Toward a Generic PRA Framework for Sodium Fast Reactors: A Comparative Review of Four Pool-Type SFR Designs

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Abstract: With renewed interest in sodium fast reactors (SFRs) and several designs now in licensing and construction, a systematic comparison of probabilistic risk assessment (PRA) features across SFR designs is valuable to both the technical community and PRA practitioners. This paper reviews four pool-type SFR reference designs: the Experimental Breeder Reactor II (EBR-II), the Power Reactor Inherently Safe Module (PRISM), ARC-100, and Natrium. It compares technical parameters, system architectures, safety features, and initiating events, identifying common characteristics that simplify PRA modeling relative to light-water reactors and key differences that require design-specific treatment. Based on the comparison, the paper proposes a generic design envelope, a safety-function structure, generic initiating-event families, and an event-sequence framework organized around protected, unprotected, and radionuclide-barrier response, with configurable modules for design-specific features. This structure is the foundation for a future generic pool-type SFR PRA model, developed in conformance with ASME/ANS RA-S-1.4-2021 and consistent with the Licensing Modernization Project methodology.

1. INTRODUCTION

Sodium fast reactors are being pursued for commercial power, demonstration plants, and test reactor applications. PRA for these designs cannot be developed by direct transfer of light-water-reactor (LWR) event trees and fault trees. Pool-type SFRs have low-pressure primary systems, sodium coolant, metallic alloy fuel, strong thermal feedback mechanisms, large primary sodium inventory, and heat-removal paths that may rely on natural circulation and passive heat rejection. These features change the initiating-event spectrum, accident progression logic, and interpretation of risk metrics.

A generic SFR PRA model addresses several near-term needs. As candidate heat and power sources for co-located industrial users, SFRs require quantitative risk characterization before siting and configuration can be optimized within an integrated energy system, and a generic model provides the risk foundation for that systems-engineering work. An independent generic model also gives regulators a reference point for reviewing and benchmarking vendor-submitted PRAs. Risk-informed, performance-based licensing pathways for non-light-water reactors favor an early PRA basis, but many SFR developers do not yet have a mature plant-specific model, so a validated generic model lowers that barrier. This paper develops the comparative framework that such a model requires; building the quantified generic PRA model from that framework is the next step.

SFR PRA has precedent. The EBR-II Level 1 PRA provided an operating-reactor treatment that credited inherent feedback and passive heat removal [1], and the PRISM and Advanced Liquid Metal Reactor (ALMR) safety documentation was later reviewed by the U.S. Nuclear Regulatory Commission (NRC) [2]. A review of U.S. SFR PRA experience found that these early studies predate current technology-inclusive methods [3], while the Versatile Test Reactor PRA recently demonstrated a Licensing Modernization Project-aligned PRA for a pool-type metallic-fuel SFR [4]. The Generation IV International Forum (GIF) has also published a generic SFR system safety assessment approach [5]. This paper extends the comparison to two current commercial designs, ARC-100 and Natrium, and organizes the result as a configurable framework rather than a retrospective review.

This paper develops a comparative basis for a generic pool-type SFR PRA framework using four reference designs: EBR-II [1], PRISM, ARC-100, and Natrium. PRISM provides a Preliminary Safety Information Document (PSID), NRC preapplication review, and recent internal-events PRA material [2, 6-9]; ARC-100 and Natrium are covered by current public design documentation [10-12]. The result is a framework for future generic model development, not a completed quantified PRA.

The advanced non-LWR PRA standard defines the technical elements a complete PRA must satisfy. This paper does not restate those requirements. Its contribution is the pool-type SFR content needed to populate them, namely the generic design envelope, the initiating-event families, the safety-function structure, and the configurable-module logic, organized so that the generic structure can be assembled before plant-specific data exist [13].

The framework also treats end-state definitions explicitly. Core damage frequency (CDF) remains an established PRA metric, but CDF alone is not sufficient for a generic non-LWR risk framework. Advanced reactor licensing guidance and Licensing Modernization Project (LMP) applications use event frequency, mechanistic source term, radionuclide barrier performance, and consequence measures to classify licensing-basis events and safety-significant structures, systems, and components; Section 7 develops the end states used here [14-16].

2. SCOPE AND METHOD

The review is limited to pool-type SFRs. The framework does not cover loop-type SFRs without modification, and it does not cover gas-cooled fast reactors, molten-salt reactors, or lead-cooled fast reactors. The four designs were selected to span maturity and design variation: EBR-II as an operating experimental reactor with a published Level 1 PRA, PRISM as a modular commercial design with both preliminary safety documentation and recent PRA publications, ARC-100 as a current small modular SFR concept, and Natrium as a current pool-type SFR design with a molten-salt energy-storage interface. The available source basis is strongest for EBR-II and PRISM PRA structure, strong for ARC-100 and Natrium design facts, and limited for public ARC-100 and Natrium event-tree or fault-tree detail.

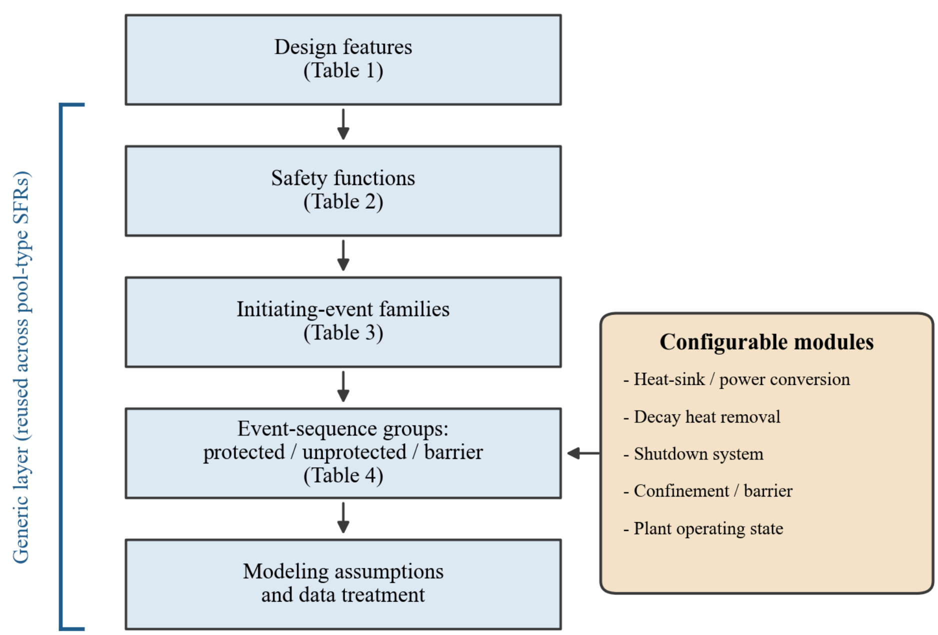

Figure 1: Method flow from design features to a configurable event-sequence framework

The review method has five steps. First, PRA-relevant design features are extracted from available design, safety, and PRA documents. These features include reactor configuration, fuel form, coolant and heat-transport architecture, shutdown systems, decay heat removal (DHR), confinement or containment approach, and power conversion interface. Second, the design features are mapped to safety functions and modeled systems. Third, initiating-event categories from available EBR-II and PRISM PRA sources are compared with design-specific features from ARC-100 and Natrium. Fourth, the initiating-event families are translated into a generic event-sequence structure. Fifth, the review identifies assumptions and data-treatment needs for a future quantified model. Figure 1 shows this flow and the relationship between the generic layer and the configurable modules.

The standards context for the framework is advanced non-LWR PRA practice and risk-informed, performance-based licensing guidance. The advanced non-LWR PRA standard provides a technology-inclusive structure for PRA technical elements, including initiating events, event sequence analysis, success criteria, systems analysis, data, human reliability analysis, common-cause failure, uncertainty, and risk integration [13, 16]; the NRC endorsed this standard for non-LWR risk-informed activities in Regulatory Guide 1.247 [17]. The LMP, documented in Nuclear Energy Institute (NEI) report 18-04 and endorsed by NRC Regulatory Guide (RG) 1.233, uses a frequency-consequence process for licensing-basis event classification, safety classification, and defense-in-depth evaluation [14-16]. The proposed framework is intended to support these evaluations by retaining both sequence frequency and release/consequence information.

In 2026 the NRC issued 10 CFR (Title 10 of the Code of Federal Regulations) Part 53, a risk-informed, technology-inclusive licensing framework for advanced reactors that allows a PRA-based risk evaluation; the framework proposed here is intended to support either the LMP route under 10 CFR Parts 50 and 52 or the new Part 53 pathway [18]. The Generation IV International Forum sodium-cooled fast reactor system safety assessment provides a complementary deterministic and probabilistic structure for SFR safety evaluation and is used here as a reference point for technology-level safety functions [5].

3. COMPARATIVE DESIGN BASIS

The four reference designs share the pool-type SFR concept, but they are not interchangeable. Common features include liquid sodium coolant, low primary-system pressure, in-vessel primary heat transport, metallic alloy fuel, and reliance on thermal inertia and passive or inherent response for some accident conditions. Differences include heat-sink interface, decay heat removal implementation, pump technology, shutdown system design, containment or functional containment strategy, and available PRA detail. Three related terms are used in this paper as follows: confinement refers to a low-leakage boundary, containment to a structure that retains pressure and limits leakage, and functional containment to the barrier-set concept of NEI 18-04 and RG 1.233, in which multiple barriers limit release without necessarily relying on a single pressure-retaining building [14-15]. The framework models radionuclide-barrier performance regardless of which strategy a design adopts.

Table 1 summarizes the design comparison at the level needed for a generic PRA framework. The designs span a wide power range, about 60 to 840 MWt, which makes decay-heat load, source-term inventory, and mission times design-specific parameters rather than generic constants. Values are intentionally limited to those supported by the source set used in this draft. For PRISM, the table separates the earlier NRC preapplication design basis from the later GE Hitachi design used in recent PRISM PRA publications, because the design vintage and power values are not identical.

The comparison leads to four modeling implications. First, the pool-type primary system changes the loss-of-coolant accident (LOCA) model. A large break in external primary piping, which is central for many LWR PRAs, does not apply in the same way to an in-vessel primary system. The generic framework should still include primary sodium boundary leaks, vessel leaks, cover-gas interactions, and inventory-retention functions, but the event definition should be specific to pool-type SFR geometry.

Table 1: PRA-relevant design comparison of the four reference designs

Feature	EBR-II	PRISM	ARC-100	Natrium
Power level	62.5 MWt, about 20 MWe	471 MWt per module (NUREG-1368); 840 MWt / 311 MWe in the later GE Hitachi design	286 MWt and 100 MWe	840 MWt
Fuel	Metallic U-Zr / U-Pu-Zr fuel	Metallic U-Pu-Zr fuel	Metallic U-10Zr fuel	Metallic U-10Zr high-assay low-enriched uranium (HALEU) fuel
Primary coolant and geometry	Liquid sodium, pool-type primary system	Liquid sodium, pool-type primary system	Liquid sodium, pool-type primary system	Liquid sodium, pool-type configuration
Heat transport and power conversion	Primary sodium, intermediate heat transport, steam/water power conversion	Intermediate sodium loop and sodium-water steam generator feeding a conventional turbine	Intermediate heat transport and conventional power conversion features in the design overview	Heat for steam and electric generation is provided through molten-salt storage tanks and is largely independent of the reactor
Decay heat removal and heat rejection	Existing plant heat-removal systems and large sodium thermal inertia; EBR-II PRA includes loss of heat sink and shutdown heat removal events	Passive safety features including reactor vessel auxiliary cooling system (RVACS) and auxiliary cooling system (ACS)	Large thermal inertia, natural circulation, guard vessel, and passive heat-removal features	Functional-containment architecture with Reactor Air Cooling (RAC) and Intermediate Air Cooling (IAC) residual heat-removal systems
Confinement or containment strategy	EBR-II plant-specific containment and release barriers	Containment vessel (guard vessel) with top dome and PRISM-specific radionuclide barrier treatment	Guard vessel and design-specific containment/confinement features	Functional containment
Public PRA support	Published Level 1 PRA with internal events, hazards, shutdown/refueling, event trees, and fault trees	Published internal-events PRA summary	Public design source supports feature comparison	Public NRC application summary and PSAR support design facts

Note. EBR-II values are from the EBR-II PRA summary report and the full Level 1 PRA report [1, 19]; PRISM from the PSID, NUREG-1368, and PRA sources [2, 7-9]; ARC-100 from the ARC-100 Technical Overview [10]; Natrium from the NRC Kemmerer application page and PSAR [11-12]. ARC-100 and Natrium support publicly documented design-feature comparison only; detailed event-tree and fault-tree claims are not made for those designs.

Second, sodium coolant changes both the heat-removal and hazard models. Sodium has high thermal conductivity and a boiling temperature well above the operating temperature, which provides margin against coolant boiling. Sodium also reacts with air and water, so internal fire, sodium leak, sodium-water reaction, and interface-leak events require explicit screening. These hazards should not be assumed to dominate every SFR PRA. Their contribution depends on plant layout, sodium inventory outside the vessel, leak detection, isolation, ventilation, fire protection, and site-specific hazard screening.

Third, metallic fuel and inherent feedback affect accident progression. Negative reactivity feedback, thermal expansion, Doppler feedback, and natural circulation can contribute to a benign response for some unprotected transients and are treated in the EBR-II and PRISM safety and PRA material. These effects should not be represented as simple hardware success states. They require success criteria and phenomenological analysis that connect reactor physics, thermal hydraulics, and damage thresholds.

Fourth, the power-conversion interface is not generic across the selected designs. EBR-II, PRISM, and ARC-100 include sodium-to-steam interfaces, while Natrium introduces a molten-salt energy-storage interface between the nuclear heat source and steam/electric generation. The Natrium PSAR identifies sodium-salt heat exchangers and a nuclear-island salt system as part of this architecture [12]. The molten-salt storage also buffers electrical output, letting the plant raise generation for limited periods independently of reactor thermal power. For PRA, this means that sodium-water reaction is not a universal generic event. It is a configurable interface hazard. Natrium-like designs require separate treatment of sodium-salt and salt-side heat-transfer interfaces when the detailed plant source supports those features.

4. GENERIC DESIGN ENVELOPE AND CONFIGURABLE MODULES

The comparison defines a generic design envelope for pool-type SFR PRA. The envelope includes features that define the reactor class and influence most initiating-event and event-sequence families:

- Pool-type primary system with the core, primary sodium inventory, primary pumps or equivalent circulation features, and intermediate heat exchangers (IHX) inside or closely coupled to the vessel.
- Liquid sodium coolant operating at low pressure relative to LWR primary systems.
- Metallic alloy fuel with design-specific composition, qualification basis, and thermal response.
- In-vessel primary heat transport with large thermal inertia.
- Intermediate heat transport or an equivalent nonradioactive heat-transfer interface.
- Passive, inherent, or naturally driven heat-removal capability for selected accident sequences.
- Guard vessel, functional containment, or an equivalent inventory-retention and radionuclide-barrier strategy.

The generic envelope should not force all reference designs into a single hardware configuration. It should provide common event-sequence logic with configurable modules for primary heat transport, reactivity control, intermediate heat transport, heat sink and power conversion, decay heat removal, confinement or release barriers, and plant operating state (POS). This structure avoids false precision for systems that differ materially between designs. For example, a generic "decay heat removal succeeds" top event may be used in early event-tree synthesis, but the reliability model under that top event must be built from the selected design's heat-removal architecture.

A simple rule separates the two layers. An element is treated as generic when the event-sequence structure, meaning the set of safety functions challenged and the order of top events, is invariant across pool-type SFRs even if the supporting hardware and the numerical values differ. An element is treated as a configurable module when the structure itself changes, for example when an added molten-salt interface introduces new initiating events and new branch points. By this rule, reactivity control, primary heat transport, and decay heat removal are generic at the function level but configurable at the reliability-model level, while the heat-sink and power-conversion interface is configurable at the structural level because it changes which initiating events and barrier challenges exist. The same rule is applied to the initiating-event families in Section 6 and the event-sequence groups in Section 7.

5. PRA-RELEVANT SAFETY FUNCTIONS AND SYSTEMS

The generic framework is organized around safety functions rather than a fixed list of systems. A safety-function structure fits advanced reactors because passive features, inherent response, and design-specific heat sinks may satisfy the same function in different ways. For the selected pool-type SFRs, the main PRA-relevant safety functions are:

- Control reactivity and shut down the reactor.
- Maintain adequate core cooling during normal operation, transients, and post-trip conditions.
- Remove decay heat to an ultimate heat sink.
- Maintain sodium inventory and primary boundary integrity.
- Limit sodium chemical reaction consequences.
- Retain radionuclides and control release paths.

Functions 1-3 and 6 are the fundamental reactor safety functions of reactivity control, heat removal, and radionuclide confinement reflected in the advanced non-LWR PRA standard and the GIF sodium-cooled fast reactor safety design criteria [13, 20]. Functions 4 and 5 are SFR-specific extensions that follow from the low-pressure sodium boundary and the sodium chemical-reaction hazards compared in Section 3.

Table 2 maps these functions to the systems or phenomena that should be represented in the PRA framework.

Table 2: Safety functions and modeling treatment

Safety function	**Representative systems or phenomena**	**Suggested PRA treatment**
Reactor trip and shutdown	Reactor protection system, trip sensors, logic, actuation, primary shutdown system, diverse shutdown system	Fault-tree model for hardware and instrumentation and control where design detail exists; anticipated transient without scram (ATWS) branch for failure to trip or insert sufficient negative reactivity
Inherent reactivity response	Doppler feedback, thermal expansion, coolant density effects, structural expansion, fuel and core restraint response	Success-criteria input from physics and thermal-hydraulic analysis; not modeled as a simple component basic event
Primary core cooling	Primary pumps, pump coastdown, natural circulation, core flow paths, sodium inventory	Event-tree top events for flow adequacy and fault-tree or phenomenological model for pump/coastdown/natural-circulation response
Intermediate heat removal	IHX, intermediate sodium pumps, isolation valves, heat-transfer path	Configurable fault-tree module; interface-specific initiating events for loop leaks or heat-transfer loss
Decay heat removal	RVACS, direct reactor auxiliary cooling system (DRACS), ACS, shutdown coolers, RAC/IAC, air heat exchangers, natural draft, active blower modes	Separate active, passive, and natural-circulation success criteria; uncertainty treatment for passive-system performance
Sodium hazard control	Leak detection, inerting, ventilation isolation, fire suppression, drainage, separation	Internal hazard event screening and detailed modeling where not screened; conditional impact on systems and release barriers
Radionuclide retention	Fuel retention, sodium pool retention, cover gas, reactor vessel, guard vessel, containment or functional containment	Confinement/release event tree or source-term binning connected to mechanistic source term analysis

Note. The safety-function structure follows the advanced non-LWR PRA standard and the GIF SFR safety design criteria [13, 20]. The sodium inventory and boundary function and the sodium chemical-reaction function reflect SFR features documented in the EBR-II and PRISM safety and PRA sources [1-2, 7].

This organization separates passive safety from the conditions needed to credit it. Passive or inherent response can reduce reliance on operator action and active equipment for some sequences, but the PRA should still model the conditions under which that response is credited. Passive heat removal depends on heat-transfer area, air path availability, sodium inventory, temperature differences, damper or valve positions where applicable, and long-term heat sink conditions. Inherent feedback depends on core state, power level, reactivity coefficients, and thermal-mechanical behavior. These are success-criteria issues, not generic assumptions that can be applied without analysis.

Human reliability analysis (HRA) remains needed even when the design uses passive features. Operator actions may affect diagnosis, manual backup shutdown, recovery of heat-removal paths, isolation of sodium leaks, transition between heat-removal modes, fire response, and shutdown/refueling operations. The generic framework should include HRA as a normal PRA technical element, while recognizing that the relevant human actions may differ from those in LWR PRAs.

6. INITIATING-EVENT SYNTHESIS

The initiating-event taxonomy should start from SFR-specific plant response rather than from LWR event labels. EBR-II and PRISM sources provide the strongest available basis for event-sequence examples. EBR-II PRA includes loss of flow, loss of heat sink, reactivity insertion or transient overpower, overcooling, local faults, shutdown and refueling events, internal fires, and seismic events [1, 19]. PRISM PSID and PRA sources organize preliminary and recent internal-event modeling around system, core-response, containment-response, protected, and unprotected sequence logic [2, 7-8].

Table 3 gives a generic initiating-event (IE) taxonomy for the framework. The table is intended for model development and screening. It is not a claim that every event family is risk-significant in every design.

Table 3: Generic initiating-event families for pool-type SFR PRA

IE family	Generic event description	Design-specific contributors	Modeling note
Loss of primary flow	Loss or degradation of forced primary circulation	Mechanical pump trip, electromagnetic pump failure, pump power loss, flow blockage, pump coastdown characteristics	Branch on protected shutdown, coastdown/natural circulation, and heat-removal success
Loss of heat sink	Loss of normal or shutdown heat rejection	Steam generator isolation, intermediate loop failure, sodium-air heat exchanger impairment, salt-side heat-removal failure, balance-of-plant trip	Heat sink definitions differ by design; Natrium-like architecture should be modeled separately from sodium-water designs
Transient overpower or reactivity insertion	Positive reactivity insertion, control rod withdrawal, loading error, feedback anomaly	Control system failure, rod motion error, fuel handling/loading error, experimental configuration	ATWS and inherent-feedback treatment require physics-based success criteria
Station blackout and support-system failures	Loss of offsite power, loss of AC/DC support, loss of control power, loss of instrument air where used	Site electrical design, battery capacity, passive DHR independence, digital I&C architecture	May be less severe where passive DHR is independent, but still affects monitoring, actuation, valves, dampers, and recovery
Primary sodium boundary leak or inventory loss	Sodium leakage within or near primary boundary, vessel leak, cover-gas or pool-level challenge	Vessel/guard vessel design, penetrations, sodium level instrumentation, leak	Large external primary-piping LOCA should not be imported from LWR models; pool-specific leak

		detection	scenarios are needed
Intermediate heat transport failure	Intermediate sodium loop leak, isolation, pump failure, IHX heat-transfer degradation	Number of loops, secondary sodium inventory, sodium-air or sodium-water interfaces	May be an initiating event or a failed mitigating function, depending on sequence definition
Interface reaction or leak	Reaction or leak at sodium-water, sodium-salt, or salt-steam interface	Steam generator tube rupture, sodium-salt heat exchanger leak, salt-to-steam generator fault	Configurable by plant; sodium-water reaction is not universal to all pool-type SFRs
Local faults	Local blockage, assembly fault, local overpower, fuel damage in limited region	Core design, fuel handling, surveillance, blockage detection	May not produce plant-wide transient initially; detection and propagation assumptions are important
Shutdown, refueling, and fuel handling events	Events during non-power states, maintenance, refueling, or fuel transfer	Refueling interval, sodium level, decay heat level, system availability, temporary configurations	Often requires a separate operating-state model; EBR-II PRA provides precedent
Internal hazards	Fire, sodium fire, flooding, internal missiles, heavy load drop	Sodium inventory outside vessel, plant layout, fire zones, ventilation, drainage, protection systems	Screen first, then model where potential impact on multiple safety functions exists
External hazards	Seismic, high wind, external flood, other site-specific hazards	Site hazard curves, seismic fragility, building layout, sodium systems, heat sink	Seismic should be treated by site-specific hazard and fragility analysis, not by generic dominance assumptions

Note. The initiating-event families are synthesized from the EBR-II Level 1 PRA and the PRISM PSID and PRA sources [1-2, 7-8] and organized using the technical elements of the advanced non-LWR PRA standard [13].

The taxonomy indicates that several event families are common, but their detailed definitions differ. A "loss of heat sink" in PRISM is not identical to a loss of heat sink in Natrium because the heat-transfer path and balance-of-plant dependencies differ. Similarly, sodium-water reaction applies to steam-generator designs but should not be treated as a universal SFR event. Shutdown and refueling exposure also varies widely, from a one-to-two-year refueling interval for PRISM to roughly a 20-year core interval for ARC-100, which changes the contribution of shutdown and fuel-handling events. These differences motivate a generic framework with configurable modules.

The taxonomy also helps avoid double counting. Intermediate loop failures, sodium leaks, heat sink loss, and balance-of-plant trips can appear as initiating events or as mitigating-system failures. The model boundary should define the initiating event as the first plant-level disturbance and then treat subsequent equipment failures or operator actions as event-tree branches or fault-tree contributors. A future quantified model should document these boundary decisions in an assumptions log.

7. EVENT-SEQUENCE MODELING APPROACH

The event-sequence framework should preserve SFR-specific plant response while remaining simple enough for a generic model. A practical structure is to organize event trees into three related groups: protected sequences, unprotected sequences, and confinement or radionuclide-barrier sequences.

The framework defines end states by barrier performance rather than by a single core-damage threshold. For metallic-fuel pool-type SFRs the analog to LWR core damage is the onset of cladding breach and fuel melting, but the risk-significant end states are release categories defined by how much radionuclide inventory escapes the fuel, the sodium pool, the cover gas, the reactor vessel and guard vessel, and the containment or functional containment. End states are therefore binned by release magnitude and mapped to the LMP licensing-basis-event classes and the frequency-consequence target [14-15].

Protected sequences begin with an initiating event followed by successful reactor trip and shutdown insertion. The main questions are whether primary flow or natural circulation remains adequate, whether decay heat removal is established, whether sodium inventory and intermediate heat transport remain available, and whether the sequence reaches a stable shutdown condition. The top events for a protected loss-of-flow sequence, for example, would include reactor trip, shutdown insertion, primary pump coastdown or natural circulation, decay heat removal, and long-term heat sink availability.

Unprotected sequences include failure of the reactor protection or shutdown function, or conditions in which shutdown is delayed or incomplete. These sequences need separate treatment because inherent feedback and passive response may strongly affect the outcome. For unprotected loss of flow, unprotected loss of heat sink, or unprotected transient overpower, the event tree should not stop at "trip failure equals core damage." It should branch through physics-based success criteria for power reduction, feedback response, heat removal, and damage thresholds. This treatment is consistent with PRISM PRA practice, where protected and unprotected sequence groups are evaluated separately.

Confinement or radionuclide-barrier sequences are needed when fuel damage, sodium leakage, or another source-term challenge occurs. These sequences connect plant damage states to mechanistic source term and consequence analysis. Mechanistic source term methods for metallic-fuel pool-type SFRs identify the radionuclide sources and the fuel-to-coolant, coolant-to-cover-gas, and cover-gas-to-environment transport barriers, and provide release fractions that populate the barrier branch points [21]. The branch points may include fuel retention, sodium pool retention, cover-gas behavior, vessel and guard-vessel retention, containment or functional containment response, building isolation, filtration, and release path. This structure is needed because the risk significance of an event depends on release category and consequence rather than core damage status alone.

Table 4 summarizes the proposed generic sequence logic across the three groups. It is an original synthesis developed for this paper, based on the EBR-II Level 1 PRA event-tree families, the PRISM PSID system/core/containment response-tree structure, the recent PRISM protected/unprotected/confinement grouping, and LMP-style frequency-consequence release-category treatment [1, 7-8, 13-16]. The top events and branch points are generic, while the configurable design-specific input in the last column is built from the selected design's architecture.

Table 4: Generic event-sequence top events by sequence group

Sequence group	Top events or branch points	Representative end states	Configurable design-specific input
Protected	Reactor trip; shutdown insertion; primary pump coastdown or natural circulation; decay heat removal; long-term heat sink	Stable shutdown; degraded shutdown; core or fuel damage	Protection logic, primary heat-transport, and decay-heat-removal modules
Unprotected	Reactor trip failure (ATWS); inherent reactivity feedback; power reduction; heat removal; damage threshold	Stable shutdown via inherent response; core or fuel damage	Physics-based success criteria from each design's reactivity coefficients and thermal-hydraulic response
Confinement or radionuclide barrier	Fuel retention; sodium pool retention; cover-gas behavior; reactor vessel and guard-vessel retention; containment or functional containment; building isolation and filtration; release path	Release-category bins linked to mechanistic source term and consequence	Barrier module: containment, guard vessel, or functional-containment strategy

The EBR-II and PRISM PRA sources provide the event-tree examples for this framework: EBR-II contributes internal, shutdown/refueling, fire, and seismic event trees [1, 19], and PRISM contributes both

PSID-era and recent protected/unprotected structures [7-8]. ARC-100 and Natrium are used here to check that the generic structure accommodates current design features; detailed event-tree or fault-tree claims for those designs require additional public PRA documentation.

A loss of forced primary flow illustrates how the generic structure and the configurable modules work together. The initiating event is the same first-level disturbance in both PRISM and Natrium: loss or degradation of primary circulation. The protected event tree is also the same in structure, with top events for reactor trip, shutdown insertion, transition to pump coastdown and natural circulation, decay heat removal, and long-term heat sink. The modules differ. For PRISM, decay heat removal is credited through the passive RVACS path acting on the reactor vessel and guard vessel, and the heat-sink interface is a sodium-water steam generator, so the loss-of-heat-sink and long-term-heat-sink branches inherit sodium-water reaction as a possible coupled hazard. For Natrium, decay heat removal is credited through the Reactor Air Cooling and Intermediate Air Cooling systems, and the heat-sink interface is a sodium-salt heat exchanger feeding molten-salt storage, so the same branches inherit sodium-salt and salt-side hazards. The molten-salt storage adds a decoupling stage that changes the loss-of-heat-sink definition. The unprotected branch, reached on failure to trip, is governed in both designs by physics-based reactivity-feedback success criteria rather than by hardware states, but the feedback coefficients and the time available are design-specific inputs [4, 21]. The event-tree skeleton is reused; the reliability models, success criteria, and coupled hazards are configured per design. This is the sense in which the framework is generic.

8. MODELING ASSUMPTIONS AND DATA TREATMENT

A generic SFR PRA framework must separate plant-generic structure from plant-specific numerical inputs. Event-tree structure, safety functions, and major initiating-event families can be generic for pool-type SFRs. Frequencies, component reliabilities, passive-system success probabilities, human error probabilities, and common-cause failure parameters should be plant-specific where possible [13, 16].

Initiating-event frequencies should follow a hierarchy. Plant-specific operating data should be used when available and relevant. Where plant-specific data are not available, data from similar sodium facilities, test reactors, or non-nuclear sodium systems may be considered if the applicability is documented. Generic LWR data should be used only when the equipment, operating conditions, and failure mechanism are comparable. For advanced designs with limited experience, uncertainty ranges should be retained rather than collapsed into a single point estimate too early.

Passive-system reliability should be treated with explicit success criteria. For heat-removal systems that rely on natural draft, natural circulation, thermal radiation, or passive heat exchange, the PRA should identify the physical conditions required for success. These may include sodium level, heat-transfer surface availability, air-path availability, damper position, fouling, and mission time. Where passive systems include active features, such as blowers, dampers, valves, or controls, those features should be modeled separately from the passive heat-transfer mechanism. Established passive-system reliability methods quantify the functional reliability of natural-circulation and natural-draft systems, including the uncertainty in the thermal-hydraulic conditions required for success, and integrate that result into the PRA; these methods provide a structured way to quantify the passive decay-heat-removal branches, although design-specific thermal-hydraulic success criteria for systems such as RVACS, DRACS, and RAC/IAC are still required [22].

Human reliability analysis, common-cause failure, and uncertainty remain standard technical elements, with SFR-specific emphases. Although some SFR sequences depend less on immediate operator action than LWR sequences, HRA is still required for pre-initiator actions, such as maintenance, system lineup, and calibration, and for post-initiator actions, such as manual shutdown, transition to backup heat removal, sodium-leak and fire response, and recovery of power or control systems. Common-cause failure groups should be identified by function and technology, with digital I&C treated separately because software, platform, sensor, logic, and actuation commonalities may not match traditional hardware groupings. Uncertainty should be built in from the start, and sensitivity cases should test whether the generic conclusions depend on a few assumptions; the dominant uncertainties for these designs include passive heat-removal performance, sodium fire progression, source-term retention, seismic fragility, initiating-event frequencies for new systems, and long-term heat-sink behavior.

9. DISCUSSION

The comparison supports a generic PRA framework for pool-type SFRs, but not a single plant model without configuration. The common basis is strongest at the level of safety functions, initiating-event families, and event-sequence logic, and it justifies common treatment of loss of flow, loss of heat sink, transient overpower, sodium leakage, local faults, and shutdown/refueling events. The comparison also shows where a single generic treatment would mislead: sodium-water reaction should be modeled only for designs with a sodium-water steam generator, Natrium's molten-salt interface needs separate sodium-salt and salt-side treatment, and decay heat removal architectures differ enough that a single DHR fault tree would hide design differences unless built as a configurable module. The framework also avoids treating CDF as the only output, connecting event frequency to plant damage state, mechanistic source term, release category, and consequence, consistent with LMP-style licensing-basis event evaluation.

This work has limits that should be stated plainly. The framework is a modeling structure, not a quantified PRA; it has not been exercised against a full plant model, and no risk-significance ranking is claimed. The success criteria that govern the unprotected and passive-heat-removal branches depend on thermal-hydraulic and reactivity analyses outside the scope of this comparison. Public information is uneven across the four designs: EBR-II and PRISM support detailed PRA discussion, while ARC-100 and Natrium are limited to publicly documented design features, so design-specific event-tree and fault-tree claims are not made for them. The framework applies to pool-type SFRs; no extension to loop-type SFRs is claimed without modification.

10. CONCLUSIONS

A comparative review of EBR-II, PRISM, ARC-100, and Natrium supports development of a generic PRA framework for pool-type SFRs. The shared modeling basis includes low-pressure sodium coolant, pool-type primary configuration, metallic alloy fuel, large thermal inertia, in-vessel primary heat transport, passive or inherent response, and sodium-specific hazards. These features support common initiating-event families and event-sequence logic.

Other features cannot be generalized and are better carried as configurable modules than forced into a single generic hardware model: the heat-sink and power-conversion interface, the decay-heat-removal architecture, the shutdown system, the confinement or functional-containment strategy, the primary-pump design, operating-state exposure, and plant-specific interface hazards.

The proposed framework organizes generic pool-type SFR PRA development around safety functions, initiating-event families, protected and unprotected sequence groups, and radionuclide-barrier response. Relative to the advanced non-LWR PRA standard, which specifies the technical elements a PRA must satisfy, the contribution of this work is the pool-type SFR content that populates those elements before plant-specific data exist [13]. The next step is to complete the generic event-tree and fault-tree set, document success criteria, establish a data hierarchy, and build an assumptions log that traces each generic or design-specific claim to its source.

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