A Framework for Probabilistic Risk Assessment of Nuclear Power Plants Coupled with Proton Exchange Membrane Electrolyzers

Stefano Marchetti[a], Samantha E. Wismer[a] and Katrina M. Groth[a]
[a]Systems Risk and Reliability (SyRRA) Lab, Center for Risk and Reliability, Reliability Engineering, University of Maryland, College Park, 20742, MD, USA.
Emails: smachet@umd.edu, swismer@umd.edu and kgroth@umd.edu

Abstract: The integration of Nuclear Power Plants (NPPs) with Proton Exchange Membrane (PEM) electrolyzers is a promising pathway for low-emission hydrogen production and expansion of the nuclear power plant capabilities. Co-located hydrogen production facilities and nuclear power plants are one aspect of the advanced integrated energy systems concepts being developed across the U.S. However, co-location of these facilities introduces new physical and functional interdependencies that are not explicitly captured in conventional Probabilistic Risk Assessments (PRA). This work presents a probabilistic framework to model risk scenarios and escalation pathways between a PEM electrolyzer facility and an NPP. Hydrogen leak frequencies and ignition probabilities are combined with overpressure consequence modeling and fragility-based damage estimation to quantify the potential impact of hydrogen explosions on shared electrical infrastructure, including switchyards, transformers, and transmission assets. In addition to overpressure effects, the framework also accounts for overcurrent events in the shared electrical infrastructure, which may lead to damages to the NPP turbine. Both overpressure and overcurrent-induced failures are mapped to site-level initiating events (i.e., loss of offsite power and loss of heat sink), which are integrated into a dynamic PRA model to estimate the increase in core damage frequency caused by the PEM integration. Preliminary results show that, while hydrogen-related escalation frequencies are generally low, shared electrical infrastructure can represent a critical vulnerability. The proposed methodology enables risk-informed siting and design decisions for nuclear-hydrogen integrated systems and provides a foundation for further extensions to broader IES configurations.

1. Introduction

The coupling of Nuclear Power Plants (NPPs) with Proton Exchange Membrane (PEM) electrolyzers as integrated energy systems is receiving increasing attention as a pathway for low-emission hydrogen production ([1], [2]). In such configurations, nuclear power can support hydrogen production during periods of reduced electricity demand, improving plant flexibility and capacity factor [3]. However, co-location of hydrogen production facilities with NPPs also introduces new physical and functional interdependencies that are not explicitly represented in conventional Probabilistic Risk Assessment (PRA) frameworks [4]. In particular, accidental hydrogen releases may generate explosions capable of damaging shared infrastructure and initiating site-level disturbances [5]. Also, hydrogen production facilities share electrical infrastructure that is relevant to nuclear safety, such as transformers, switchyards, and transmission assets [6]. These interdependencies can create initiating events at the nuclear site even when the original disturbance originates outside the nuclear plant itself. As a result, conventional PRA frameworks that treat the nuclear plant as an isolated system may underestimate the risk implications of coupled nuclear-hydrogen configurations [7].

This work presents a novel PRA framework to model escalation scenarios between a PEM electrolyzer facility and an NPP, and to propagate their effects to assess the site-level impact. The proposed approach integrates hydrogen release frequencies, ignition probabilities, overpressure consequence modeling, and fragility-based damage assessment to estimate the frequencies of PEM-induced initiating events. The impact of these initiators is then evaluated using a physical simulation model integrated within a Monte Carlo simulation for uncertainty propagation, allowing the estimation of the increase in core damage frequency caused by the PEM integration.

The contribution of the paper is therefore twofold. First, it provides a framework to embed PEM-induced escalation pathways into an existing PRA structure through additional site-wide initiating events. Second, it quantifies how these pathways affect core damage frequency and identifies the dominant escalation mechanisms under the current assumptions.
The remainder of the paper is organized as follows: Section 2 presents the methods used, Section 3 describes the proposed framework, Section 4 presents the case study, Section 5 shows the results of the application to the case study, and in Section 6, conclusions are drawn.

2. Methods

2.1. Blast modeling

The ignition of hydrogen can either happen immediately upon release, leading to a jet fire, or after it has accumulated, leading to either a detonation or deflagration event. In this work, jet fires are neglected because they cannot damage the nuclear reactor and its surrounding infrastructures, thanks to the presence of the fire walls surrounding each electrolyzer unit ([4], [7]).
Regarding explosions, the model used to calculate the overpressure is an engineering fit of the Baker blast correlation curves, which is defined by the following equations ([8], [9]):

$$W_{TNT} = \alpha \cdot m_{H_2} \cdot \frac{E_{H_2}}{E_{TNT}} \tag{1}$$

$$Z = \frac{R}{W_{TNT}^{1/3}} \tag{2}$$

$$\Delta P = \frac{1772}{Z} + \frac{114}{Z^2} + \frac{108}{Z^3} \tag{3}$$

where W_{TNT} is the TNT-equivalent mass, m_{H_2} is the mass of ignited hydrogen, E_{H_2} is the equivalent blast energy of hydrogen, E_{TNT} is the equivalent blast energy of TNT, R is the distance from the target, ΔP is the change in pressure due to the explosion, and α is the conversion coefficient (which represents the fraction of potential chemical energy converted into shock energy).

2.2. Fragility modeling

To assess the impact of explosion-induced overpressure on NPP structures and components, fragility curves are used to relate the blast overpressure to the probability of failure. In general, a fragility curve expresses the conditional probability that a component reaches or exceeds a given damage state F_k for a specified overpressure level [10].

In this work, fragility is represented in lognormal form for a component k given an overpressure ΔP:

$$P(F_k|\Delta P) = \Phi\left(\frac{\ln(\Delta P) - \ln(\Delta P_{k,50})}{\beta_k}\right) \tag{4}$$

where $\Phi(\cdot)$ is the standard normal cumulative distribution function, $\ln(\Delta P_{k,50})$ is the median overpressure capacity of the component and β_k is the logarithmic standard deviation representing the uncertainty in its capacity.

This formulation provides a flexible way to translate blast loads into component failure probabilities, which can then be propagated within the overall risk assessment framework.

2.3. Uncertainty propagation

The estimation of the impact of PEM-induced hazards on nuclear safety is affected by multiple sources of uncertainty, including the occurrence of failures and their timing, the severity of the associated physical effects, and the resistance of the exposed structures and components. To account for these uncertainties, a Monte Carlo simulation is performed [11]. Let **X** denote the vector of uncertain input parameters characterizing the hazard scenarios, the consequence models, and the component fragilities. For each *m-th* simulation trial, one realization $\mathbf{X}^{(m)}$ is sampled from the corresponding probability distributions and fed to a simulation model to obtain the output of interest. By repeating this procedure for M trials, an empirical distribution of the output is obtained from which statistical quantities such as expected values, percentiles, and uncertainty bounds can be estimated.

3. The proposed framework

Let us consider a Nuclear Power Plant (NPP) coupled with a PEM electrolyzer through electrical coupling to provide power.
The proposed framework consists of the following steps (**Fig. 1**):

1. identification and characterization of the hazards introduced by the coupling;
2. characterization of the system response to the hazards;
3. core damage frequency increase estimation.

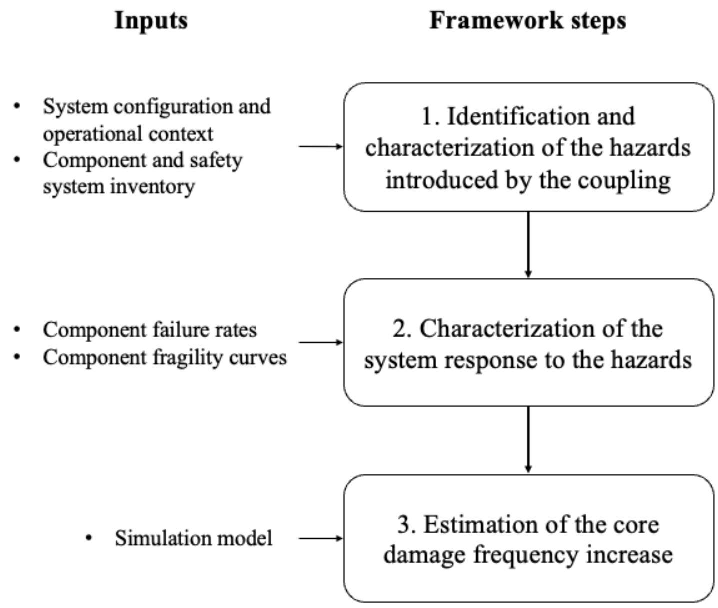

Figure 1: Steps of the proposed framework.

3.1. Identification and characterization of the hazards introduced by the coupling

The first step of the framework consists of identifying and characterizing the hazards introduced by the coupling of the two systems, namely failures originating in the PEM facility or in the shared coupling infrastructure that may affect nuclear power plant safety. Hazard identification can be performed using established system safety analysis approaches, such as Failure Mode and Effects Analysis (FMEA) and functional hazard analysis, which enable the systematic identification of component failure modes and the evaluation of their potential effects at the system level ([12], [13]).
The literature provides a useful basis for this step. Wismer et al. [14] identified and characterized more than 850 relevant failure scenarios for nuclear-PEM electrolyzer systems at the modal level and assigned qualitative risk rankings to them through an FMEA. In a subsequent study, Wismer et al. [15] used fault trees to organize the failure logic leading to system-level failure events, such as hydrogen leaks and ruptures, and to estimate their occurrence frequencies. In addition, Tamburini et al. [16] developed a Bayesian network to estimate the ignition probability of released hydrogen under a wide range of system-specific conditions and operating parameters.
Building on these works, the present framework first identifies the PEM and coupling-induced failure scenarios that may generate escalation pathways relevant to NPP safety, and then characterizes each retained scenario in terms of failure mode, occurrence frequency, escalation mechanism and potentially affected assets. The output of this step is therefore a set of coupling-induced hazard scenarios and the related frequencies, which forms the basis for the consequence assessment performed in the following steps.

3.2. Characterization of the system response to the hazards

The second step of the framework consists of characterizing the response of the system to the hazards introduced by the coupling, with the objective of determining whether such hazards can lead to damage to structures or components relevant to the NPP safety. Starting from each identified hazard, the corresponding physical effects on the exposed assets are evaluated and translated into component-level failure probabilities.

For hazards related to hydrogen leakages, the system response is characterized by first estimating the explosion overpressure generated by delayed ignition of the released and accumulated hydrogen using the blast model described in Section 2.1. The resulting overpressure at the location of each exposed asset is then combined with the corresponding fragility model introduced in Section 2.2 to estimate the probability of failure of that asset. Consequently, the response of the system is quantified as a function of both the severity of the explosion and the vulnerability of the potentially affected infrastructures.

For electrical hazards, the system response is characterized by assessing how faults or overcurrent events originating in the shared electrical infrastructure propagate to the equipment relevant to plant operation. This includes identifying the affected components and evaluating whether the disturbance can impair the availability or functioning of assets relevant to the NPP, such as transformers, switchyards, transmission elements, or the turbine.

The obtained failures are then mapped to the corresponding site-level initiating events of the nuclear power plant. In particular, failures affecting the shared electrical infrastructure may generate initiating events such as loss of offsite power or station blackout, while failures affecting systems involved in heat removal may lead to loss of heat sink or to other relevant initiators, depending on the considered coupling configuration. The output of this step is therefore the set of coupling-induced initiating events, together with their associated frequencies, which constitute the input for the final step of the framework.

3.3. Core damage frequency increase estimation

The final step of the framework consists of integrating the coupling-induced initiating events into the site-level PRA model to quantify the impact of the PEM integration on nuclear safety. In this work, the core damage frequency, defined as the expected number of core damage events per reactor-year, is estimated by propagating the uncertainties affecting the hazard scenarios, consequence models, and component fragilities through a Monte Carlo simulation procedure with M trials. Each m-th trial proceeds as follows:

- sample the energy demand to determine the PEM operating conditions;
- sample the occurrence of electrical failures and overcurrent events;
- sample the release diameter d;
- sample the hydrogen accumulation time t_{acc};
- determine the quantity of accumulated hydrogen using the HyRAM+ software [17];
- calculate the explosion overpressure affecting all the relevant nuclear structures and components and sample their failure;
- sample the failure times of all components and safety systems;
- determine the initiating events originated by the failures and feed them to the simulation model to assess the impact on the nuclear reactors and whether core damage occurs by checking the related threshold of the relevant safety variable $S \leq S_{threshold}$.

The core damage frequency is eventually calculated as follows:

$$F_{CDF} = \frac{1}{M} \sum_{m=1}^{M} 1\{S_m \geq S_{threshold}\} \quad (5)$$

where $1\{S_m \geq S_{threshold}\}$ is a function equal to 1 when the failure criterion is met and 0 otherwise.

4. Case study

The layout of the example coupled facility is sketched in **Fig. 2**.

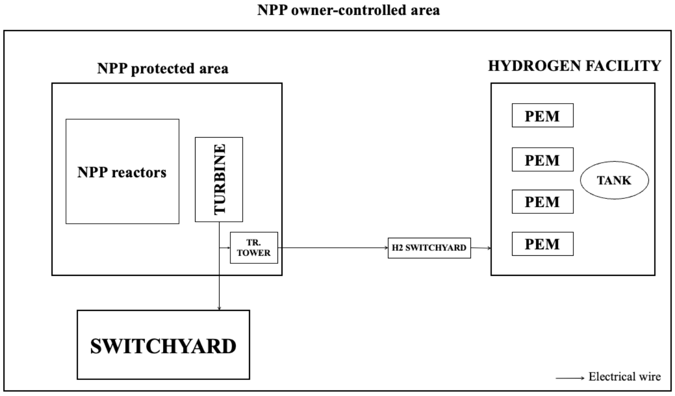

Figure. 2. Layout of the considered illustrative case study.

The system comprises a NPP with $N = 4$ SMDFR, each with a nominal power of 250 MW, and a hydrogen production facility with nominal rating (i.e., power input at full hydrogen production) of 30 MW. The NPP turbine is connected to both the power grid, through a high-voltage switchyard adjacent to the NPP protected area, and to the hydrogen facility, through a transmission tower located inside the NPP protected area. Electrical energy, in the form of alternating current, is diverted from the output of the turbine to the hydrogen facility, where most of the required power is converted to rectified direct current with a transformer. The operating parameters of the NPP are reported in **Table 1**.

Table 1. NPP operating parameters [18].

Parameter	Value
Mean linear power density	$609\ W/cm$
Fuel inlet temperature	$1300\ K$
Coolant inlet temperature	$973\ K$
Steam flow rate	$75\ kg/s$
Steam temperature	$700\ °C$
Steam pressure	$0.4\ MPa$

The hydrogen facility is composed of several PEMs and is located outside of the NPP protected area, but inside of the owner-controlled area. To assess the impact of the layout on the risk, a variable separation distance $d \in [10, 250]\ m$ is considered between the hydrogen facility and the NPP, and between the hydrogen facility and the NPP high-voltage switchyard. To determine the operating conditions of the hydrogen facility at the time of the accident, and therefore the hydrogen leakage rate, we assume that the NPP is always operating at full power, whereas the variability in the energy demand is accounted for using the normalized demand of nuclear energy in the US in 2025, whose cumulative distribution function is shown in **Fig. 3** [19]. The hydrogen facility operates on the unused portion of the produced steam and electricity, creating a dynamic dependency between grid demand and hydrogen production.

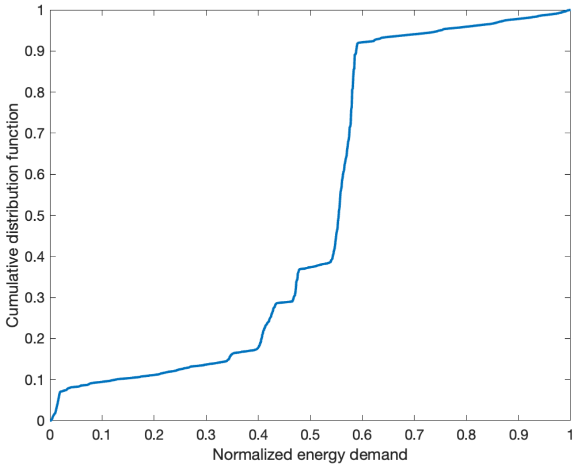

Figure. 3. Normalized energy demand cumulative distribution function.

Regarding hydrogen leakages, we considered a frequency of minor hydrogen leakage (associated with leakage diameters smaller than $5\ mm$) equal to $F_{minor} = 9.8 \times 10^{-6}\ 1/y$ and a frequency of major hydrogen leakage (associated with leakage diameters larger than $5\ mm$) equal to $F_{major} = 1.55 \times 10^{-8}\ 1/y$ [15]. The associated delayed ignition probabilities are 0.068 for minor releases and 0.091 for major releases [16]. The fragility curves used to assess the impact of explosions on the switchyard, transmission tower and turbine building are shown in **Fig. 4** (for the switchyard, the transmission tower and the turbine building) in line with [6]. The failure of the switchyard or the transmission tower leads to a loss of offsite power event, while the collapse of the turbine building leads to a loss of heat sink accident.

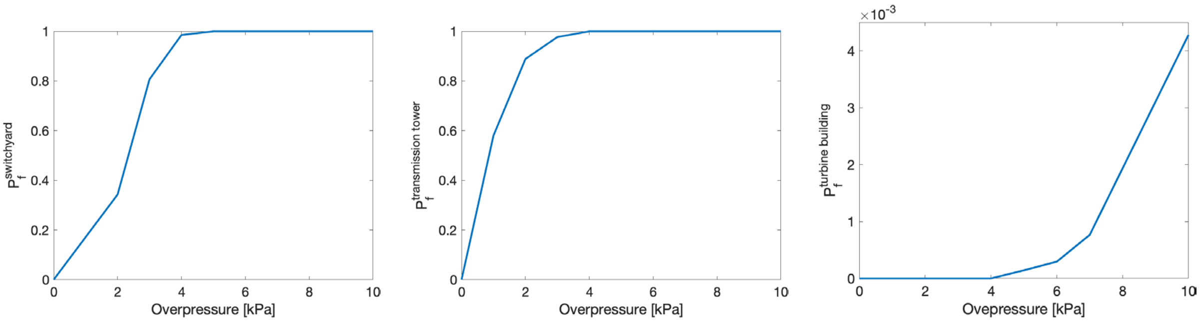

Figure. 4. Fragility curve of the switchyard (left), transmission tower (center) and turbine building (right).

Regarding electrical failures, an overcurrent event in the hydrogen facility can be caused by the failure of the transformer. To protect the NPP turbine, three identical breakers are installed: one in the $H2$ island, one within the NPP boundary and one near to the turbine [6]. Each breaker consists of a parallel configuration of two relays in series with one high-voltage circuit breaker. In case of an unmitigated overcurrent event (i.e., failure of all breakers), the NPP turbine is damaged, leading to a loss of heat sink accident in which the heat extracted from the coolant is progressively reduced. The failure rates of electrical components are reported in **Table 2**.

Table 2. Failure rates of electrical components.

Component	Failure rate [1/y]
Transformer	4.10×10^{-3} [20]
Circuit breaker	1.24×10^{-4} [21]
Relay	4.38×10^{-3} [22]

Finally, to assess the impact of the hazard on the system, the one-dimensional lumped parameter model presented in [23] is used to simulate safe and accidental conditions transients that might occur in the system following the occurrence of the initiating events. The model consists of coupled neutronic and thermal-hydraulic models. The neutronic behaviour is described using a modified point-kinetics model with six delayed neutron groups, which accounts for fuel flow. The thermal-hydraulic model represents the heat transfer within the reactor core assuming that the fuel, piping wall, and coolant are each divided into three sections with lumped properties. The model takes the failure times of the components and the safety systems as input and provides the peak cladding temperature $T_{w,i}$ of each $i = 1,2,3,4$ reactor unit as output, which are the safety parameters of interest here considered. The duration of the accidental scenario is $1200\ s$, which is the time needed to safely drain the fuel from the reactor. The safety threshold not to be exceeded by $T_{w,i}$ during the accident is $T_{w,fail} = 1244\ K$ [23]. The uncertain parameters to be sampled in the Monte Carlo simulation are the leak diameter d, which is assumed to follow the uniform distribution $U(1, 25)\ mm$, the accumulation time t_{acc}, which is assumed to follow the uniform distribution $U(1,120)\ s$, where t_{acc}^{max} is the maximum hydrogen sensor response time, and the failure times of the components and safety systems, which are assumed to follow the uniform distribution $U(0,1200)\ s$.

5. Results

Results obtained considering $M = 10^5$ trials for each distance considered are shown in **Fig. 5**. It can be seen that the increase in core damage frequency due to the coupling decreases monotonically as the hydrogen plant is moved farther from the NPP switchyard and remains below the acceptability threshold (i.e., $CDF_{threshold} = 10^{-6}$ [24]) for all distances considered.

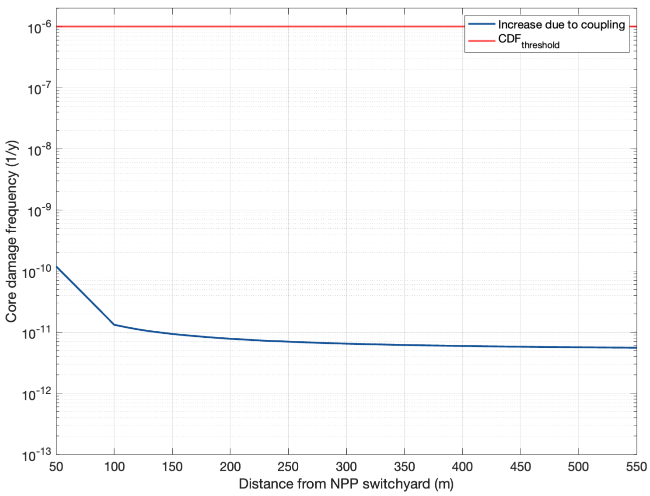

Figure. 5. Core damage frequency increase as a function of the distance from the NPP switchyard.

To better interpret this trend, the total core damage frequency increase is decomposed into the contributions associated with hydrogen explosions and overcurrent events, as shown in **Fig. 6**. As expected, the observed distance dependence is driven by the hydrogen-explosion contribution, which decreases as the separation distance from the NPP switchyard increases. By contrast, the overcurrent contribution remains constant with distance because it is represented through a fixed electrical-failure logic that is independent of hydrogen-plant siting. Consequently, at short separation distances, the total risk increase is dominated by hydrogen-explosion-induced escalation. At larger distances, the explosion contribution becomes progressively smaller, and the total increase approaches the constant contribution associated with overcurrent events. This decomposition indicates that increasing the stand-off distance between the hydrogen plant and the NPP switchyard is effective in reducing the explosion-induced escalation pathway, while it has no effect on the overcurrent-related contribution.

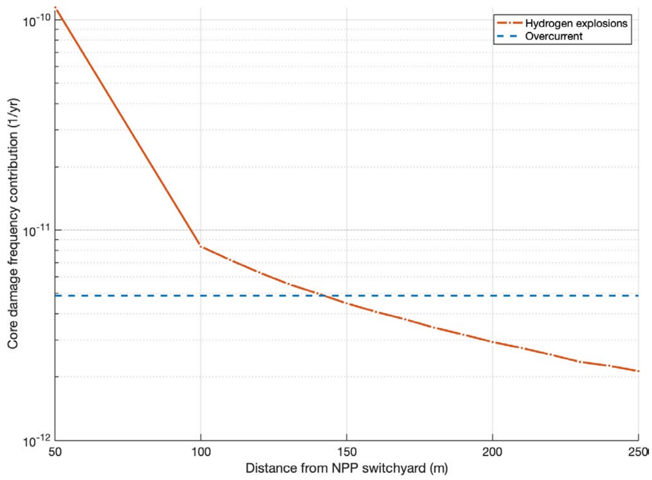

Figure. 6. Contribution of hydrogen explosions and overcurrent events to the core damage frequency increase as a function of the distance from the NPP switchyard.

6. Conclusions

This work proposed a probabilistic risk assessment framework to quantify the safety impact of coupling nuclear power plants with PEM electrolyzer facilities. The framework models hazards introduced by the coupling, maps them to nuclear initiating events, and propagates their effects through a dynamic PRA model to estimate the resulting increase in core damage frequency.

The illustrative case study shows that the core damage frequency increase remains below the considered acceptability threshold for all analyzed separation distances. The results also show that increasing the stand-off distance from the NPP switchyard reduces the explosion-induced contribution, while the overcurrent contribution remains constant because it depends on the shared electrical configuration rather than on physical separation.

Overall, the framework supports risk-informed siting and design of nuclear-hydrogen integrated systems by identifying the dominant coupling-induced risk contributors. Future work will extend the analysis to additional coupling configurations, including applications to existing nuclear reactors, and will incorporate economic considerations to support risk-informed plant design.

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