Risk Assessment of Co-Siting Small Modular Reactors and Microreactors with Nuclear Fuel Cycle Facilities

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Abstract: The concepts of Small Modular Nuclear Reactors and Micromodular Reactors have generated interest for decentralized electricity and heat generation in the industrial sector, presenting new co-siting challenges for existing nuclear safety assessment frameworks. Nuclear fuel cycle facilities, which cover conversion, enrichment, fuel fabrication, and reprocessing, considered in this paper, are unique industrial sites. They concentrate radiological and chemical hazards in their perimeters that make the co-location of a fuel cycle facility and a reactor system a safety challenge due to the potential dependent nature of accident scenarios, demanding more than a qualitative judgment. Any credible probabilistic safety assessment of such a proposition requires an empirical risk baseline by assessing and quantifying previous accidents and their causes. Currently, such a facility-driven baseline does not exist in a systematic form. A database of 184 nuclear fuel cycle facilities across 16 countries was compiled and validated, totaling roughly 3,881 facility-years of operational experience, which increases to 25,836 facility-years when production capacity-weighted normalization is applied. Both methods were kept separate as they capture different aspects of exposure, and there is a 58.7% missing end-date rate for decommissioned facilities. Accident frequency and severity were examined across temporal, regional, and facility-type dimensions, with 90% confidence intervals estimated via Monte Carlo simulation over 1,000 iterations. The industry-wide fatality rate is markedly elevated for reprocessing, which carries the heaviest burden, more than five times the per-facility average. A 99.7% reduction in fatality rates was observed from a peak in the 1950s to the 2010s, with no fatalities recorded in the 2020s. The USA exhibits the strongest historical record; post-1990, no fatal accidents have been recorded at U.S. nuclear fuel cycle facilities. Using Poisson-based frequency modeling, a strong correlation between accidents and fatalities was observed in the historical data. The conducted risk-informed analysis seeks to leverage facility-specific hazard profiles, assess regional regulatory maturity, and integrate combined source-term evaluations. This approach is designed to serve as a foundational framework for the subsequent development of a more comprehensive probabilistic safety assessment.

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1. INTRODUCTION

Small modular reactors (SMRs) and Micromodular reactors (MMRs) have attracted significant attention not only from state entities but also from the industrial sector over the last few years — driven by reliability, stability, and captive heat and power for energy-intensive facilities. The licensing record reflects this directly: U.S. NRC design certification of NuScale's VOYGR (2023), Oklo's Aurora combined license application, Kairos Power's construction permit, and active demonstration programs for Westinghouse's eVinci heat pipe microreactor, and Global First Power's U-Battery represent a momentum for the next year's commercial deployment. For early SMR/MMR deployments across different industrial sectors, nuclear fuel cycle (NFC) facilities are among the natural candidates, as they are among the most electricity-intensive operations in the nuclear sector and already operate within a licensed nuclear site.

For the purposes of this study, co-siting refers to the placement of an SMR or MMR within or immediately adjacent to an operating NFC facility, providing dedicated process heat and/or grid-semi-independent electrical power within the existing nuclear security perimeter. The distinction from a grid-connected nuclear power plant extends beyond operation; the entire risk and licensing pathways change when the reactor serves solely one captive industrial customer. A co-sited reactor serves a captive industrial load; it does not export power to the electric grid, or exports only in case of excess, and it operates entirely within a licensed and secure nuclear environment. History demonstrates the successful co-location of reactors and non-reactors in industrial operations, such as the Bruce Nuclear Generating Station in Canada, Tricastin in France, and the Paducah Gaseous Diffusion Plant in Tennessee. More discussions are presented in Section 2.3.

The co-siting of SMR/MMR with NFC facilities raises questions not fully answered by existing precedents. NFC processes—conversion, enrichment, fuel fabrication, and reprocessing—have unique hazard profiles in which radiological and chemical risks vary by stage. A credible probabilistic safety assessment (PSA) for a co-located reactor requires prior characterization of the NFC facility's risk profile, yet no systematic risk database exists for the global NFC industry. Current PSA literature focuses on reactor accidents in isolation and lacks the empirical basis to evaluate the combined risks of co-siting SMR/MMR with NFC, or to rank the viability of the four fuel cycle stages.

This paper addresses that knowledge gap. Specifically, the contributions of this study are:

1. Construction of the first systematic, facility-resolved global NFC risk database (184 facilities, 16 countries, 80+ years of operational history);
2. Development of two facility-years calculation methodologies with Monte Carlo uncertainty quantification (3,881 status-based; 25,836 capacity-weighted facility-years);
3. Derivation of a multi-dimensional NFC risk hierarchy across facility type, temporal period, and geographic region, and
4. Development of stage-specific hazard profiles, separation distance screening criteria, and a co-siting risk matrix.

Section 2 provides background on NFC facility hazard profiles, co-siting rationale, historical precedents, and SMR/MMR safety characteristics. Section 3 describes the database construction and analytical methodology. Section 4 presents and discusses the risk results, the co-siting risk matrix, and study limitations. Finally, Section 5 states the conclusion of this study and future work to be performed.

2. BACKGROUND

2.1. Nuclear Fuel Cycle Facility Types and Hazard Profiles

The four NFC stages (i.e., conversion, enrichment, fuel fabrication, and reprocessing) are not interchangeable from a radiological and chemical hazard standpoint. Before co-siting interaction pathways can be assessed, the baseline risk of each must be established on its own terms.

Conversion. At conversion facilities, uranium hexafluoride (UF_6) reacts violently with atmospheric moisture ($UF_6 + H_2O \rightarrow UO_2F_2 + 2HF$), leading to the primary accident scenario. Under worst-case conditions, a hydrogen fluoride (HF) gas plume can reach dangerous concentrations 1-2 km away [1]. The Sequoyah Fuels Corporation accident in January 1986 released about 13,000 kg of UF_6, resulting in one fatality due to acute HF inhalation and numerous injuries [2]. A 2014 HF release at Honeywell Metropolis Works was classified as alert-level emergency by a U.S. NRC inspection [3].

Enrichment. Modern gas centrifuge plants are primarily electrical loads but contain large pressurized UF_6 inventories in rotating equipment, posing unique chemical-mechanical hazards. Operating mainly in a vacuum enhances safety compared to older gaseous diffusion plants. The U.S. NRC addressed these hazards in NUREG-1827, the safety evaluation report for the URENCO USA National Enrichment Facility, which set a regulatory precedent for external hazard screening and probabilistic analysis of natural gas pipeline ruptures at distances around 500 m [4]. Historically, gaseous diffusion operations, such as those at the Paducah and

Portsmouth facilities, were conducted at even greater U.S. enrichment scales.

Fuel Fabrication. Fuel fabrication is distinct from other NFC stages due to criticality concerns. Bulk processing of UO_2, UF_4, and MOX powders poses risks of accumulating fissile material in unfavorable geometries, leading to potential accidents. Three key incidents illustrate these risks: the Y-12 Oak Ridge excursion in June 1958, the Wood River Junction accident in July 1964, and the JCO Tokaimura accident in September 1999, which resulted in two fatalities and significant radiation exposure for over 100 individuals [5, 6]. Despite these events, fuel fabrication has a better overall incident record compared to other NFC stages. A 2016 incident at the Westinghouse Columbia facility, where approximately 100 kg of uranium-bearing material exceeded criticality safety limits, highlights the ongoing regulatory attention to criticality safety [7].

Reprocessing. Reprocessing plants present the most complex NFC hazard profile. PUREX aqueous chemistry involves concentrated nitric acid, organic solvents, and high-level liquid waste (HLW) containing cesium and strontium fission products. No other NFC stage approaches the combined radiological and chemical source term that results. The worst-case historical reference for this facility class is the September 1957 explosion at the Mayak production facility near Chelyabinsk-65, USSR, which dispersed an estimated 70-80 tonnes of high-level radioactive waste across approximately 20,000 km^2 [8].

2.2. SMR/MMR Co-Siting Rationale and Regulatory Enablers

MMRs with outputs of 2 to 25 MWe and coolant temperatures of 300 to 550°C align well with NFC process demands: enrichment uses pure electrical loads, conversion facilities need 300-400°C processing heat, and reprocessing requires high-pressure steam. Co-siting MMRs at licensed NFC sites offers benefits such as a nuclear security perimeter, a qualified workforce, and safeguards, ensuring mission-critical operations and supporting low-carbon heat generation. The applicable regulatory framework is detailed in Section 3.5.

2.3. Historical Co-Siting Precedents

Before examining the regulatory pathway for SMR/MMR-NFC co-siting, three historical cases with over 50 years of combined operating records illustrate its feasibility: Bruce, Paducah, and Tricastin. The Bruce Nuclear Generating Station in Ontario supplied steam to the adjacent Bruce Heavy Water Plant from the early 1970s until 1997, demonstrating reactor-to-fuel-cycle integration [9]. At Tricastin, UF_6 conversion, enrichment, and four Électricité de France (EDF) power reactors have operated under a single licensing security perimeter since the 1970s, with the French Nuclear Safety Authority (ASN) enforcing consistent safety standards [10]. In the United States, the Paducah Gaseous Diffusion Plant operated entirely on electricity from the nearby TVA Shawnee Fossil Plant, representing a non-nuclear co-siting model that is now being proposed for SMRs and MMRs to deliver clean energy [11].

2.4. SMR/MMR Hazard Profile by Technology Family

Sections 2.1 to 2.3 detail the NFC-related hazards, which vary by reactor type across four technology families. **LWR-based designs** (e.g., NuScale VOYGR) feature an integral reactor vessel that prevents large-break loss-of-coolant accidents, with U.S. NRC certification indicating negligible off-site doses during design-basis and beyond design-basis events [12]. **Gas-cooled modular reactors** (e.g., BWXT-BARN, X-Energy XE100) utilize TRISO-coated fuel to retain fission products at high temperatures, with inert helium coolant that presents no reaction hazard in UF_6 or HF environments, confirming suitability for co-siting with conversion and enrichment facilities, as shown by recent tests on China's HTR-PM [13]. **Liquid-metal fast reactors** (e.g., Oklo Aurora Powerhouse) use sodium coolant, demonstrated by the EBR-II test in 1986 to passively resolve loss-of-flow and loss-of-heat-sink transients [14]. Lastly, **heat-pipe microreactors** (e.g., Westinghouse eVinci) feature sealed sodium heat pipes and self-regulating designs, reducing accident risk compared to larger reactors [15]. Under RG 1.233, all four SMR/MMR technologies' small source terms qualify them for brownfield co-siting scenarios [16].

3. METHODOLOGY

3.1. Database Construction

The database for accident records was completed from four source categories: the IAEA National Nuclear Fuel Cycle Information System (INFCIS), specifically the Nuclear Fuel Cycle Facilities Database (NFCFDB) [17], the U.S. NRC Events Notification Reports [18], OECD/NEA facility reports, and peer-reviewed nuclear safety literature [8]. The database covers all four nuclear fuel cycle stages: conversion (47 facilities), fuel fabrication (86), enrichment (22), and reprocessing (29), totaling 184 facilities across 16 countries. Each record carries the following fields: country, facility name, facility type (subcategory with stage), fuel type, operational status, scale, design capacity, start year, end year, and source comments.

The design capacities for conversion, fuel fabrication, and reprocessing facilities are expressed in tonnes of Heavy Metal per year (tHM/year). The enrichment capacities, originally reported in metric tonnes of Separative Work Units per year (MTSWU/year), are converted to tHM/year at 5.0 tHM per MTSWU, which is consistent with standard enrichment material balance practice, to allow cross-state normalization in Mechanism 2 (see Section 3.2). Eight operation status categories are used: In Operation, Decommissioning, Shutdown, Standby, Under Construction, Decommissioned, Refurbishment, and Others.

Data completeness for facility name, country, status, and capacity is 100%; start-of-operation dates are presented for 93.5% of records; end-of-operation dates for only 41.3%. End-of-operation dates are absent for 58.7% of non-operational facilities, introducing significant uncertainty in the facility-years (FY) calculation. Section 3.2 details the imputation method and Monte Carlo propagation employed to address this issue.

3.2. Facility-Years Calculation

Two independent facility-years mechanisms were implemented as a cross-check. Mechanism 1 counts facility-years weighted by operational status. Mechanism 2 goes further, adjusting each facility's contribution by its production capacity relative to other facilities in the same stage, the idea being that a large enrichment plant and a small reprocessing plant should not contribute equally to the risk denominator, as presented in Table 1.

Mechanism 1 - Status-Based. The facility-years contribution of each record is the product of its operational capacity and a status factor reflecting the degree of active operations, which is: *In Operation*: 1.0; *Decommissioning*: 0.5; *Refurbishment*: 0.3; *Stand By*: 0.25; *Shutdown/Under Construction/Others*: 0.1; and *Decommissioned*: 0.0. Status factors are expert-elicited proxies for process activity intensity relative to full operation: Decommissioning (0.5) reflects reduced but active hazardous material management; Refurbishment (0.3) and Stand By (0.25) reflect offline states with residual on-site inventory; Shutdown, Under Construction, and Others (0.1) reflect minimal NFC process activity; and Decommissioned (0.0) reflects complete cessation of operations. A sensitivity analysis varying all intermediate factors simultaneously by ±50% of their base values yielded total facility-years within 5.9% of the base case (3,881 FY), confirming that the risk hierarchy and headline fatality rates are not materially sensitive to the specific factor assignments. For facilities with confirmed start and end dates, the duration period is the calendar difference between the two dates. For facilities currently *In Operation* with no end date, the analysis reference year (2024) is used as the cut-off, freezing the database at the end of 2024 for reproducibility. For non-operational facilities that do not have a confirmed end date, the end year is estimated by adding a stage-specific expected operational lifetime to the start year. These lifetimes are drawn from truncated Normal distributions (TN)[1] as follows: Conversion TN(40, 10^2, 10, 60), Enrichment TN(35, 8^2, 10, 60), Fuel Fabrication TN(38, 7^2, 10, 60), and Reprocessing TN(30, 10^2, 10, 60) years. The truncation bounds [10, 60] years are empirically grounded in the subset of 73 facilities with confirmed start and end dates: the observed maximum operational period is 58 years, establishing the upper bound of 60 years as a marginally conservative ceiling; the lower bound of 10 years excludes anomalously short operational periods, observed minimum of 1 year, which are associated

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[1] A truncated normal distribution is defined by the mean (μ), standard deviation (σ), and the minimum and maximum boundaries, a and b, that clip the distribution's tails and rescale the remaining probability to 1.

with pilot-scale or commercially unsuccessful facilities not representative of the missing-end-date population. The truncation affects only the extreme tails of the stage-specific distributions, which place the bounds approximately 2–4 standard deviations from the respective means. As illustrated in Table 1, the status-based facility-year total, across all 184 records, comes to **3,881 FY**.

Mechanism 2 - Capacity-Weighted. The capacity weight for each facility is its standardized design capacity divided by the reference median for its subtype; this weight multiplies the Mechanism 1 facility-years, scaling exposure by production intensity relative to within-stage peers. Mechanism 2 yields **25,836 FY**, a 6.66× multiplier over the status-based total, reflecting the concentration of production capacity in a smaller number of large facilities. Enrichment exhibits the strongest amplification (14.3× M1), driven by the dominance of a small number of very large centrifuge plants; reprocessing is the only stage where M2 < M1 (0.92×), as depicted in Table 1, indicating below-reference-average capacity utilization.

Monte Carlo Uncertainty Propagation. Two Monte Carlo analyses were performed. The first (1,000 iterations) propagates the missing end-date uncertainty: for each iteration, non-operational facilities without confirmed end dates are assigned lifetimes sampled from the stage-specific truncated Normal distribution above, Mechanism 1 is recomputed, and the resulting distributions of the total facility-years; the 5th and 95th percentiles of this distribution define the 90% confidence interval on the facility-years denominator of the primary risk metric, the fatality rate per 1,000 FY (see Eqs. 1 and 2). The second (250 iterations) propagates uncertainty in the fatality rate metric: fatality counts are drawn from a Normal approximation to the Poisson count distribution, facility-years are sampled with a coefficient of variation of 15%. The 15% CV is composed of: end-date sampling uncertainty (~0.19%), status-factor sensitivity (±5.9%), and a conservatively assigned residual (~8.9%) covering capacity-estimation variability across 16 countries and potential under-reporting of events. The fatality rate per 1,000 FY is recomputed at each draw. In both analyses, the 90% CI boundaries are taken as the 5th and the 95th percentiles of the simulation output distribution.

Table 1. Database Summary — Facility Counts, Data Completeness, and Facility-Years by Stage

Stage	Facilities	% End Dates Known	Status-Based FY (M1)	Capacity-Weighted FY (M2)	M2/M1
Conversion	47	23.4%	1,090	2,862	2.63×
Enrichment	22	27.3%	628	8,988	14.3×
Fuel Fabrication	86	47.7%	1,806	13,657	7.56×
Reprocessing	29	62.1%	357	328	0.92×
Total	**184**	**41.3%**	**3,881**	**25,836**	**6.66×**

3.3. Accident Data Collection

Accident data were compiled from three source categories: the IAEA INES database, the U.S. NRC event report and augmented inspection team findings, and peer-reviewed nuclear safety literature. In total, 35 accident events, accounting for 101 direct fatalities, were identified across all NFC stages, spanning from the 1940s to the 2020s, and each event was classified by facility type, decade, and country of occurrence. Events captured include the Windscale fire (UK, 1957), the Mayak (Chelyabinsk-65) explosion (USSR, 1957), the Sequoyah Fuels UF_6 cylinder rupture (the USA, 1986), the Wood River Junction criticality accident (USA, 1964), and the JCO Tokaimura criticality event (Japan, 1999). The database is restricted to events resulting in fatalities or confirmed radiation exposure above background; near-misses and events without documented consequences were excluded, introducing a potential under-reporting bias acknowledged in Section 4.6.

3.4. Risk Metrics

Accident arrivals are modeled as independent Poisson processes: accidents are the discrete, countable events that form the statistical foundation of the analysis. Fatalities are a spin-off consequence: each accident produces a number of fatalities governed by per-event severity. As demonstrated by Eq. 1, the accident rate per 1,000 FY is therefore the base metric:

$$\text{Accident Rate} = (\text{Total Accidents}/\text{Total Facility-Years}) \times 1,000 \tag{1}$$

The fatality rate per 1,000 FY is derived from the accident rate by incorporating the severity index (mean fatalities per accident event), as demonstrated by Eq. 2:

$$\text{Fatality Rate} = \text{Accident Rate} \times \text{Severity Index} = (\text{Total Fatality}/\text{Total Facility Years}) \times 1,000 \tag{2}$$

The strong Pearson correlation between accident counts and fatality counts across all dimensions ($r = 0.9958$) confirms that severity is approximately constant at the aggregate level, which means the fatality rate faithfully tracks the underlying Poisson accident process and can serve as the headline risk metric integrating both frequency and consequence. Additional metrics include the per-facility relative fatality risk (fatalities per facility normalized to the industry average). All metrics are computed independently for each facility type, decade, and geographic region, and are reported with 90% CI derived from the Monte Carlo procedure described in Section 3.2.

3.5. Separation Distance and Co-Siting Hazard Screening

The screening criteria discussed below are derived from the U.S. NRC (e.g., 10 CFR 70.22(i) and 10 CFR 70.61 [19]) and IAEA safety standards (e.g., SSR-4 [20] and SSG-35 [21]), both of which have direct applicability to NFC co-siting scenarios. Under 10 CFR 70.22(i), NFC licensees define a controlled area boundary within which access is managed, and emergency response can be executed with the off-site dose criterion set at ≤ 1 rem (10 mSv) per credible accident. The 10 CFR 70.61 quantitative health objectives establish three frequency tiers: 1) high-consequence event (exposure exceeding 25 rem, or life-threatening chemical exposure) must be highly unlikely at $\leq 10^{-5}$ to 10^{-6} per year; 2) intermediate-consequence events, i.e., exposures of 5-25 rem or immediately dangerous to life or health (IDLH)-level chemical exposures, must be unlikely at $\sim 10^{-4}$ per year; and 3) beyond design-basis events (those falling outside all defined consequence tiers) must fall below 10^{-7} per year. The emergency planning zone (EPZ) required for large power reactors is substantially less prescriptive than the 10-mile requirement [19]. U.S. NRC Regulatory Guide 1.233 (finalized from draft DG-1353 [22]) extends this logic to advanced non-light-water reactors, tying EPZ dimensions explicitly to the actual extent of source-term and consequence analysis rather than to any fixed radius [16]. NuScale's 2023 design certification (FR Doc. 2023-0079) demonstrated a site boundary EPZ for a 600 MWe Integral LWR, providing the clearest regulatory precedence to date for source-term-proportional EPZ sizing [12]. On the IAEA side, SSR-4 Requirement 11 establishes that external human-induced hazards are within scope for the NFC safety analysis. Additionally, the graded site evaluation procedure that operationalizes the aforementioned requirement is present in SSG-35.

For overpressure screening, peak overpressure below 0.01-0.02 bar at the SMR/MMR boundary requires no explicit analysis. Events generating overpressure above the threshold require demonstrated compliance with the 10 CFR 70.61 risk goals [19]. Reference blast thresholds from structural response data are 1 psi (0.069 bar) for glass breakage and 5 psi (0.345 bar) for unreinforced wall collapse.

For HF toxic gas dispersion from conversion or enrichment releases, worst-case atmospheric dispersion modeling places dangerous HF concentrations (IDLH: 30 ppm) at 1-2 km from the release points, depending on UF_6 inventory and release scenario [1]. SMR/MMR control room ventilation intakes must be sited outside this plume radius or equipped with a filtered, isolated air supply [23].

For external pipeline hazards, the U.S. baseline rupture frequency is approximately $3.5 - 4.5 \times 10^{-5}$ per km-year [24]. NUREG-1827 provides the directly applicable regulatory precedent: a natural gas pipeline at approximately 545 m separation was evaluated, and the combined probability of rupture, ignition, and impact to the enrichment facility was demonstrated to be below 10^{-6} per year, satisfying the 10 CFR 70.61 high-consequence threshold [4].

Table 2. Risk Metrics by NFC Facility Stage

Stage	Facil-ities	Fatal-ities	Acci-dents	FY	Fatality Rate	Accident Rate	Severity Index	Relative Risk
Conversion	47	15	8	1,090	13.77	7.34	1.88	0.58
Enrichment	22	8	4	628	12.74	6.37	2.00	0.66
Fuel Fabrication	86	25	12	1,806	13.84	6.64	2.08	0.53
Reprocessing	29	53	11	357	148.33	30.81	4.82	3.33
Total	**184**	**101**	**35**	**3,881**	**25.48**	**8.98**	**2.89**	**1.00**

Note: All FY values are database-integrated, 2024 reference year. Per-Facility Relative Risk = (stage fatalities per facility) / (industry-average fatalities per facility = 0.549), is normalized by facility count, not by operational exposure (FY); see §4.2 for interpretation.

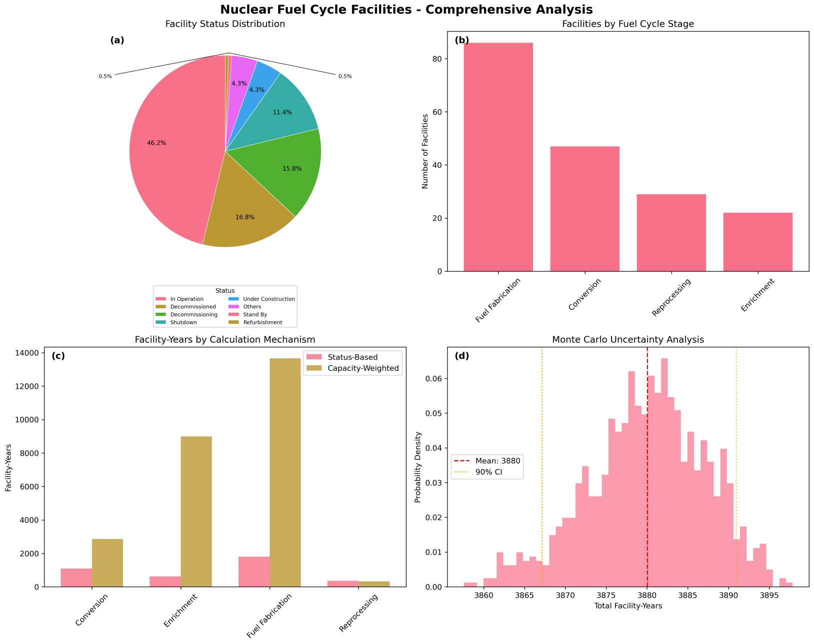

Fig. 1. Global NFC historical risk analysis covers 184 facilities, with the reference year of 2024. (a) Most facilities are either In Operation or Decommissioned/Shutdown, indicating a mature inventory. (b) Facility count by fuel cycle stage shows Fuel Fabrication has the highest number at 86 (46.7%), followed by Conversion (47), Reprocessing (29), and Enrichment (22). (c) Comparison of status-based (M1) and capacity-weighted (M2) facility-years. (d) Monte Carlo analysis (1,000 iterations) for total status-based facility-years indicates a mean of about 3,881 FY with a 90% confidence interval of [3,869 — 3,892].

The brownfield scenario starts with assets that a greenfield application would need to create: a licensed controlled area boundary, nuclear-qualified personnel, existing emergency plans, and established safeguards infrastructure. From a regulatory process standpoint, brownfield co-siting at a licensed NFC site does not trigger a full new-site safety evaluation; instead, an Integrated Safety Analysis (ISA) updated under NUREG-1520 [25] is the applicable mechanism.

4. RESULTS AND DISCUSSION

4.1. Overall Industry Risk Profile

The historical occurrences related to NFC facilities include a total of 35 documented incidents, resulting in 101 fatalities from the 1940s to the 2020s. These events are distributed across all four stages of the fuel cycle and span 16 different countries. The overall fatality rate obtained across the industry is 25.48 per 1,000 facility-years (90% CI: 19.61–35.12), based on 3,881 status-based facility-years (2024 reference year). The Monte Carlo simulation, with 250 iterations, yielded a mean fatality rate of 26.39 per 1,000 FY (σ = 5.02; Coefficient of Variation (CV) = 19.0%), confirming a reasonable precision given the 35-event sample size. The accident rate is 9.02 per 1,000 FY (90% CI: 6.20–13.03). The industry-wide severity index (i.e., mean fatalities per accident event) is 2.89, reflecting the mixture of single-fatality criticality accidents and multi-fatality catastrophic events in the historical record.

The Pearson correlation between accident and fatality counts, computed across all facility-type and temporal dimensions, is $r = 0.9958$, which supports the use of Poisson-based accident frequency as a reliable proxy for fatality risk throughout this study. The strong correlation indicates that the fatality rate is primarily driven by accident frequency instead of being influenced by event severity at the aggregate level, although reprocessing represents a clear exception, as discussed in Section 4.2.

4.2. Risk Hierarchy by Facility Type

Table 2 presents the full risk metric profile for each of the four NFC stages. The risk hierarchy is unambiguous and has direct consequences for co-siting evaluation.

As illustrated by Fig. 1, reprocessing exhibits the highest risk profile across all metrics, with a fatality rate of 148.33 per 1,000 FY, which is 10.7× higher than the next closest stage, alongside an accident rate of 30.81 per 1,000 FY, a severity index of 4.82, and a per-facility relative risk of 3.33× above the industry average. These findings align with the hazard profile discussed in Section 2.1, where the PUREX aqueous chemistry and high-level liquid waste containing long-lived cesium and strontium fission products create a radiological and chemical source term that significantly exceeds all other NFC stages. The Mayak (Chelyabinsk-65) explosion incident is the main contributor to the reprocessing aggregate and the Russian regional statistics noted in Section 4.4.

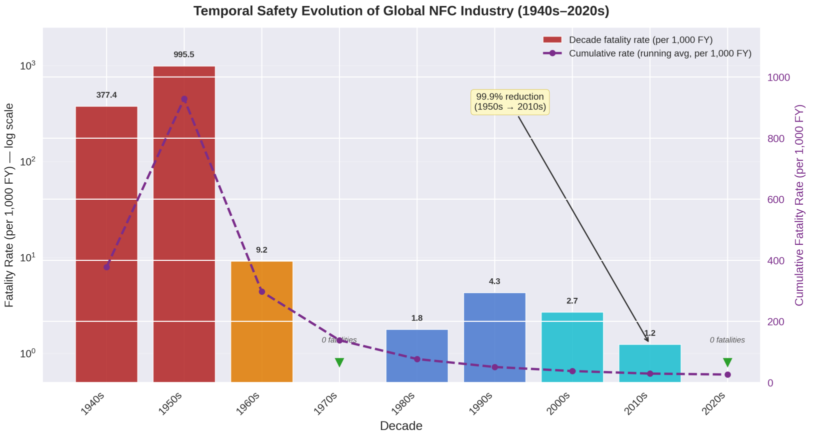

Fig. 2. Temporal safety evolution of the global NFC industry (1940s–2020s). Bars show the decade fatality rate per 1,000 FY (left axis, log scale); the dashed line shows the cumulative running-average fatality rate per 1,000 FY (right axis, linear scale). The 1970s and 2020s recorded zero fatalities (▼). The industry achieved a 99.9% reduction in fatality rate from the 1950s peak (995.5/1,000 FY) to the 2010s (1.2/1,000 FY).

The three remaining stages—Fuel fabrication (13.84/1,000 FY), Conversion (13.77/1,000 FY), and Enrichment (12.74/1,000 FY)—are statistically equivalent due to the uncertainty in accident counts, varying by less than 1.1 per 1,000 FY. Despite equivalent rates, per-facility relative risks differ (0.53 for fuel fabrication, 0.58 for conversion, and 0.66 for enrichment) because of variations in facility count and fatality distribution. Fuel fabrication's lower relative risk (0.53) arises from 25 fatalities across 86 facilities, leading to fewer fatalities per facility. Adjusted for operational exposure, these facilities have 1,806 FY, with a fatality rate of 13–14 per 1,000 FY, indicating similar risks to conversion and enrichment. This suggests that fuel fabrication is not inherently safer but shares the risk among more facilities. The risk pathways also differ: conversion and enrichment risks primarily stem from UF_6 chemical toxicity and pressurized inventory, whereas fuel fabrication risk is largely driven by criticality accidents, notably the JCO Tokaimura event (Japan, 1999), which contributed two of the 25 fatalities.

4.3. Temporal Safety Evolution

Figure 2 presents the temporal trajectory of NFC safety performance over 80 years. The 1950s represent the peak-risk decade, with 995.5 fatalities per 1,000 FY, accounting for 88 of the 101 total recorded fatalities (87.1%) across only 88.4 facility-years of operational experience. Two events are dominant: the Windscale Fire (UK, October 1957) and the Mayak (Chelyabinsk-65) explosion (USSR, September 1957). The preceding 1940s also demonstrate an elevated risk profile (377.4/1,000 FY), attributable to the comparatively less rigorous formal safety guidelines and organizational safety culture that characterized the early development of the nuclear fuel cycle industry.

The transition period from the 1950s to the 1960s represented the steepest single-decade improvement in the record: a reduction from 995.5 to 9.21 per 1,000 FY, a 99.1% reduction within one decade. This expressive reduction came along with the establishment of guidelines for nuclear safety governance, such as the IAEA Safety Fundamentals series, the creation of the U.S. AEC/NRC regulatory framework, and the formal adoption of criticality safety standards following the Y-12 Oak Ridge (1958) and Wood River Junction (1964) accidents. The 1970s achieved a completely clean decade across 362.6 facility-years with zero recorded fatalities.

The 1990s show a limited reversal to 4.32 per 1,000 FY, mainly attributable to the JCO Tokaimura criticality accident in Japan. The 2000s (2.71/1,000 FY) and 2010s (1.25/1,000 FY) continue the downward trend. At the present moment, the 2020s have recorded zero fatalities. From the 1950s peak to the 2010s, the global NFC industry has achieved a 99.9% reduction in fatality rate — a performance trajectory that closely tracks the maturation of the modern nuclear regulatory state.

4.4. Regional Risk Distribution

Figure 3 presents the regional distribution of fatality risk, revealing a strong correspondence between regulatory maturity and risk outcomes. Russia/USSR registers the highest regional rate at 76.06 per 1,000 FY, representing 54.5% of all recorded global NFC fatalities across 723.1 database-consistent facility-years; the Mayak (Chelyabinsk-65) explosion dominates, with Cold War-era industrial safety practices contributing secondary events. The UK follows at 60.32 per 1,000 FY — driven entirely by the Windscale Fire, which accounts for an estimated 33 radiation-linked fatalities from 547.1 UK facility-years. Together, the Soviet reprocessing incidents and the UK Windscale event account for 87.1% of all recorded NFC fatalities while representing only 32.7% of the database facility-years, underscoring the degree to which the aggregate global risk statistics are dominated by two catastrophic events from a single decade (1957).

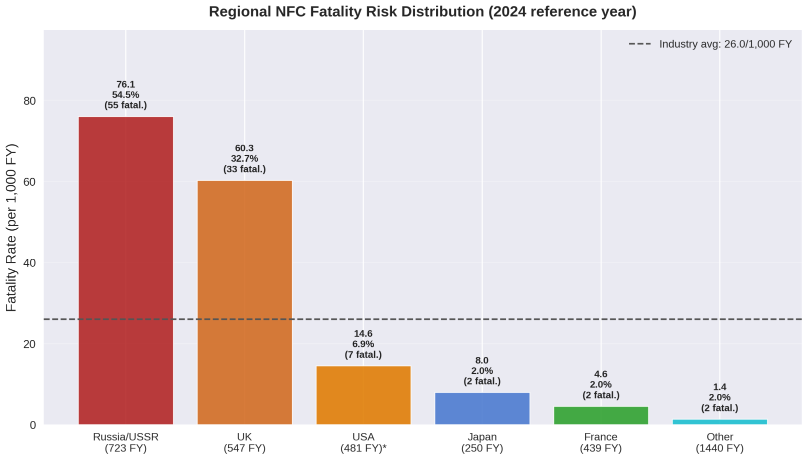

Fig. 3. Regional nuclear fuel cycle fatality risk distribution (2024 reference year). Bar height indicates the fatality rate per 1,000 facility-years (FY), with annotations for rate value, percentage of total fatalities, and absolute counts. Bar colors represent regulatory maturity (firebrick: Soviet-era; chocolate: legacy UK; amber: U.S.; blue-green: mature regimes). The dashed line shows the industry average (26.0/1,000 FY). *See Section 4.6 for U.S. rate caveat.

The United States records 14.56 per 1,000 FY across 480.9 database-consistent facility-years, the third-highest regional rate. Such an outcome reflects the methodology's treatment of fully decommissioned facilities (status factor = 0.0), which excludes many completed U.S. operational periods; the rate should therefore be interpreted as a lower bound on historical U.S. exposure (see Section 4.6). Japan (7.98/1,000 FY) reflects JCO Tokaimura as its primary historical driver across 250.5 facility-years. France (4.56/1,000 FY) and the remaining countries collectively (1.39/1,000 FY across 1,440.2 FY) demonstrate the lowest rates, consistent with well-developed regulatory regimes (ASN, EURATOM, ONR) and a large and growing international NFC base. The regional results inform co-siting site selection: French, Japanese, and other well-regulated environments carry the lowest empirical NFC baseline risks, with the caveat that U.S. rates are subject to the methodological limitation noted above.

4.5. Co-Siting Risk Matrix: Stage-Specific Implications

Combining the empirical risk hierarchy from Table 2, the hazard profiles from Section 2.1, and the separation distance criteria from Section 3.5, Table 3 presents the Co-Siting Risk Matrix. This matrix illustrates the link between the empirical NFC risk hierarchy and the regulatory framework for co-siting. Fuel fabrication sites emerge as the most viable entry point due to the lowest historical fatality rate, the highest global facility count (46.7% of all NFC facilities), and a defined regulatory mechanism, namely the ISA update under NUREG-1520, supplemented by a criticality safety evaluation for UO_2 powder accumulation near reactor ventilation intakes [25].

Enrichment sites are the second most suitable option for co-siting, necessitating two analyses: a common-cause utility-failure assessment (evaluating scenarios where an external event like a grid outage disables both the enrichment plant and the co-located MMR) and a UF_6 plume consequence evaluation, as per the NUREG-1827 Urenco USA SER framework, the main regulatory precedent for U.S. centrifuge enrichment facilities [4]. Conversely, conversion sites need an additional HF gas dispersion analysis, and their SMR/MMR control room ventilation must filter up to 30 ppm IDLH for the maximum UF_6 release scenario [26].

Finally, reprocessing sites demand a specific analytical approach, the combined radiological source term from PUREX high-level liquid waste and the reactor coolant system, potential domino failure sequences, and the requirement to demonstrate 10 CFR 70.61 high-consequence event frequencies at or below 1×10^{-6}/year before co-siting approval, deferred to the funded follow-on scope described in Section 4.6.

4.6. Study Limitations

Three major limitations are acknowledged in this methodology. First, the 58.7% missing end-date rate is the primary source of uncertainty in the dataset. Second, Cold War-era Soviet incident data is demonstrably incomplete, and near-miss events are excluded globally; the true pre-1970 accident frequency is likely underestimated. Third, the Mechanism 1 status factor of 0.0 for fully decommissioned facilities contributes no facility-years to the denominator, while any associated fatalities continue to be counted in the numerator.

Table 3. Safety Parameters and Regulatory Requirements by Nuclear Fuel Cycle (NFC) Stage

NFC Stage	Fatality Rate (per 1,000 FY)	Primary Hazard Mechanism	Key Co-Siting Concern	Minimum Separation	Required Regulatory Action
Conversion	13.77	UF_6 + HF chemistry; fire	HF toxic plume to MMR control room (1–2 km worst-case range)	400–1,700 m (inventory dependent)	Toxic gas dispersion analysis; filtered air supply
Enrichment	12.74	UF_6 pressurized; electrical load	Shared utility failure; UF_6 plume near reactor	≥400 m from UF_6 systems	10 CFR 70.64(b) analysis; pipeline frequency
Fuel Fabrication	13.84	UO_2 powder handling; criticality	Combined criticality; UO_2 dust near intake	Brownfield viable (controlled area)	ISA update; criticality safety evaluation
Reprocessing	148.33	PUREX HLW; criticality; Cs/Sr	Combined radiological source term; domino failures	Site-specific; full PSA required	10 CFR 70.61 demonstration ($\leq 10^{-6}$/yr)

This results in an inflated computed rate; therefore, the U.S. regional rate of 14.56 per 1,000 FY represents an upper bound, as crediting the full operational periods of these decommissioned sites would yield a larger FY base and a lower computed rate.

5. CONCLUSIONS AND FUTURE WORK

The rising demand for thermal and electrical power in sectors like nuclear fuel cycle facilities is accelerating the adoption of localized generation through SMRs and MMRs. This study makes four key contributions to risk assessment for co-siting SMRs/MMRs with nuclear fuel cycle facilities: 1) a global NFC risk database of 184 facilities across 16 countries with over 80 years of operational history; 2) a dual facility-years methodology with Monte Carlo uncertainty quantification; 3) a multi-dimensional risk hierarchy by facility type, time period, and region; and 4) stage-specific hazard profiles along with a co-siting risk matrix.

This paper highlights key findings regarding historical risks in the nuclear fuel cycle (NFC) industry, noting an overall fatality rate of 26.02 per 1,000 facility-years, a drastic 99.9% reduction from the 1950s to the 2010s, and zero fatalities recorded in the 2020s. This trend indicates that modern NFC facilities have achieved a risk level suitable for co-siting, though stage-specific hazard interactions, especially at reprocessing sites, necessitate a dedicated probabilistic risk assessment prior to deployment. This reflects the efficacy of contemporary regulatory frameworks in improving operational safety within the NFC sector. Existing regulations, including 10 CFR 70.22 and international guidelines like IAEA SSR-4, will underpin upcoming co-siting projects. Future research will examine the interactions between small modular/micro modular reactors and NFC facilities, using classical probabilistic safety assessment methods to assess potential accident scenarios.

CONFLICT OF INTEREST DECLARATION

The authors declare that they have no conflicts of interest regarding the subject matter or materials discussed in this manuscript.

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REFERENCES

[1] U.S. Nuclear Regulatory Commission (U.S. NRC), Nuclear fuel cycle facility accident analysis handbook, Tech. Rep. NUREG/CR-6410, Science Applications International Corporation (SAIC), Washington, DC (March 1998).

[2] U.S. Nuclear Regulatory Commission (U.S. NRC), Release of UF6 from a ruptured model 48y cylinder at sequoyah fuels corporation, Tech. Rep. NUREG-1198, U.S. Nuclear Regulatory Commission (1986).

[3] U.S. Nuclear Regulatory Commission (U.S. NRC), Honeywell metropolis works HF release, Tech. Rep. Inspection Report 40-3392/2014-005, U.S. Nuclear Regulatory Commission (January 30 2015).

[4] U.S. Nuclear Regulatory Commission (U.S. NRC), Safety evaluation report for the national enrichment facility, lea county, NM, Tech. Rep. NUREG-1827, U.S. Nuclear Regulatory Commission (June 2005).

[5] T. P. McLaughlin, S. P. Monahan, N. L. Pruvost, V. V. Frolov, B. G. Ryazanov, V. I. Sviridov, A review of criticality accidents: 2000 revision, Tech. Rep. LA-13638, Los Alamos National Laboratory (LANL) (May 2000).

[6] International Atomic Energy Agency (IAEA), Report on the preliminary fact finding mission following the accident at the nuclear fuel processing facility in tokaimura, japan, Tech. rep., IAEA, Vienna (1999).

[7] U.S. Nuclear Regulatory Commission (U.S. NRC), Augmented inspection team report, westinghouse columbia fuel fabrication facility, Tech. Rep. NRC Inspection Report 70-1151/2016-007, U.S. Nuclear Regulatory Commission (September 2016).

[8] B. V. Nikipelov, G. N. Romanov, L. A. Buldakov, N. Babaev, Y. B. Kholina, E. I. Mikerin, A radiation accident in the southern Urals in 1957, At Energy 67 (1989) 569–576.

[9] Bruce Power, Guide to bruce power (Report No. 160072, Rev. R003) and bruce power fact sheets (Report No. 230188, Rev. R001), Tech. rep., Bruce Power (2023).

[10] ASN / Orano, Tricastin site information report, Tech. rep., ASN / Orano (2019).

[11] U.S. Department of Energy, Finding of No Significant Impact for the Paducah Gaseous Diffusion Plant Final Environmental Assessment for Disposition of Waste and Materials (DOE/EA-2116), Tech. rep., U.S. Department of Energy (2020).

[12] U.S. Nuclear Regulatory Commission (U.S. NRC), NuScale Small Modular Reactor Design Certification, Federal Register Doc. 2023-00729 (March 2 2023).

[13] Z. Zhang, Y. Dong, F. Li, X. Huang, Y. Zheng, Z. Dong, H. Zhang, Z. Chen, X. Li, Loss-of-cooling tests to verify inherent safety feature in the world's first HTR-PM nuclear power plant, Joule 8 (2024).

[14] H. Planchon, J. Sackett, G. Golden, R. Sevy, Implications of the EBR-II inherent safety demonstration test, Nuclear Engineering and Design 101 (1987) 75–90.

[15] U.S. Nuclear Regulatory Commission (U.S. NRC), eVinci, Tech. Rep. NRC Docket 99902079, U.S. Nuclear Regulatory Commission (2026).

[16] U.S. Nuclear Regulatory Commission (U.S. NRC), Regulatory Guide 1.233, Rev. 0 (finalized from DG-1353), Tech. rep., U.S. Nuclear Regulatory Commission (2020).

[17] International Atomic Energy Agency, Nuclear Fuel Cycle Facilities Database (NFCFDB), `https://infcis.iaea.org/NFCFDB/`, accessed: November 15, 2025 (2026).

[18] U.S. Nuclear Regulatory Commission (U.S. NRC), Event Notification Reports, `https://www.nrc.gov/reading-rm/doc-collections/event-status/event/index`, accessed: November 10, 2025 (2026).

[19] Code of Federal Regulations, 10 CFR 70.22(i) and 10 CFR 70.61, U.S. Government Publishing Office (2016).

[20] International Atomic Energy Agency (IAEA), Safety of nuclear fuel cycle facilities, Tech. Rep. SSR-4, IAEA, Vienna (2017).

[21] International Atomic Energy Agency (IAEA), Safety aspects in siting for nuclear installations, Tech. Rep. SSG-35, IAEA, Vienna (2015).

[22] U.S. Nuclear Regulatory Commission (U.S. NRC), Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors, Tech. Rep. DG-1353, U.S. Nuclear Regulatory Commission (April 2019).

[23] U.S. Nuclear Regulatory Commission (U.S. NRC), Technical Basis for Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", Tech. Rep. NUREG/CR-2260, U.S. Nuclear Regulatory Commission (October 1981).

[24] Y. Shen, W. Zhou, A comparison of onshore oil and gas transmission pipeline incident statistics in Canada and the United States, International Journal of Critical Infrastructure Protection 45 (2024).

[25] U.S. Nuclear Regulatory Commission (U.S. NRC), Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, Tech. Rep. NUREG-1520, Rev. 2, U.S. Nuclear Regulatory Commission (May 2015).

[26] U.S. Nuclear Regulatory Commission (U.S. NRC), Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation, Tech. Rep. NUREG-1391, U.S. Nuclear Regulatory Commission (February 1991).