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Seismic resilience assessment of Small Modular Reactors by a Three-loop Monte Carlo Simulation

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ABSTRACT

We develop a three-loop Monte Carlo Simulation (MCS) framework for the seismic resilience assessment of Small Modular Reactors (SMRs), embedding Probabilistic Seismic Hazard Analysis (PSHA), seismic fragility evaluation and multiple SMR units accident sequence analysis. A set of metrics are computed to capture different aspects of SMR resilience to earthquakes, specifically the ability to withstand seismic disruption, mitigate consequences and restore normal operation. The MCS framework allows accounting for the aleatory and epistemic uncertainties of the PSHA and fragility parameters. An application is given with regards to an advanced Nuclear Power Plant (aNPP) consisting of four reactor units of NuScale SMR design. A comparison is made to a conventional NPP (cNPP), i.e., a typical large reactor of equivalent generation capacity. Both plants are fictitiously located on the Garigliano nuclear site (southern Italy). The results show that resilient features of SMRs overcome cNPPs in terms of post-accident scenario mitigation and restoration capabilities.

1. Introduction

The increased frequency and severity of extreme natural events calls for reliable, safe and resilient energy systems (Schaeffer et al., 2012; Perera et al., 2020; Di Maio et al., 2021). This is of great concern also for the nuclear power industry, due to the potentially catastrophic consequences of accidents (Kemeny, 1979; Lipscy et al., 2013; Eddy and Sase, 2015) and the requirements on post-accident response (Ahn et al., 2017; Funabashi and Kitazawa, 2012).

To address this concern Small Modular Reactor (SMR) designs have been introduced to benefit from small size, modularity, and novel inherent (e.g., integral reactor vessel layout) and passive (e.g., natural circulation of primary coolant) safety characteristics (Di Maio et al., 2022). These advanced Nuclear Power Plants (aNPPs) can meet the set of functional requirements for resilience, intended to ensure that they are little vulnerable to and readily recoverable from Natural hazard triggered Technological (NaTech) accidents, such as Loss Of Offsite Power (LOOP) and Loss Of Coolant Accidents (LOCAs).

This paper presents the development of an approach to quantitatively assess the resilience of SMRs to natural hazards, in general, and to earthquakes, specifically. This is done by embedding Probabilistic Seismic Hazard Analysis (PSHA), seismic fragility evaluation and multiple SMR units accident sequence analysis, i.e., a full Seismic Probabilistic Risk Assessment (SPRA), into a three-loop Monte Carlo Simulation (MCS) framework. Metrics are computed that capture different aspects of system resilience to earthquakes: the ability to withstand a seismic disruption, mitigate its consequences and restore normal operation. Uncertainty associated with the parameters of PSHA and the fragility models is also propagated (Baker, 2013; Park et al., 1998).

SPRA has been widely used to compute the frequency of reactor core damage and release of radiation from nuclear plants (IAEA, 2021; Choi et al., 2021). Initially based on ground motion fragility curves (Commission, 1983; Pickard, 1981; Smith, 1981; Cornell, 1968), SPRA eventually became independent of seismic hazard and included correlations in responses-damages directly in the risk assessment through the use of (i) response- rather than ground-motion-based fragility models and (ii) MCS (possibly multi-stage with benefits on computational efficiency (Choi et al., 2021) to determine damage states of components (Huang et al., 2011). Following the methodology set out in Huang et al. (Huang et al., 2011), in Kumar et al. (Kumar et al., 2017) and Yawson and Lombardi (Yawson and Lombardi, 2018) SPRAs for seismically isolated (i.e., independent) nuclear facilities located at eight sites across the United States, and a nuclear reactor in a hypothetical rock site in the United Kingdom are presented, respectively. In Zhou et al. (Zhou et al.,

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Nomenclature

Acronyms

aNPP	advanced Nuclear Power Plant
cNPP	conventional Nuclear Power Plant
DG	Diesel Generator
DHRS	Decay Heat Removal System
ECCS	Emergency Core Cooling System
GTST-dMLD	Goal Tree Success Tree-dynamic Master Logic Diagram
G-R	Gutemberg-Richter
LOCA	Loss Of Coolant Accident
LOOP	Loss Of Offsite Power
MCS	Monte Carlo Simulation
MUCC	Multi-Unit Correction Coefficient
NaTech	Natural hazard triggered Technological
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
PSHA	Probabilistic Seismic Hazard Analysis
PZR	PressuriZeR
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
RPV	Reactor Pressure Vessel
RRV	Reactor Recirculation Valve
RSV	Reactor Safety Valve
RVV	Reactor Vent Valve
SG	Steam Generator
SGTF	Steam Generator Tube Failure
SIP	Safety Injection Pump
SMR	Small Modular Reactor
SPRA	Seismic Probabilistic Risk Assessment

List of symbols

$A_{m,\gamma}$	Median acceleration capacity of component γ in the fragility model of Eq. (6)
$A^*_{m,\gamma}$	Median acceleration capacity of component γ in the fragility model of Eq. (7)
b	Parameter of the Gutemberg-Richter recurrence laws
c_γ	Random number from $U[0,1)$ for seismic fragility evaluation of component γ in multiple SMR units
d_{epi}	Distance of the earthquake epicenter from the site of interest
E	Disruptive event
E_{DHRS}	Failure event of both trains of the DHRS
E_{ECCS}	Failure event of the ECCS
E_{LOCA}	Event of Loss Of Coolant Accident within the RCS
E_{LOOP}	Event of Loss Of Offsite Power
E_{RHRS}	Failure event of the RHRS
E_{SGTF}	Event of Steam Generator Tube Failure
$Exp(\bullet)$	Exponential distribution
$F_M(\bullet)$	Cumulative distribution function of the earthquake magnitude M
$F_{Ra}(\bullet)$	Cumulative distribution function of d_{epi} (radius Ra)
$Fr_{\gamma,PGA}$	Fragility of component γ for a given PGA
f_γ	Random number from $U[0,1)$ for seismic fragility evaluation of component γ in one SMR unit
k	Index of the MCS outer loop
l	Dimension of the vector t_{GR}
lb	Lower bound
$MTTR$	Mean Time To Repair
$MUCC_{\gamma,PGA}$	Multi Unit Correction Coefficient of component γ for a given PGA
M	Earthquake magnitude
m	Earthquake magnitude value
m_{max}	Upper bound of M
m_{min}	Lower bound of M
$N(\bullet)$	Normal distribution
N_S	Sample size of the MCS middle loop
N_U	Sample size of the MCS outer loop
n_M	Counter associated to p_M
n_R	Counter associated to p_R
n_W	Counter associated to p_W
PGA	Peak ground acceleration
p_M	Probability of mitigating the accident consequences
p_R	Probability of restoring the system
p_W	Probability of withstanding the earthquake shock
Q	Confidence level of not exceeding $Fr_{\gamma,PGA}$
Ra	Radius of circular area of the area source model
R_0	Set of system states of permanent damage
S_e	Counter of seismic events
T	Designed lifetime of the nuclear power plant
T_{lim}	Prescribed period for restoration
t_{GR}	Vector of earthquake recurrence times
t_R	Restoration time
t_S	Earthquake occurrence time
$U[0,1)$	Standard uniform distribution
ub	Upper bound
X	Discrete random indicator variable of the system state
$X_{\bullet,aNPP}$	Discrete random indicator variable of the aNPP state
X_{cNPP}	Discrete random indicator variable of the cNPP state
X_0	State of nominal performance of the nuclear power plants
$X_{\gamma,\bullet}$	Discrete random indicator variable of the state of component γ
$\beta_{a,\gamma}$	Logarithmic standard deviation of $A^*_{m,\gamma}$ due to aleatory uncertainty
$\beta_{e,\gamma}$	Logarithmic standard deviation of $A^*_{m,\gamma}$ due to epistemic uncertainty
β_γ	Logarithmic standard deviation of $A_{m,\gamma}$ due to total uncertainty
γ	Index of a component of the nuclear plant
λ_m	Annual rate of earthquakes with magnitude greater than m
σ_b	Standard deviation of b
$\Phi(\bullet)$	Standard Gaussian cumulative distribution function
Γ	Number of seismically uncorrelated components of the nuclear power plant
$<\bullet>$	Mean value

2018), an improved SPRA is proposed for multi-unit sites, where unit-to-unit dependencies are considered based on a combination of copulas, importance sampling and parallel MCS.

On the other hand, resilience assessment in the nuclear industry is still limited and not fully explored, especially for advanced reactor concepts (Commission, 1975; Park et al., 2013). Few examples of application exist for conventional Nuclear Power Plants (cNPPs). In Ferrario and Zio (Ferrario and Zio, 2014); a Goal Tree Success Tree–dynamic Master Logic Diagram (GTST-dMLD) and MCS are combined for seismic resilience assessment. In Du et al. (Du et al., 2019) and Zeng et al. (Zeng et al., 2021), a resilience modeling and analysis framework is based on a Markov reward process. In Santhosh and Patelli (Santhosh and Patelli, 2020) and Estrada-Lugo et al. (Estrada-Lugo et al.), resilience assessments of the reactivity control system and the main heat transport system of a new generation large reactor are carried out using Bayesian and Credal networks, respectively. In Yan and

Dunnett (Yan and Dunnett, 2022), the resilience of a single unit Pressurized Heavy Water Reactor (PHWR) exposed to external events is assessed using Petri net modeling.

To the authors knowledge, this work is the first attempt to develop a resilience assessment framework for SMRs. The proposed framework will be exemplified with respect to the seismic resilience of multiple SMR units of Pressurized Water Reactor (PWR) NuScale design. The SMR units are assumed to be located in the nuclear site of Garigliano (southern Italy) and a comparison is made with a single, large PWR unit of equivalent generation capacity.

The remainder of the paper is organized as follows: in Section 2, the specifics of SPRA for SMRs are discussed; in Section 3, the metrics and the MCS framework for seismic resilience assessment are presented; in Section 4, the case study is worked out and the results discussed; in Section 5, conclusions are drawn.

2. Seismic probabilistic risk assessment of small modular reactors

SPRA consists of (Budnitz et al., 1997):

1) Seismic hazard analysis to estimate the probability of occurrence of different levels of earthquake ground motion at the (nuclear) site of interest;
2) Seismic fragility evaluation to quantify the conditional probability of failure of components for any ground motion level;
3) Accident sequence analysis to combine the hazard and fragility analyses outcomes to evaluate the impact of probable earthquakes on the site of interest.

Seismic hazard analysis is specific on the site where the nuclear power plant is located. It is developed as PSHA that consists of (Baker, 2013; Cornell, 1968):

1. The characterization of the stochasticity of the earthquake magnitude in terms of the rate at which earthquakes of various magnitudes are expected to occur. Specifically, the *Gutemberg-Richter recurrence law* is used to describe λ_m, which is the annual rate of occurrence of earthquakes with magnitude M larger than m (Baker, 2013):

$$\log_{10}\lambda_m = a - b \bullet m \tag{1}$$

where a and b are constants estimated based on historical observations and represent the overall rate of earthquakes in a region and the relative ratio of small and large magnitude earthquakes, respectively (typically, b is expressed in terms of mean $\bar{b}$ and standard deviation σ_b as $\bar{b} \pm \sigma_b$ (de Santis et al., 2011).

Eq. (1) implies that M follows the cumulative exponential distribution:

$$F_M(m) = 1 - 10^{-b\bullet m} \tag{2}$$

This distribution is commonly modified into a *bounded Gutemberg-Richter recurrence law* (Baker, 2013), to realistically bound the magnitude values to a finite size of earthquake sources [m_{min}, m_{max}]:

$$F_M(m) = \frac{1 - 10^{-b(m-m_{min})}}{1 - 10^{-b(m_{max}-m_{min})}} \tag{3}$$

2. The characterization of the distribution of the source-to-site distance associated with potential earthquakes. Generally, the distribution of the distance between the earthquake epicenter and the nuclear site of interest is modeled assuming that the earthquakes occur randomly with equal likelihood anywhere within a circular area of radius Ra, so that the probability of an epicenter being located at a distance less than d_{epi} is equal to the area of a circle of radius d_{epi} divided by the area of a circle of radius Ra, and the probability of an epicenter being located at a distance larger than Ra is zero, giving the cumulative distribution function of d_{epi} (Baker, 2013):

$$F_{Ra}(d_{epi}) = \begin{cases} \dfrac{d_{epi}^2}{Ra^2} & d_{epi} \le Ra \\ 0 & d_{epi} > Ra \end{cases} \tag{4}$$

3. The estimation of the resulting ground motion levels as a function of earthquake magnitude, distance, etc, typically with reference to the peak ground acceleration PGA (Ambraseys et al., 2005):

$$\log_{10}PGA = a_1 + a_2 m + (a_3 + a_4 m) \bullet \log_{10}\sqrt{d_{epi}^2 + a_5^2} + a_6 S_S + a_7 S_A + a_8 F_N + a_9 F_T + a_{10} F_O \tag{5}$$

where m and d_{epi} are the earthquake magnitude value and epicenter distance, a_i, $i = 1, \cdots, 10$, are constants, S_S and S_A indicate the types of soil (i.e., soft, still or rock) and F_N, F_T and F_O describe the faulting mechanism (i.e., normal, thrust or odd).

Seismic fragility evaluation depends on the specificities of the nuclear power plant located on the site considered. Typically, it is evaluated as (Cover et al.):

$$Fr_{\gamma,PGA} = \Phi\left[\frac{\log\left(\frac{PGA}{A_{m,\gamma}}\right)}{\beta_\gamma}\right] \tag{6}$$

where $Fr_{\gamma,PGA}$ is a standard Gaussian cumulative probability distribution (Φ) of failure of the γ-th component of the plant conditional to ground motion level (i.e., PGA), $A_{m,\gamma}$ and β_γ are the γ-th component fragility parameters (i.e., the median acceleration capacity, corresponding to the PGA that implies $Fr_{\gamma,PGA} = 0.5$, and its logarithmic standard deviation due to uncertainty). In our analysis of SMRs, to account for the limited functional knowledge of design-phase SMRs:

1) We assume the lognormal fragility model of Eq. (7) where, following (Kim et al., 2011), we decompose β_γ into an aleatory component $\beta_{a,\gamma}$ (that defines the slope of the cumulative distribution) and an epistemic component $\beta_{e,\gamma}$ (that describes the uncertainty of $A_{m,\gamma}$ along the PGA axis by the term of Q, the confidence level of not exceeding $Fr_{\gamma,PGA}$), i.e., $\beta_\gamma = \sqrt{\beta_{a,\gamma}^2 + \beta_{e,\gamma}^2}$ (Park et al., 1998).

$$Fr_{\gamma,PGA} = \Phi\left[\frac{\log\left(\frac{PGA}{A^*_{m,\gamma}}\right)}{\beta_{a,\gamma}}\right] \tag{7}$$

where $A^*_{m,\gamma} = A_{m,\gamma} \pm \beta_{e,\gamma} \bullet \Phi^{-1}(Q)$.

2) A 100% seismic correlation is conservatively assumed for components of the same type, located at the nuclear site in the same building with the same elevation, e.g., redundant components. For example, being V_1, V_2 and V_3 three redundant safety valves, a single fragility model $Fr_{V(3),PGA}$ is considered for the three of them, rather than one different fragility model for each of them $Fr_{V_1,PGA}$, $Fr_{V_2,PGA}$ and $Fr_{V_3,PGA}$.

Accident sequence analysis is performed with logic models, such as event trees that define the accident sequences triggered by seismic-induced initiators, linked with fault trees that describe the basic events that might lead to components and system failures (IAEA, 2021). In the case of SMRs, one must account for the fact that multiple reactor units are located on the same site (Alrammah, 2022). In this work, we then adopt a multi-unit logic model (US Nuclear Regulatory Commission, 2020) to extend the single unit logic model by correcting the basic events probabilities to account for the likelihood of extension to multiple units. The corrections are made by introducing Multi-Unit Correction Coefficients (*MUCCs*) (US Nuclear Regulatory Commission, 2020), for estimating the probability that if a failure occurs in the γ-th component of a unit (1U) ($X_{\gamma,1U} = 1$, where $X_{\gamma,\bullet}$ is the state variable of the γ-th component and '1' indicates the failure state), following an earthquake of a given PGA, this will also occur in the same γ-th component of the multiple SMR units (MU) on the site ($X_{\gamma,MU} = 1$): $MUCC_{\gamma,PGA} = P(X_{\gamma,MU} = 1 | X_{\gamma,1U} = 1, PGA)$. Given that earthquakes, by nature, affect multiple units simultaneously and that the conditional probability of inducing damage in one unit, as well as in the other units, increases with PGA (US Nuclear Regulatory Commission, 2020), for the SPRA of SMRs considered later in the case study we assume $MUCCs$ to rapidly grow as PGA increases.

3. The resilience metrics and Monte Carlo simulation framework

Three metrics are considered in this work for measuring the resilience of nuclear power plants (Section 3.1) and a three-loop MCS framework is designed for their evaluation within the seismic resilience assessment of SMRs (Section 3.2).

3.1. Resilience metrics

Resilience is agreed to represent the ability of a system to withstand, absorb and recover from an accident or disruptive event (Zio, 2018; Hosseini et al., 2016). Withstanding is the ability of a system to resist to disruptions without degrading performance. A system with high withstanding capability is one capable to operate at nominal performance also when the disruptive event occurs, with no need of restoration. Absorption is the ability of a system to resist the impact of disruptions without suffering permanent damages, so that it can be restored to nominal performance after. Recovery is the ability of a system to be brought back to normal performance, within required time limits. In this paper, we use three metrics to quantitatively describe the above characteristics of resilience of a system (Linkov et al., 2014; Yodo and Wang, 2016):

1) *Probability of withstanding.* The conditional probability $p_W = P(X = X_0|E)$ that the (discrete) random indicator variable X of the system state remains at the nominal performance X_0, given that a disruptive event E has occurred;
2) *Probability of mitigating.* The conditional probability $p_M = P(X \notin R_0|E, X \neq X_0)$ that system state X does not reach the set R_0 of the states of permanent damage from which the system cannot be restored, given that earthquake E has occurred and has led to a system state X below X_0;
3) *Probability of restoring.* The conditional probability $p_R = P(t_R \leq T_{lim}|E, X \neq X_0)$ that the system is restored in a restoration time t_R (i.e., the period spent by the system in states other than X_0) smaller than a prescribed period T_{lim}, given that earthquake E has occurred and has led to a system state X below X_0.

3.2. Three-loop Monte Carlo Simulation framework

To calculate p_W, p_M and p_R, a three-loop MCS is implemented to sample the stochastic events occurrence during the life of the nuclear power plant exposed to earthquakes up to a time horizon T (inner loop) for N_S different scenarios (middle loop), this for N_U different alternatives (outer loop) of the seismic hazard and fragility inputs (i.e., the b parameter of the Gutemberg-Richter laws, Eqs. (1) and (3), and the median acceleration capacities $A^*_{m,\gamma}$ of the Γ seismically uncorrelated components of the nuclear plant, respectively). In the inner loop, at each earthquake occurrence time t_S (lower than the nuclear power plant lifetime T), PSHA and seismic fragility evaluation are combined to conduct accident sequence analysis and if the system state is degraded by the earthquake impact (i.e., $X \neq X_0$), a restoration time t_R is sampled.

The pseudocode of the developed simulation framework is as follows (Fig. 1, where the inner, middle and outer loops are coloured in red, blue

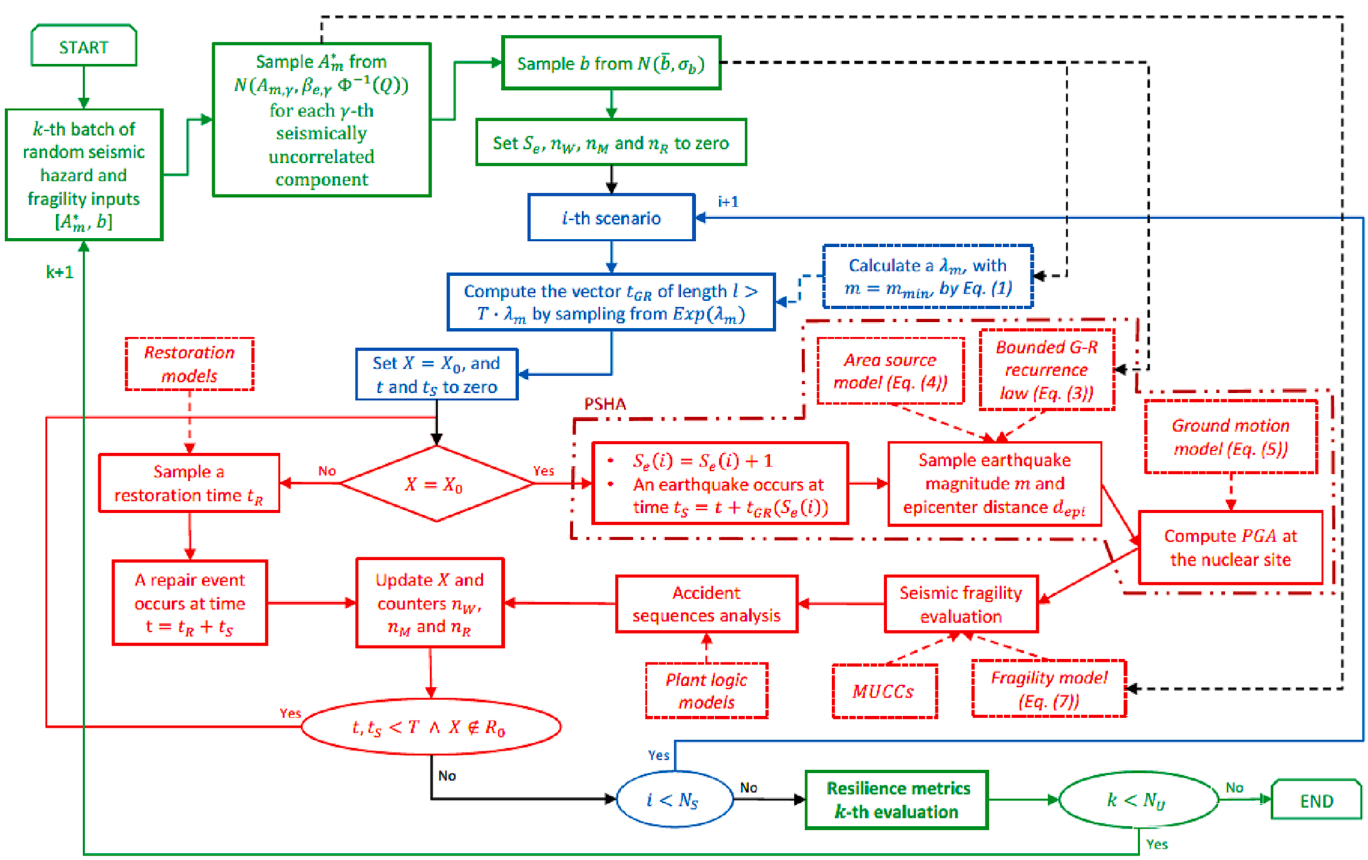

Fig. 1. The flowchart of the three-loop MCS framework.

and green, respectively):

(1) for $k = 1$ to N_U
(2) sample $A^*_{m,\gamma}$ from $N(A_{m,\gamma}, \beta_{e,\gamma} \bullet \Phi^{-1}(Q))$ to be used in Eq. (7), for each γ-th seismically uncorrelated component of one SMR unit
(3) sample b from $N(\overline{b}, \sigma_b)$ to be used in Eqs. (1) and (3)
(4) compute λ_m, with $m = m_{min}$, by Eq. (1)
(5) set $S_e = 0$, $n_W = 0$, $n_M = 0$, $n_R = 0$,
(5.1) for $i = 1$ to N_S
(5.2) compute t_{GR}, the vector of dimension $l > T \bullet \lambda_m$ of earthquake recurrence times, by sampling its entries from $Exp(\lambda_m)$
(5.3) set $t = 0$, $t_S = 0$, $X = X_0$
(5.3.i) while $t, t_S < T$ and $X \notin R_0$
(5.3.ii) if $X = X_0$, a seismic event is to be simulated:
(5.3.iii.a) increase the counter of seismic events for the i-th scenario: $S_e(i) = S_e(i) + 1$
(5.3.iii.b) conduct the PSHA:

- compute the earthquake occurrence time t_S: $t_S = t + t_{GR}(S_e(i))$ (i.e., the $S_e(i)$-th e of entry of t_{GR} is added to current time t);
- sample:
  - the earthquake magnitude value m from the *bounded Gutemberg-Richter recurrence law* by Eq. (3)
  - the epicenter distance d_{epi} from the area source model by Eq. (4);
- compute the *PGA* at the nuclear site by Eq. (5).

(5.3.iii.c) seismic fragility evaluation:

- compute $Fr_{\gamma,PGA}$ by Eq. (7), for each γ-th seismically uncorrelated component of one SMR unit.
- for $\gamma = 1$ to Γ:
  - sample f_γ from $U[0,1)$
  - if $f_\gamma < Fr_{\gamma,PGA}$, $X_{\gamma,1U} = 1$ (the γ-th component and those seismically correlated to it fail due to the seismic disruption in one SMR unit)
    - sample c_γ from $U[0,1)$
    - if $c_\gamma < MUCC_{\gamma,PGA}$, $X_{\gamma,MU} = 1$ (the failure of the γ-th component and those seismically correlated to it will occur in more than one SMR unit)
    - if $c_\gamma > MUCC_{\gamma,PGA}$, $X_{\gamma,MU} = 0$ (the failure of the γ-th component and those seismically correlated to it will not occur in more than one SMR unit)
  - if $f_\gamma > Fr_{\gamma,PGA}$, $X_{\gamma,1U} = X_{\gamma,MU} = 0$ (the γ-th component and those seismically correlated to it withstand the seismic disruption in all the SMR units)

(5.3.iii.d) accident sequence analysis:

- determine the state X of one SMR unit by setting $X_{\gamma,1U}$, $\gamma = 1, \cdots, \Gamma$, in plant logic model;
- determine the state X_{MU} of multiple SMR units by setting $X_{\gamma,MU}$, $\gamma = 1, \cdots, \Gamma$, in plant logic model.
- if $X = X_0$: $n_W = n_W + 1$
  - return to 5.3.i.
- if $X \neq X_0$ and $X \notin R_0$: $n_M = n_M + 1$
  - return to 5.3.i.

(5.3.iii) if $X \neq X_0$, a repair event is to be simulated:
(5.3.iii.a) sample a restoration time t_R
(5.3.iii.b) if $t_R \leq T_{lim}$: $n_R = n_R + 1$
(5.3.iii.c) $t = t_R + t_S$
(5.3.iii.d) $X = X_0$
(5.3.iii.e) return to 5.3.i.
(5.4) return to 5.1.
(6) compute the k-th set of values of the resilience metrics as follows:
a. $\boldsymbol{p_W(k)} = \boldsymbol{n_W} / \sum_{i=1}^{Ns} \boldsymbol{S_e}$
b. $\boldsymbol{p_M(k)} = \boldsymbol{n_M} / (\sum_{i=1}^{Ns} \boldsymbol{S_e} - \boldsymbol{n_W})$;
c. $\boldsymbol{p_R(k)} = \boldsymbol{n_R} / (\sum_{i=1}^{Ns} \boldsymbol{S_e} - \boldsymbol{n_W})$;
(7) return to 1.

It is worth noting that, without loss of generality, the Γ seismically uncorrelated components of the nuclear power plant are assumed with binary states: fully operative (i.e., $X_{\gamma,\bullet} = 0$) or completely damaged (i.e., $X_{\gamma,\bullet} = 1$). Also, the simulation framework allows considering different stochastic restoration models (from which restoration times t_R are sampled in point 5.3.iii.a.) and prescribed periods for repair T_{lim}, based on the state of one SMR unit (i.e., X) and multiple SMR units (i.e., X_{MU}). Finally, we assume that no damage is caused by earthquakes during the restoration time t_R (i.e., the earthquake source following a major seismic event has quenched).

4. Case study

In this Section, we show the application of the proposed framework for the assessment of the seismic resilience of a four SMR units plant (the aNPP). The results are compared to those obtained for a single unit plant of equivalent generation capacity (the cNPP). All units are based on PWR technology and it is assumed that they are located at the site of the Garigliano nuclear power plant in southern Italy (Di Maio et al., 2022), for which the seismic data are available. In Section 4.1 and Appendix B a brief description of the aNPP and cNPP is provided. Section 4.2 presents the PSHA. The seismic fragility evaluation and accident sequence analysis for the aNPP are given in Section 4.3 and Section 4.4, respectively; those of the cNPP are reported in Appendix C and Appendix D, respectively. Finally, Section 4.5 presents the comparison of the seismic resilience assessment outcomes for the aNPP and the cNPP.

4.1. The advanced nuclear power plant

The aNPP is based on the NuScale design and comprises four identical SMR units placed in a water pool, which serves as ultimate heat sink. All units are connected to a power station and a substation through power lines, as shown in Fig. 2. We assume that each SMR unit has a natural circulation Reactor Coolant System (RCS), so that it does not require any circulation pumps (Di Maio et al., 2022), and is equipped with two safety systems, the Decay Heat Removal System (DHRS) and the Emergency Core Cooling System (ECCS). Fig. 3 shows the scheme of the RCS (adapted from (US Nuclear Regulatory Commission, 2020) that consists in: the reactor pressure vessel, the integral pressurizer, two once-through helical-coil steam generators, two reactor safety valves and the RCS injection and discharge pipelines (US Nuclear Regulatory Commission, 2020). The DHRS provides the reactor secondary side passive cooling for non-LOCAs (e.g., Steam Generator Tube Failure (SGTF) (US Nuclear Regulatory Commission, 2020) and includes four actuation valves and two trains of decay heat removal equipment, each connected to one Steam Generator (SG) loop and sized to completely remove the decay heat load (Iaea, 2020). To actuate the DHRS, the main steam valves and the feedwater valves close, and the actuation valves open: steam, then, flows from the helical-coil SGs to the DHRS heat exchangers (external to the reactor vessel), from which the heat is eventually transferred to the water pool, where cold condensate is produced that returns to the SGs by gravity-driven circulation.

The ECCS includes three Reactor Vent Valves (RVVs) and two Reactor Recirculation Valves (RRVs); its operation ensures that the core remains covered and the decay heat is passively removed under both non-LOCA and LOCA conditions (US Nuclear Regulatory Commission, 2020). To actuate the ECCS, the main steam valves and the feedwater valves close, and the RVVs open: steam is, then, discharged from the reactor vessel to the containment vessel, so that the reactor pressure decreases and the containment pressure increases, until equilibrium is reached. Steam condenses on the inside surface of the containment vessel and floods the containment bottom, while the liquid level within the reactor decreases until level equilibrium is reached thanks to the RRVs. By this way, the nuclear fuel remains covered, thus ensuring a stable and safe shutdown.

The aNPP states are defined as follows:

- Normal operation ($X_{aNPP} = 0$): the plant is operating normally at its full generation capacity;
- Reduced power output ($X_{aNPP} = 1$): in case of a LOOP event, the plant allocates reactor units to supply housekeeping power, while still retaining as much generation capacity as possible (Di Maio et al., 2022);
- Safe shutdown: the plant is shutdown and the safety systems are working to ensure safe shutdown cooling in one ($X_{aNPP} = 2$) or more ($X_{MU,aNPP} = 2$) reactor units (Di Maio et al., 2022);

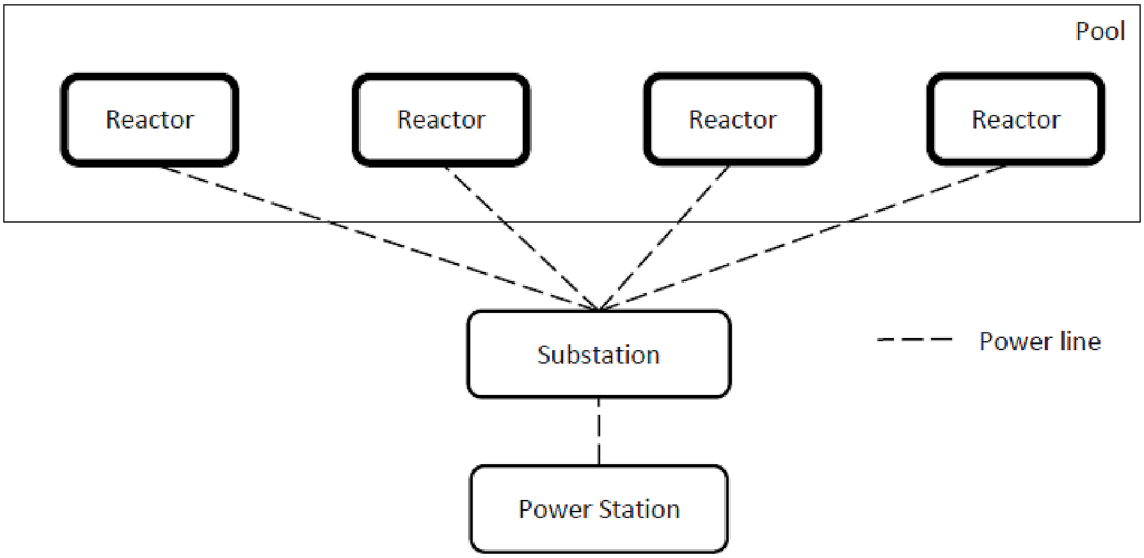

Fig. 2. The aNPP power connection scheme.

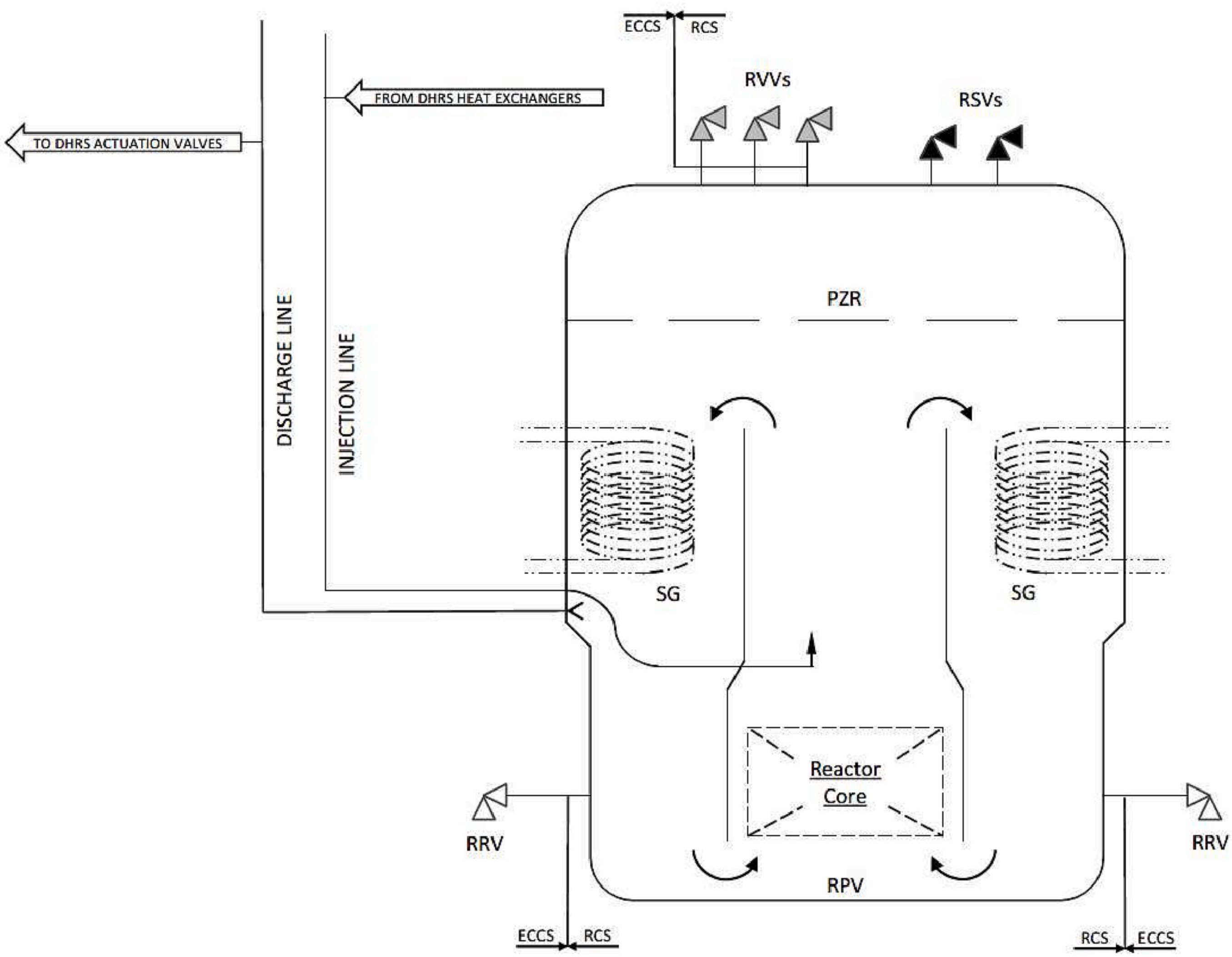

Fig. 3. Simplified RCS scheme.

- Core damage: the core of one ($X_{aNPP} = 3$) or more ($X_{MU,aNPP} = 3$) reactor units is damaged due to failure of safety systems.

4.2. Probabilistic seismic hazard analysis

The aNPP and cNPP are assumed to be located in Garigliano (southern Italy). Following the approach described in (Gutemberg-Richter relationship: Magnitude vs. frequency of occurrence, 2019), the parameters a and b of the Gutemberg-Richter laws of Eqs. (1) and (3) are set so that $a = -b\log(1.5 \bullet 10^{-6})/2.7$ (with $b = 0.89 \pm 0.03$ as estimated by (de Santis et al., 2011), to fit Italy earthquake data between 1963 and 2021 shown in Fig. 4 (Rovida et al., 2022). PGA magnitudes $m < m_{min} = 6.5$ and $m > m_{max} = 9$ are neglected because they cannot cause any damage (Reed and Kassawara, 1990) or have negligible recurrence rates, respectively. The area source model of Eq. (4) considers a radius $Ra = 100km$, whereas for the PGA calculation by Eq. (5), $S_S = 1$, $S_A = 0$, $F_N = 1$, $F_T = 0$ and $F_O = 0$ since the Garigliano nuclear site lies on a soft soil type (Forte et al., Jul. 2019) and a normal faulting mechanism is assumed.

4.3. Seismic fragility evaluation for the advanced nuclear power plant

We assume that: all the SMR units in the plant share the same design, so that their components have the same fragility parameters $A_{m,\gamma}$, $\beta_{a,\gamma}$ and $\beta_{e,\gamma}$ of the lognormal fragility model of Eq. (7) (listed in Table 1). In each SMR unit, the components of same type, location and elevation (i.e., steam generators, reactor safety valves, DHRS actuation valves and heat exchangers, and ECCS reactor vent and recirculation valves) are 100% seismically correlated and, therefore, a single fragility $Fr_{\gamma(\#),PGA}$ is defined where # is the cardinality of the set γ (see Section 2). For example, being V_1, V_2 and V_3 three redundant safety valves, a single fragility model $Fr_{V(3),PGA}$ is considered for the whole set, rather than independent fragility models $Fr_{V_1,PGA}$, $Fr_{V_2,PGA}$ and $Fr_{V_3,PGA}$. The value $Q = 0.95$ means 95% confidence that the actual probability of failure is less than the fragility calculated by Eq. (7).

4.4. Accident sequence analysis for the advanced nuclear power plant

Fig. 5 shows by the occurrence of an earthquake the event tree for the accident sequences involving one or more SMR units. The sequences are initiated of a certain PGA within the circular area of radius Ra. The

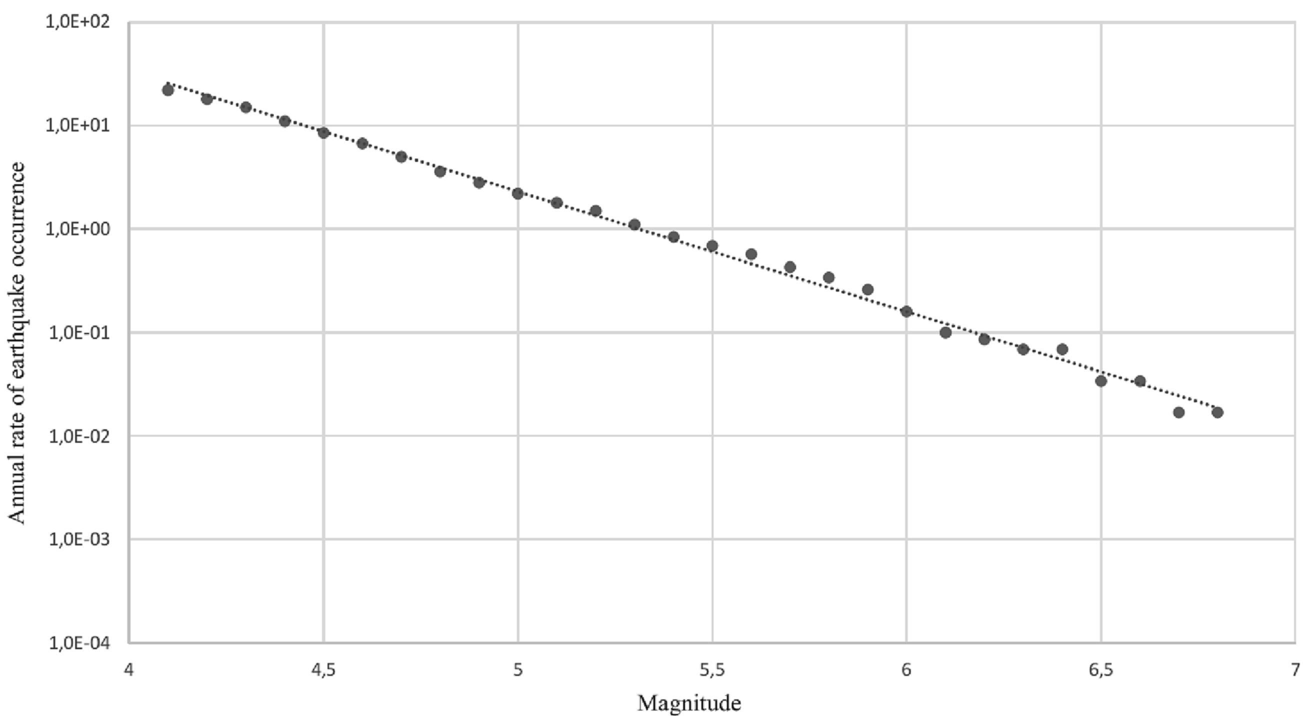

Fig. 4. Seismic hazard curve of Italy.

Table 1
Fragility parameters for the aNPP.

γ-th component	$A_{m,\gamma}$	$\beta_{a,\gamma}$	$\beta_{e,\gamma}$	Reference
Reactor pressure vessel	3.83	0.23	0.39	(Cover et al.)
Steam generator	2.53	0.28	0.36	(US Nuclear Regulatory Commission, 2020)
RCS injection/discharge pipeline	1.88	0.43	0.48	(Zio and Ferrario, 2013)
Reactor safety valve (spurious-open)	3.37	0.24	0.32	(US Nuclear Regulatory Commission, 2020)
Reactor vent valve (spurious-open)	2.38	0.28	0.50	(US Nuclear Regulatory Commission, 2020)
Reactor vent valve (fail-to-open)	17.45	0.27	0.37	(US Nuclear Regulatory Commission, 2020)
Reactor recirculation valve (spurious-open)	3.32	0.24	0.32	(US Nuclear Regulatory Commission, 2020)
Reactor recirculation valve (fail-to-open)	9.52	0.27	0.37	(US Nuclear Regulatory Commission, 2020)
DHRS heat exchanger	2.34	0.32	0.51	(US Nuclear Regulatory Commission, 2020)
DHRS actuation valve	0.57	0.32	0.52	(US Nuclear Regulatory Commission, 2020)
Power station	0.70	0.30	0.10	(Zio and Ferrario, 2013)
Substation	0.90	0.40	0.30	(Zio and Ferrario, 2013)

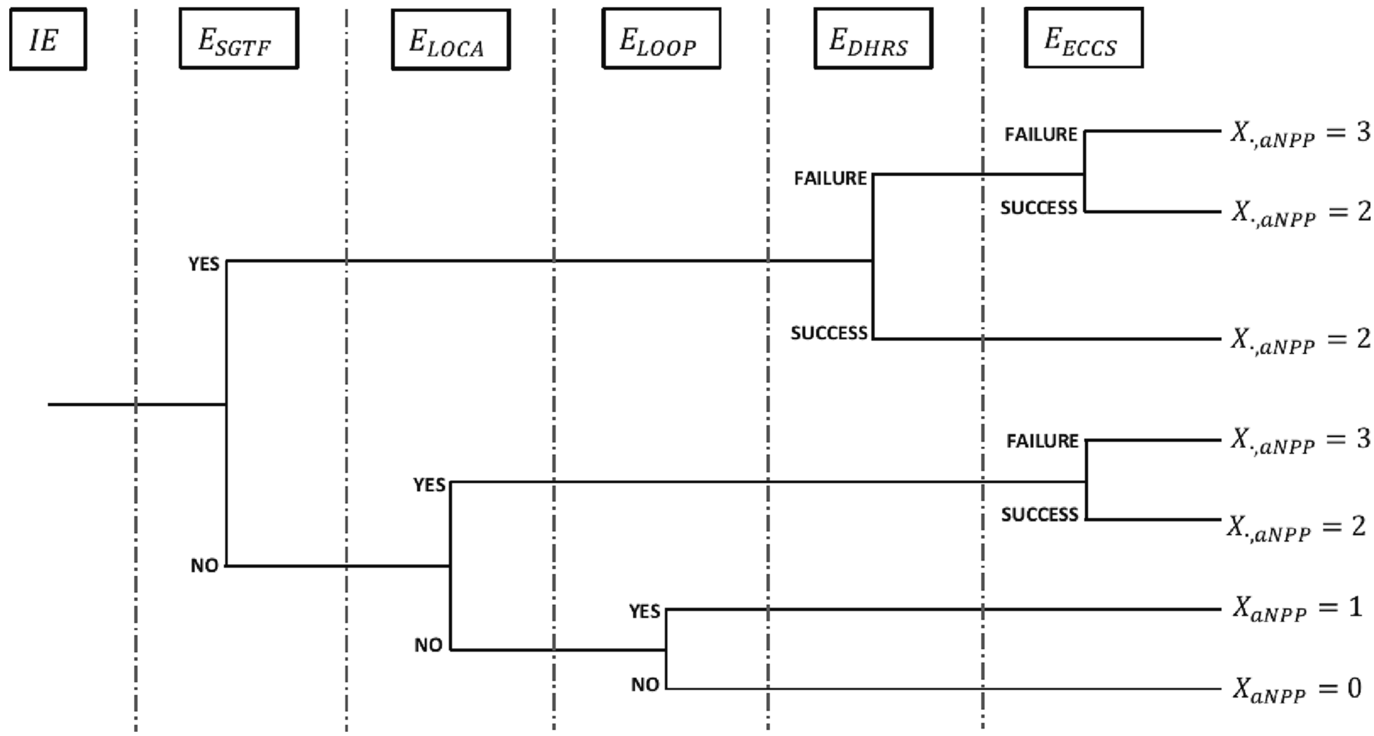

Fig. 5. Event tree analysis for the aNPP.

initiating event may induce the occurrence of other events such as Steam Generator Tube Failure (E_{SGTF}), Loss Of Coolant Accident within the RCS (E_{LOCA}) and Loss Of Offsite Power (E_{LOOP}). In case of occurrence of the latter events, the modular aNPP can allocate one or more SMR units to supply housekeeping power (mainly to the circulation pumps of the secondary coolant), while the other SMR units continue to supply as much power capacity as feasible to the grid (Di Maio et al., 2022). If E_{SGTF} or E_{LOCA} occurs, the safety systems come in action. The DHRS provides passive cooling of the reactor secondary side for non-LOCAs (i. e., E_{SGTF}) and is actuated by the closing of the main steam and feedwater valves and the opening of the actuation valves. The ECCS ensures that the core remains covered and the decay heat is passively removed under both non-LOCA and LOCA conditions, and is actuated by closing the main steam and feedwater valves and opening of RVVs. Failure of both trains of the DHRS (E_{DHRS}) and ECCS (E_{ECCS}) would drive the system to reactor core damage (US Nuclear Regulatory Commission, 2020). Fig. 6 shows the fault tree models developed to identify the basic events for E_{LOCA}; E_{LOOP}, E_{DHRS} and E_{ECCS}. Since event E_{ECCS} is conditional on the occurrence or non-occurrence of E_{LOCA}, that might result from the spurious opening of ECCS reactor vent or recirculation valves, one or both branches (marked by dotted lines) of the fault tree model for E_{ECCS} can be removed as appropriate, i.e., whenever E_{LOCA} occurs and it is due to the spurious opening of ECCS reactor vent or recirculation valves. To account for multiple SMR units, the *MUCCs* listed in Table 2 are assigned based on engineering judgment, for different levels of *PGA*, to classes of components according to their failure mode, i.e., normal failures (including the steam generators, the power station and the substation), structural failures (including the reactor pressure vessel, and the injection and discharge pipelines) and common cause failures (including the reactor safety, vent, recirculation and actuation valves, and the heat exchangers) (US Nuclear Regulatory Commission, 2020). Upon disruption, different restoration models are considered: 1) t_R [days]$\propto N(3, 1.5)$ if $X_{aNPP} = 1$: the restoration time t_R is normally distributed with a Mean Time To Repair (MTTR) of 3 days and a standard deviation of 1.5 days if the aNPP is recovering from a LOOP event (Reed and Kassawara, 1990); 2) t_R [years]$\propto Exp(1/1.32)$ if $X_{aNPP} = 2$: the t_R is exponentially distributed with a MTTR = 1.32 years if one SMR unit is recovering from either a LOCA within the RCS or a SGTF (Forte et al., Jul. 2019); 3) t_R [years]$\propto Exp(1/2.64)$ if $X_{MU,aNPP} = 2$: the previous MTTR is assumed doubled in value if either E_{LOCA} or E_{SGTF} extend to multiple SMR units, 4) $t_R \rightarrow \infty$ if $X_{aNPP} = 3 \vee X_{MU,aNPP} = 3$: the aNPP is never recovered if one or more SMR units run into core damage.

4.5. Results of seismic resilience assessment and discussion

The simulation framework presented in Section 3.2 is used to assess the resilience of the nuclear power plants under analysis for a time horizon $T = 50$ years, which is the chosen designed life. The parameters needed to simulate the cNPP behavior are given in Table 3 (the interested reader may refer to Appendix B for a thorough description of the related modelling assumptions). The sizes of the outer (i.e., N_U) and middle (i.e., N_S) loops of the MCS are 10^3 and 10^5, respectively. In the resilience metrics, the disruptive event E is represented by the occurrence of an earthquake, and the prescribed periods for repair T_{lim} are assumed to be 50% larger than the corresponding *MTTR*. Mean values and 95% confidence intervals of p_W, p_M and p_R, for the aNPP and cNPP, are presented in Fig. 7 and Table 4, respectively.

It can be seen that: 1) p_W is 0.97 ± 0.01 and 0.96 ± 0.01 for the aNPP and cNPP, respectively: it is therefore very likely that none of the two nuclear power plants will be disrupted by the assumed possible earthquakes of random magnitude occurring during their entire lifetime; in other words, the conditional probability that the nuclear power plant

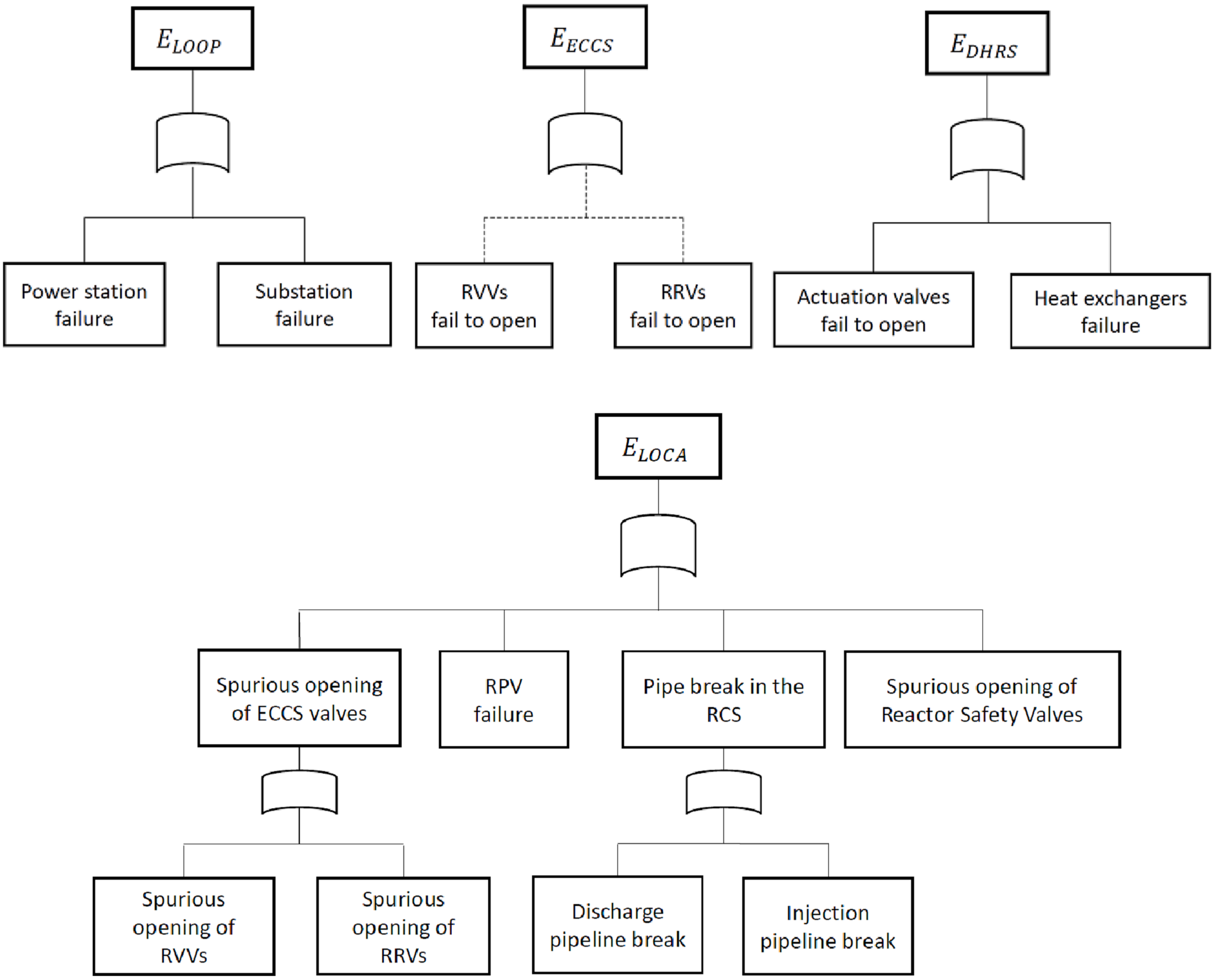

Fig. 6. Fault tree analysis for the aNPP.

Table 2
Multi-Unit Correction Coefficients for multi-unit modeling.

PGA[g]	Multi-Unit Correction Coefficient ($MUCC_{\gamma,PGA}$)		
	Normal failures	*Structural failures*	*Common cause failures*
< 0.1	0.10	0.01	0.30
$0.1 \div 0.2$	0.15	0.02	0.35
$0.2 \div 0.4$	0.25	0.05	0.45
> 0.4	0.40	0.10	0.60

Table 3
Fragility parameters for the cNPP.

γ-th component	$A_{m,\gamma}$	$\beta_{a,\gamma}$	$\beta_{e,\gamma}$	Reference
Reactor pressure vessel	3.83	0.23	0.39	(Cover et al.)
Steam generator	2.45	0.24	0.37	(Cover et al.)
Pressurizer	2.00	0.21	0.34	(Cover et al.)
RCS pipeline	1.88	0.43	0.48	(Zio and Ferrario, 2013)
Reactor coolant pump	2.64	0.24	0.37	(Park et al., 1998)
Safety injection pump	5.47	0.33	0.30	(Park et al., 1998)
Pool	4.23	0.27	0.37	(Park et al., 1998)
Diesel generator	0.70	0.40	0.20	(Zio and Ferrario, 2013)
Power station	0.70	0.30	0.10	(Zio and Ferrario, 2013)
Substation	0.90	0.40	0.30	(Zio and Ferrario, 2013)

exposed to random earthquakes keeps operating throughout its design life is larger than 0.95, 2) p_M is 1.00 ± 0.00 and 0.80 ± 0.02 for the aNPP and cNPP, respectively: this means that if an earthquake has degraded the system state, the aNPP will very likely ($lb = ub = 1.00$) not suffer from reactor core damage, whereas for the cNPP the probability is on average 0.80 ± 0.02, which might not be satisfactory (i.e., reactor core damage probability would be high), 3) p_R is 0.80 ± 0.01 and 0.63 ± 0.02 for the aNPP and cNPP, respectively: this indicates that if an earthquake has degraded the system state, the aNPP will likely be restored in a restoration time t_R lower than the prescribed period T_{lim}, whereas for the cNPP, t_R will not meet the requirements in about one third of the occurrences; since the restoration models considered for the two nuclear power plants are similar, this behavior is mainly due to the probability of entering the states with severe performance degradation (i.e., $1 - p_M$), hence requiring long t_R, larger for the cNPP, than for the aNPP. In addition, it is worth noting that the sojourn time of the aNPP in state $X_{aNPP} = 1$ is not so detrimental for its performance since, in that degradation state, it still retains part of its generation capacity, unlike the cNPP in the corresponding state $X_{cNPP} = 1$. Also, the confidence intervals are narrow, which means that the sample sizes used in the simulation are adequate to obtain accurate estimates. Fig. 7 shows that

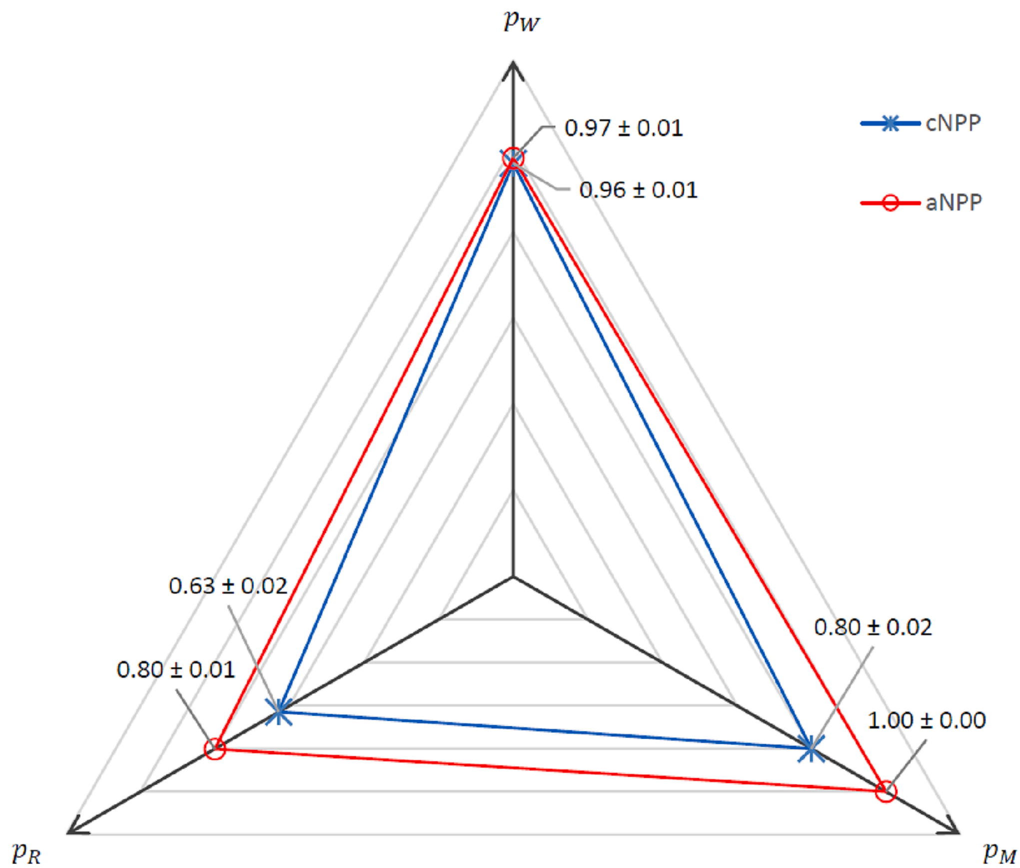

Fig. 7. Results of seismic resilience assessment.

Table 4
95% confidence interval of the resilience metrics.

		p_W	p_M	p_R
cNPP	*lb*	0.95	0.78	0.61
	< • >	0.96	0.80	0.63
	ub	0.97	0.82	0.65
aNPP	*lb*	0.96	1.00	0.79
	< • >	0.97	1.00	0.80
	ub	0.98	1.00	0.81

the resilience triangle of the cNPP is entirely contained in that of the aNPP: this means that, with respect to the metrics considered in this work, the latter exhibit a greater resilience to seismic disruptions, than the former. More specifically, the results show that the aNPP overcomes the cNPP in terms of post-accident scenario mitigation and restoration, as expected given the design safety features of SMRs (Di Maio et al., 2022).

5. Conclusions

The resilience of nuclear power plants to NaTech accidental scenarios is fundamental for their acceptance in the current and future conditions of increasingly frequent and severe extreme natural events due to climate change. In this paper, we have presented a framework for seismic resilience assessment of SMRs. Seismic fragility evaluation and accident sequence analysis are performed, tailored on the SMR peculiarities. Three dimensionless metrics have been defined to capture the system behavior during the three phases of the accident: withstanding, mitigating and restoring. A three-loop MCS, designed to evaluate the metrics is implemented for the resilience assessment of SMRs, while addressing the uncertainties of PSHA and of the seismic fragility evaluation. By application to a realistic aNPP, the framework has been shown able to provide a comprehensive description of the resilience of nuclear power plants to earthquakes for a given time horizon. The results have been compared to those of a cNPP, showing that the aNPP exhibits an overall greater resilience because of superior post-accident scenario mitigation and restoration. To address the computational demand of MCS, future work will consider other advanced sampling methods.

CRediT authorship contribution statement

Francesco Di Maio: Conceptualization, Validation, Writing – review & editing, Visualization, Supervision, Project administration. **Lorenzo Bani:** Conceptualization, Software, Formal analysis, Investigation, Writing – original draft. **Enrico Zio:** Conceptualization, Validation, Writing – review & editing.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper. This study was developed within the research project “Assessment of Cascading Events triggered by the Interaction of Natural Hazards and Technological Scenarios involving the release of Hazardous Substances” funded by MIUR - Italian Ministry for Scientific Research under the PRIN 2017 program (grant 2017CEYPS8).

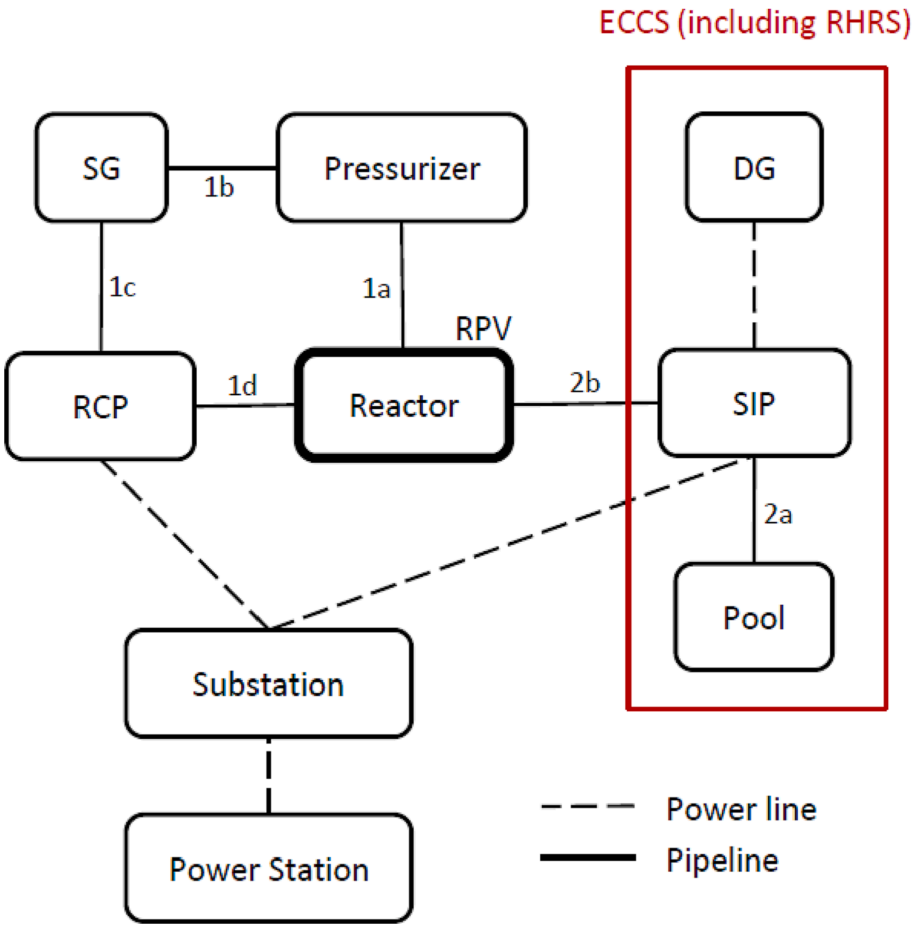

Fig. 8. Physical representation of the cNPP.

Data availability

The authors do not have permission to share data.

Appendix A. Description of the conventional nuclear power plant

The cNPP comprises a single, large reactor unit that bears the entire plant generation capacity. We assume that the unit has a forced circulation Reactor Coolant System (RCS) equipped with two safety systems, the Emergency Core Cooling System (ECCS) and the Residual Heat Removal System (RHRS), as shown in Fig. 8. The RCS consists in the reactor pressure vessel, the reactor coolant pump (supplied with offsite power, i.e., a power station and substation), the steam generator, the pressurizer and the piping system. The ECCS includes an emergency diesel generator, a pool and a safety injection pump (that can be supplied from either the offsite power or the internal emergency diesel generator) (Zio and Ferrario, 2013). Both the RHRS and the ECCS can ensure decay heat removal under accident conditions, and the latter returns coolant to the RCS in case of Loss Of Coolant Accidents (LOCAs).

The cNPP states are defined as follows:

- Normal operation ($X_{cNPP} = 0$): the plant is operating normally at its full generation capacity;
- Safe shutdown, following a LOOP ($X_{cNPP} = 1$): in case of a LOOP event, the plant is shutdown and the RHRS is working to ensure safe shutdown cooling of the reactor;

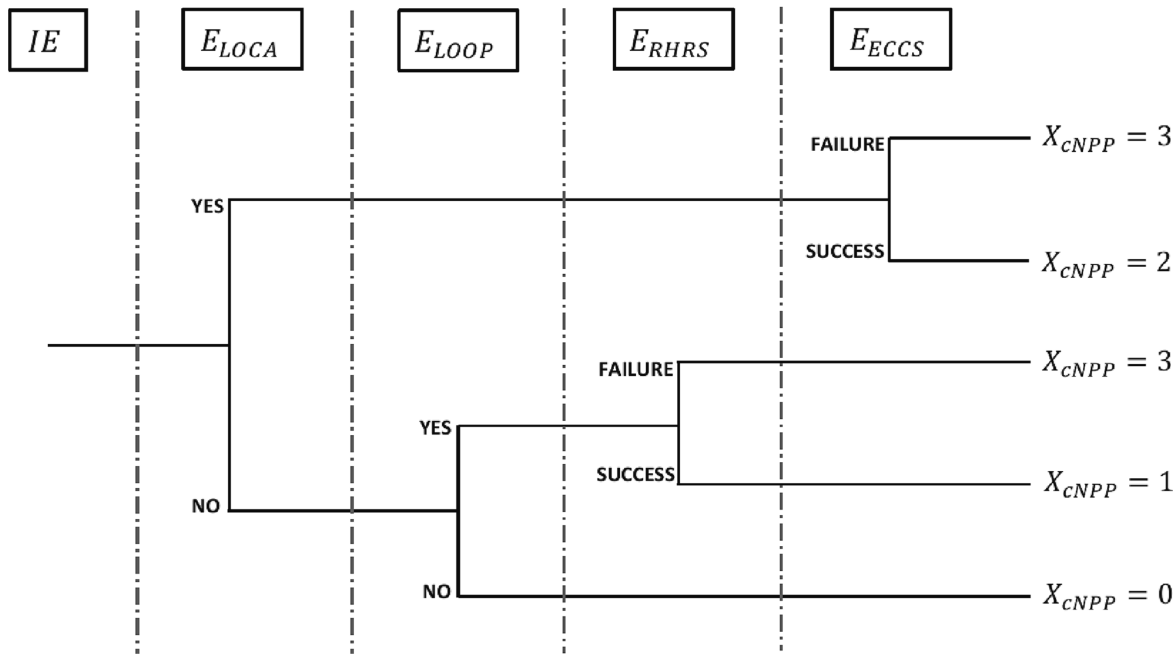

Fig. 9. Event tree analysis for the cNPP.

- Safe shutdown, following a LOCA ($X_{cNPP} = 2$): in case of a LOCA, the plant is shutdown and the ECCS is working to ensure safe shutdown cooling of the reactor;
- Core damage ($X_{cNPP} = 3$): the reactor core is damaged due to failure of safety systems.

Appendix B. . Fragility evaluation for the conventional nuclear power plant

The fragility parameters $A_{m,\gamma}$, $\beta_{a,\gamma}$ and $\beta_{e,\gamma}$ of the lognormal fragility model of Eq. (7) for the components of the cNPP are listed in Table 3. The value $Q = 0.95$ means 95% confidence that the actual probability of failure is less than the fragility calculated by Eq. (7).

Appendix C. . Accident sequence analysis for the conventional nuclear power plant

Fig. 9 shows by the occurrence of an earthquake the event tree for the accident sequences involving the single, large reactor unit. The sequences are initiated of a certain PGA within the circular area of radius Ra. The initiating event may induce the occurrence of other events such as Loss Of Coolant Accident (E_{LOCA}) and Loss Of Offsite Power (E_{LOOP}). If E_{LOOP} or E_{LOCA} occurs, the safety systems come in action. Both the RHRS and the ECCS can ensure decay heat removal, and the latter returns coolant to the RCS in case of E_{LOCA}. Failure of the RHRS (E_{RHRS}) and ECCS (E_{ECCS}) would drive the system to reactor core damage. Fig. 10 shows the fault tree models developed to identify the basic events for E_{LOOP}, E_{LOCA}, E_{RHRS} and E_{ECCS}. Upon disruption, different restoration models are considered: 1) $t_R[days] \propto N(3, 1.5)$ if $X_{cNPP} = 1$: the restoration time t_R is normally distributed with a Mean Time To Repair *MTTR* of 3 days and a standard deviation of 1.5 days if the cNPP is recovering from a LOOP event (Committee, et al., 2013); 2) $t_R[years] \propto Exp(1/1.32)$ if $X_{cNPP} = 2$: t_R is exponentially distributed with $MTTR = 1.32$ years if the cNPP is recovering from a LOCA (International Atomic Energy Agency, 2011); 3) $t_R \to \infty$ if $X_{cNPP} = 3$: the cNPP is never recovered if it runs into reactor core damage.

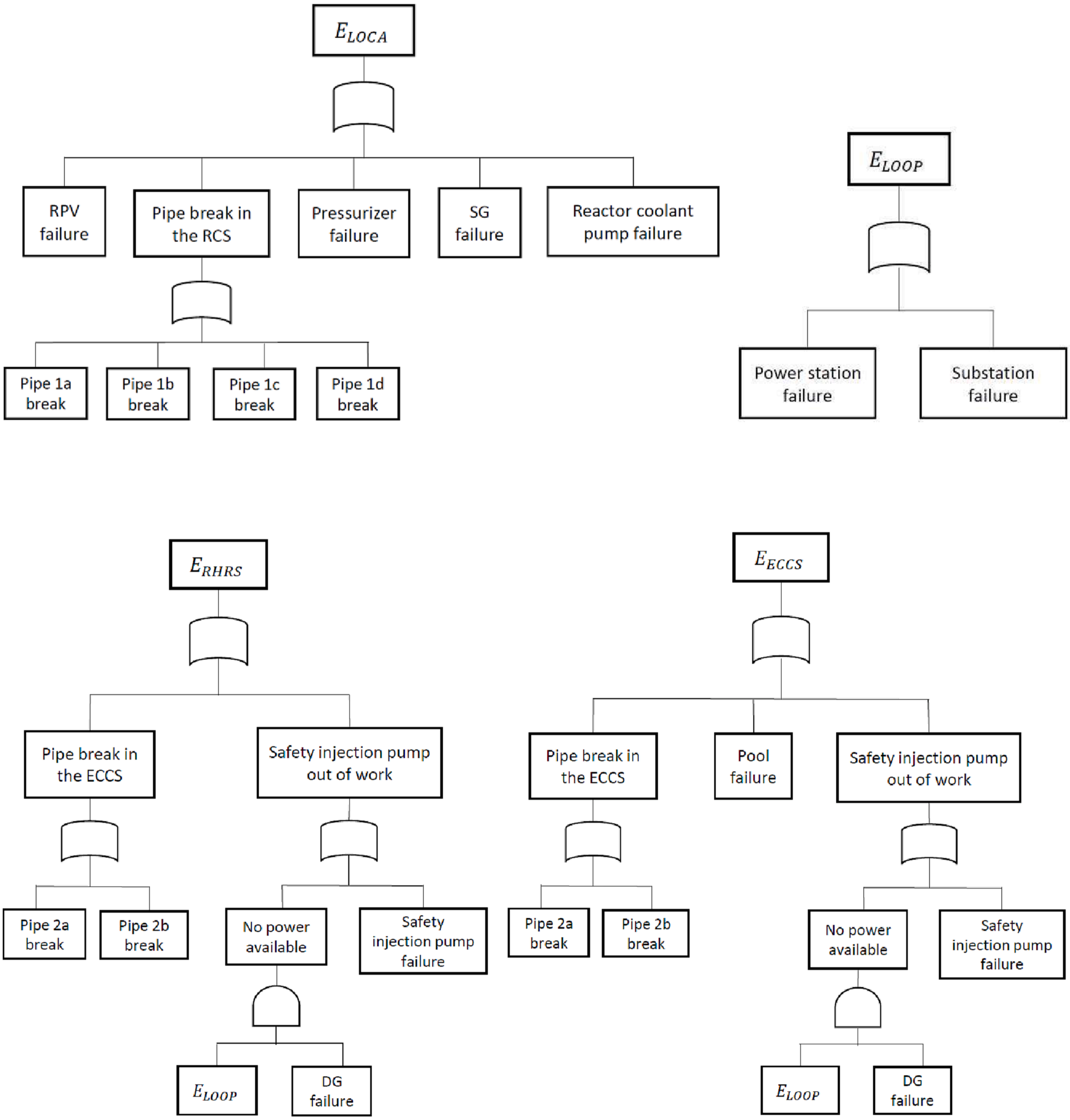

Fig. 10. Fault tree analysis for the cNPP.

References

Ahn, J., Guarnieri, F., Furuta, K., 2017. Resilience: A New Paradigm of Nuclear Safety: From Accident Mitigation to Resilient Society Facing Extreme Situations. Springer International Publishing, Cham.

Alrammah, I., 2022. Application of probabilistic safety assessment (PSA) to the power reactor innovative small module (PRISM). Nucl. Eng. Technol. 54 (9), 3324–3335.

Ambraseys, N.N., Douglas, J., Sarma, S.K., Smit, P.M., 2005. Equations for the estimation of strong ground motions from shallow crustal earthquakes using data from Europe and the middle east: horizontal peak ground acceleration and spectral acceleration. Bull. Earthq. Eng. 3 (1), 1–53. https://doi.org/10.1007/s10518-005-0183-0.

Baker, J.W., 2013. An introduction to probabilistic seismic hazard analysis. White Paper Version 2 (1), 79.

Budnitz, R.J., Apostolakis, G., Boore, D.M. 1997. "Recommendations for probabilistic seismic hazard analysis: guidance on uncertainty and use of experts," US Nuclear Regulatory Commission (NRC), Washington, DC (United States). Div.…

Choi, E., Ha, J.-G., Hahm, D., Kim, M.K., 2021. A review of multihazard risk assessment: progress, potential, and challenges in the application to nuclear power plants. Int. J. Disaster Risk Reduct. 53, 101933.

Choi, E., Kwag, S., Ha, J.-G., Hahm, D., 2021. Development of a two-stage DQFM to improve efficiency of single-and multi-hazard risk quantification for nuclear facilities. Energies (Basel) 14 (4), 1017.

U. S. N. R. Commission, *Reactor Safety Study: An Assessment of Accident Risks in US Commercial Nuclear Power Plants. Draft*, vol. 1. US Government Printing Office, 1975.

U. S. N. R. Commission. 1983. "PRA Procedures Guide (NUREG/CR 2300)," *Washington, DC*.

Committee, E., *et al.* 2013. *HAZUS-MH 2.1 Earthquake Model Technical Manual*.

Cornell, C.A., 1968. Engineering seismic risk analysis. Bull. Seismol. Soc. Am. 58 (5), 1583–1606.

Cover, L.E., Bohn, M.P., Campbell, R.D., Wesley, D.A. "Handbook of Nuclear Power Plant Seismic Fragilities.".

de Santis, A., Cianchini, G., Favali, P., Beranzoli, L., Boschi, E., 2011. The Gutenberg-Richter law and entropy of earthquakes: two case studies in central Italy. Bull. Seismol. Soc. Am. 101 (3), 1386–1395. https://doi.org/10.1785/0120090390.

Di Maio, F., Bani, L., Zio, E., 2022. The contribution of small modular reactors to the resilience of power supply. J. Nucl. Eng. 3 (2), 152–162. https://doi.org/10.3390/jne3020009.

Di Maio, F., Morelli, S., Zio, E., 2021. "A Simulation-Based Framework for the Adequacy Assessment of Integrated Energy Systems Exposed to Climate Change. In: Fathi, M., Zio, E., Pardalos, P.M. (Eds.), Handbook of Smart Energy Systems. Springer International Publishing, Cham, pp. 1–35.
Di Maio, F., Tonicello, P., Zio, E., 2022. "A Modeling and Analysis Framework for Integrated Energy Systems Exposed to Climate Change-Induced NaTech Accidental Scenarios," in Sustainability 14 (2), 786. https://doi.org/10.3390/su14020786.
Du, S., Zeng, Z., Fang, Y.-P., Zhai, Q., 2019. Resilience analysis of multistate systems based on Markov reward processes. In: in *2019 4th International Conference on System Reliability and Safety (ICSRS)*, pp. 436–440.
Eddy, C., Sase, E., 2015. Implications of the Fukushima nuclear disaster: man-made hazards, vulnerability factors, and risk to environmental health. J. Environ. Health 78 (1), 26–33.
Estrada-Lugo, H.D., Santhosh, T.V., de Angelis, M., Patelli, E. "Resilience assessment of safety-critical systems with credal networks".
Ferrario, E., Zio, E., 2014. Goal Tree Success Tree-Dynamic Master Logic Diagram and Monte Carlo simulation for the safety and resilience assessment of a multistate system of systems. Eng. Struct. 59, 411–433.
Forte, G., Chioccarelli, E., De Falco, M., Cito, P., Santo, A., Iervolino, I., 2019. Seismic soil classification of Italy based on surface geology and shear-wave velocity measurements. Soil Dyn. Earthq. Eng. 122, 79–93. https://doi.org/10.1016/j.soildyn.2019.04.002.
Funabashi, Y., Kitazawa, K., 2012. Fukushima in review: a complex disaster, a disastrous response. Bull. At. Sci. 68 (2), 9–21.
"Gutemberg-Richter relationship: Magnitude vs. frequency of occurrence," 2019. https://www.eoas.ubc.ca/courses/eosc256/Feb7_2011_GR.pdf (accessed Nov. 01, 2022).
Hosseini, S., Barker, K., Ramirez-Marquez, J.E., 2016. A review of definitions and measures of system resilience. Reliab. Eng. Syst. Saf. 145, 47–61.
Huang, Y.-N., Whittaker, A.S., Luco, N., 2011. A probabilistic seismic risk assessment procedure for nuclear power plants:(I) Methodology. Nucl. Eng. Des. 241 (9), 3996–4003.
IAEA, "ADVANCES IN SMALL MODULAR REACTOR TECHNOLOGY DEVELOPMENTS 2020 Edition A Supplement to: IAEA Advanced Reactors Information System (ARIS)," 2020. [Online]. Available: http://aris.iaea.org.
IAEA. 2021. *PROBABILISTIC SAFETY ASSESSMENT FOR SEISMIC EVENTS.* International Atomic Energy Agency.
International Atomic Energy Agency, 2011. Earthquake Preparedness and Response for Nuclear Power Plants. Internat. Atomic Energy Agency.
Kemeny, J.G. 1979. *Report of the President's Commission on the Accident at Three Mile Island: The Need for Change: the Legacy of TMI.*, vol. 41. The Commission.
Kim, J.H., Choi, I.-K., Park, J.-H., 2011. Uncertainty analysis of system fragility for seismic safety evaluation of NPP. Nucl. Eng. Des. 241 (7), 2570–2579. https://doi.org/10.1016/j.nucengdes.2011.04.031.
Kumar, M., Whittaker, A.S., Kennedy, R.P., Johnson, J.J., Kammerer, A., 2017. Seismic probabilistic risk assessment for seismically isolated safety-related nuclear facilities. Nucl. Eng. Des. 313, 386–400.
Linkov, I., Bridges, T., Creutzig, F., Decker, J., Fox-Lent, C., Kröger, W., Lambert, J.H., Levermann, A., Montreuil, B., Nathwani, J., Nyer, R., Renn, O., Scharte, B., Scheffler, A., Schreurs, M., Thiel-Clemen, T., 2014. Changing the resilience paradigm. Nat. Clim. Chang. 4 (6), 407–409.
Lipscy, P.Y., Kushida, K.E., Incerti, T., 2013. The Fukushima disaster and Japan's nuclear plant vulnerability in comparative perspective. Environ. Sci. Tech. 47 (12), 6082–6088.
Park, Y.J., Hofmayer, C.H., Chokshi, N.C., 1998. Survey of seismic fragilities used in PRA studies of nuclear power plants. Reliab. Eng. Syst. Saf. 62 (3), 185–195. https://doi.org/10.1016/S0951-8320(98)00019-2.
Park, J., Seager, T.P., Rao, P.S.C., Convertino, M., Linkov, I., 2013. Integrating risk and resilience approaches to catastrophe management in engineering systems. Risk Anal. 33 (3), 356–367.
Perera, A.T.D., Nik, V.M., Chen, D., Scartezzini, J.-L., Hong, T., 2020. Quantifying the impacts of climate change and extreme climate events on energy systems. Nat. Energy 5 (2), 150–159.
Pickard, L. 1981. "Garrick, Inc., and Westinghouse Electric Corporation, Fauske & Associates, Inc.," *Zion Probabilistic Safety Study. Prepared for Commonwealth Edison Company, Chicago*.
Reed, J.W., Kassawara, R.P., 1990. A criterion for determining exceedance of the operating basis earthquake. Nucl. Eng. Des. 123 (2–3), 387–396.
Rovida, A., Locati, M., Camassi, R., Lolli, B., Gasperini, P., Antonucci, A., "Catalogo Parametrico dei Terremoti Italiani (CPTI15), versione 4.0," 2022. [Online]. Available: https://doi.org/10.13127/CPTI/CPTI15.4.
Santhosh, T., Patelli, E., 2020. A Bayesian network approach for the quantitative assessment of resilience of critical systems. in *Proceedings of the 30th European Safety and Reliability Conference and the 15th Probabilistic Safety Assessment and Management Conference*.
Schaeffer, R., Szklo, A.S., Pereira de Lucena, A.F., Moreira Cesar Borba, B.S., Pupo Nogueira, L.P., Fleming, F.P., Troccoli, A., Harrison, M., Boulahya, M.S., 2012. Energy sector vulnerability to climate change: a review. Energy 38 (1), 1–12.
Smith, P.D. *et al.* 1981. "Seismic safety margins research program. Phase I final report-Overview".
US Nuclear Regulatory Commission, "Probabilistic Risk Assessment and Severe Accident Evaluation," *NuScale Standard Plant Design Certification Application - Chapter 19 - Part 2 - Tier 2 - Revision 5*, Jul. 2020.
US Nuclear Regulatory Commission, "Engineered Safety Features," *NuScale Standard Plant Design Certification Application - Chapter 6 - Part 2 - Tier 2 - Revision 5*, 2020.
US Nuclear Regulatory Commission, "Reactor Coolant System and Connecting Systems," *NuScale Standard Plant Design Certification Application - Chapter 5 - Part 2 - Tier 2 - Revision 5*, Aug. 2020.
Yan, R., Dunnett, S., 2022. Resilience assessment for nuclear power plants using Petri nets. Ann. Nucl. Energy 176, 109282.
Yawson, P.Y., Lombardi, D., 2018. Probabilistic seismic risk assessment of nuclear reactor in a hypothetical UK site. Soil Dyn. Earthq. Eng. 113, 278–285.
Yodo, N., Wang, P., 2016. Engineering resilience quantification and system design implications: a literature survey. J. Mech. Des. 138 (11), 111408.
Zeng, Z., Fang, Y.-P., Zhai, Q., Du, S., 2021. A Markov reward process-based framework for resilience analysis of multistate energy systems under the threat of extreme events. Reliab. Eng. Syst. Saf. 209, 107443. https://doi.org/10.1016/j.ress.2021.107443.
Zhou, T., Modarres, M., Droguett, E.L., 2018. An improved multi-unit nuclear plant seismic probabilistic risk assessment approach. Reliab. Eng. Syst. Saf. 171, 34–47.
Zio, E., 2018. The future of risk assessment. Reliab. Eng. Syst. Saf. 177, 176–190. https://doi.org/10.1016/j.ress.2018.04.020.
Zio, E., Ferrario, E., 2013. A framework for the system-of-systems analysis of the risk for a safety-critical plant exposed to external events. Reliab. Eng. Syst. Saf. 114 (1), 114–125. https://doi.org/10.1016/j.ress.2013.01.005.