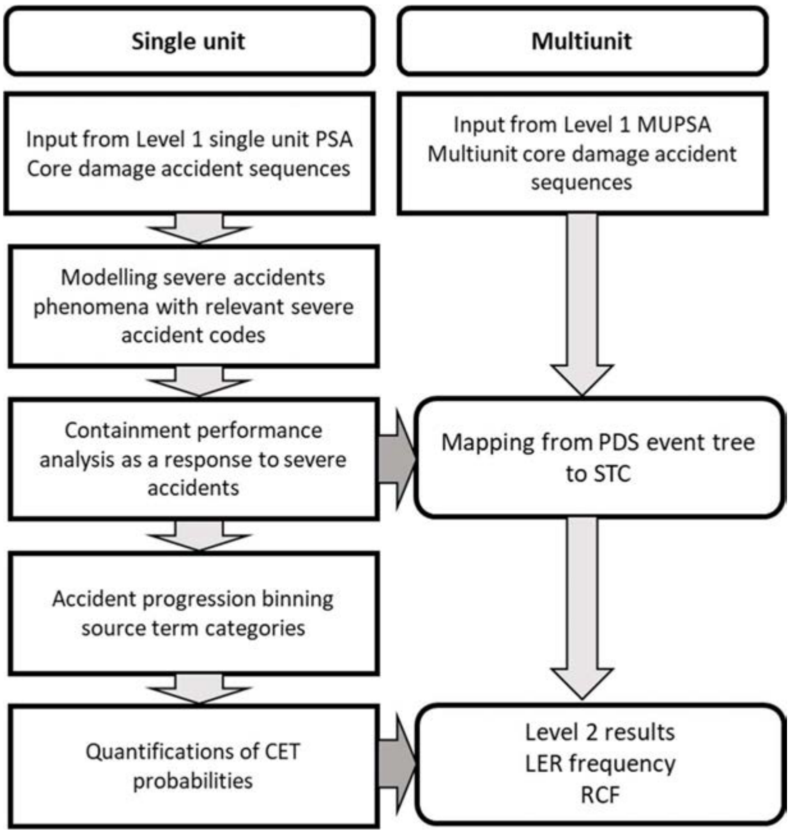


FIG. 78. Description of Level 2 MUPSA modelling approach.

The example to demonstrate the above modelling approach is selected from NUREG–1150 study [31] and applied to a conceptual two unit site having two PWRs of the same design. The initiating event is a SBO resulting from a LOOP. The most probable cut set as determined in the NUREG–1150 study is analysed and quantified. The severe accident phenomena analysis is based on MELCOR simulations. Table 51 gives the top event probabilities obtained from the quantification of fault trees used to quantify the PDSs event tree shown in Fig. 79.

TABLE 51. TOP EVENT PROBABILITY USED TO QUANTIFY PDS

Event Name	Probability	Description
IE–T1	0.0994	SBO caused by LOOP
DG–01–FS	0.0133	DG1 fails to start
DG–03–FS	0.0133	DG3 fails to start
AFW	0.0762	Failure of operator to open manual valve from AFW (auxiliary feedwater) pump suction to CST2
QS–SBO	0.0675	Stuck open safety relief valve in the secondary system
NRAC–60 minutes	0.44	Failure to restore offsite power within 1 hour
NOTQ	0.973	RCS PORV(s) successfully reclose during SBO

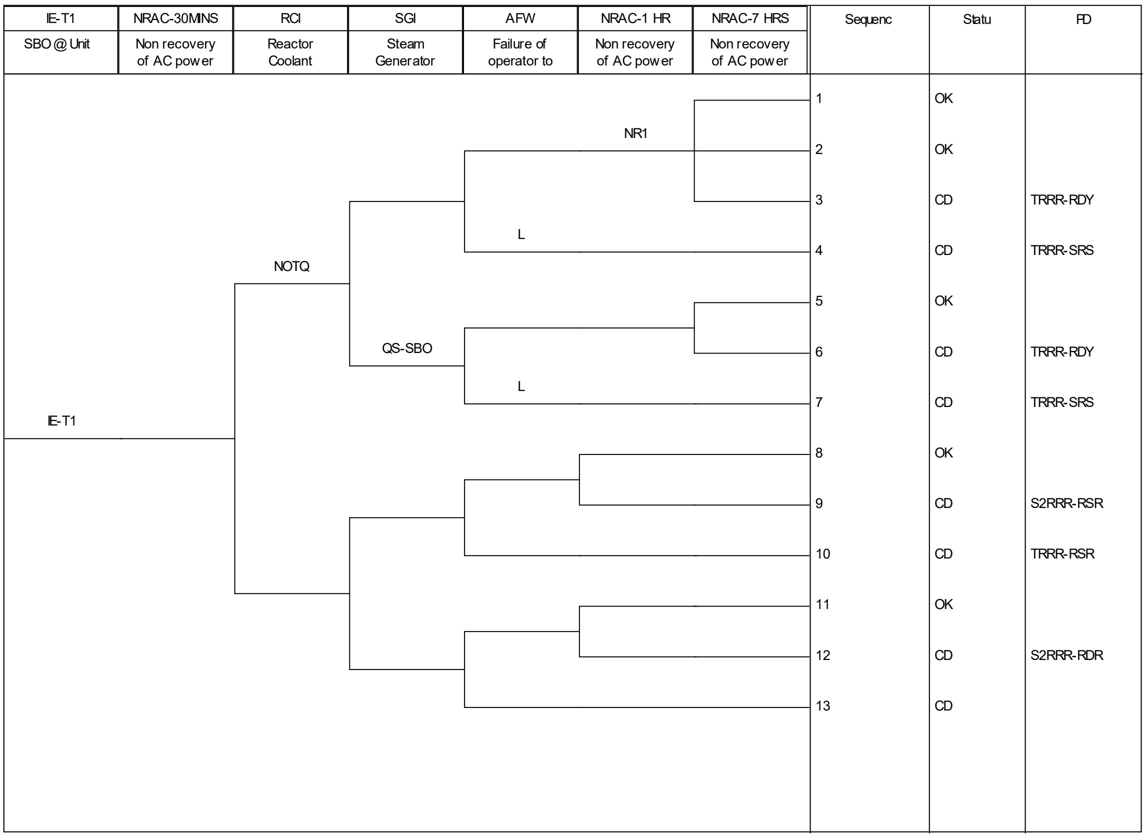


FIG. 79. PDS event Tree for SBO initiating event.

The PDSs corresponding to the most probable cut set is sequence #7 in the PDS event tree and means the following:

- T – RCS intact at the onset of core damage;
- R – Emergency core cooling is recoverable;
- R – Containment heat removal is recoverable;
- R – AC power is recoverable from offsite sources;
- R – RWST contents have not been injected into the containment but can be injected if AC power is recovered;
- S – Steam–turbine–driven auxiliary feedwater system failed at the start of the accident; the electric motor–driven auxiliary feedwater system is recoverable; and
- R – Cooling for the reactor coolant pump seals is recoverable.

The accident sequence followed for this example is T1S–QS–L = T1×NRAC–30 min × /Q × QS–SBO × L, which is the highest frequency sequence that leads to PDS TRRR–RSR. This sequence was determined to be the most probable of several sequences that involve SBO and early failure of the auxiliary feedwater system. T1S–QS–L contributes about 75% to the mean frequency of the PDS TRRR–RSR. Sequence T1S–QS–L is comprised of 216 cut sets.

The CET or accident progression event tree is shown in part in Fig. 80 The top events in the CET given the occurrence of the PDS group TRRR–RSR and used to estimate the containment failure probability are: containment bypass, RCS failure, core melt progression stopped, alpha mode failure, amount of corium ejected in CCI, early containment failure, late recirculation sprays, debris bed cooled, late containment failure and basemat melt through. The CET results in 97 end states considering the top events and time dependencies between major severe accident phenomena. Most of the CET end states terminate or lead to probabilities lower that the set point.

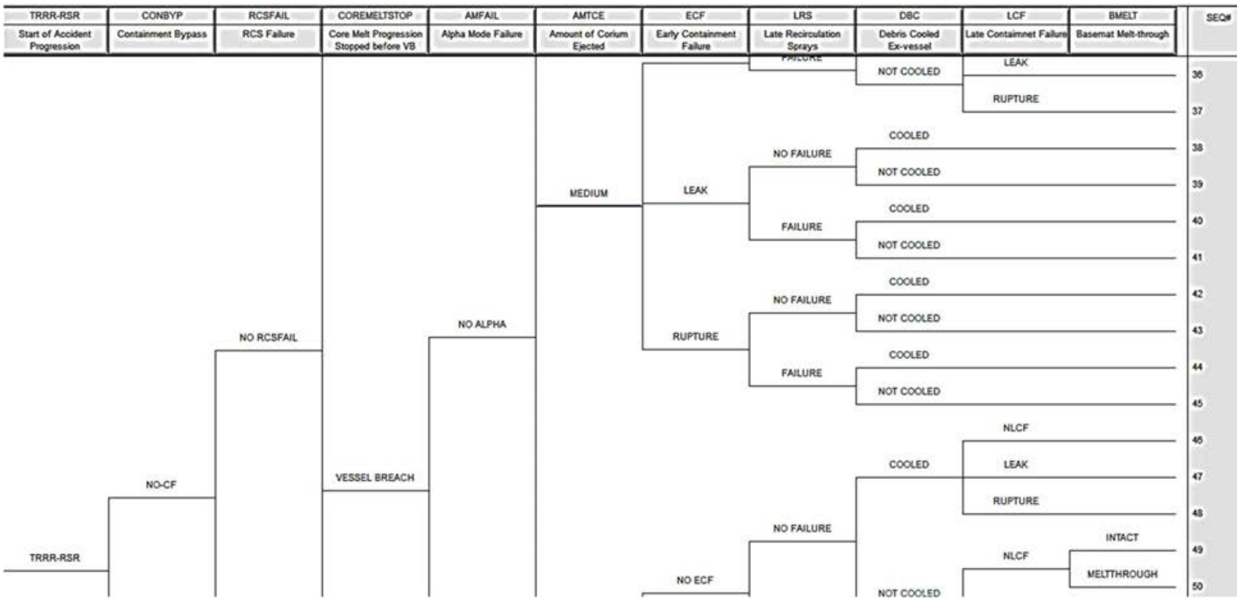


FIG. 80. Partial containment event tree.

4.3.5.1. *Accident progression bins*

The CET end states or sequences are grouped together into accident progression bins (APB). The APB are determined by the status of containment integrity and the timing of containment failure.

The containment failure modes evaluated in this example are: no containment failure (NCF), early containment failure (ECF), basemat melt through/ late containment leak (BMT) and containment bypass (CB). These containment failure modes are used to determine the mapping fractions for Level 2 SUPSA from PDS event tree to CET to source term category (STC) as well as to estimate LER and LLR probabilities.

4.3.5.2. *Results and insights*

Level 2 SUPSA results: these results are based on Level 1 PSA results in the Seabrook study [20] and the APET quantification results in NUREG–1150 study [31]. For LOOP initiating event, the core damage frequency estimate is 4×10^{-4}/reactor–year. Table 52 shows details of the various APB groups and corresponding containment failure probabilities. From Table 52 it follows that the single unit LER frequency = ECF probability+ CB probability =1.1%, ~ 4.4×10^{-6}/reactor year.

TABLE 52. CONTAINMENT FAILURE PROBABILITIES FOR LEVEL 2 SUPSA

Containment Failure Modes	CDF Fractions	Containment Failure Probability
No containment failure	91%	3.6×10^{-4}
Basemat melt–through	7.9%	3.2×10^{-5}
Early containment failure	0.8%	3.2×10^{-6}
Containment bypass	0.3%	1.2×10^{-6}

Multiunit Level 2 PSA results: Table 53 shows the containment failure modes of multiunit LER frequency for Level 2 MUPSA.

TABLE 53. CONTAINMENT FAILURE MODES FOR LEVEL 2 MUPSA

		Unit 2			
		NCF	**ECF**	**BMT**	**CB**
	NCF	NCF&NCF	NCF&ECF	NCF&BMT	NCF&CB
Unit 1	**ECF**	ECF&NCF	ECF&ECF	ECF&BMT	ECF&CB
	BMT	BMT&NCF	BMT&ECF	BMT&BMT	BMT&CB
	CB	CB&NCF	CB&ECF	CB&BMT	CB&CB

Multiunit LER frequency = (NCF&ECF) + (ECF&ECF) + (BMT&ECF) + (CB&ECF) + (ECF&NCF) + (CB&NCF) + (ECF&BMT) + (CB&BMT) + (NCF&CB) + (ECF&CB) + (BMT&CB) + (CB&CB) = (3.6E–04*3.2E–06) + (3.2E–06*3.2E–06) + (3.2E–05*3.2E–06) + (1.2E–06*3.2E–06) + (3.2E–06*3.6E–04) + (1.2E–06*3.6E–04) + (3.2E–06*3.2E–05) + (1.2E–06*3.2E–05) + (3.6E–04*1.2E–06) + (3.2E–06*1.2E–06) + (3.2E–05*1.2E–06) + (1.2E–06*1.2E–06) = 1.2E–09 + 1.0E–11 + 1.0E–10 + 3.8E–12 + 1.2E–09 + 3.8E–12 + 3.8E–11 + 3.8E–12 + 3.8E–12 + 3.8E–12 + 3.8E–11 + 1.4E–12 = 2.6E–09/site–year.

Site LER frequency = LER freq. unit1 + LER freq. unit 2 – LER freq. for unit1&unit2 = 4.4E–06 + 4.4E–06 – 2.6E–09 ~8.8E–06/site–year.

The insights are summarized as follows:

— Containment failure probability for both units is low given the occurrence of LOOP that leads to fast SBO.
— Estimated site LER frequency is less than the proposed large offsite release safety goal.
— Contribution of multiunit LER frequency to site LER frequency is negligible.
— Recovery of AC power within 24 hr following core damage would ensure there is no containment failure for both units.
— In reality, accident sequences at different units are likely to differ, this implies that the approach presented in this study for estimating site LER simplifies the problem however, valuable insights of accident progression are gained.

4.3.5.3. Human reliability analysis for Level 2 MUPSA

Human reliability analysis in MUPSA may pose additional challenges for analysts. These challenges may arise from limited human resources, complexity in managing multiple scenarios from a site, shared system prioritization, and prioritizing the deployment of portable equipment. There may be need to modify existing human events modelled in a SUPSA to account for multiunit challenges as well as consider new HFEs to support a MUPSA. Considerations for determining what new HFEs need to be added in MUPSA are as follows:

- Identify additional complexities for human errors due to shared MCR and additional opportunities for recovery managed by other unit's operator resources in MCR;
- Identify additional opportunities for human errors in determining where to arrange for shared systems initially or when to alternate in between the units (for example the shared EDG or makeup water systems);
- Identify if there is an opportunity for human error for field workers performing a task on the wrong unit;
- In case a shared TSC is dealing with multiple accidents at different accident phases, identify if there is potentially an increased likelihood for human error to be induced by staff changing the focus in between the units;
- Decide how decisions are to be directed from SAMGs related to mitigation strategies, and then be modelled.

The approach adopted in this study for performing the HRA for MUPSA is as follows:

- For identified initiators to decide on the existing relevant operator actions;
- Identify new actions (if any);
- Characterize multiunit conditions and HFE requantification using dependency action.

HEP quantification using the dependency approach: this approach is derived from the THERP [27] and SPAR–H [12] methods for modelling operator action dependencies in SUPSA. The THERP method is primarily based on using a task analysis (step–by–step decomposition of an activity into simple items, such as read meter, turn switch) to provide a data base and rules for application to HRA. The SPAR–H approach is focused on using eight performance shaping factors to account for human error initiators and quantify the associated HEP. The performance shaping factors are, available time, stressors, task complexity, experience/training, procedures, ergonomics/HMI, fitness for duty and work processes.

In this study it is assumed that there are two identical HFEs: HFE1 at unit 1 and HFE2 at unit 2. The degree of dependency in this case is dependent on the shared actors (operators or staff) involved in the HFE. The estimation of HEP is based on the contribution of each actor on diagnosis, execution and recovery actions.

Table 54 shows various dependency categories with corresponding conditional HEPs for SUPSA are presented based on HEPs in NUREG CR–1278 resulting from the use of THERP approach. For MUPSA application, HEPs are adjusted using factors suggested in SITRON together with scenario descriptions presented in the 4th column of Table 54. The adjusted conditional HEPs together with other approaches such as penalty factors can then be added in integrated fault trees and event trees for MUPSA.

TABLE 54. DEPENDENCY AND HEP ADJUSTMENT FROM SUPSA TO MUPSA

Dependency Level	SUPSA HEP, p_1	Adjusted HEP for MUPSA	Scenario Description
Zero	N	$N * N$	No common actors in accident management
Low	$\frac{1+19N}{20} \sim 0.05$	$2p_1 * p_1 \approx 0.005$	Shared recovery. A TSC can follow up on decisions made at units. Different scenarios assumed at both units, but a level of dependency exist.
Medium	$\frac{1+6N}{7} \sim 0.15$	$\sqrt{p_1} * p_1 \approx 0.06$	Shared diagnosis. A TSC supports MCR with diagnosis. Different scenarios assumed at both units, but a level of dependency exist.
High	$\frac{1+N}{2} \sim 0.5$	$0.5 * p_1 \approx 0.25$	Same personnel perform recovery actions for both units. A high dependency is assumed as the failure/success at one unit implies similar outcome for the other unit.
Full	1.0	1.0	Common action for unit 1 & unit 2.

4.3.6. India/BARC

Level 2 PSA of a severe accidents is based on a combination of probabilistic and deterministic approaches, to determine the release of radionuclides from containment, including physical processes describing loss of reactor core structural integrity. The probabilistic approach emphases are on the performance reliability of NPP containment safety systems. This information is combined with Level 1 PSA, to obtain release frequencies of different release categories from the NPP containment safety systems. The deterministic approach focuses on the analysis of physical processes involved in an accident (timing and magnitude of radioactive release) and expected response of the containment. The probabilistic approach requires development and quantification of containment logic models to include:

1) Grouping and categorization of accident sequences into PDSs;
2) Development of CETs, modelling of containment safety functions failure and containment failure due to various phenomena related to H2, heat up, corium etc., which defines a spectrum of containment damage or release states;
3) Development of CET top event definitions and the quantification of failure probabilities by fault tree analysis; and
4) Collapsing the CET release modes into a few release/consequence categories. The release categories are containment failure bins, for which fission product releases are calculated.

Level 2 interface with Level 1 PSA includes redefining the Level 1 end states, modifying the event trees and binning of PDSs and estimating the frequency. In order for the quantification of LERF, the release categories were formulated based on the amount of release of fission products (e.g. large and small) and time of release (e.g. early and late) as shown in Table 55.

TABLE 55. RADIOACTIVITY RELEASE CATEGORISATION

Inventory of release		Time of release	
Large	More than 10^{14} Bq of ^{137}Cs or 10^{15} Bq of ^{131}I	Early	within 24 hr
Small	Less than 10^{14} Bq of ^{137}Cs or 10^{15} Bq of ^{131}I	Late	beyond 24 hr

For SFF, PSA estimates the FDF. Initiating events that could occur were initially identified after studying design basis reports, discussions with the designers and literature survey. The FDF has been estimated using small event tree and large fault tree approach. Consequences are defined in terms of damaged states/activity release i.e., fuel damage, fuel/pool water temperature high (100°C) and ground release of activity.

The methodology adopted for Level 2 MUPSA is analogous to SUPSA. However, special attention is given when modelling the inter–unit correlations, CCF and HRA in the multiunit regime and in risk aggregation. With respect to seismic correlations in Level 2 PSA, the same methodology adopted in the Level 1 PSA is used.

As the advanced reactor under consideration would have many passive safety features, not much human operator intervention is anticipated. However, the SFP facility in the event of loss of pool water would require the human operator to valve in the emergency water system to keep the fuel bundles submerged to prevent the fuel damage and the subsequent release of the activity. The following human actions are anticipated in the case of SFF:

— **Category 1**: Initiating events affecting only individual units/ source resulting in site release: Loss of SFF water would require human operator intervention to mitigate this event to prevent the fuel damage and activity release.
— **Category 2**: Initiating events which can affect both reactor units: No immediate human operator action is envisaged.
— **Category 3**: Initiating events which can affect one reactor unit and SFP: No immediate human operator action is envisaged.
— **Category 4**: Initiating events which can affect both reactor units and SFF simultaneously: In the case of seismic event, if the SFF tank gets damaged, then replenishment of the pool water would require human operator intervention to prevent the fuel damage and activity release.

As part of the Level 2 PSA work has been carried out both for internal as well as external event (seismic hazard) for both the reactor units. The summary of the results is shown in Table 56.

TABLE 56. RESULTS FOR SIRF

S. No.	Consequence	Description	% Contribution
1.	LER1	Large Early Release from Reactor 1	**46% of SiRF**
2.	LER2	Large Early Release from Reactor 2	**46% of SiRF**
3.	LER12	Large Early Release from Reactor 1 & 2	**8% of SiRF**
4.	LER12–SERSF	Large Early Release from Reactor 1 & 2 and Small Early Release from Spent Fuel Facility	**0.00021% of SiRF**

The specific conclusions from the Level 2 PSA point of view are as follows:

— Site LERF
  - o Main contribution from LLOCA related events and earthquake PGA level above 0.5g;
  - o Common initiating event – Turbine building structural failure leading to main steam line break outside the reactor building in both reactors leads to large early release.

— SiRF
  - Release from SFP leads to small release only, it is considered as part of SiRF;
  - SiRF consists of Site LERF and small release from SPF.

4.3.7. Pakistan/PAEC

The methodology for MUPSA Level 2 adopted in the study is as follows:

— Interfacing of MUPSA Level 1 and MUPSA Level 2;
— Accident progression analyses;
— Containment performance analyses;
— Development of CET;
— Source term analyses;
— Quantification of frequencies for release categories.

Details about each step is provided as follows:

4.3.7.1. Interfacing of MUPSA Level 1 and MUPSA Level 2

Bridge tree is developed to map multiunit LOOP scenario of MUPSA Level–1 study. The bridge tree is used to map the core damage sequences into PDSs. The top functional events considered in the development of multiunit LOOP bridge tree are (see Fig. 81 that shows developed bridge tree and resulting six PDSs):

— Post core damage depressurization is the operator action to depressurize the primary system pressure. In the study, credit of operator action is not considered conservatively.
— Containment isolation system is modelled to account for containment isolation failure due to major penetrations. The CCF of isolation valves are considered.
— Containment heat removal: containment spray system is considered in injection phase and re–circulation phase.

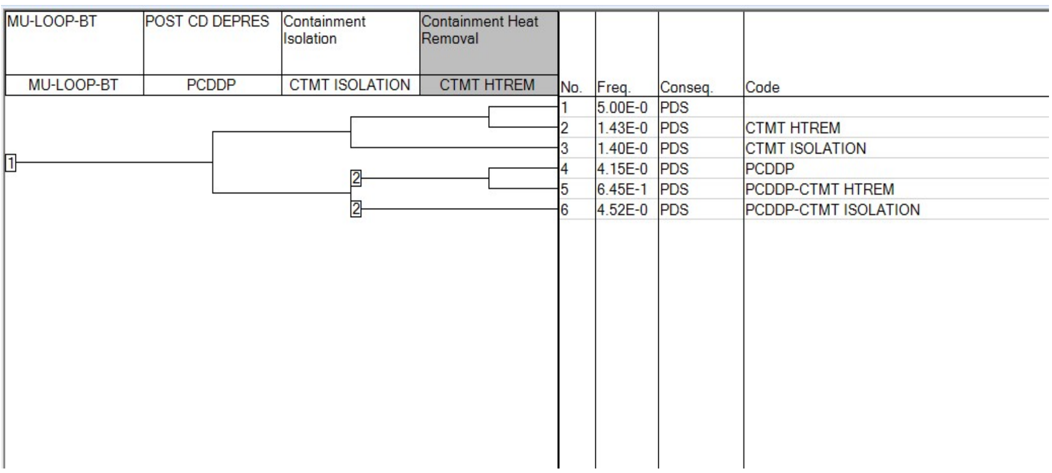


FIG. 81. Multiunit LOOP bridge tree.

4.3.7.2. Accident progression analysis

MELCOR (1.8.5) is a fully integrated code that models the progression of accidents in light water reactor NPPs. In this analysis, the auxiliary feedwater system is assumed unavailable resulting in total loss of feedwater flow to steam generators. Steam generator relief valves are assumed unavailable whereas steam generator safety valves are available to remove core decay heat from primary loop. This causes drop in the primary side pressure and temperature.

As the steam generator begins to deplete due to the opening of steam valves, the steam generators heat removal capacity also decreases, increasing the temperature as well as pressure in RCS until the opening of the pressurizer safety valves. Loss of coolant in RCS due to discharge from the opening valve causes the core to uncover gradually, consequently leading to a series of core degradations, including core heating, melting, slumping down to lower head of RPV, until lower head failure.

All core hot debris ejects from the vessel into the reactor cavity under the reactor vessel. Molten core concrete interactions generated non–condensable gases and decay heat of radionuclides in containment contributed to rise of the containment pressure.

4.3.7.3. Initial conditions and assumptions

Initially, the plant is postulated to operate at full power. The sequence is initiated with a complete loss of all AC powers.

The assumptions that are considered in the analysis are:

- At 0.0 sec, the reactor trip, reactor coolant pumps trip, turbine trip and loss of main feed water as well as auxiliary feed water;
- Diesel driven auxiliary feed water pumps are unavailable;
- EDGs and AAC are assumed to be unavailable;
- Pressurizer relief valves are unavailable due to station blackout and its safety valves (SVs) are available;
- Relief valves of secondary system are unavailable due to station blackout and its safety valves are available;
- Safety injection system and safety injection system in recirculation mode are unavailable;
- Accumulators are available;
- Containment spray system and containment spray system in recirculation mode are unavailable;
- Cavity flooding system is unavailable;
- Dedicated motor–operated throttle bleed valve to depressurize the primary system is unavailable.

Main parameters for full power operation and initial conditions in the analysis are shown in Table 57.

TABLE 57. PARAMETERS DURING FULL POWER OPERATION AND INITIAL CONDITIONS

Parameter	Unit	Value
Reactor thermal power	MWt	998.6
Primary coolant temperature	^{o}C	302.4
Pressurizer pressure	MPa	15.3
Steam generator pressure	MPa	5.6
Primary coolant flow rate	kg/s	3,493
Main feedwater flow rate	kg/s	269
Main steam flow rate	kg/s	269
Containment free volume	m^3	49,000

4.3.7.4. Nodalization

The basic nodalization used for the primary and secondary systems has been shown in Fig. 82 The MELCOR model for these calculations contains 54 control volumes (6 for reactor vessel and internals, 12 for primary loops, 16 for secondary system, 6 for containment, and one for the environment). These include detailed discretization of all important components of the primary and secondary side. Detailed reactor core nodalization is shown in Fig. 83. It consists of 14 axial levels and four radial rings. The lower plenum has been divided into three axial levels. Barrel base plate, flow diffuser plate and the lower core support plates have been specified up to ninth axial level. Axial levels from 10 to 13 make ups the active core region, whereas the 14th axial level is for the upper non heated part of the core.

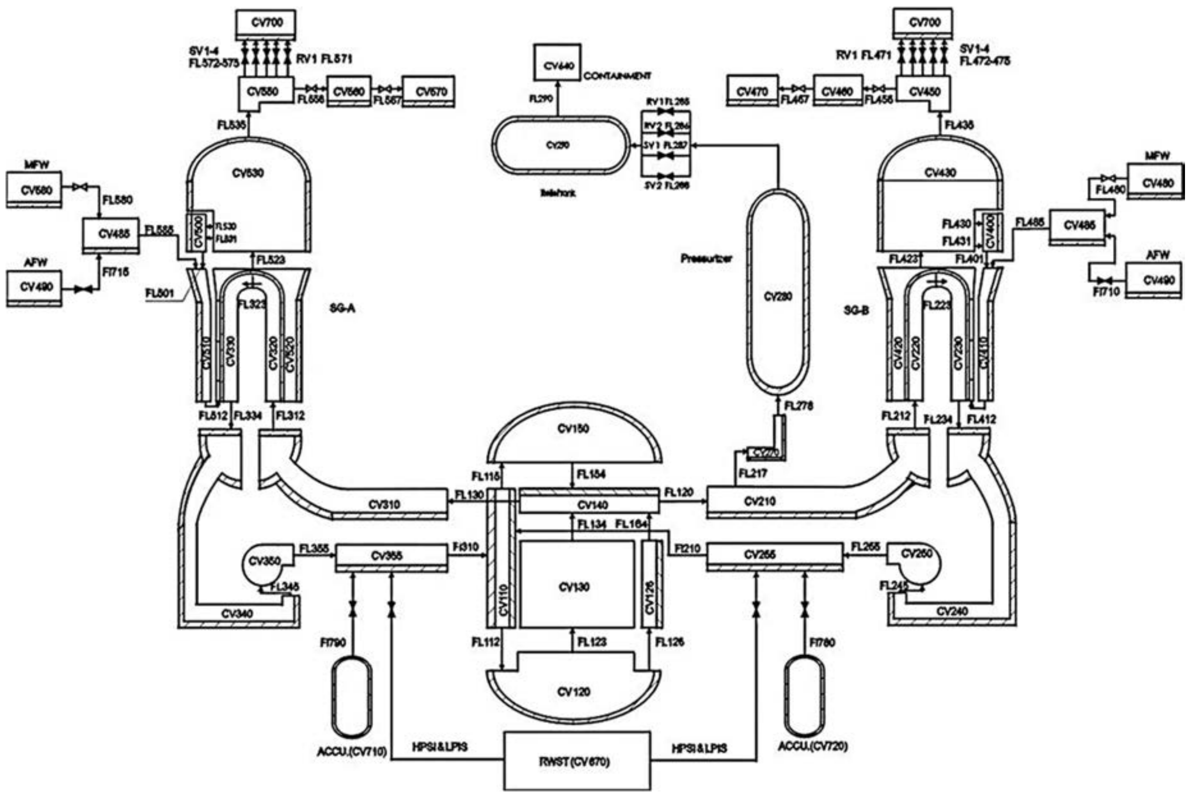


FIG. 82. Plant nodalization.

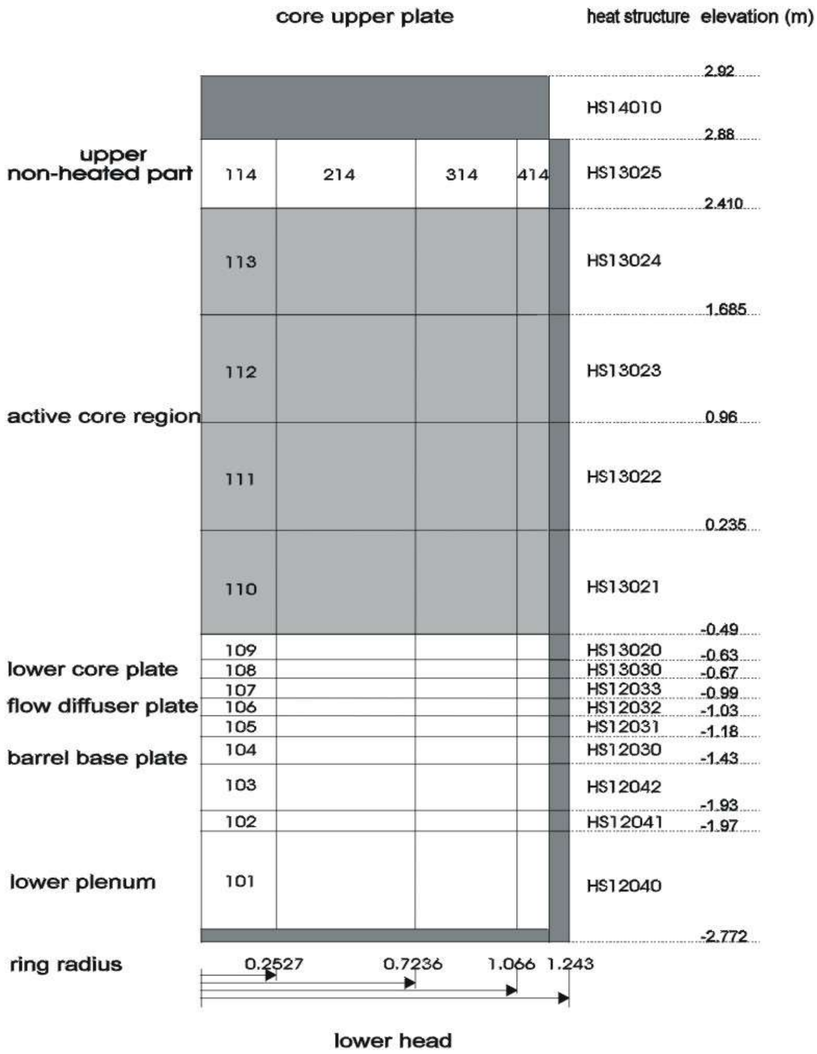


FIG. 83. Core axial nodalization.

Figure 84 shows the nodalization of containment. There are 6 control volumes used to simulate the sub–compartments of containment. The environment is simply modelled as a time dependent volume. All control volumes are specified to use non–equilibrium thermodynamics and vertical volumes. The steady state temperature gradient self–initialization option has been used in all heat structure. All heat structures in the containment are modelled at their corresponding control volume.

4.3.7.5. Thermal hydraulic response

The SBO scenario leads to events in the primary and secondary steam system of RCS and containment, transportation, deposition of aerosols in primary system, core heat up, fuel melting, slumping until failure of RPV lower head, and pressurization of containment.

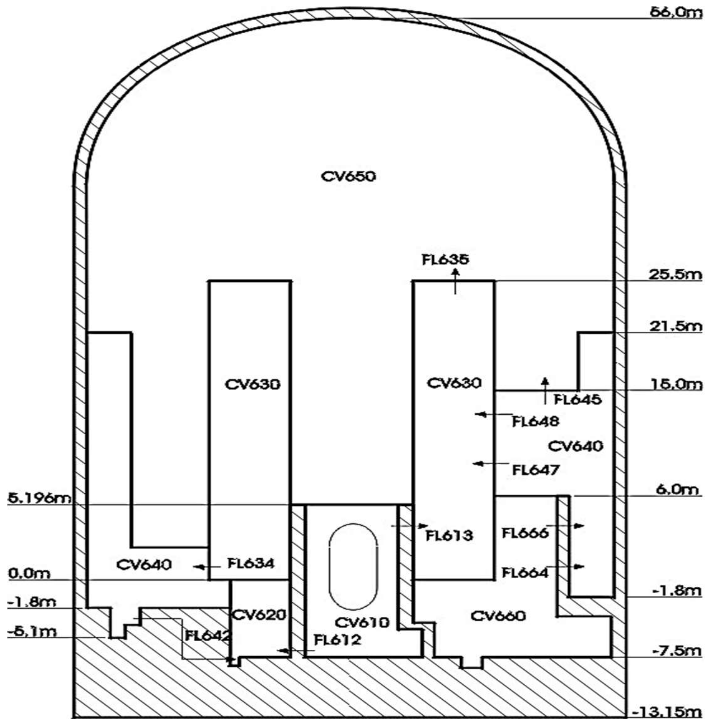


FIG. 84. Containment nodalization.

4.3.7.6. Containment performance analyses

Containment performance analysis has been performed on seismic fragility and risk assessment of structures housing radiological sources on multiunit NPP sites in Pakistan. The results show that leakage starts from the containment to the environment at ~0.68 MPa after 5.7 days of initiation of Multiunit LOOP. It clearly demonstrates that containment is a robust structure and could hold radioactivity for a long duration for the accident sequence analysed.

4.3.7.7. Development of containment event tree

The PDSs are mapped into release categories. Each PDS is propagated through CET with the CET for multiunit LOOP shown in Fig. 85.

The following top functional events are considered:

— No isolation failure: it evaluates containment isolation system during the course of accident. The accident sequence analyzed considers containment isolation system successfully isolates containment and thus preclude containment bypass scenario;
— No hydrogen burn: hydrogen burning is not credited in analyzed accident sequence as passive autocatalytic recombiner could successfully burn the generated hydrogen;
— In–vessel recovery: operator action to depressurize primary system is not considered;
— Containment integrity: is analyzed.

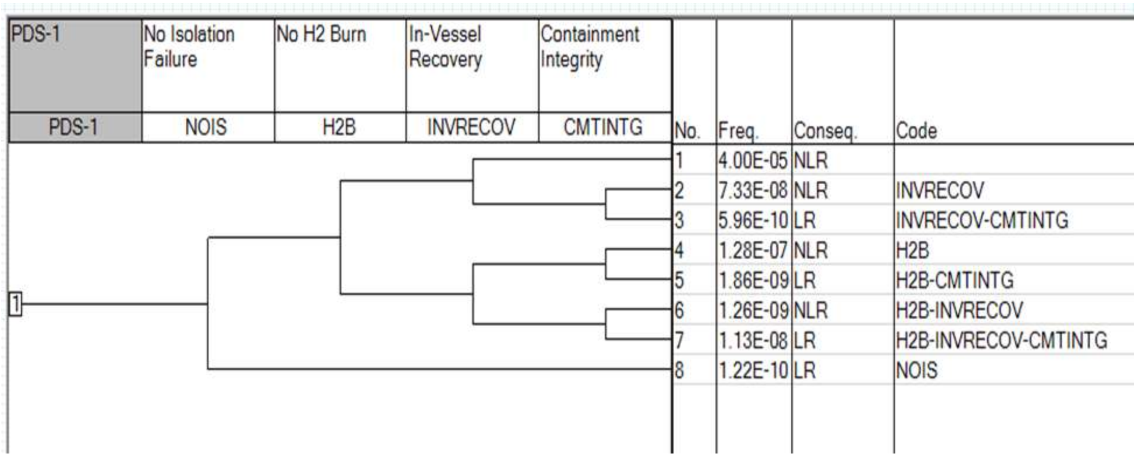


FIG. 85. Containment event tree.

4.3.7.8. Source term analyses

The fission product starts to release from the fuel following the failure of the fuel cladding after the uncover of the top of the fuel rods. The in–vessel fission product release phase continued through vessel failure. Initially, the fission product releases from the fuel circulating in the primary system and released to the containment through the pressurizer safety relief valves. The pressurizer relief tank rupture disk opened before the start of the fission product releases. Following vessel failure, the resultant blowdown of the vessel immediately discharged most of the release to the containment and the fission product releases continued from the ex–vessel fuel in the reactor cavity. Total mass of fission products produced by the reactor core and its internals is shown in Fig. 86.

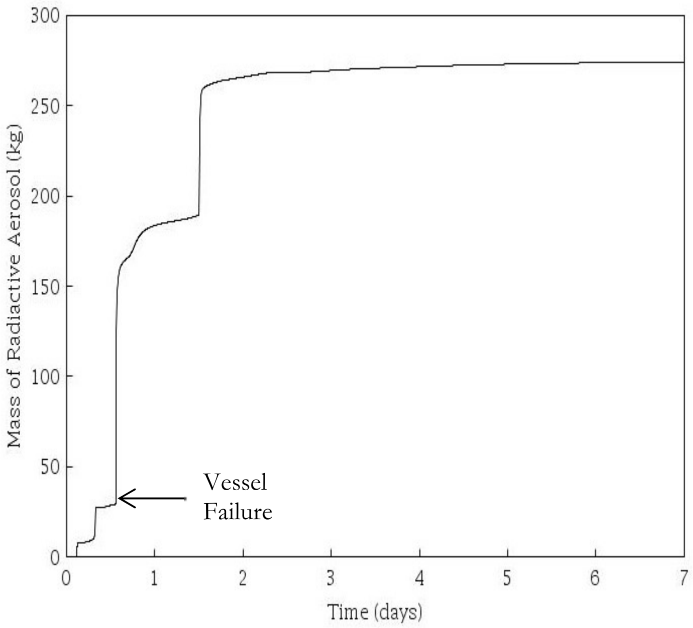


FIG. 86. Mass of radioactive aerosol.

4.3.7.9. Quantification of frequencies for release categories

The risk metrics considered in the study are single unit LRF, multiunit LRF and SLRF for MUPSA Level 2. Therefore, large release frequency for single unit, multiunit and finally at site level are calculated. Result of large release frequency due to core damage in all four reactor cores concurrently due to multiunit LOOP is considered for multiunit LRF. Both, LRF and multiunit LRF are used in the computation of single unit LRF and SLRF.

4.3.8. Republic of Korea/KHNP

Multiunit Level 2 PSA aims to quantify the integrity of containment and evaluate the radioactive source terms released into the environment for multiunit core damage sequences identified from Level 1 MUPSA. The results obtained from Level 2 MUPSA are as follows:

— Frequency of multiunit severe accident
— Combinations of containment failure modes for multiunit core damage sequences (e.g. unit 1–No containment failure * unit 2–LCF)
— Combinations of radioactive source term for multiunit core damage sequences (e.g. unit 1–STC1 * unit 2–STC5)

Figure 87 shows the flowchart of Level 2 MUPSA. For developing Level 2 MUPSA model, core damage accident sequences resulting from Level 1 MUPSA model are extended to Level 2 using Level 2 SUPSA information. Details related to model development are covered in subsequent sections.

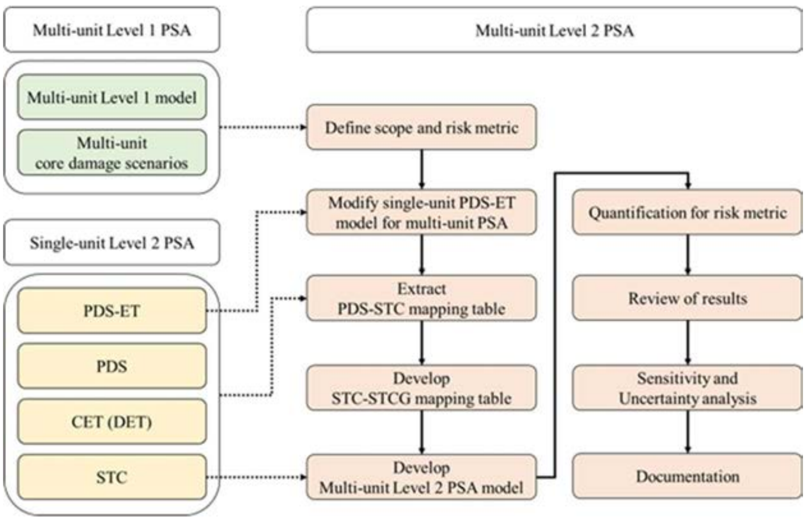


FIG. 87. Multiunit Level 2 PSA flowchart.

4.3.8.1. Definition of scope and risk metric for Level 2 MUPSA

As Level 2 MUPSA model is developed by expansion of Level 1 PSA model, the scope of Level 2 MUPSA is limited to that of Level 1 MUPSA. If some multiunit initiating event considered in Level 1 MUPSA are negligible in terms of risk metrics of Level 1 PSA (e.g. multiunit CDF), it can be excluded in the scope of Level 2 MUPSA. After defining the scope of Level 2 MUPSA, risk metric has to be defined. This is an essential step to determine the model structure, modelling details, etc. KHNP defined site LERF and multiunit LERF as a risk metric and developed the approach for Level 2 MUPSA for them.

4.3.8.2. Extraction of PDS–STC mapping table for each unit

Level 2 PSA models of KHNP consist of PDS, CET, and STC based on IAEA SSG–4 framework. PDS grouped by PDS event tree is quantified through CET and DET, and based on this, the branch probability for each STC is determined. Each core damage sequence identified by Level 1 MUPSA has PDS information. In a Level 2 SUPSA, a CET is developed to analyze accident progression including consideration of severe accident phenomena, where each PDS is used as the initial condition of the CET analysis. Each PDS is mapped to all CET sequences, and each CET sequence is mapped to a specific STC. Therefore, for each PDS, the fraction of each STC can be calculated from a Level 2 SUPSA model. Figure 88 represents an example of extending Level 1 scenarios (given the occurrence of a multiunit LOOP initiating event) to Level 2 scenarios using the PDS–STC fraction table obtained from a base Level 2 SUPSA model with 39 PDSs and 21 STCs. If each unit has a separate Level 2 SUPSA model, a separate PDS–STC fraction table can be obtained for each unit.

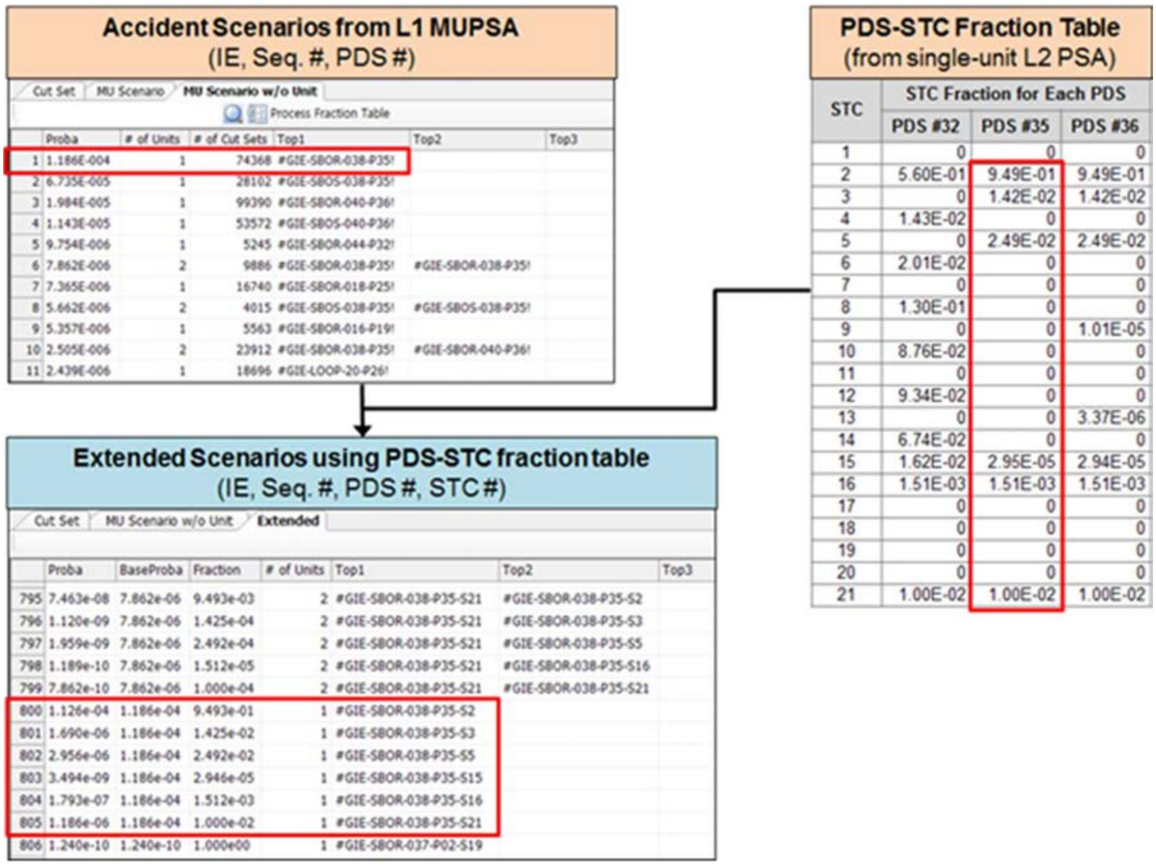


FIG. 88. Example of extending Level 1 scenarios to Level 2 scenarios.

4.3.8.3. Simplification of source term categories and modelling approach

The STC combinations may increase extremely depending on the number of STCs in each unit. If there are three units with 15, 20, and 30 STCs, possible combinations are 9,000 (15 * 20 * 30) in theory. It is practically impossible to deal with all these combinations in Level 2 MUPSA and makes Level 3 MUPSA difficult to manage. Hence, KHNP developed a simplification approach to perform Level 2 MUPSA efficiently. This approach adopted Source Term Category Group (STCG) concept. STCG is grouping STCs into similar release characteristics. This can reduce the number of combinations of STCs to be handled practically. The STCGs defined by KHNP are as follows:

— STCG–1: NOCF (No Containment Failure)
— STCG–2: ECF (Early Containment Failure)
— STCG–3: LCF (Late Containment Failure)
— STCG–4: BMT (Basement Melt Through)
— STCG–5: CFBRB (Containment Failure Before Reactor Vessel Breach)
— STCG–6: NOISO (No Isolation Failure)
— STCG–7: BYPASS

Among the seven STCG above, LERF defined by KHNP as a risk metric corresponds to STCG–2, STCG–6, and STCG–7. It is noted that STCG grouping criteria may vary depending on the defined risk metric and may also vary depending on the structure and characteristics of the model. Through the simplification of STCs, each PDS has one to seven branch probability of STCG.

Figure 89 shows modelling example of Level 2 MUPSA. Each scenario of the PDS event tree has one top model extended from Level 1 MUPSA model and includes the STCG branch probability of the corresponding PDS.

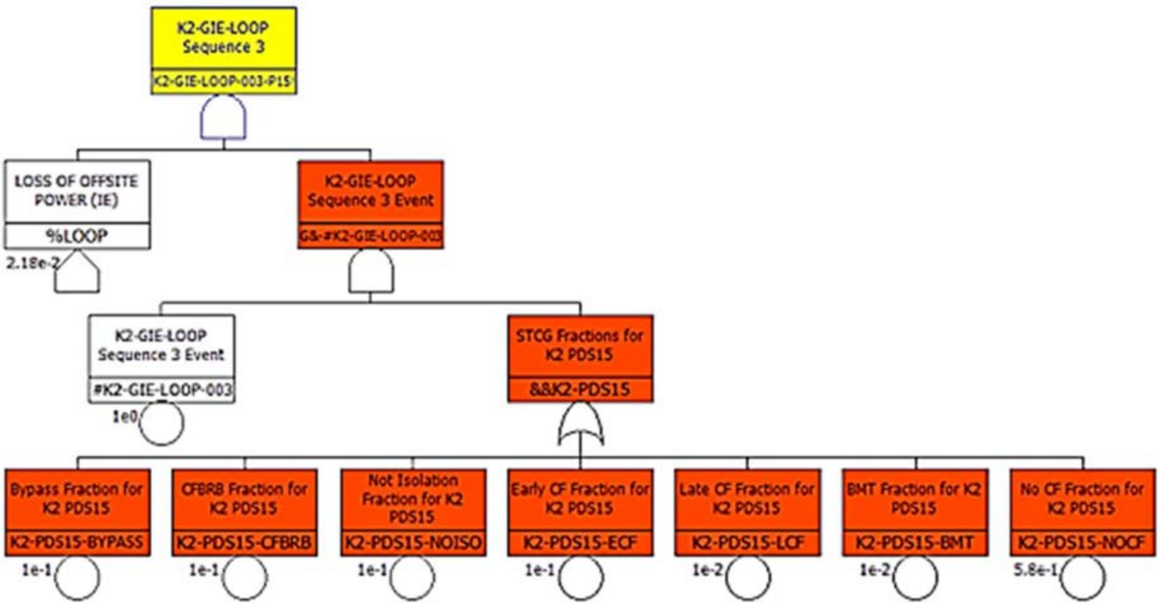


FIG. 89. Example of Level 2 MUPSA modelling.

4.3.9. Republic of Korea/Hanyang University

Figure 90 shows the overall structure for the Level 2 MUPSA process. The first step is the development of the SUPSA model such as a PDS, a CET with a decomposition event trees, a source term logic tree and the frequency quantification of the containment failure and the source term categories. In this step a PDSs to STC mapping table is generated for the single unit. The second step is to develop the MUPSA model from the SUPSA information mainly considering the inter–unit effect. This multiunit model generates the multiunit plant damage sequences and quantifies the frequency of multiunit PDSs. The third step is to propagate the multiunit PDSs through the PDSs to STC mapping table in a similar way to the SUPSA. The final step is grouping the end state of mapping table into the multiunit containment failure sequences or the multiunit source term categories. In this process the major assumption is that the severe accident progression in one plant does not affect the other plants. This assumption is reasonable in typical PWRs since they are designed not to share the important safety systems between units.

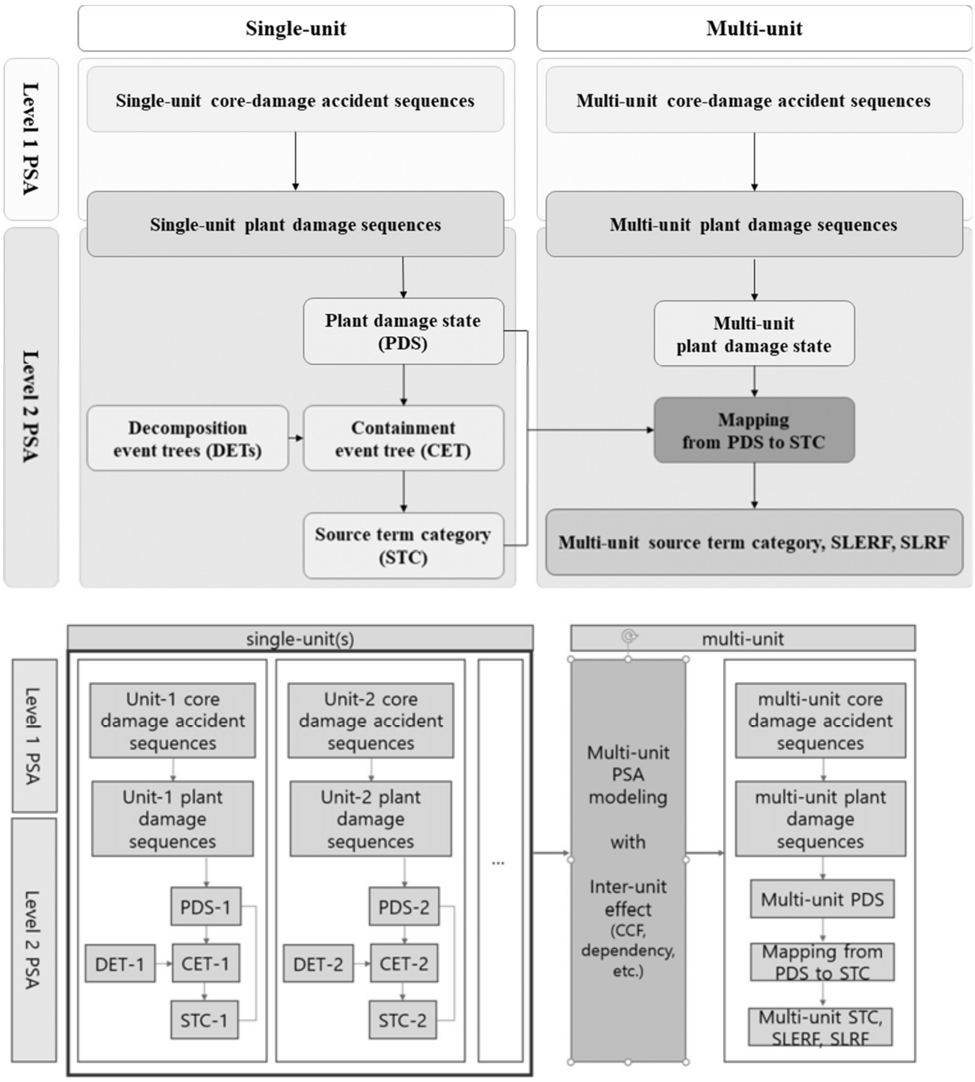


FIG. 90. Multiunit Level 2 PSA framework.

Development of model for multiunit plant damage sequences (Fig. 91): multiunit Level 1 PSA developed the multiunit core damage accident sequences with the truth–table method using AIMS–PSA and FTREX program. This SRA model for Level 1 PSA was explained in the previous section. The model for the multiunit plant damage sequences is developed based on the SRA model for Level 1 PSA. The event trees have been expanded to include the containment systems at or after the core damage. The included systems are, for example, the containment isolation system, the hydrogen control system. This process is similar to the process for the interface with Level 1 PSA and Level 2 PSA in the SUPSA. The combined PDSET is expressed as $PDSET_{Xi}$ * $PDSET_{Yj}$. In this expression the subscript X and Y mean the plant number and the subscript i and j mean the PDSET number for the plant X and the plant Y, respectively.

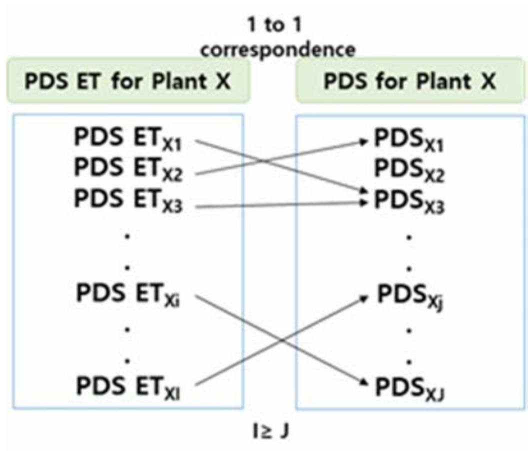


FIG. 91. Relationship between PDSET and PDS for plant X.

Multiunit PDS: after multiunit plant damage sequences are quantified, each plant damage sequence is classified into PDS. Figure 92 illustrates the relationship between PDSET and PDSs for plant X. When the core damage occurred at the only one plant, the PDSs assignment is same as the single unit PDSs assignment. When the core damage occurred at two or more plants, this plant damage sequence is classified into the combined PDSs using PDS logic trees for the plant which the core damage occurred. For example, when the core damage occurred at the plant X and Y simultaneously, the plant damage sequence for plant X is classified in to the PDSs for the plant X using the PDSs logic tree for the plant X and the plant damage sequence for the plant Y is classified in to the PDS for the plant Y using the PDSs logic tree for the plant Y. The combined PDSs is expressed as PDS_{Xj} * PDS_{Yk}. In this expression the subscript X and Y mean the plant number and the subscript j and k mean the PDSs number for the plant X and the plant Y, respectively.

Multiunit source term category: Figure 90 shows the conceptual diagram for obtaining the Level 2 PSA result for plant X. The PDSs–STC mapping table for plant X is created through this process. In the PDS–STC mapping table, the split fractions from PDS_{Xj} into every CET end state are expressed as f_{Xj1} to f_{Xjk}. The sum of f_{Xj1} to f_{Xjk} is equal to one. When the core damage occurred at the only plant X, the frequency of PDS_{Xj} of a plant X is divided into K containment failure sequences and L source term categories of a Level 2 MUPSA. When the core damage occurred at the two plant X and Y, the combined PDSs is expressed as PDS_{Xj} * PDS_{Yk} and is divided into the source term category of each plant's source term category. This mapping table idea is originally proposed by KAERI. This idea is based in the assumption that the severe accident progression at the one plant does not affect the severe accident progression at other plants.

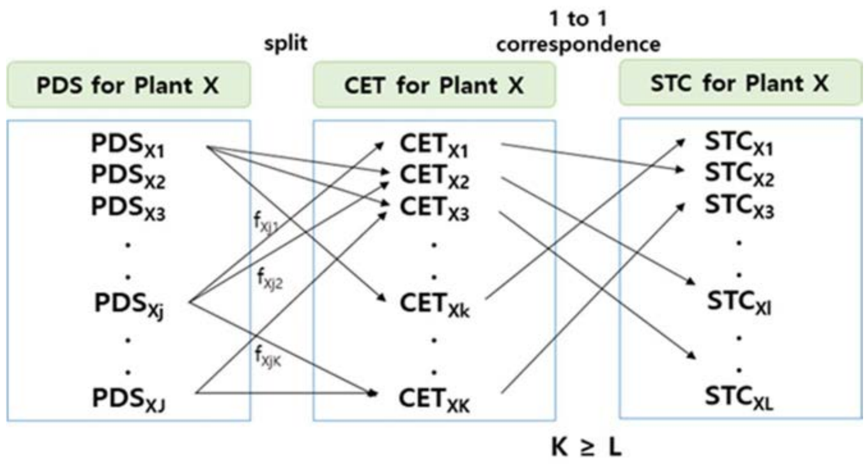


FIG. 92. PDS mapping to STC through CET for plant X.

Containment failure mode for MUPSA Level 2: Source terms for a Level 2 SUPSA can be classified into 6 categories: no containment failure, early containment failure, late containment failure, containment bypass, containment isolation failure, and basemat melt–through. With these six source term categories for each plant, multiunit source term categories would become very complex, e.g. leading to 216 category combinations for a three unit site. When at least one plant in the site experiences early containment failure, containment bypass, or containment isolation failure, it is considered a site large early release (SLER).

Computer code: the plant damage frequency is calculated using FTREX (fault tree Reliability Evaluation eXpert) code for the internal event. BeEAST code for the seismic event instead of FTREX which uses the rare event approximation and overestimates when quantifying the seismic event. BeEAST (Boolean equation Evaluation Analysis, and Sensitivity Tool) code is developed by Prof. Jung. CET is quantified by COFUN (COntainment Failure probability and UNcertainty analysis program) code which is developed at Hanyang university. COFUN code used quantifying Level 2 MUPSA as well as quantifying the Level 1 SUPSA. It also has capability to calculate the uncertainty of Level 2 SUPSA result and Level 2 MUPSA result.

4.3.10. Romania/CNCAN

From one side the PSA Level 2 has an increased technical basis for plant behavior in severe accidents than the PSA Level 1, which counts on Design Basis Accidents, proven and validated. The PSA Level 2 scenarios are defined by severe accidents software, under intensive scrutiny and review from experimental and validation aspects, having as a result a high degree of uncertainty in the decisions on possible scenarios is barriers for Level 1 are triggered. Therefore, the decision points, called in CAFTA ‘headings’ and in RiskSpectrum ‘Function events’ have a degree of uncertainty to be considered in a series of sensitivity analyses. From another side the MUPSA type initiating event relate the interface between units and sources not only at the level of impact on their scenarios if reacting to challenges similar to initiating event ones, but also a set of possible correlating failures throughout the site for all sources/units. If this impact may be not significant in the case of PSA Level 1 MUPSA< for PSA Level 2 MUPSA is of very high impact. However, the review and iterations on those assumptions are based on intensive severe codes evaluations (in case of CANDU for instance MAAP 4 CANDU) and/or experiments. The practical implementation of the MUPSA model building is based on the methodology for building MUPSA model is applied for the list of multiunit initiating event (Tables 58 and 59).

TABLE 58. MULTIUNIT INITIATING EVENT SUPSA HEADINGS (CAFTA MODELLING)

Release category	Multiunit initiating event SUPSA headings	U1	U2	U3	U4
REL0	IEM1_GT_1	GT–S1*;S2*	GT–S1*;S2*		
	IEM4_LOCLIV_1	LOCLIV–S1*;S2*;S3*	GT–S1**;S2**	GT–S1**;S2**	GT–S1**;S2**
REL1	IEM2_GT_2	GT*	GT*	GT*	GT*
	IEM3_GT_2	GT*	LOCLIV–S1*;S2*;S3*	LOCLIV–S1*;S2*;S3*	LOCLIV–S1*;S2*;S3*
	IEM5_LOCLIV_2	LOCLIV–S1*;S2*;S3*	LOCLIV–S1*;S2*;S3*	LOCLIV–S1*;S2*;S3*	LOCLIV–S1*;S2*;S3*

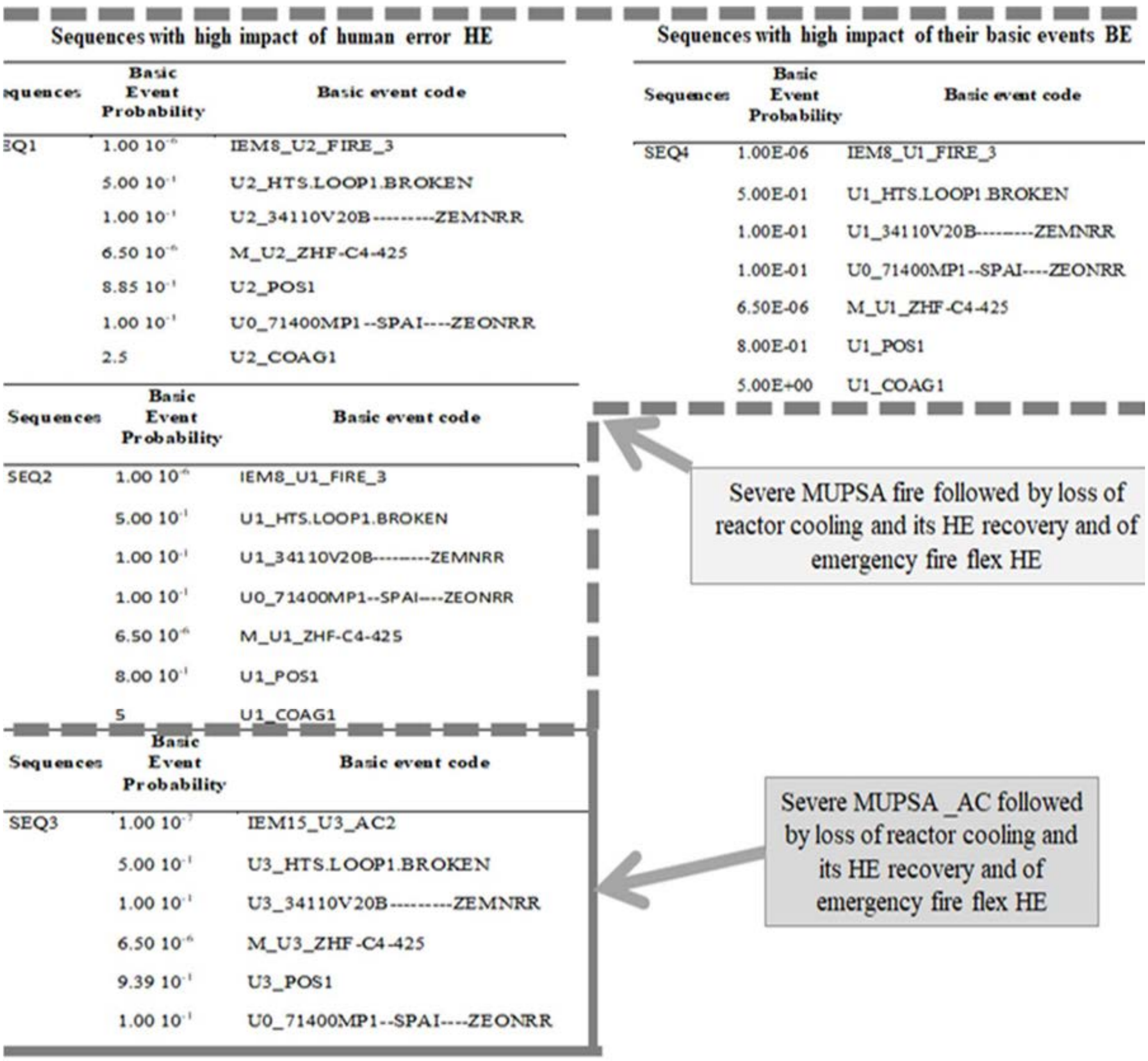
TABLE 59. MULTIUNIT INITIATING EVENT NEW SEQUENCES

Release category	Multiunit initiating event new sequences	U1	U2	U3	U4
REL0	IEM6_FIRE_1	GT**	GT**	GT**	GT**
	IEM7_FIRE_2	LOCLIV–S1**;S2**;S3**	GT–S1***;S2***	GT–S1**;S2**	GT–S1**;S2**
	IEM9_DROUGHT_1	GT**	GT**	GT**	GT**
	IEM10_S2	GT**	GT**	GT**	GT**
	IEM13_EW	LOCLIV–S1**;S2**;S3**	LOCLIV–S1**;S2**;S3**	LOCLIV–S1**;S2**;S3**	LOCLIV–S1**;S2**;S3**
	IEM16_CTRF	GT**	GT**	GT**	GT**
	IEM17_DICA	GT**	GT**	GT**	GT**
REL1	IEM8_Fire_3	SFB	SFB	SFB	SFB
	IEM11_S3	GT** LOCLIV–S1**;S2**;S3**	GT** LOCLIV–S1**;S2**;S3**	GT** LOCLIV–S1**;S2**;S3**	GT** LOCLIV–S1**;S2**;S3**
	IEM14_AC_1	GT** LOCLIV–S1**;S2**;S3**	GT** LOCLIV–S1**;S2**;S3**	GT** LOCLIV–S1**;S2**;S3**	GT** LOCLIV–S1**;S2**;S3**
REL2	IEM12_S4	IEM8_FIRE_3 LOCLIV–S1**;S2**;S3**	IEM8_FIRE_3 LOCLIV–S1**;S2**;S3**	IEM8_FIRE_3 LOCLIV–S1**;S2**;S3**	IEM8_FIRE_3 LOCLIV–S1**;S2**;S3**
	IEM8_FIRE_3	LOCLIV–S1**;S2**;S3**	LOCLIV–S1**;S2**;S3**	LOCLIV–S1**;S2**;S3**	LOCLIV–S1**;S2**;S3**
	IEM15_AC_2	IEM8_FIRE_3 IEM17_DICA	IEM8_FIRE_3 IEM17_DICA	IEM8_FIRE_3 IEM17_DICA	IEM8_FIRE_3 IEM17_DICA

The rules applied for modifying SUPSA model to MUPSA are divided into two categories of approaches, depending how the amended sequences in SUPSA are implemented, by:

— Minimal changes and impact on existing SUPSA model for category;
— Extensive changes to SUPSA for category.

Figure 93 and Table 60 illustrate the results of the implementation process of the matrices from Tables 57 and 58, by using control files to a CAFTA model. In this process the existing SUPSA model defines the barriers of the plant reaction to a certain initiating event (in the illustrated case they are noted as for instance U1_GT_ALLSEQ meaning the barrier to the unit 1 at a general transient (GT). A MUPSA initiating event as represented in the orange circled area will trigger the same barriers as for SUPSA. However, new basic events specific to the multiunit impact will be introduced. In the meantime for some MUPSA initiating event (multiunit initiating event or IEM) there will be cases of high degree of change of barriers



Sequences	Basic Event Probability	Basic event code
SEQ1	$1.00\ 10^{-6}$	IEM8_U2_FIRE_3
	$5.00\ 10^{-1}$	U2_HTS.LOOP1.BROKEN
	$1.00\ 10^{-1}$	U2_34110V20B---------ZEMNRR
	$6.50\ 10^{-6}$	M_U2_ZHF-C4-425
	$8.85\ 10^{-1}$	U2_POS1
	$1.00\ 10^{-1}$	U0_71400MP1--SPAI----ZEONRR
	2.5	U2_COAG1

Sequences	Basic Event Probability	Basic event code
SEQ2	$1.00\ 10^{-6}$	IEM8_U1_FIRE_3
	$5.00\ 10^{-1}$	U1_HTS.LOOP1.BROKEN
	$1.00\ 10^{-1}$	U1_34110V20B---------ZEMNRR
	$1.00\ 10^{-1}$	U0_71400MP1--SPAI----ZEONRR
	$6.50\ 10^{-6}$	M_U1_ZHF-C4-425
	$8.00\ 10^{-1}$	U1_POS1
	5	U1_COAG1

Sequences	Basic Event Probability	Basic event code
SEQ3	$1.00\ 10^{-7}$	IEM15_U3_AC2
	$5.00\ 10^{-1}$	U3_HTS.LOOP1.BROKEN
	$1.00\ 10^{-1}$	U3_34110V20B---------ZEMNRR
	$6.50\ 10^{-6}$	M_U3_ZHF-C4-425
	$9.39\ 10^{-1}$	U3_POS1
	$1.00\ 10^{-1}$	U0_71400MP1--SPAI----ZEONRR

Sequences	Basic Event Probability	Basic event code
SEQ4	1.00E-06	IEM8_U1_FIRE_3
	5.00E-01	U1_HTS.LOOP1.BROKEN
	1.00E-01	U1_34110V20B---------ZEMNRR
	1.00E-01	U0_71400MP1--SPAI----ZEONRR
	6.50E-06	M_U1_ZHF-C4-425
	8.00E-01	U1_POS1
	5.00E+00	U1_COAG1

FIG. 93. Sample of dominant sequences results for MUPSA by developing from SUPSA to MUPSA.

TABLE 60. DOMINANT MUPSA SEQUENCES DERIVED BY USING THE METHOD TO DEVELOP FROM SUPSA TO MUPSA

Sequences	Basic Event Probability	Basic event code
SEQ1	1.00 10^{-6}	IEM8_U2_FIRE_3
	5.00 10^{-1}	U2_HTS.LOOP1.BROKEN
	1.00 10^{-1}	U2_34110V20B————ZEMNRR
	6.50 10^{-6}	M_U2_ZHF–C4–425
	8.85 10^{-1}	U2_POS1
	1.00 10^{-1}	U0_71400MP1—SPAI——ZEONRR
	2.5	U2_COAG1
SEQ2	1.00 10^{-6}	IEM8_U1_FIRE_3
	5.00 10^{-1}	U1_HTS.LOOP1.BROKEN
	1.00 10^{-1}	U1_34110V20B————ZEMNRR
	1.00 10^{-1}	U0_71400MP1—SPAI——ZEONRR
	6.50 10^{-6}	M_U1_ZHF–C4–425
	8.00 10^{-1}	U1_POS1
	5	U1_COAG1
SEQ3	1.00 10^{-7}	IEM15_U3_AC2
	5.00 10^{-1}	U3_HTS.LOOP1.BROKEN
	1.00 10^{-1}	U3_34110V20B————ZEMNRR
	6.50 10^{-6}	M_U3_ZHF–C4–425
	9.39 10^{-1}	U3_POS1
	1.00 10^{-1}	U0_71400MP1—SPAI——ZEONRR

4.3.10.1. Process of building solutions

The modelling of MUPSA is used to build the solutions of the plant reaction to multiunit type of challenges. This is a process comprising the model quantification followed by intensive sensitivity analyses and with indications on the issues for the next iterations. Most frequently for PSA Level 2 iteration are related to the extensive need for deterministic recalculations of various scenarios.

The enveloping results for MUPSA are compared to those for SUPSA, to identify specific new issues for multiunit case. A set of sample results for CAFTA type runs for SUPSA case A are represented in Fig. 94.

The results of MUPSA are in a format like in Table 61, and Fig. 95 shows the sample case A results from CAFTA. The results are used to compare with the SUPSA case and to identify the new elements of impact (and their ranking) in case of the multiunit model.

The generic sample runs as shown in Figs. 96 and 97 are the results of MUPSA in case A in one iteration before sensitivity evaluations and are based on detailed set of ranked sequences for each release category, as illustrated in Tables 62–64 in a CAFTA run set.

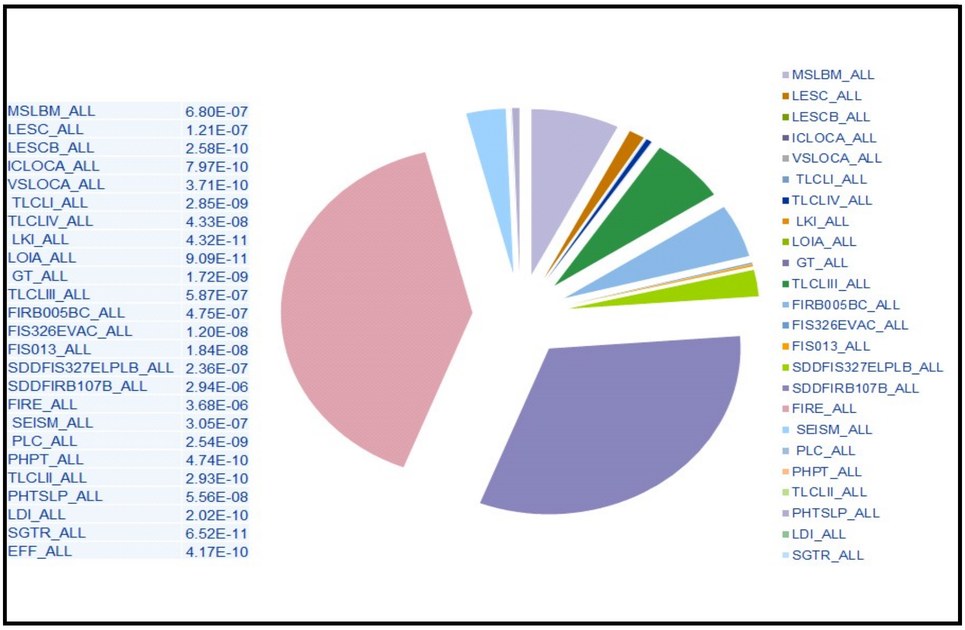


FIG. 94. Sample results for SUPSA simplified model case A in one iteration.

TABLE 61. RESULTS ON SOME DOMINANT IN MUPSA COMPARED TO SUPSA

RANK		**RANK**	**DELTA MULTI**	**TOTAL**
I	FIRE_ALL	4	3.0E+00	3.8×10^{-6}
II	SEISM_ALL	3	1.0E+01	3.1×10^{-7}
II	TLCLI–IV_ALL	3	2.0E+00	6.0×10^{-7}
III	GT_ALL	2	6.0E+00	1.8×10^{-9}
IV	LKI_ALL	1	2.0E+00	4.5×10^{-11}
IV	LOIA_ALL	1	1.0E+00	9.1×10^{-11}

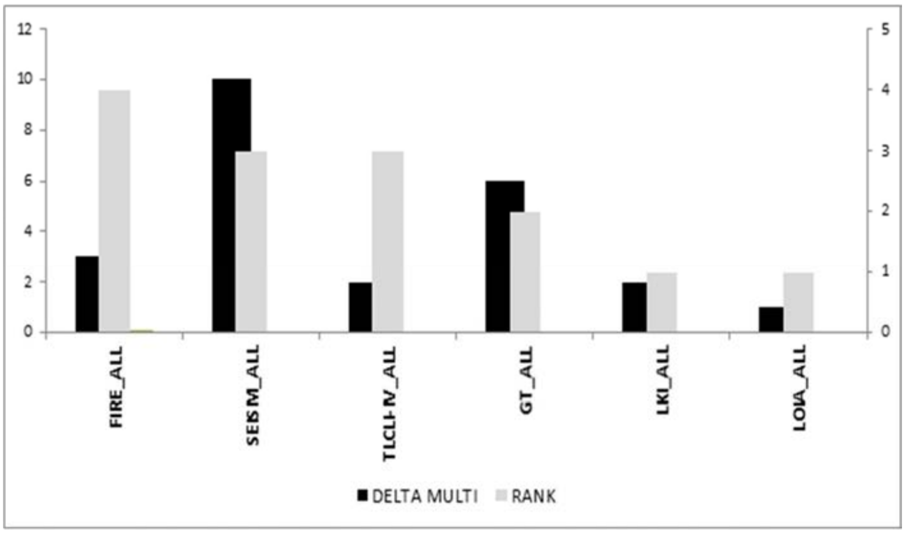


FIG. 95. Sample results of MUPSA in case A in one iteration before sensitivity evaluations.

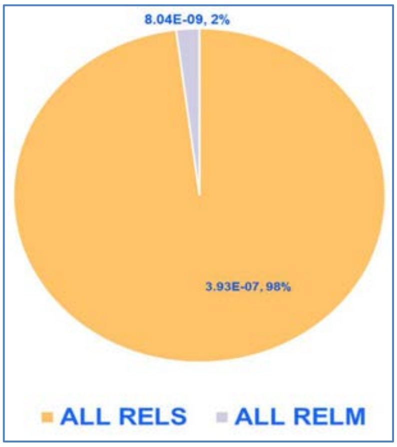


FIG. 96. Sample results of MUPSA in case A in one iteration before sensitivity evaluations.

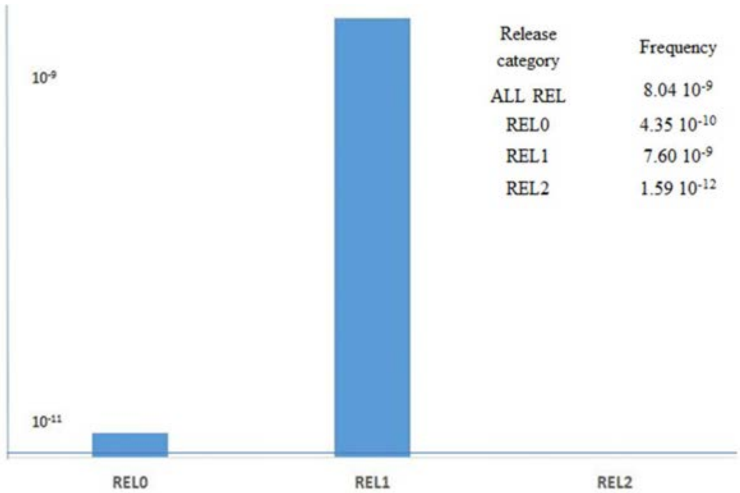


FIG. 97. Sample results of MUPSA in case A in one iteration before sensitivity evaluations.

TABLE 62. SAMPLE DETAILED RESULTS FOR REL0 CATEGORY OF RELEASE

REL0	Initiating event and basic event	Description
SEQ1	IEM4_U1_LOCLIV_1	
	U1_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U1_34110V20B–ZEMNRR	Failure to manually open valve 34110V20B
	U0_71400MP1–SPAI–ZEONRR	Operator fails to start 71400MP1 motor driven pump
	M_U1_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP si EPS
	U1_POS1	At power operation
	U1_COAG1	Ageing factor
SEQ2	IEM4_U1_LOCLIV_1	
	U1_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U1_52300DG2–DG3–UM	Class III 52300 DG2 unavailable due to maintenance

TABLE 62. SAMPLE DETAILED RESULTS FOR REL0 CATEGORY OF RELEASE (CONT.)

REL0	Initiating event and basic event	Description
	M_U1_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U1_34110V20B–ZEMNRR	Failure to manually open valve 34110–V20B
	U0_71400MP1–SPAI–ZEONRR	Operator fails to start 71400MP1 motor driven pump
	U1_POS1	At power operation
	U1_COAG1	Ageing factor
SEQ3	IEM10_U1_S2_SD	S2 initiating event FOR MUPSA MODEL
	U1_DR04.IS.CLOSED.SD	R/B Personnel Door DR04 is closed (in shutdown)
	U1_DR03.IS.CLOSED.SD	R/B Equipment Door DR03 is closed (in shutdown)
	U1_21601V3–VXCBIL	EAL R/B Equalizing valve internal leaks
	M_ALL_Z2OFFSITEPOWER	Loss of offsite power, fire induced due to seismic event
	U1_POS9.2	Sub–POS9.2 – Minor mtce. activities
	U1_POS9	HTS cold, drained and open (2 SDC pumps and 2 SDC HXs)
	U1_COAG1	Ageing factor
SEQ4	IEM1_U1_GT_1	General transient initiating event for MUPSA model
	M_U1_52900EPS.FAIL.ZESNR1	Operator fails to start EPS (as per APOP–G01)
	M_U1_CLIII.ALL.DG.FS.CCF	DG1, DG2, DG3 and DG4 fail to start due to common cause
	U1_POS1	At power operation
	ALL_CONSEQ.LCIV	Consequential loss of class IV
	U1_DR06.NORMAL.OPERATION.FP	Normal Operation of DR06
	U1_21601DR06–ZESNRR	Operator fails to close DR06 (using handwheel)
	U1_COAG1	Ageing factor
SEQ5	IEM10_U1_S2_SD	S2 initiating event FOR MUPSA MODEL
	U1_DR04.IS.CLOSED.SD	R/B Personnel Door DR04 is closed (in shutdown)
	U1_DR03.IS.CLOSED.SD	R/B Equipment Door DR03 is closed (in shutdown)
	U1_21601V3–VXCBIL	EAL R/B Equilizing valve internal leaks
	M_U1_Z2–RSW–RLCH	Relay Chatter
	M_U1_RECLOSE_3	Failure to Reclose HTS Opening within Recall time curve #3 in MULTIUNIT event
	U1_POS9.2	Sub–POS9.2 – Minor mtce. activities
	U1_POS9	HTS cold, drained and open (2 SDC pumps and 2 SDC HXs)
	U1_COAG1	Ageing factor

TABLE 63. SAMPLE RESULTS OF SOME RUNS FOR REL1 CATEGORY OF RELEASE

REL1	Initiating event and basic event	Description
SEQ1	IEM18_U1_LOCLIII_SD	Total Loss of Class III Power Supply FOR MUPSA MODEL
	M_U1_REC_COOL_PHTS	Failure to recover PHTS cooling in POS9 in Multiple unit event
	U1_DR04.IS.CLOSED.SD	R/B Personnel Door DR04 is closed (in shutdown)
	U1_DR03.IS.CLOSED.SD	R/B Equipment Door DR03 is closed (in shutdown)
	U1_21601V3–VXCBIL	EAL R/B Equalizing valve internal leaks

TABLE 63. SAMPLE RESULTS OF SOME RUNS FOR REL1 CATEGORY OF RELEASE (CONT.)

REL1	Initiating event and basic event	Description
	U1_POS9.2	Sub–POS9.2 – Minor mtce. activities
	U1_POS9	HTS cold, drained and open (2 SDC pumps and 2 SDC HXs)
	U1_COAG1	Ageing factor
SEQ2	IEM18_U1_LOCLIII_SD	Total Loss of Class III Power Supply FOR MUPSA MODEL
	M_U1_REC_COOL_PHTS	Failure to recover PHTS cooling in POS9 in Multiple unit event
	U1_21602SEALS–1/2SF——NCCF	Both seals fail to seal due to common cause
	U1_POS9.2	Sub–POS9.2 – Minor mtce. activities
	U1_POS9	HTS cold, drained and open (2 SDC pumps and 2 SDC HXs)
	U1_COAG1	Ageing factor
SEQ3	IEM11_U1_S3_SD	S3 initiating event FOR MUPSA MODEL
	U1_DR06.IS.CLOSED.SD	R/B Personnel Door DR06 is closed (in shutdown)
	M_U1_21601V6——————VXCBIL	PAL R/B Equalizing valve internal leaks in MULTIUNIT
	U1_POS9.2	Sub–POS9.2 – Minor mtce. activities
	U1_POS9	HTS cold, drained and open (2 SDC pumps and 2 SDC HXs)
	U1_COAG1	Ageing factor
SEQ4	IEM5_U1_LOCLIV_2	
	U1_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U1_34110V20B————ZEMNRR	Failure to manually open valve 34110–V20B
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
	M_U1_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP or EPS
	U1_POS1	At power operation
	U1_COAG1	Ageing factor
SEQ5	IEM2_U1_GT_2	General Transient initiating event FOR MUPSA MODEL
	U1_DR04.IS.CLOSED.FP	R/B Personnel Door DR04 is closed
	U1_DR03.IS.CLOSED.FP	R/B Equipment Door DR03 is closed
	U1_21601DR03SEALS–1/2SFNCCF	DR03 Both seals fail to seal due to common cause
	U1_52900EPS.FAIL——ZESNR1	Operator fails to start EPS (as per APOP–G01)
	M_U1_CLIII.ALL.DG.FS.CCF	DG1, DG2, DG3 and DG4 fail to start due to common cause
	U1_POS1	At power operation
	ALL_CONSEQ.LCIV	Consequential loss of class IV
	U1_COAG1	Ageing factor
SEQ6	IEM11_U2_S3	
	U2_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U2_OMKCVL	operator fails to initiate makeup to calandria vault

TABLE 63. SAMPLE RESULTS OF SOME RUNS FOR REL1 CATEGORY OF RELEASE (CONT.)

REL1	Initiating event and basic event	Description
	U2_52300DG4————— DG3–UM	Class III 52300 DG4 unavailable due to maintenance
	M_U2_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U2_POS1	At power operation
	U2_COAG1	Ageing factor
SEQ7	IEM14_U2_AC1	
	U2_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U2_34110V20B————— ZEMNRR	Failure to manually open valve 34110–V20B
	M_U2_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP si EPS
	U2_POS1	At power operation
	U0_71400MP1—SPAI—— ZEONRR	Operator fails to start 71400–MP1 motor driven pump
	U2_COAG1	Ageing factor
SEQ8	IEM14_U4_AC1	
	U4_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U4_52300DG2————— DG3–UM	Class III 52300 DG2 unavailable due to maintenance
	M_U4_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U4_34110V20B————— ZEMNRR	Failure to manually open valve 34110–V20B
	U4_POS1	At power operation
	U0_71400MP1—SPAI—— ZEONRR	Operator fails to start 71400–MP1 motor driven pump

TABLE 64. SAMPLE RESULTS OF SOME RUNS FOR REL2 CATEGORY OF RELEASE

REL2	Initiating event and basic event	Description
SEQ1	IEM8_U1_FIRE_3	
	U1_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U1_34110V20B————— ZEMNRR	Failure to manually open valve 34110–V20B
	U0_71400MP1—SPAI—— ZEONRR	Operator fails to start 71400–MP1 motor driven pump
	M_U1_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP si EPS
	U1_POS1	At power operation
	U1_COAG1	Ageing factor
SEQ2	IEM12_U1_S4	
	U1_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U1_34110V20B————— ZEMNRR	Failure to manually open valve 34110–V20B

TABLE 64. SAMPLE RESULTS OF SOME RUNS FOR REL2 CATEGORY OF RELEASE (CONT.)

REL2	Initiating event and basic event	Description
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
	M_U1_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP si EPS
	U1_POS1	At power operation
	U1_COAG1	Ageing factor
SEQ3	IEM8_U2_FIRE_3	%FIRE_2 FOR MUPSA MODEL
	U2_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U2_34110V20B————ZEMNRR	Failure to manually open valve 34110–V20B
	M_U2_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP si EPS
	U2_POS1	At power operation
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
	U2_COAG1	Ageing factor
SEQ4	IEM8_U3_FIRE_3	FIRE_3 FOR MUPSA MODEL
	U3_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U3_52300DG2—————DG3–UM	Class III 52300 DG2 unavailable due to maintenance
	M_U3_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U3_34110V20B————ZEMNRR	Failure to manually open valve 34110–V20B
	U3_POS1	At power operation
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
SEQ5	IEM8_U4_FIRE_3	FIRE_3 FOR MUPSA MODEL
	U4_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U4_52300DG2—————DG3–UM	Class III 52300 DG2 unavailable due to maintenance
	M_U4_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U4_34110V20B————ZEMNRR	Failure to manually open valve 34110–V20B
	U4_POS1	At power operation
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
SEQ6	IEM12_U1_S4	
	U1_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U1_52300DG2—————DG3–UM	Class III 52300 DG2 unavailable due to maintenance
	M_U1_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U1_34110V20B————ZEMNRR	Failure to manually open valve 34110–V20B

TABLE 64. SAMPLE RESULTS OF SOME RUNS FOR REL2 CATEGORY OF RELEASE (CONT.)

REL2	Initiating event and basic event	Description
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
	U1_POS1	At power operation
	U1_COAG1	Ageing factor
SEQ7	IEM12_U3_S4	
	U3_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U3_52300DG2——————DG3–UM	Class III 52300 DG2 unavailable due to maintenance
	M_U3_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U3_34110V20B—————ZEMNRR	Failure to manually open valve 34110–V20B
	U3_POS1	At power operation
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
SEQ8	IEM12_U4_S4	
	U4_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U4_52300DG2——————DG3–UM	Class III 52300 DG2 unavailable due to maintenance
	M_U4_ZHF–C4–017	Operator fails to start not–running feed pump, to initiate ECCS and EWS to PHTS
	U4_34110V20B—————ZEMNRR	Failure to manually open valve 34110–V20B
	U4_POS1	At power operation
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
SEQ9	IEM15_U3_AC2	General Transient initiating event FOR MUPSA MODEL
	U3_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U3_34110V20B—————ZEMNRR	Failure to manually open valve 34110–V20B
	M_U3_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP si EPS
	U3_POS1	At power operation
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
SEQ10	IEM15_U2_AC2	General Transient initiating event FOR MUPSA MODEL
	U2_HTS.LOOP1.BROKEN	PHTS Break located on LOOP 1
	U2_34110V20B—————ZEMNRR	Failure to manually open valve 34110–V20B
	M_U2_ZHF–C4–425	Operator fails to initiate SOLID mode, ECCS, ECCS LP si EPS
	U2_POS1	At power operation
	U0_71400MP1—SPAI——ZEONRR	Operator fails to start 71400–MP1 motor driven pump
	U2_COAG1	Ageing factor

A sample set of results indicate as important ranked sequences for each category of releases defined for PSA Level 2, as ranked before the phase of extensive sensitivity case and at first runs of the model, before the review of the deterministic safety re-evaluations of the assumptions, are the following:

For MUPSA case A REL2

- High intensity earthquake followed by loss of containment leak tightness during open reactor in shutdown mode;
- Loss of class IV followed by loss of leak tightness and cooling reactor due to human error;
- General transient followed by multi loss of class IV followed by loss of leak tightness, cooling reactor and emergency power;
- Severe site fire followed by loss of reactor cooling and human error to restore it and to assure power supply and fire flex equipment;
- Severe earthquake followed by loss of reactor cooling and human error to restore it and to assure power supply and fire flex equipment.

For MUPSA case A REL1

- High intensity earthquake followed by loss of containment leak tightness during open reactor in shutdown mode without other human error;
- Loss of class IV followed by multi loss of class IV followed by loss of leak tightness, cooling reactor and emergency power with existing recovery possible;
- General transient followed by multi loss of class IV followed by loss of leak tightness, cooling reactor leading to medium release and loss emergency power.

For MUPSA case A REL0

- Medium intensity earthquake followed by loss of containment leak tightness leading to low releases during open reactor in shutdown, LOOP and without other human error during recovery;
- General transient followed by multi loss of class IV followed by loss of leak tightness and human error to recover it, CCF of class III Diesels and consequential loss of class IV and human error to start emergency power;
- Medium intensity earthquake followed by loss of containment leak tightness leading to low releases during open reactor in shutdown and power relay control chatter due to earthquake.

- There are some generic issues identified at the first iteration of runs, which indicate aspects of impact in the case of MUPSA, as follows:
- The impact of multiunit initiating event is within 10 % increase for risk metrics of Level 2 (LERF as reflected in REL0,1,2) for some dominant multiunit initiating event like Multiunit LOOP, multiunit general transient;
- The Seismic impact needs more detailed analysis. There are some lists of sequences for Level 2 that need review from MAAP runs point of view to confirm assumed behaviours;
- A nest iteration is done to cross check model assumptions and differences in codes results;
- The sensitivity analyses is complemented by more factors especially on the correlation ones for multiunit impact modelling;

— New methodologies are useful for the process of checking the results of building MUPSA starting from SUPSA, as for instance the approaches based on operational research.

4.3.10.2. Sensitivity analysis

Sensitivity analysis for a CAFTA MUPSA case A model: a series of sensitivity analyses are performed to check the stability of the solutions for the CAFTA model. They illustrate also specific insights on general issues, as for instance HRA impact on the model. The sensitivity analyses cases consist of variation by orders of magnitude the frequencies for the HRA actions, as shown in Tables 65–71.

Sensitivity analyses cases indicate the impact of the dominating element evaluated on the set of results. In the case of HRA sensitivity analyses a set of insights are derived and may be considered important for the next iterations and eventually to the decision making process. It can be concluded for the HRA sensitivity analyses results that the impact on various release categories of the human error is important and improvements in human performance has significant impact as follows:

— For REL0 by improving the training program for operators the frequency of REL0 is decreased with 99%;
— For REL1 by improving the training program for operators and by decreasing the POS9 duration, the results are significantly improved, the frequency is decreased with 86.42%;
— For REL2 by improving the training program for operators the frequency is decreased with 96%.

— In the case of the basic event sensitivity analyses another set of results indicate significant dominating factors for various release categories, as follows:
— For the REL2 the dominant sequences (Fig. 98) are severe MUPSA fire followed by loss of reactor cooling and its human error recovery and of emergency fire flex human error;
— Severe MUPSA _AC followed by loss of reactor cooling and its human error recovery and of emergency fire flex human error;
— For REL1 the dominant sequences are shown in Table 68;
— Loss of CLASS III in shutdown with reactor open and actions to assure cooling with average releases followed by loss of containment leak tightness;
— For REL0 the dominant sequences are shown in Table 71;
— Loss of class IV in MUPSA during operation followed by loss of ESC in a broken loop with small releases and loss of containment leak tightness.

TABLE 65. HRA FOR CASE 1 FOR REL2 MULTIPLE UNIT (MUPSA) MODEL

Case No	Event	Fussell–Vesely	Probability (initial)	Modified Probability change	REL2 (Initial)	Modified REL2 Increase/Decrease
1.	U1_REC_COOL_PHTS	0.638	2.5×10^{-01}	$1/2.5\times10^{-02}$	6.37×10^{-07}	$1.86\times10^{-06}/2.71\times10^{-07}$ +192%/ –57%
2.	U0_71400MP1—SPAI——ZEONRR	0.348	1×10^{-01}	$1/1\times10^{-02}$	6.37×10^{-07}	$2.63\times10^{-06}/4.37\times10^{-07}$ +313%/–31%
3.	U1_34110V20B————ZEMNRR	0.348	1×10^{-01}	$1/1\times10^{-02}$	6.37×10^{-07}	$2.63\times10^{-06}/4.37\times10^{-07}$ +313%/–31%
4.	U1_52900EPS.FAIL–FI_C	0.348	3.3×10^{-02}	$3.3\times10^{-01}/3.3\times10^{-03}$	6.37×10^{-07}	2.63×10–06/4.37×10^{-07} +313%/–31%

TABLE 66. HRA CASE 2 FOR REL2 MULTIPLE UNIT MODEL

Case No	Event	Fussell–Vesely	Probability (initial)	Modified Probability Increase/ Decrease	REL2 (Initial)	Modified REL2 Increase/Decrease
1.	U0_71400MP1—SPAI––—ZEONRR	1	1×10^{-01}	$1/1\times10^{-02}$	2.05×10^{-13}	$2.05\times10^{-12}/2.05\times10^{-14}$ +900%/–90%
2.	U1_34110V20B———––—ZEMNRR	0.634	1×10^{-01}	$1/1\times10^{-02}$	2.05×10^{-13}	$1.37\times10^{-12}/8.8\times10^{-14}$ +568%/–57%
3.	U2_34110V20B———––—ZEMNRR	0.350	1×10^{-01}	$1/1\times10^{-02}$	2.05×10^{-13}	$8.52\times10^{-13}/1.4\times10^{-13}$ +315%/–32%
4.	U3_34110V20B———––—ZEMNRR	0.014	1×10^{-01}	$1/1\times10^{-02}$	2.05×10^{-13}	$2.32\times10^{-13}/2.02\times10^{-13}$ +14%/–1%

TABLE 67. HRA FOR CASE 3 FOR REL1 MULTIPLE UNIT MODEL

Case No	Event	Fussell–Vesely	Probability (initial)	Modified Probability Increase/Decrease	REL1 (Initial)	Modified REL1 Increase/Decrease
1.	M_U1_REC_COOL_PHTS	0.759	2.5×10^{-02}	$2.5\times10^{-01}/2.5\times10^{-03}$	1.62×10^{-09}	$1.27\times10^{-08}/5.15\times10^{-10}$ +665%/–68%
2.	U0_71400MP1—SPAI——ZEONRR	0.241	1×10^{-01}	$1/1\times10^{-02}$	1.62×10^{-09}	$5.15\times10^{-09}/1.27\times10^{-09}$ +218%/–23%
3.	U1_34110V20B—————ZEMNRR	0.241	1×10^{-01}	$1/1\times10^{-02}$	1.62×10^{-09}	$5.15\times10^{-09}/1.27\times10^{-09}$ +218%/–23%

TABLE 68. HRA FOR CASE 4 FOR REL0 MULTIPLE UNIT MODEL

Case No	Event	Fussell–Vesely	Probability (initial)	Modified Probability Increase/ Decrease	REL0 (Initial)	Modified REL0 Increase/Decrease
1.	U0_71400MP1—SPAI——ZEONRR	0.992	1×10^{-01}	$1/1\times10^{-02}$	3.94×10^{-10}	$3.92\times10^{-09}/4.20\times10^{-11}$ +895%/–89%
2.	U1_34110V20B———–—ZEMNRR	0.992	1×10^{-01}	$1/1\times10^{-02}$	3.94×10^{-10}	$3.92\times10^{-09}/4.20\times10^{-11}$ +895%/–89%
3.	M_U1_RECLOSE_3	0.006	8.15×10^{-03}	$8.15\times10^{-02}/8.15\times10^{-04}$	3.94×10^{-10}	$4.19\times10^{-10}/3.92\times10^{-10}$ +6%/–1%

TABLE 69. RESULTS FROM MUPSA SENSITIVITY ANALYSES BASIC EVENT REL2 CASE

Sequence	Probability of basic event	Basic event code
SEQ1	$3.01\ 10^{-4}$	IEM18_U1_LOCLIII_SD
	$2.50\ 10^{-1}$	M_U1_REC_COOL_PHTS
	9.2010^{-1}	U1_DR04.IS.CLOSED.SD
	$9.89\ 10^{-1}$	U1_DR03.IS.CLOSED.SD
	$5.00\ 10^{-4}$	U1_21601V3——————VXCBIL
	$7.20\ 10^{-1}$	U1_POS9.2
	$1.00\ 10^{-2}$	U1_POS9
	2.5	U1_COAG1

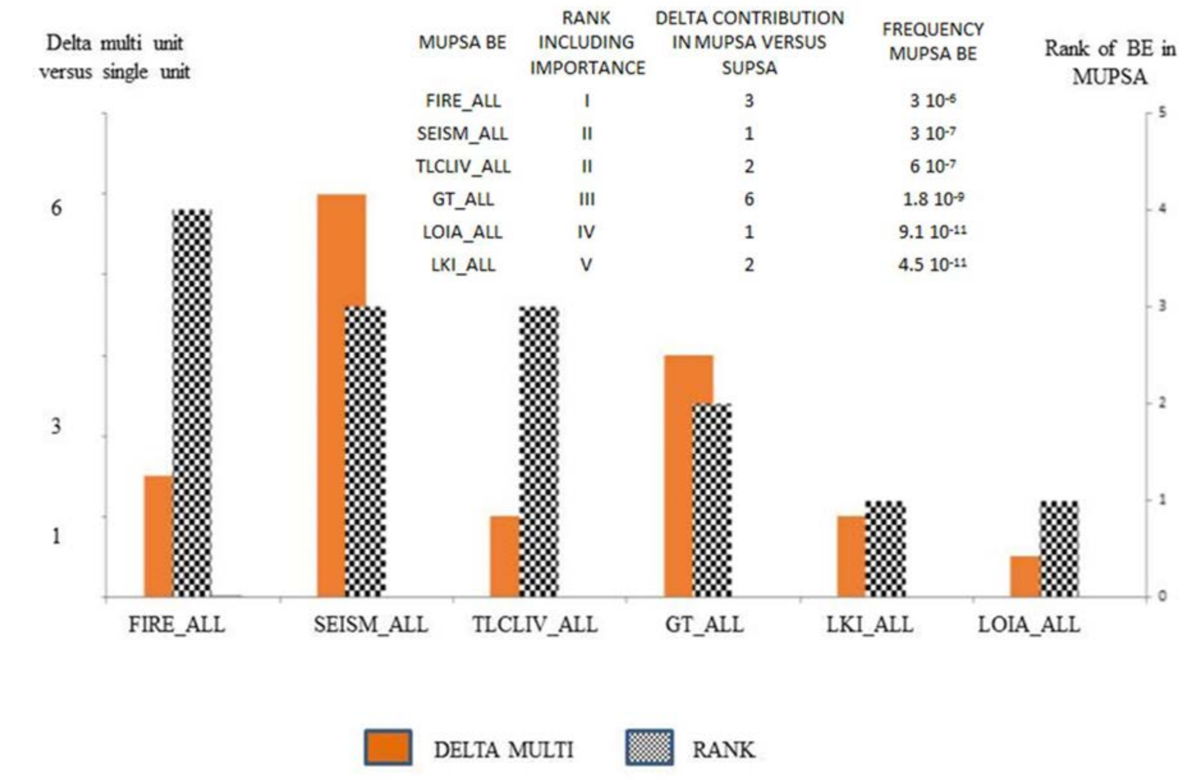


FIG. 98. Sample of MUPSA SA basic event REL2 case.

TABLE 70. RESULTS FROM MUPSA SENSITIVITY ANALYSES BASIC EVENT REL2 CASE

Sequence	Probability of basic event	Basic event code
SEQ1	$3.01\ 10^{-4}$	IEM18_U1_LOCLIII_SD
	$2.50\ 10^{-1}$	M_U1_REC_COOL_PHTS
	9.2010^{-1}	U1_DR04.IS.CLOSED.SD
	$9.89\ 10^{-1}$	U1_DR03.IS.CLOSED.SD
	$5.00\ 10^{-4}$	U1_21601V3——————VXCBIL
	$7.20\ 10^{-1}$	U1_POS9.2
	$1.00\ 10^{-2}$	U1_POS9
	2.5	U1_COAG1

TABLE 71. RESULTS FROM MUPSA SENSITIVITY ANALYSES BASIC EVENT REL1 CASE

Sequence	Probability of basic event	Basic event code
SEQ2	$3.01\ 10^{-3}$	IEM4_U1_LOCLIV_1
	$5.00\ 10^{-1}$	U1_HTS.LOOP1.BROKEN
	$1.00\ 10^{-1}$	U1_34110V20B—————ZEMNRR
	$1.00\ 10^{-1}$	U0_71400MP1—SPAI——ZEONRR
	$6.50\ 10^{-6}$	M_U1_ZHF–C4–425
	$8.00\ 10^{-1}$	U1_POS1
	5	U1_COAG1

Other sensitivity cases are related to the correlation factors for POSs and ageing impact (COAG): sensitivity analyses cases are run by variation of the correlation factors and frequency in a POS impact on the results.

Table 72 shows the results from MUSA sensitivity analysis. The results of basic runs at first iteration and of the severe accident cases lead to a set of issues for the next iteration review.

TABLE 72. EXTRACT ON RESULTS FROM MUPSA SENSITIVITY ANALYSES– POS CORRELATION FACTORS AND AGEING CASE

Coefficient Case 1	**Value**	**Coefficient Case 2**	**Value**	**Coefficient Case 3**	**Value**
POS1	0.9	POS1	0.8	POS1	0.8
POS9	0.1	POS9	0.2	POS9	0.3
COAG1	5	COAG2	7.5	POS1C	0.6
COAG2	2.5	U2_EPS_HE	$1.8\ 10^{-1}$	POS9C	0.4
U1_EPS_HE	$1.8\ 10^{-1}$	U2_FIRE_HE	$1.0\ 10^{-1}$	COAG2	10
U1_FIRE_HE	$1.0\ 10^{-1}$	U2_MKUP_CAL_HE	$2.0\ 10^{-3}$	U2_EPS_HE	$1.8\ 10^{-1}$
U1_MKUP_CAL_HE	$2.0\ 10^{-3}$	U2_ECCS_DOUS_HE	$1.0\ 10^{-1}$	U2_FIRE_HE	$1.0\ 10^{-1}$
U1_ECCS_DOUS_HE	$1.0\ 10^{-1}$	U2_ECCS_U1_EWS_HE	$3.0\ 10^{-5}$	U2_MKUP_CAL_HE	$2.0\ 10^{-3}$
U1_ECCS_U1_EWS_HE	$5.0\ 10^{-5}$	U1_COOL_HE	$3.0\ 10^{-5}$	U2_ECCS_DOUS_HE	$1.0\ 10^{-1}$

The next iteration is performed based on the re-evaluation of the deterministic codes results assumed for some scenarios, as follows:

For the REL2

— High intensity earthquake followed by loss of containment leak tightness during open reactor in shutdown;
— loss of class IV followed by loss of leak tightness and cooling reactor due to human error;
— General transient followed by multi loss of class IV followed by loss of leak tightness, cooling reactor and emergency power;
— Severe site fire followed by loss of reactor cooling and human error to restore it and to assure power supply and fire flex equipment;
— Severe earthquake followed by loss of reactor cooling and human error to restore it and to assure power supply and fire flex equipment.

For the REL1

— High intensity earthquake followed by loss of containment leak tightness during open reactor in shutdown without other human error;
— loss of class IV followed by multi loss of class IV followed by loss of leak tightness, cooling reactor and emergency power with existing recovery possible;
— General transient followed by multi loss of class IV followed by loss of leak tightness, cooling reactor leading to medium release and loss emergency power.

For the REL0

- Medium intensity earthquake followed by loss of containment leak tightness leading to low releases during open reactor in shutdown, LOOP and without other human error recovery of it;
- General transient followed by multi loss of class IV followed by loss of leak tightness and human error to recover it, CCF of class III diesels and consequential loss of class IV and human error to start emergency power;
- Medium intensity earthquake followed by loss of containment leak tightness leading to low releases during open reactor in shutdown and power relay control chatter due to earthquake.

Alternative solutions verifications approaches: CAFTA model case A are verified for solution stability and convergence by evaluating solutions with alternative codes and approaches, as for instance the use of binary digital diagram for the master fault tree and using Lambert Function for sequences ranking checks.

4.3.10.3. Use of the evaluation insights in the decision process

The MUPSA results are included in a decision–making process on safety and licensing and it is connected with other safety related projects, as for instance the technical basis for the emergency plan and the systematic review of critical operator actions in emergencies.

4.3.11. Russian Federation/JSC A

A methodology for Level 2 MUPSA is not yet developed and there is no risk metric established for Level 2 PSA of multiunit site. However, Level 2 PSA was performed for two radiation sources: fuel in the reactor and fuel in spend fuel pool for singe unit. The modelling approach was based on the feature of risk spectrum PSA software and includes the following steps:

- **Step 1**: core damage frequency was quantified for fuel in the reactor. All internal initiation events and internal and external hazards were included in the scope of the assessment for all modes of operation (full power, shutdown for partial refuelling, shutdown for full refuelling accidental hot or cold shutdown for maintenance;
- **Step 2**: MCS obtained from Step 1 stored in CD–R consequence analyses case are used as an input to event trees to initiate events LHR–POND–RP–LOOP and LHR–POND–RP–LOOP (loss of heat removal from SFP) developed for multi sources FDF quantification (Fig. 99). These event trees are constructed based on single source event tree LOOP–POND with the following modification:
    - o End state CD was changed to end state CD–R–P to distinguish that this end state associates with fuel damage in both reactor and spend fuel pool
    - o Functional event POWER–REC (restoration of power) is removed to avoid duplication of recoveries of external power modelled for the reactor and fuel pond.

  The structure of the events is identical, but event tree LHR–POND–RP–LOOP is quantified in conditions of LOOP (boundary condition set LOOP is enabled), where the frequency of fuel melt in the reactor and SFP was assessed.

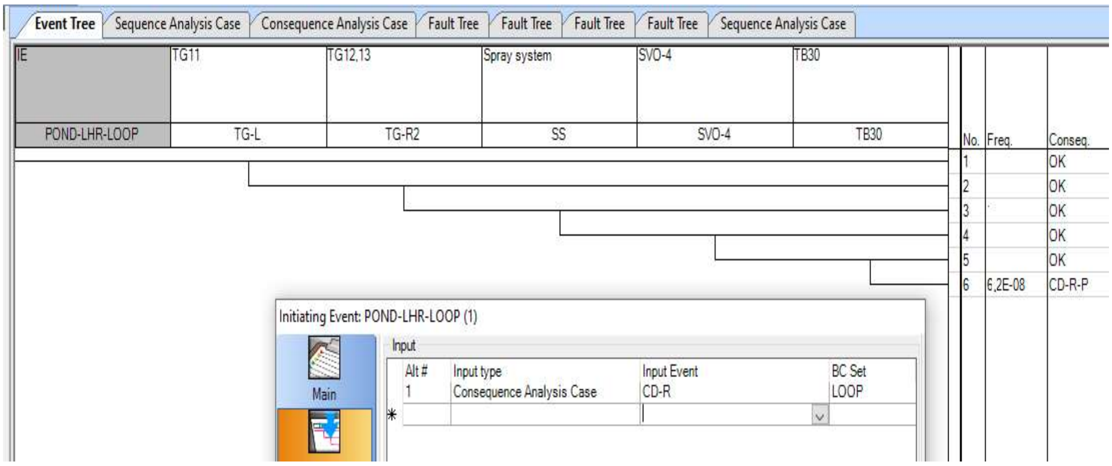


FIG. 99. Example of evet tree for multi source FDF quantification.

— **Step 4**: Steps 1 through 3 allows to obtain LRF for fuel in the reactor and the SFP for Balakovo NPP single unit that start from the accidents that initially affect fuel in the reactor or both fuel in the reactor and SFP. Accidents that initially affect fuel in SFP and may challenge safety of the reactor are possible and present a risk associated with these accidents. Primarily these accidents deal with heavy loads dropped into the SFP and/or leaks from SFP and connected systems. Note that accident that are caused by loss of heat removal from SFP due to heat removal system failures were already accounted for at previous steps. The FDF quantified for all accident with heavy load drops and leaks from SFP appears to be below 10^{-7}/yr. Knowing that neither of the systems used to compensate leak in SFP is needed to remove heat from the reactor and leaks from SFP cannot challenge reactor status it was decided to neglect such accidents in Level–2 PSA scope.

4.3.12. Tunisia/STEG

The technical considerations are specific to each study. However, there are in this moment recommended practices for building the models in MUPSA. They are included in documents under development at national and international (IAEA) levels and they fall under two approaches:

— To build master fault trees for the entire model; or
— To build event trees and to assure their interface with the plant reaction (function events) by carefully considering the correlation (switches) for the elements that do not have to be accounted for in any type of multiunit initiating event. This approach also implies either to evaluate all the possible combinations for failures for the units or to consider one combination as conservative.

Combinations of the two methods is also possible and the individual reports reflect the fact that this methodological aspect of MUPSA is yet under review and testing. Therefore, the information on particular experience is very important. Fig. 100 illustrates the steps included in our analysis and modelling approach.

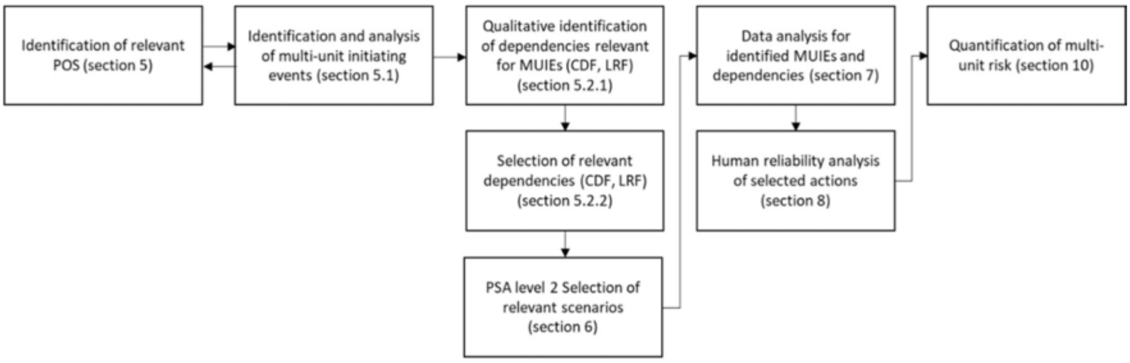


FIG. 100. Overview of the analysis process.

4.3.12.1. General screening principles

Even at the site with only two reactor units, the number of expected multiunit scenarios is anticipated to be large. Thus, introducing and applying qualitative / quantitative screening principles, several scenarios needed to be analysed can be decreased; qualitative screening principles result in classified dependencies, while quantitative screening principles rely on probabilistic criteria to screen out insignificant dependencies. When applied to site PSA, quantitative screening principles are based on risk importance of basic events associated with dependencies in SUPSAs. In this context, a dependency can address a potential multiunit initiating event, a combination of POSs or potential dependency between the systems, structures, components or operator actions. It is a good practice that similar initiating events are grouped together. It was found that reasonable individual screening criterion for Level 1 PSA risk importance can be set to 1×10^{-8}/yr, while for Level 2 PSA to 1×10^{-9}/yr. This is explained as follows: if the contribution of scenarios where dependency is present in SUPSA, is $<1.0\times10^{-8}$/yr for Level 1 PSA and $<1.0\times10^{-9}$/yr for Level 2 PSA, with good confidence can be stated from the site risk perspective that the dependency is not significant. This screening principle can be used to rationalise why in a site PSA it is not meaningful to study combinations of independent initiating events or to justify why there are only few relevant POS combinations worth analysing.

4.3.12.2. Plant operational state impact

In the assessment of multiunit scenarios, the units' various combinations of POSs can be analysed. Different POSs bring different safety systems and recovery actions. Thus, it the approach is to identify relevant site configurations. However, full consideration of all possible combinations of POSs between several units would generate many site POS combinations. Thus, the POSs are merged into fewer POS groups to reduce a number of combinations. Knowing that multiunit scenarios have impact on core cooling and residual heat removal functions, regrouping of POSs is based on configuration of residual heat removal systems. Thus, the analysis of multiunit POSs includes the following steps:

- Evaluate to estimate the time shares of the larger POS groups;
- In the case of loss of residual heat removal in each POS group need to define the time windows for core / fuel damage;
- Exclude POS group with short duration or with very long time window to fuel damage;
- Remove irrelevant combinations; season dependency is related to external hazards with different

likelihoods (e.g. during winter compared to summer season, while longer outages are typically in Nordic NPPs during summer).

4.3.12.3. Identification of relevant initiators

In the existing SUPSAs, the initiating events need to be reviewed to identify those that can affect only one unit and those that impact multiple units concurrently to categorize them as follows:

— Single unit initiating events occur only in one unit and do not affect other units or radioactive sources (except possibly in a later phase of the accident);
— Multiunit initiating events challenge two or more units or radioactive sources concurrently (e.g. seismic events and other external hazards);
— Partial multiunit initiating events occur on a single unit or impact multiple units, depending on the cause; one such example is LOOP potentially affecting a single unit or any combination of units. Events in this category are placed into one of the previous initiating event categories.

Conservatively, partial multiunit events may be considered as multiunit events in thus limiting the scope of analysis. If a single unit event has a potential to propagate, it may be relevant to multiunit events (example: if single unit event is causing a secondary loss of offsite grid or if fire spreads between the units). Additionally, a single unit event through severe accident, may potentially cause an initiating event in other units.

4.3.12.4. Identification and selection of dependencies

For each relevant initiator relevant dependencies need to be identified. The dependencies can be:

— Shared SSCs;
— Identical components (CCFs);
— Spatial dependencies;
— Human and organizational dependencies;
— Simultaneous maintenance;
— State–of–knowledge dependencies.

The dependencies related to shared SSCs, inter–unit CCFs and operator actions might be important for multiunit analysis. If the initiating event has a potential to spread to another unit (such as for example fire) or if the accident sequence causes damage affecting the adjacent unit/s, then the dependencies through spatial interactions are considered as relevant. Simultaneous maintenance is likely to be screened out (such as for example simultaneous a scheduled maintenance).

Dependency is created by the epistemic uncertainty in the estimation of events probabilities. An example is phenomenological events, such as a steam explosion, as considered in the Level 2 PSA. If two identical reactor units are under same severe accident conditions, same probabilities are applied for phenomenological events reflecting both the epistemic uncertainty and randomness of such an event. The identification is suggested to be carried out in two steps, qualitative screening, and selection of dependencies. In the qualitative screening, the importance of multiunit dependencies relevant for identified initiators is ranked qualitatively.

The dependencies are ranked as ‘very important’, ‘important’, ‘less important’ and ‘insignificant,’ as summarized in Table 73, to:

- Ensure that considered dependencies are likely to be relevant are captured correctly in the quantitative analysis;
- Screen out dependencies that do not require further analysis.

TABLE 73. IMPORTANCE CATEGORIES FOR QUALITATIVE IDENTIFICATION OF DEPENDECIES

Category	Description of Dependencies
Very important	No additional SSCs are available to cope with initiating event (shared water intake)
Important	Limited number of additional SSCs is available to cope with initiating event (diesel generators at a site with SBO gas turbine system)
Less important	Number of additional SSCs is available to cope with initiating event (shared fire water system)
Insignificant	No risk for core damage or a radioactive release (shared domestic water system)

Dependencies ranked as ‘very important’ or ‘important’ are relevant when selecting dependencies for further analysis with quantitative evaluation to select those important for quantitative screening when all dependencies are identified as ‘very important’, ‘important’ or ‘less important’ and analysed with respect to representative basic event(s) making it possible to quantitatively evaluate dependencies.

4.3.13. Ukraine/ENERGORISK

To evaluate multiunit LRF occurring simultaneously at several reactors, existent Level 2 SUPSA models available for ZNPP units 1–6 has been adopted and modified similarly to approach used for MUPSA Level 1. Since separate PSA models for each ZNPP unit use different component reliability and CCF data bases, it was decided to use one harmonized and actual statistical data (reliability parameters, initiating event frequency) for all models. As well, PSA models has been modified to use unique names for basic events and top events at PSA for each unit. Approaches used for SUPSAs on development bridge trees to define PDSs, and on development large CETs and small decomposition fault trees have been retained for MUPSA. Event tree technique to integrate bridge trees/CET for different units and to consider different unit combinations is similar to the Level 1 MUPSA.

Multiunit effects were modelled using linked event tree approach, where PDSs and release categories are linked for several NPP units in such way that the final states of one tree for the first considered unit became input for sequences for other units. Examples are shown in Figs. 101 and 102.

LRF due to damage of the fuel in reactor core and SFP at single ZNPP unit was also evaluated. Since for VVER–1000 SFP is located within the reactor building, accidents at SFP can have impact on the reactor and vice versa. Flowchart for evaluation of combined reactor–SFP LRF is shown on Fig. 103.

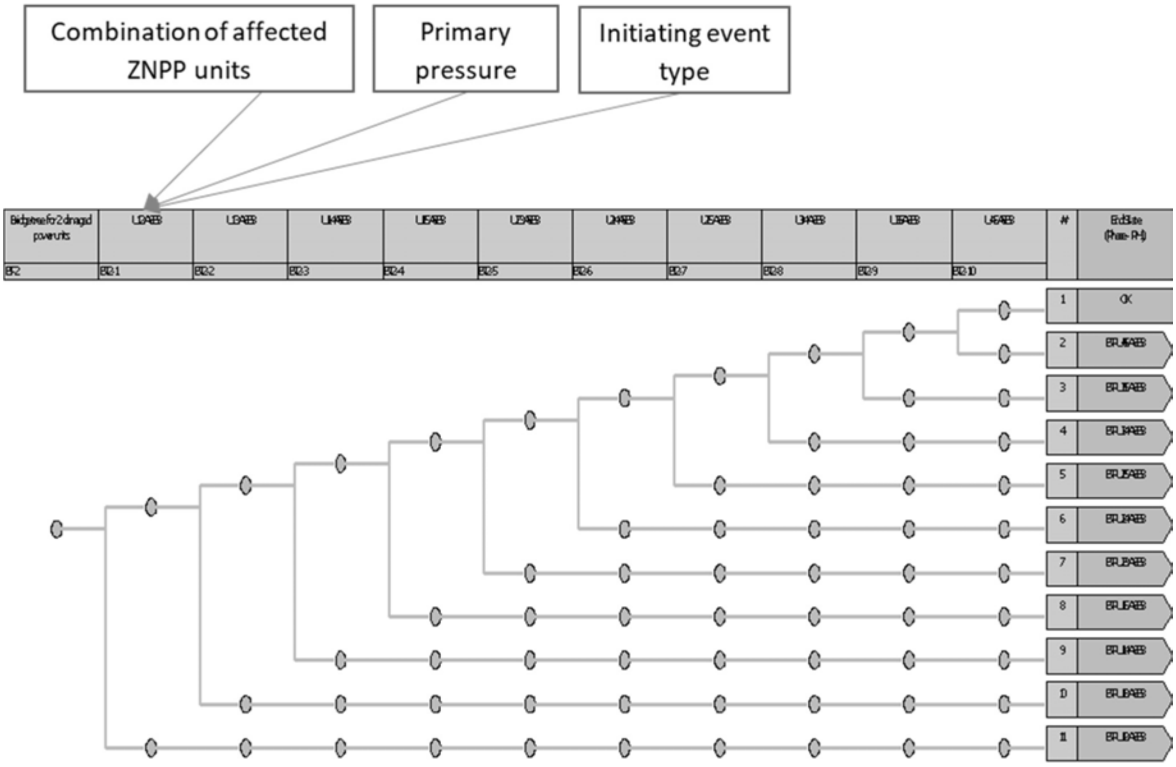


FIG. 101. Bridge tree for two ZNPP units combination.

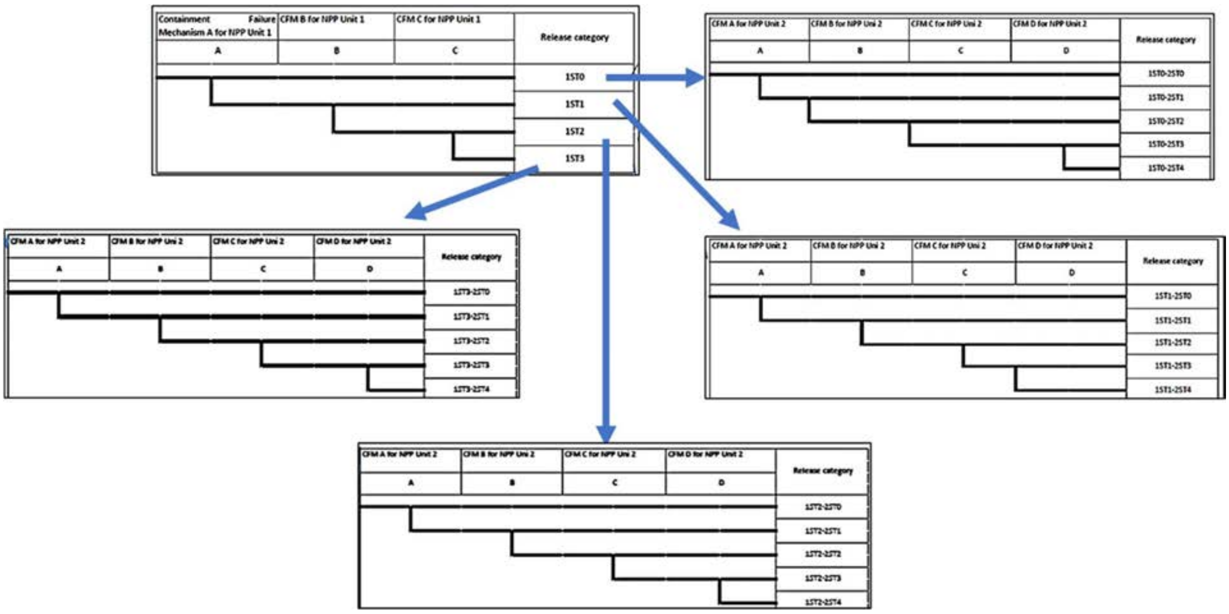


FIG. 102. Example for CETs for two units combination.

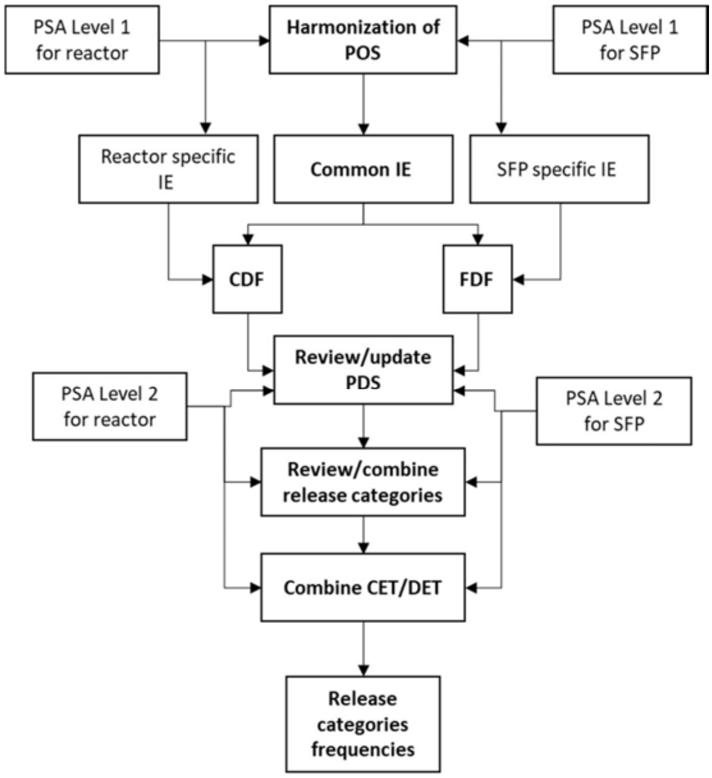


FIG. 103. Flowchart for evaluation of combined reactor SFP LRF.

The following aspects are considered: mutual impact of the reactor and SFP; differences in accident progression timing in the reactor core and the pool; and preventing double counting of accident sequences and SSC failures. Mutual impact of the reactor and SFP can include:

a) Impact from shared SSC between the reactor and SFP:

- ESWS (supply of cooling water to SFP cooling system, to ECCS heat exchangers, diesel generators, other consumers significant for accident progression and mitigation);
- emergency core cooling system (to provide coolant to the reactor vessel during accidents; SFP emergency makeup using water from the sump tank of ECCS systems);
- spray system (containment cooling as well SFP emergency makeup);
- other shared systems (power supply, ventilation).

b) Impact of accidents in the reactor on SFP: e.g., containment isolation due to LOCA, and necessity to re–open of pneumatic valves for SFP cooling system;
c) Impact of accidents in SFP on the reactor: e.g., flooding of SSC due to the SFP system piping rupture; accidents in SFP may lead to conditions which require emergency NPP shutdown by MCR staff (to be taken into account during frequency calculation of events which may lead to scram);
d) Impact of simultaneous accidents, e.g. generation of additional quantity of hydrogen in SFP;
e) Consideration of accident management for the reactor and SFP: e.g., to arrange containment

venting it is necessary to consider that pressure decrease may lead to more intensive water boiling in SFP, to increased evaporation of SFP water and, therefore, to reduced time window to the beginning of uncovering and heating of spent nuclear fuel.

During development of common reactor–SFP probabilistic model several integration aspects need to be appropriately considered, including: prevention of double counting of initiating events and SSC failures (mainly by using adequate naming scheme for basic events representing initiating event frequencies and components failures); harmonization of POSs for reactor and SFP developed under PSAs for single radiation sources; combination of release categories for reactor and SFP; combined progression of initiating event and/or severe accidents. Harmonization of POSs for reactor and SFP is performed as part of preparatory actions under PSA Level 1. It is needed to ensure consistency between timing of POSs and associated POS–specific frequency to provide additivity. Under harmonization, POSs (and their duration) for SFP have been correlated with POSs for reactor. It is also needed to review identification of initiating events to determine reactor–specific, SFP specific and common initiating events. Common initiating event means event that can simultaneously impact both reactor and SFP, or event that occurs at one radiation source but can influence on accident progression at another radiation source. For example, LOCA will lead to containment localization and to necessity to open localization valves to provide SFP cooling. As result of preparatory activities, CDF and FDF need to be updated. It can be noted, that for ZNPP MUPSA study, CDF was not changed due to consideration of SFP impact, while FDF was slightly increased due to impact of reactor accidents on SFP operation, Table 74.

TABLE 74. HARMONIZATION OF PLANT OPERATIONAL STATES

Initiating event	**Containment state**															
	Containment gates sealed							**Containment min gate open**				**Containment gates sealed**				
	POSs for reactor															
	POS0	**POS1**	**POS2**	**POS3**	**POS4**	**POS5**	**POS6**	**POS7**	**POS8**	**POS9**	**POS10**	**POS11**	**POS12**	**POS13**	**POS14**	**POS15**
	POSs for SFP															
	POS2							**POS1**				**POS2**				
Loss of ESW	×	×	×	×	×	×	×							×	×	×
SFP Loss of ESW	×	×	×	×	×	×	×	×	×	×	×	×	×	×	×	×

New event trees were developed for initiating events that are common for reactor and SFP (LOOP, loss of ESWS) to represent in one event tree all accident sequences leading to loss of service water to reactor consumers, to SFP consumers and to both. As a result of CDF and FDF frequencies re–quantification, CDF was not changed while FDF was slightly increased due to impact of reactor accidents on SFP operation. Review/development of the release categories is based primarily on a severe accident scenario, which includes: availability of systems involved in mitigating a severe accident, severe accident phenomena, human actions on severe accident management, recovery actions. It was assumed that if during severe accident progression (CET accident sequences) the same SSCs are used both for reactor and SFP, the same phenomena of severe accident are considered and the same human actions are performed, then with some extent the final states of CET accident sequences will also be similar. This is only modelling assumption with some uncertainty since the time characteristics of accident processes in the reactor and SFP differ significantly, and, therefore, both the isotopic composition and the mass of fission products released outside containment will be different. Based on this assumption was defined combined reactor–SFP release categories (see Table 75). Combination of release categories for reactor and SFP was performed using comparison of severe accident phenomena at reactor, SFP and containment, based on analyses available from individual PSA Level 2 for reactor and SFP.

TABLE 75. COMBINED RELEASE CATEGORIES

Release category for reactor		**Integrated LRF**
ST0	No containment failure, containment leaks	Reactor/SFP
ST1	Containment early failure, core melt within reactor vessel, failed spray system	Reactor/SFP
ST2	Containment early failure, core melt within reactor, spray system is operable	Reactor/SFP
ST3	Containment early failure, core melt retained in reactor shaft, failed spray system	Reactor/SFP
ST4	Containment early failure, core melt through reactor shaft, failed spray system	Reactor/SFP
ST5	Intake containment dome, early core melt through reactor shaft, failed spray system	Reactor/SFP
ST6	Containment late failure, core melt retained in reactor shaft, failed spray system	Reactor/SFP
ST7	Containment late failure, failed spray system	Reactor/SFP
ST8	Intake containment with late core melt through reactor shaft, failed spray system	Reactor/SFP
ST9	Containment bypass (primary–to–secondary leaks), at open secondary steam dump valves	Reactor
ST10	Containment bypass (primary–to–secondary leaks), at secondary steam dump valves operation	Reactor
Release category for SFP		
ST0	No containment failure, containment leaks	n/a
ST11	Containment failure due to hydrogen explosion	n/a
ST12	Release due to open containment	n/a

To develop common set of release categories, as well to develop CETs comprehensive (or complex) consideration of severe accident processes is needed. Such consideration was based on existent severe accident analysis and analytical justifications performed to support SAMG for reactor and SAMG for SFP. It was done by mapping of important points of accident progression on one timeline for visualization and better understanding (see Fig. 104). Conservative approach was used to identify end states of common accident progression and to find time when severe accident progression and phenomena for reactor covers SFP severe accidents. Such analysis are done for each accident. As a result, combined CET for reactor/SFP were constructed and combined reactor–SFP LRF was quantified.

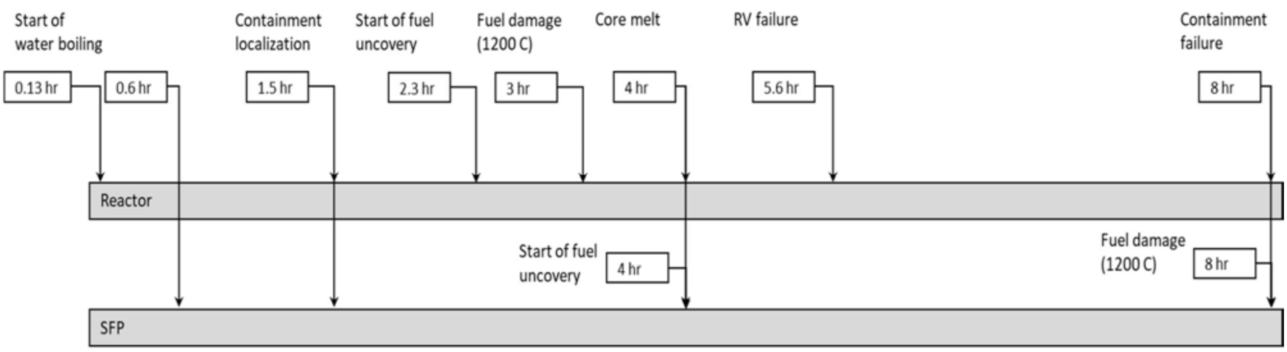


FIG. 104. Combined timeline for accidents at reactor and SFP.

4.3.14. Ukraine/ SSTC NRS

The existing PSA–2 model for internal initiating events during reactor full power operation at unit 1 was corrected for further combination of the unit 1 and 2 models and estimation of the multiunit early release frequency. The codes of basic events that modelled radioactive release categories RC1S – RC7 in PSA–2 model for RNPP–1 was extended with unit identifier. As a result, the codes of the specified events obtained the following view U1–Release_Category. PSA–2 models were combined through the integration of the event trees of the PSA–2 model for RNPP–2 into PSA–2 model for RNPP–1. The integration was implemented using built–in means of SAPHIRE 8 computer code applying the 'Integrate Project' function. The example CETs that combined radioactive release categories for RNPP–1, 2 for initiating event T8 were developed. The description of the developed CET is provided in Fig. 105 and Fig. 106.

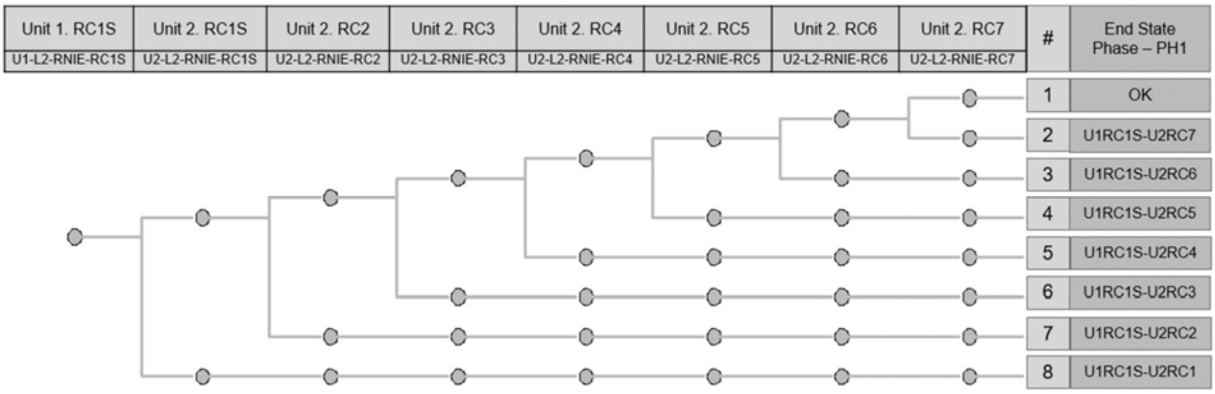


FIG. 105. Event tree U1–L2–RNIE–RC1S.

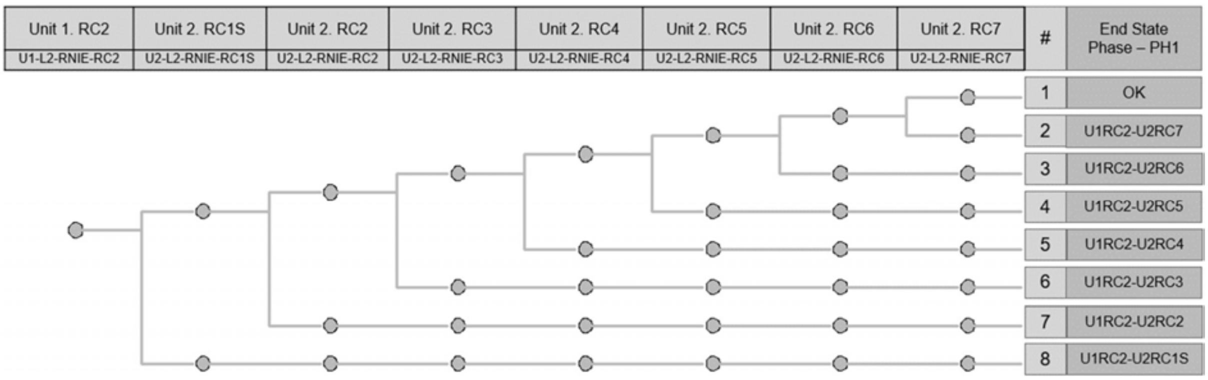


FIG. 106. Event tree U1–L2–RNIE–RC2.

HRA: SAMGs, dependencies from external organizations (e.g., TSC could be common for all units), L1–L2 PSA dependencies, applicability of HRA methods to multiunit context, timing considerations, impact from affected unit (e.g., radiation impact).

Release categories: the results of the analysis of possible initiating event T8 progression show that several categories of radioactive release are possible, which are described in Table 76.

TABLE 76. RADIOACTIVE RELEASE CATEGORIES FOR INITIATING EVENT T8

Release category	Description
RC1	Release through design containment leakages for severe accident progression with operating spray system.
RC1S	Release through design containment leakages for severe accident progression with failed spray system.
RC2	Late containment failure (containment basemat melt–through).
RC3	Late containment failure with operating spray system.
RC4	Late containment failure with failed spray system.
RC5	Early containment failure/loss of localization with operating spray system.
RC6	Early containment failure/loss of localization with failed spray system.
RC7	Containment bypass resulting from the primary to secondary breaks.

The RC1 category, which represented release through design containment leakages with operating spray system was not considered in the early release frequency evaluation. Based on this assumption, the matrix of radioactive releases, presented in Table 77, is developed. This matrix combines radioactive release categories RC1S, RC2 – RC7 of RNPP–1, 2.

TABLE 77. MATRIX OF MULTIUNIT RADIOACTIVE RELEASE CATEGORIES FOR RNPP 1 AND 2[9]

Release categories		Unit 2						
		U2–RC1S	U2–RC2	U2–RC3	U2–RC4	U2–RC5	U2–RC6	
Unit 1	U1–RC1S	U1–RC1S_U2–RC1S	U1–RC1S_U2–RC2	U1–RC1S_U2–RC3	U1–RC1S_U2–RC4	U1–RC1S_U2–RC5	U1–RC1S_U2–RC6	U1–RC1S_U2–RC7
	U1–RC2	U1–RC2_U2–RC1S	U1–RC2_U2–RC2	U1–RC2_U2–RC3	U1–RC2_U2–RC4	U1–RC2_U2–RC5	U1–RC2_U2–RC6	U1–RC2_U2–RC7
	U1–RC3	U1–RC3_U2–RC1S	U1–RC3_U2–RC2	U1–RC3_U2–RC3	U1–RC3_U2–RC4	U1–RC3_U2–RC5	U1–RC3_U2–RC6	U1–RC3_U2–RC7
	U1–RC4	U1–RC4_U2–RC1S	U1–RC4_U2–RC2	U1–RC4_U2–RC3	U1–RC4_U2–RC4	U1–RC4_U2–RC5	U1–RC4_U2–RC6	U1–RC4_U2–RC7
	U1–RC5	U1–RC5_U2–RC1S	U1–RC5_U2–RC2	U1–RC5_U2–RC3	U1–RC5_U2–RC4	U1–RC5_U2–RC5	U1–RC5_U2–RC6	U1–RC5_U2–RC7
	U1–RC6	U1–RC6_U2–RC1S	U1–RC6_U2–RC2	U1–RC6_U2–RC3	U1–RC6_U2–RC4	U1–RC6_U2–RC5	U1–RC6_U2–RC6	U1–RC6_U2–RC7
	U1–RC7	U1–RC7_U2–RC1S	U1–RC7_U2–RC2	U1–RC7_U2–RC3	U1–RC7_U2–RC4	U1–RC7_U2–RC5	U1–RC7_U2–RC6	U1–RC7_U2–RC7

[9] The radioactive release categories shown in the table combine the categories of radioactive releases RC1S, RC2, RC3, RC4, RC5, RC6, RC7 for RNPP-1, 2. For example, category U1-RC5_U2-RC6 describes a SA scenario with an early containment failure/loss of localization of unit 1 with operating spray system and an early containment failure/loss of localization of unit 2 with failed spray system. The description of the radioactive release categories is provided in Table 59.

4.4. RESULTS OVERVIEW

The main assumptions and technology specific dependencies are related to the adopted risk metrics, the specifics of the model, and how the dependency analyses were performed, as well as the time frame and the release categories for which the Level 2 progression calculations were performed.

Independent initiators may potentially lead to multiunit core damage; however, in case when the sequences are completely independent, the probability of multiple core damage becomes negligible. Scenarios with two independent initiators are insignificant from the perspective of multiunit risk development; thus, they are generally screened out from further analysis. The probability of simultaneous initiating events can be studied by analysing for example two independent initiators that are occurring within 72 hr (based on the assumption that NPP is in a stable state after 72 hr). The highest frequency for a generic single unit initiating event is typically in the order of 1.0×10^{-1}/yr at power, while the frequency for a second initiating event occurring in another unit within 72 hr is already fairly low. The frequency of the events' combination leading to an undesired end state would also require a failure of safety systems. The results generated by the participants correspond to the selected set of dominant sequences for single unit and for multiunit sequences, with sample quantified results in each case and a comparison (ratio) of the contribution from multiunit consequences versus the single unit cases. The specific aspects that all participants were asked to consider, and report were:

- For the dependency analysis it is provided a type of dependency considered relevant in the specific reported study, which could be the significant shared SSCs, the inter–unit CCF, the consideration of the hazard fragilities and/or HRA. For the relevant aspect considered the report includes more details on the models adopted;
- In the time frame column, the information of significance for the benchmark is the value of this timeframe and the indication on how the dynamics of the plant change during this time frame was considered (by detailed deterministic calculations from the emergency plans, or from Level 3 for a different configuration of the plant beyond the time mission considered in MUPSA Level 1, for instance of 24, 48 or 72 hr);
- Important aspect to be mentioned is the manner of developing the MUPSA model (in fault tree or event tree approaches and/or combined ones);
- Results on the releases (by isotopes and for time durations) are crucial for the understanding;
- Sequences describe in their turn the dominant scenarios leading to single unit type of impact or of multiunit impact (of cascading and/or common site impact). In these categories the cascading description is understood in the sense of a series of consequential initiators from one unit to another, not mandatory of the same type and magnitude (a core damage started in one unit may lead to inhabitability of a control room and releases in a second unit, and they could induce a blackout of the next unit for instance);
- There could be more dominant/representative sequences considered for each group and results are presented as a ratio of multiunit impact versus single unit impact.

For example, in a dependency case where the most important aspects are HRA and shared flex equipment in multiunit case the HRA information and management of flex are reviewed based on independent plant specific studies on critical operator actions and the operator actions are modelled considering those results. For the input to the study in this case a set of evaluations as resulted from emergency operating procedures technical database on the plant configuration after more than three days is performed by reviewing the MCS from the sequences (to adapt them for the case of unavailability of some more functions beyond 3 days, as assumed in MUPSA level1. Definition of a release category (large, small and very small, including other sources than the reactor – tritium removal facility for instance) is modelled in a fault tree. However, study may provide modelling and/or independent peer review with another approach (event tree and/or combined or even binary digital diagram type). As a result of those assumptions and technical specific aspects, a set of results is obtained, containing dominant sequences for single unit (LOCA for large releases) and/or of multiunit type (cascading effect of a set of blackouts due to units' common connection to a switch yard for small and very small releases) and/or external hazards (earthquake for small and large releases).

Argentina/CNEA: the MUPSA approach was applied to a case study for a hypothetical multiunit site (two SMR units and a shared SFP). Modelling features for Level 1, Level 2 and Partial Level 3 MUPSA were analysed. It was considered the LOOP as IEM. The proposed site and MUPSA risk metrics have been calculated. The lessons learned can be summarized in the following points:

- Need to extend the evaluation time to 48 hr, given the posed strategy to cope with the initiating event, a first stage based on passive safety systems, and a second one with active systems.
- MUPSA Level 2 event trees, based on CET considering all sources (unit 1, 2 and SFP in this case) can be large. It is considered that the main challenge is to model dependences among radioactive sources. To reduce its complexity, assumptions and simplifying hypothesis on phenomenological aspects, headers and consequences are made;
- Deterministic characterization of multiunit release categories, based on the combination of release category from different radioactive sources, represents a challenge to analyze;
- Partial Level 3 development effort is low in comparison with the obtained results. The Partial Level 3 calculations can be developed by software and they do imply low calculation resources;
- The proposed IRR as site and MUPSA risk metric is a metric established from the point of view of the public, regarding their radiological risk from the site. It is no focused on plant characterization, like CDF or early release category frequency. It captures the strengths and weaknesses of NPPs. It can evaluate in an integral way consequences for members of the public and frequencies of release category and multiunit release category occurrences. The IRR evaluates all range of consequences. Then, it allows identify scenarios that can implies higher risk for individuals of public, due to higher exposure frequencies and lower consequences.
- Level 1 and Level 2 MUPSA development can be a power tool for emergency management for events that can affect more than one radioactive source in the site.

Canada/COG: the Canadian PSAs have always been MUPSA in the sense that they explicitly account for multiunit interactions, even though PSA results are expressed on a per–unit, per–hazard basis. Developing the methodology for a whole site PSA allowed to gain new perspective on the issue of whole site risk and the role of whole site PSA. It also allowed to shed light on relative contributions of purely single vs. multiunit risks across the site and on the relative risk of different hazards from a site perspective.

China/INET: since HTR–PM does not apply CDF and LERF as risk metrics, it adopts an integrated modelling framework, starting from initiating events and ending with release categories as well as dose estimates, i.e., Level 1, Level 2 and part of Level 3 are performed with an overall modelling approach of event tree and fault tree linking. Shared SSCs are explicitly modelled in the event trees and fault trees. Both inter–reactor and intra–reactor CCFs are considered.

Release categories are concluded at the end of event tree branches. Each release category is then analysed to determine its source term and dose estimates. Event tree and fault tree models are developed by using RiskSpectrum PSA software. Dose consequence assessment is done by the software ARCAT which is developed by INET for the purpose of multiple source releases. The technical issues related to modelling are basically the same as those of Level 1.

Hungary/NUBIKI: it was foreseen from the beginning of this research effort that risk could not be quantified and, consequently, the results could not be interpreted for all the hazards within the framework of the CRP due to some unresolved issues, existing uncertainties, unknowns, and time and resource limitations. For benchmark purposes, risk has been preliminarily quantified for LOOP at power operation of units 1 and 2. The LOOP induced single as well as multiunit risk results relevant to Level 1 and Level 2 PSA are given in Table 78 (it is noted that the Level 1 PSA results differ slightly from the results presented earlier due to some refinements performed in the model recently). To help avoid misinterpreting the results from the table it is noted that SCDF and SLERF do not equal to the mathematical summation of all the other calculated CDF based or LERF based risk metrics.

TABLE 78. MULTIUNIT LEVEL 1 AND LEVEL 2 RISK RESULTS FOR LOOP INITIATING EVENT

Risk Metrics	**Frequency (1/yr)**	**Ratio to SCDF or SLERF**
CDF (core damage on at least unit 1)	4.95×10^{-07}	54%
Multiunit CDF (core damage on both units)	6.47×10^{-08}	7%
Single unit CDF (core damage only on unit 1)	4.30×10^{-07}	46%
SCDF (Single unit CDF1+Single unit CDF2+ Multiunit CDF)	9.25×10^{-07}	100%
LERF (LER on at least unit 1)	6.28×10^{-09}	56%
Multiunit LERF (LER on both units)	1.39×10^{-09}	12%
Single unit LERF (LER only on unit 1)	4.89×10^{-09}	44%
SLERF (Single unit–LERF1+Single unit–LERF2+Multiunit–LERF)	1.12×10^{-08}	100%

Moreover, the following ratios were also quantified:

— LERF/CDF = 1.27%
— Multiunit LERF/ multiunit CDF = 2.15%
— SLERF/SCDF = 1.21%

The LOOP event was selected as the subject of the limited scope Level 2 MUPSA of the Paks NPP, because:

— LOOP is generally a consequence of several external hazards such as seismic events or extreme weather conditions (wind, snow, temperature, etc.);
— LOOP affects all the reactors simultaneously at the Paks site.

This pilot study did not only establish a methodology on how to perform a Level 2 MUPSA for the LOOP event at the Paks site, but also pointed out some weaknesses and strengths of the Paks NPP in such situations. Some lessons learned are highlighted hereby that can be utilized later when performing a full scope Level 2 MUPSA for the Paks NPP. An integrated PSA model was developed for the two units. The SUPSA event trees were modified first to establish a good basis for the MUPSA model. Subsequently, these event trees were linked together. It appeared a straightforward task to perform modifications in the PSA model of the second unit, since both units were designed as identical in terms of severe accident management. It can be concluded that the single unit deterministic severe accident calculations of the Paks NPP can in principle be used for the purposes of Level 2 MUPSA for the plant. The timeline of the key events as well as the time available to perform operator actions (including recoveries) have paramount importance in the Level 2 MUPSA. The availability of appropriate severe accident analyses are essential to accurately model the complexity of multiunit accidents as well as the radiological impact of a severe accident at one unit on operator interventions and accident management at the other unit. The single unit CETs can be used directly in the MUPSA, if the multiunit specific features of an accident are already taken into account in the single unit CETs (i.e. in the availability of severe accident management equipment as well as in the HEPs).

The interpretation of the results presented above does not seem fully sufficient to characterize multiunit releases in terms of the release magnitude. The same risk metric (LERF) is the basis of Level 2 SUPSA and MUPSA, since there is no distinction between single unit LERF and multiunit LERF in terms of release magnitude and timing. We assume that a Level 3 MUPSA could characterize appropriately the substantial difference between single unit and multiunit risk with respect to the likelihood of environmental and health consequences of various degrees. Shared systems provide some additional redundancy for each unit of the Paks NPP; however, the analysis revealed that not all systems or human resources will be sufficient to mitigate an accident after a multiunit accident initiator. It was pointed out that multiunit risk assessment needs to be carried out in an integrated manner rather than evaluating each unit separately.

India/BARC, following are some of the observations made during Level 2 MUPSA analysis:

— Risk aggregation is done by considering all the releases from various units in the site.
— Risk metrics such as SCDF can be evaluated if the releases are considered only from the nuclear reactors. If the site also includes units other than nuclear reactors (for example spent fuel facility) it is better to evaluate SRF considering all the releases from all the facilities.
— Risk results can be well represented with frequency and consequence curve. Hence, the work is under progress to predict the dose at the plant boundary due to the releases from various facilities in the site (partial Level 3).

Republic of Korea/KHNP, from Level 2 MUPSA, KHNP has identified the following lessons learned and insights:

— Site LERF for multiunit LOOP was estimated at ~5% of SCDF for multiunit LOOP.
— The portion of multiunit LERF for multiunit LOOP to site LERF was identified as 0.2%, which is ten times lower than that of multiunit CDF and, multiunit LERF from any unit combination of two out of nine units has almost all contribution to total multiunit LERF for multiunit LOOP, which is in a range of 10^{-9}/site–yr.
— Based on these insights, KHNP could consider that multiunit risk for multiunit LOOP is insignificant.
— As for seismic events, we considered four kinds of seismic PGA bins; bin 1 (0.1g~0.2g), bin 2 (0.2g ~ 0.3g), bin 3 (0.3g ~ 0.5g) and bin 4 (0.5g ~ 1.0g). In Level 1 MUPSA, the contribution of Bin 4 to the total seismic multiunit CDF was estimated about 70%, and that of Bin 3 was about 26%. As

a result of performing Level 2 seismic MUPSA, the contribution of Bin 4 was about 93%, and that of bin 3 was about 6%.

— Multiunit risk for seismic events is not sensitive to inter–unit dependencies. multiunit CDF and LERF are dependent on seismic induced failures of each SSC in multiunit.

— In Level 2 PSA perspective, the only concern about multiunit risk was identified as seismic events, especially for bin 4 (0.5g~1.0g). The uncertainty of PSHA (Probabilistic Seismic Hazard Analysis) is, however, much larger than any other uncertainty sources in PSA. Moreover, the uncertainty of bin 4, in which multiunit LERF is totally dependent on the mean value of PSHA, is tremendously increased. Therefore, it is important to understand the results of multiunit risk with considering the large uncertainty.

Republic of Korea/Hanyang University, most of core damage frequency comes from core damage of only one plant regardless of the initiating event. Table 79 and Fig. 107 show that the frequency of no containment failure, SLERF, and Non SLERF occupies 79.9%, 4.8%, and 15.3% of the core damage sequences, respectively.

TABLE 79. PRELIMINARY RESULT OF 9 UNIT LEVEL 2 PSA FOR THE INTERNAL EVENT

	Number of units where core damage occurred													
	Single unit initiating event		**CCI**										**Site**	
	1 unit		**1 unit**		**2 units**		**3 units**		**4 units**		**5 or more units**			
Containment failure mode	CDF/yr	%	CDF/yr	%	CDF/yr	%	CDF/yr	%	CDF/yr	%	CDF/yr	%	CDF/yr	%
No failure	1.2×10^{-4}	57.5	4.6×10^{-5}	22.2	8.6×10^{-7}	0.4	2.5×10^{-9}	<0.1	0	0.0	0	0.0	1.7×10^{-4}	79.9
No SLERF	2.2×10^{-5}	10.7	8.5×10^{-6}	4.1	5.8×10^{-7}	0.3	4.1×10^{-9}	<0.1	2×10^{-12}	<0.1	0	0.0	3.1×10^{-5}	15.3
SLERF	7.2×10^{-6}	3.5	2.7×10^{-6}	1.3	8.7×10^{-8}	0.0	3×10^{-10}	<0.1	0	0.0	0	0.0	1.0×10^{-5}	4.8
Total	1.5×10^{-4}	71.7	5.7×10^{-5}	27.6	1.5×10^{-6}	0.7	7.0×10^{-9}	<0.1	2×10^{-12}	<0.1	0	0.0	2.1×10^{-4}	100

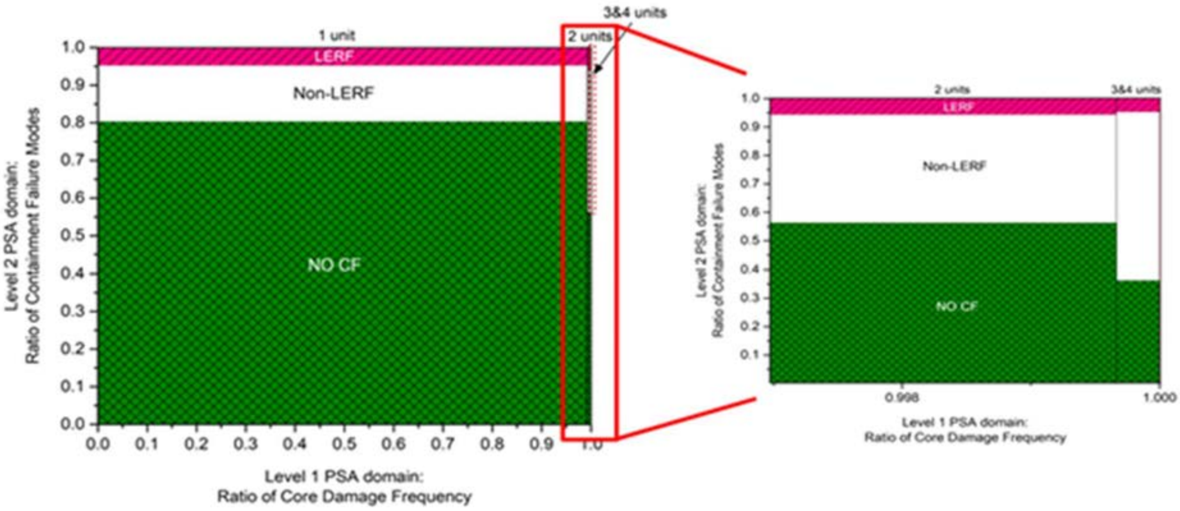


FIG. 107. Preliminary result of 9 unit Level 2 PSA for the internal event (LERF means site LERF).

Romania/CNCAN, in summary:

- PSA Level 2 is of significant relevance for multiunit, multisource site impact. However, it has to be in accordance with the imposed regulatory metrics;
- It is meaningful to use the initial PSA Level 2 MUPSA modelling based on SUPSA for screening purposes to identify the major expected contributors to site metrics;
- PSA Level 2 requires extensive sensitivity analyses and consideration of iterative evaluations using deterministic codes of Level 2 and the inputs from SAMG and EP;
- The use of diverse codes confirms that a modelling approach, in our case fault tree for MUPSA model starting from SUPSA and consider the barrier to challenges as defined in SUPSA is useful and leads to convergent results. However, the strengths of a code (fault tree in CAFTA versus event tree in RiskSpectrum) may be used for sample crosschecks of results;
- In the ranking of contributors, the use both of frequencies and importance is very useful, as well as independent review by using a different approach, like binary digital diagramss;
- Construction of the model was used for young team members training and for preparing applications to plant actions and activities (SAMG, emergency planning).

Russian Federation/JSC A*,* multiunit Level 2 PSA was not performed in Russia due to extremely low frequency of multiunit core damage. Instead, multisource Level 2 PSA has been performed. The major insights from this study were as follow:

- Results obtained show relatively high LRF associated with simultaneous damage of fuel in the reactor and SFP. LRF is only 1.5 times lower than safety goal established in Russian regulation; however, it is more than 80 times lower that LRF from single sources of radioactivity.
- Majority of accident with simultaneous damage of fuel in the reactor and SFP directly leads to Large Release. The reason for this is that SFP damage occurs only when spray system fails and thus containment integrity is always challenged.
- Main contribution to LRF from simultaneous damage of fuel in the reactor and SFP provides two types of accidents: accident with total loss of power and accidents with ventilation system failures. The latest leads to failure of power sources and frontline systems located in the compartment cooled by these systems.
- When the term ‘simultaneous’ is used it does not mean ‘close in time’. Due to high amount of water damage in spent duel pool occurs 50–70 hr later than in the reactor. However, Russian regulations require consideration of doses for population during 10 days after any accident and thus even occurred with significant time differences such accident still led to Large Release (according to definition of large release in Russian regulations). For other countries, where regulations do not require long term assessment of off site consequences, multisource Level 2 PSA may not be needed.

Tunisia/STE,*:* multiunit risk assessment is based on the lookup approach to connect frequency data and consequence data with the objective to not to perform a detailed Level 3 MUPSA but rather to focus on developing a new approach to facilitate Level 3 MUPSA. Four representative multiunit initiating events are considered: multiunit LOOP, multiunit LOUHS, multiunit TSUNAMI (multiunit tsunami induced event), and multiunit SEISMIC–0.3 g/0.5 g/0.7 g/0.9 g/1.1 g (multiunit seismic induced event by various magnitudes). An inter–unit seismic correlation coefficient of 0.3 is used in the Level 1 MUPSA. Due to restrictions of the used PSA tool, only 10,000 frequency data could be connected to consequence data. All scenarios in multiunit LOOP, multiunit LOUHS, and multiunit–single unit–0.3 are covered by scenarios numbered below 10,000. The percentage distributions of CDF (core damage frequency), EFR, and LCFR for multiunit accident are summarized in Table 80. It was found that EFR and LCFR are distributed consistently with CDF. The single unit accident takes the largest proportion in case of multiunit LOOP, multiunit LOUHS, and multiunit–single unit–0.3.

TABLE 80. DISTRIBUTION OF CDF, EFR, AND LCFR FOR MULTIUNIT LOOP, MULTIUNIT LOUHS, AND MULTIUNIT SEISMIC–0.3 G FROM THE RESULTS OF LEVEL 3 MUPSA

Initiating Event	**Multiunit LOOP**			**Multiunit LOUHS**			**Multiunit SEISMIC–0.3 g**		
Cores damaged /Total	**CDF (%)**	**EFR (%)**	**LCFR (%)**	**CDF (%)**	**EFR (%)**	**LCFR (%)**	**CDF (%)**	**EFR (%)**	**LCFR (%)**
1/3	92.1%	87.9%	87.9%	98.0%	94.7%	95.6%	96.8%	92.9%	93.9%
2/3	7.1%	10.7%	10.8%	98.0%	94.7%	95.6%	96.8%	92.9%	93.9%
3/3	0.8%	0.4%	0.3%	1.6%	3.9%	3.2%	3.1%	6.9%	5.9%
Total	100.0%	100.0%	100.0%	0.0%	0.0%	0.0%	0.0%	0.0%	0.0%

Comparison of CDF distributions between full coverage case and the cut off case is provided in Table 81. It can be seen that full coverage case and cut off case show quite different CDF distributions implying that including an adequate amount of multiunit accident scenarios in the Level 3 MUPSA is not optional but rather mandatory to estimate more comprehensive risk of multiunit accidents.

TABLE 81. COMPARISON OF CDF DISTRIBUTION BETWEEN FULL COVERAGE AND CUT OFF CASES

Initiating event	**Multiunit TSUNAMI**		**Multiunit SEISMIC 0.5 g**		**Multiunit SEISMIC 0.7 g**		**Multiunit SEISMIC 0.9 g**		**Multiunit SEISMIC 1.1 g**	
# of Scenarios	Total > 1,048,562	Cut off below 10,000th	Total 290,379	Cut off below 10,000th	Total > 1,048,562	Cut off below 10,000th	Total 579,590	Cut off below 10,000th	Total 257,467	Cut off below 10,000th
Cores damaged /Total	CDF (%)	CDF (%)	CDF (%)	CDF (%)	CDF (%)	CDF (%)	CDF (%)	CDF (%)	CDF (%)	CDF (%)
1/3	41.6%	73.1%	47.1%	49.2%	62%	56.4%	48.7%	38.7%	38.7%	38.7%
2/3	3.3%	1.1%	34.8%	36.0%	24.2%	12.0%	10.2%	10.2%	10.2%	10.2%
3/3	55.1%	25.8%	18.1%	14.8%	13.8%	31.6%	41.1%	51.1%	51.1%	%51.1
Total	100%	100%	100%	100%	100%	100%	100%	100%	100%	100%

In this study, the starting time of release was categorized into early and late release and therefore, the resulting lookup table has two dimensions. In future studies, the release category may be divided into early, intermediate, and late release categories, or even more. It is expected that lookup tables with higher dimensions will have diminished uncertainties in thus providing more reliable consequence results. In consequence analysis, the long term exposures such as ingestion are not considered; the focus was on the exposure during specified emergency phase (one week). Long term exposure needs to be included in developing updated lookup tables, after developing long term exposure models such as for example, a food chain model.

Ukraine/Energorisk, from the results obtained the following insights can be drawn:

— Development of integrated MUPSA Level 2 model requires a comprehensive set of deterministic safety assessments;
— Integrated reactor/SFP LRF on 2 % lesser that sum of individual LRF for reactor and SFP for considered initiating events;
— It can be predicted that for full scope PSA integrated LRF would be 20–30% lesser that sum of individual LRFs;
— Multiunit LRF ≤ 1% of total LRF for NPP sites with low inter–unit dependencies, truncation value is to be significantly decreased, comparing to SUPSA.

Ukraine/SSTC NRS, the quantitative assessment was performed by means of probabilistic code SAPHIRE 8 through calculating the frequency of accident scenarios leading to the radioactive release categories described in Table 76. The multiunit early release frequency was calculated as the sum of categories U1–RC1S_U2–RC1S – U1–RC7_U2–RC7. The separation degree of the MCS is equal to 1×10^{-30}.

According to the calculation results, the multiunit early release frequency for initiating event T8 ‘Loss of the ESWS’ is $3{,}330\times10^{-16}$ 1/year. The dominant radioactive categories are U1–RC5_U2–RC6 и U1–RC6_U2–RC6. The U1–RC5_U2–RC6 frequency is 2.2×10^{-16} while the U1–RC6_U2–RC6 frequency is 1.1×10^{-16}. According to the results of PSA–2 quantitative assessment for initiating event T8, the early release frequency for RNPP– 1, 2 is 2.2×10^{-07} (for each unit). Table 82 contains the results of the multiunit early release frequency estimation for initiating event T8 ‘Loss of the ESWS’.

TABLE 82. PSA–2 RESULTS FOR INTERNAL INITIATING EVENT T8

Dependency Analysis		**PSA Model Approach**	**Unit–specific PSA risk metric**	**Single unit Results**	**MUPSA Results**
Shared SSCs and support systems	Failures of shared SSCs in fault trees.	Integrated event tree end state calculations.	LRF	SLRF 2.2×10^{-07}/yr	multiunit LRF 3.3×10^{-16}/yr

5. LESSONS LEARNED

While individual participants’ insights gained, and lessons learned, are summarized in some detail in Sections 3.5.2. and 4.4.1 for the MUPSA Level 1 and Level 2 analyses, respectively, below are some of the most important insights and lessons learned that are common to many of the participants.

Since Level 1 MUPSA event trees can be very large, a simplified approach, based on only a few initiators that are considered potentially significant for MUPSA and grouping of systems in headers is required for a practicable event tree approach. However, seismic hazards need to be included in the scope of MUPSA and would benefit from a common approach for seismic fragility correlations. With respect to the analysis effort and utilization of software, MUPSA Level 2 requires extensive sensitivity analyses and iterative evaluations using deterministic codes along with inputs from SAMG. The use of diverse codes can confirm that the chosen modelling approach, e.g. fault tree for MUPSA starting from SUPSA, is appropriate and can lead to convergent results. Also, the strengths of a code (fault tree in CAFTA versus event tree in RiskSpectrum) may be used for crosschecks of results.

Multiunit CDF is in most cases driven by inter units CCFs for similar equipment and by external hazards disabling common equipment at the site for sites with low inter–unit dependencies. But even for innovative reactor sites, such as a multi-module HTR, common initiator and CCFs dominate the releases involving multiple modules Therefore, the importance of CCFs and the large uncertainties in inter–unit CCF and large SSC groups requires detailed justification of inter–unit CCF parameters. Most analyses used conservative inter–unit CCF probabilities due to lack of multiunit data. Dependencies analyses among radiological sources was also identified as a key task for MUPSA development. Hence, the results could be made more realistic by collecting sufficient multiunit data on CCF probabilities and dependencies. On the other hand, some participants reported that multiunit risk for seismic events is not sensitive to inter–unit dependencies but rather that multiunit CDF and LERF are dependent on seismic induced failures of each SSC in multiple units.

In some benchmarks, human errors were identified as a significant contribution to CDF, e.g. at units that are in forced shutdown due to an event at another unit on the same site. HRA needs to consider the size of crews on site and the human system interfaces implemented, especially for new reactors. Performance shaping factors used for SUPSA may not be directly adopted for MUPSA and HRA methods need to be adapted with the objective of calculating probabilities of HFE.

In terms of the results obtained, all participants found that multiunit CDF is well below 10% of the total CDF for all sites, and around 1% for sites with low inter–unit dependencies. For large sites with more than to or three units, the two unit CDF is moderately important, but the frequency of core damage at more units simultaneously is in most cases negligible. Multiunit LRF also was found to be generally ≤ 1% of SLRF for NPP sites with low inter−unit dependencies; therefore, truncation values need to be significantly decreased in MUPSA compared to SUPSA.

Multiunit module and single unit module releases are physically similar in terms of source term category, but multi module risks contribute generally less than 5% to the total risk with respect to frequency and have a stepwise effect with respect to the consequence part of risk.

6. CONCLUSIONS

The CRP on Probabilistic Safety Assessment Benchmarks for Multiunit Multi Reactor Sites brought together experts from IAEA Member States with multi unit NPP sites to utilize, test and further develop their MUPSA analyses, and identify and discuss main risk contributors and specific safety related insights on multiunit risk, by conducting and comparing results of MUPSA benchmark exercises. Developing the methodology for a whole site PSA allowed participants to gain new perspectives on the issue of whole site risk and the role of MUPSA. It shed light on relative contributions of single vs. multiunit risks across the site and on the relative risk of different hazards from a site perspective.

Level 1 and Level 2 MUPSA development can also be a powerful tool for emergency management of events that can affect more than one radioactive source on the site and to enable and facilitate a (partial) Level 3 MUPSA covering multiunit or multi source accident scenarios with practical modelling effort. Participants with innovative reactor sites, such as SMRs and modular HTRs felt that it necessary to extend MUPSA into Level 2+, or partial Level 3, realm in their benchmarks. While the same Level 2 risk metric (LERF) is the basis of Level 2 SUPSA and MU PSA (since there is no distinction between single unit LERF and multiple unit LERF in terms of release magnitude and timing), a multiunit (partial) Level 3 PSA could characterize appropriately the potentially substantial difference between single unit and multiunit risk with respect to the likelihood of environmental and health consequences.

For benchmark purposes, results could generally not be generated or interpreted for all the hazards and/or all scenarios, within the framework of the CRP, due to unresolved issues, existing uncertainties, unknowns, and time and resource limitations. Therefore, risk is preliminarily quantified for limited hazards or initiating events (e.g. the obvious multi-unit LOOP at power operation of all units). Then the calculated single unit and multiunit risk results relevant to Level 1 and Level 2 PSA, and in a few cases partial Level 3 (or Level 2+), were evaluated and compared on a relative basis.

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ABBREVIATIONS

AAC	Alternate alternating current
AHWR	Advanced heavy water reactor
CANDU	Canada deuterium uranium
CCDP	Conditional core damage probability
CCF	Common cause failure
CCFG	Common cause failure group
CDF	Core damage frequency
CDS	Core damage state
CET	Containment event tree
CHASNUPP	Chashma NPP
COG	CANDU owners’ group
CRP	Coordinated research project
CTRF	Tritium removal facility
CVS	Containment venting system
DEC	Design extension condition
DICA	Dry spent fuel storage
DiD	Defense in depth
EDG	Emergency diesel generator
ECCS	Emergency core cooling system
EPRC	External plant release category
ESWS	Essential service water system
EWSS	External water supply system
FDF	Fuel damage frequency
GIS	Gas insulated switchgear
HEP	Human error probability
HFE	Human failure events
HRA	Human reliability analysis
HTGR	High temperature gas cooled reactor
HTR–PM	High temperature gas cooled reactor pebble–bed module
IEM	Initiating event for multiple units
IES	Initiating event for single unit
IFB	Irradiated fuel bay
IRR	Individual radiological risk
ISFS	Intermediate spent fuel storage
LOCA	Loss of coolant accident
LOOP	Loss of offsite power

LOUHS	Loss of ultimate heat sink
LER	Large early release
LERF	Large early release frequency
LRF	Large release frequency
MCR	Main control room
MCS	Minimum cut sets
MUPSA	Multi–unit PSA
NPP	Nuclear Power Plant
NSSS	Nuclear steam supply system
OBE	Operating basis earthquake
PARS	Passive autocatalytic hydrogen recombiner system
PDS	Plant damage state
PHWR	Pressurized heavy water reactor
PNGS	Pickering nuclear generating station
POS	Plant operating state
PRA	Probabilistic risk Assessment
PSA	Probabilistic safety assessment
PSHA	Probabilistic seismic hazard assessment
PWR	Pressurized water reactor
RCF	Release category frequencyy
RNPP	Rivne NPP
RPV	Reactor pressure vessel
RPVECS	RPV external cooling system
SAMG	Severe accident management guidelines
SBO	Station blackout
SCDF	Site core damage frequency
SDC	Shutdown cooling
SFB	Spent fuel bay
SFF	Spent fuel facility
SFP	Spent fuel pool
SiRF	Site release frequency
SLERF	Site LERF
SMR	Small modular reactor
SPAR	Standardized plant analysis risk
SSCs	Structures, systems, and components
SSE	Safe shutdown earthquake
SRA	Site risk assessment

SRCF	Site RCF
STC	Source term category
STEG	Tunisian Company of Electricity and Gas
STCG	Source term category group
SUNPP	South Ukraine NPP
SUPSA	Single unit PSA
SWS	Service water system
THERP	Technique for human error prediction
TSC	Technical Support Centre
UFDS	Used fuel dry storage
VVER	Water water energetic reactor
ZNPP	Zaporizhzhya NPP

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