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ARC-100 Conceptual Design Phase Level 1 PSA Early Draft

September 2022

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ABSTRACT

This report represents the initial structure and content created for the ARC-100 nuclear reactor initial conceptual design Probabilistic Safety Assessment. The objective of the study is to assess the safety of the ARC-100 nuclear reactor conceptual design using a risk-informed approach. This early draft report is the first transmittal of the Probabilistic Safety Assessment approach and will be expanded over the remainder of the projects.

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ACRONYMS

DOE	Department of Energy
INL	Idaho National Laboratory
LOCA	loss-of-coolant accident
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
SSC	structures, systems, and component

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1. INTRODUCTION

1.1 Objective

The focus of performing a “risk-informed” assessment is to be able to describe scenarios as having low or high importance as measured by a risk metric—as described in technical reports (Szilard et al, 2015). The objective of the study described in this report is to assess the safety of the ARC-100 nuclear reactor conceptual design using a risk-informed approach.

1.2 Probabilistic Safety Assessment Scope

1.2.1 Summary of Probabilistic Safety Assessment Approach

1.2.1.1 Classical Probabilistic Safety Assessment Approaches

Probabilistic Safety Assessment (PSA) is the systematic process of constructing and quantifying a model representing risk, wherein either (or both) the likelihood and the consequences are treated in a probabilistic fashion. PSA identifies, probabilistically, the scenarios initiating and leading to the undesired outcome, the likelihood of said scenarios, and the magnitude of the consequences. These three items, the scenario, the likelihood, and the negative consequences, form the basis of the so-called risk triplet for the i’th scenario (Kaplan and Garrick, 1981).

While the terms Probabilistic Risk Assessment (PRA) and PSA are frequently interchanged, risk and safety are not the same entities. As noted in NUREG/CR-4350, “safety can, in a sense, be thought of as being the degree to which risk is absent.” (Breeding et al, 1985). PSA provides an important analysis tool for decision making since it attempts to answer the three questions (1) what can go wrong, (2) how likely is it to occur, and (3) what are the outcomes?

Note that the scope of PSA has traditionally covered incidents which, by definition, occur leading up to a core melt or core damage events. For PSA, we discriminate between the "what can go wrong" scenarios by first categorizing upset conditions (called initiating events). Then, for each initiating event scenario, we determine the mechanisms that respond to the upset condition. Secondly, one identifies the likelihood of the defined scenarios. The execution of this step requires the determination of both the initiating event likelihood (which can take the form of either a probability or frequency) and the likelihood that the plant fails to respond adequately to the upset condition. The plant response characterization includes structures, systems, and component (SSC) failures in addition to human error conditions. Traditionally, the plant response modeling has been carried out via logic-based models which describe the conditions where the plant fails to prevent the occurrence of the undesired outcome. Lastly, for the completion of the PSA model, one identifies the scenario consequences. For nuclear power plants, we are typically concerned with the dispersal of radionuclides from the damaged core.

In a PSA, analysts determine lists of upset conditions (initiating events), the plant response to said upsets (accident sequences), and the performance of specific plant systems (typically captured in fault trees). Further, as the PSA is decomposed into additional layers of detail, one reaches the lowest level of the PSA, representing individual component behavior (basic events). These component modules generally contain either (1) subjective information about a component's likelihood of not performing its intended function; (2) actual failure data; or a combination of (1) and (2). The realm of subjective modeling using probabilistic information falls under the umbrella of Bayesian methods.

At a high-level, our Bayesian PSA model is a mixture of deterministic and stochastic (better described by the term aleatory) modules. For example, both a fault tree and its underlying system success criteria are deterministic. But, because we do not know when a particular component in the system will be inoperable, failures of the component are represented via an aleatory model. These deterministic and aleatory models have parameters associated with them, where each parameter may be uncertain. This second "type" of uncertainty is classified as epistemic, indicating that our state of knowledge about a portion of the model is incomplete.

To better understand the techniques that make up classical nuclear power plant PSAs, the major parts of the analysis will be described. In general, a full-scope PSA involves three "levels." The first level contains the logic models (e.g., fault trees and event trees) and probability data representing the outcome of damage to the reactor core. The second level concerns the plant response to the core damage progression (primarily the containment and associated systems). The third level focuses on the off-site consequences resulting from the damaged core and containment. These levels are called Level 1, Level 2, and Level 3, respectively (U.S. NRC 1988a).

For any of the three PSA levels, an additional subdivision has been historically used to define the type of upset or initiating event condition. Specifically, the breakdown between internal events and external events is used to identify initiators that occur within the plant or outside the plant, respectively. Note though that this distinction becomes somewhat blurred since initiators such as a loss-of-offsite-power may happen at the plant (e.g., in the switch yard) or may occur at a geographically distant location. Further, events such as floods and fires can occur either internal or external to the plant, but these have typically been lumped exclusively in the category of external events.

Due to the complex models required, computer tools have been developed to provide a modeling framework for traditional PSA tasks. For example, event trees can be built to determine accident sequences using initiating events and systems. The individual systems—as named on the event trees—can be modeled using logic fault tree editors. Initiating events and other failure events that comprise each system can be assigned frequencies or probabilities. Minimal cut sets (i.e., a minimally sufficient group of failures that can lead to an undesired outcome) can be generated to quantify fault trees and sequences.

An initiating event is a departure from a desired operational envelope to a system state where a control response is required either by human or machine intervention. Enabling events are conditions that provide the opportunity to challenge system safety, potentially leading to an accident. These events are illustrated in Figure 1 where they are used to define the initial boundary conditions for off-normal scenarios.

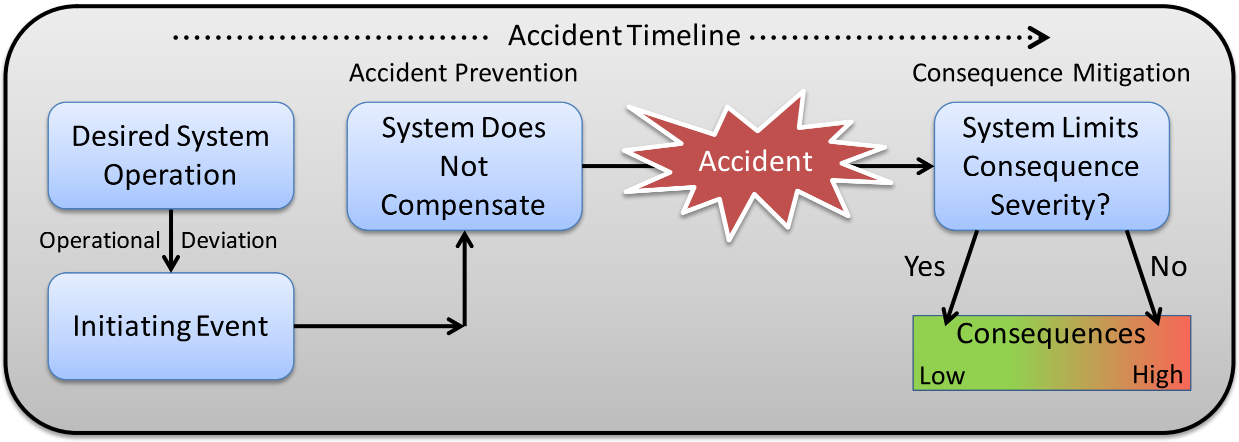

Figure 1. The elements of context in an accident scenario framework.

The concept of a scenario is used to define the safety context. As indicated in Figure 1, adverse consequences occur when initiating events occur, system control responses fail, and the consequence severity are not limited. Hazards may impinge on the system in several ways, including the following:

- They may provide enabling events (conditions that permit the scenario to proceed)
- They may affect the occurrence of initiating events (a departure from a desired operational envelope to a state where a control response is required)
- They may challenge system controls or safety functions
- They may defeat mitigating systems.

Once the starting point of a scenario is known, classical PSA represents the accident sequences using an event tree model. For each potential accident scenario, we identify the plant safety systems which are designed to respond to the upset condition. These system responses are denoted in the event tree model where, according to traditional nomenclature, an up branch point in the tree signifies that the system functions properly, while a down branch indicates that the system is not functional. Each branch point represents the system response, based upon previous occurrences in the event tree sequence.

The PSA analyst represents the up or down branch point in the event tree model by using either an associated fault tree model or simply a probability for the system performance. A probability for the branch (also called a split-fraction) is used when:

- The system is quite complex to model via a fault tree, and it is independent of other systems within the same event tree
- Sufficient operational data exists to estimate an overall system failure probability.

Classical PSA models typically use static fault tree models to represent system performance. Further decomposing the fault tree model leads the analyst to the component and human behavior level of the system. At this level, operational experience tends to play an important role in the determination of component and human failure probabilities. The performance of individual SSCs is modeled such that relevant failure mechanisms (e.g., fails to start, fails plugged, fails to run, fails to energize, inoperable due to maintenance) become a part of the PSA. For each failure mechanism, a failure probability will be determined by the analyst and will include the potential for repair or restoration if possible.

Event Tree Models

An event tree provides a tree-like structure that represents desired and undesired outcomes as one moves from left-to-right in the tree. Each branch point in the event tree represents the pivotal events (system failure/success). Following each path through the tree results in a distinct scenario. In general, the event tree model contains four elements, the initiating event, top events, the branching logic, and outcomes (called end states). The initiator is the first point on the event tree. For each event tree there is generally only a single initiating event. Scenarios with multiple initiators (e.g., a loss-of-coolant accident (LOCA) at the same time as a loss-of-offsite-power) may only be considered in the case of “special” situations, such as seismic events or fires, where the possibility exists of having multiple impacts due to the same event. Each top event in the event tree represents system of operator success/failure probabilities. These tops may be modeled as a single entity (a “chance node”) with its probability or may represent a placeholder for a more complex model such as a fault tree. The branches under top events are generally binary (one up, one down), but may be multi-way wherein each branch point represents a different failure context for the system. The branches for a specific node under a top event should sum to a probability of 1.0. Lastly, the terminal end of a sequence—the outcome or end state—completes the sequence picture. Specific outcomes will be assigned to each sequence through the event tree. For nuclear power plant PSA, these end state assignments are items such as core damage or no core damage for Level 1 PSA or the type of core damage, i.e., plant damage state including status of the containment, for

Level 2 PSA. Other applications may have different outcome metrics and could include entities such as dollars, radiation dose, loss of property, or fatalities.

End State

An end state is the textual description assigned to the expected consequences of an accident sequence. End states may be described at a high-level (core damage, no core damage, release of radionuclides) or may describe details of the accident scenario (type of core damage, specific release characteristics).

When creating an event tree, it is standard that a "down branch" represents a failure (e.g., failure of hardware, software, a human, or a combination of these). Conversely, an "up-branch" represents success. An example of an event tree is shown in Figure 2. For this event tree, the initiating event, IE-OB, is depicted by an initial horizontal line, with systems (top events) connected in a branching manner. The event tree top events are questioned from left-to-right (and generally represent a chronological scenario). A top event can also be passed indicating that success/failure was not questioned the next system top event is questioned. The unique combination of system failures and/or successes from the initiating event to the end state defines the "sequence" for that path. An "end state" is a group of accident sequences, which share certain characteristics that the analyst determines. For this example, there is only one end state called "RELEASE."

TOP Event

The TOP event is a place holder to indicate the system that is needed to function at that particular point in the accident sequence. An up-branch under the top event indicates successful operation of the system while a down branch indicates failure of the system.

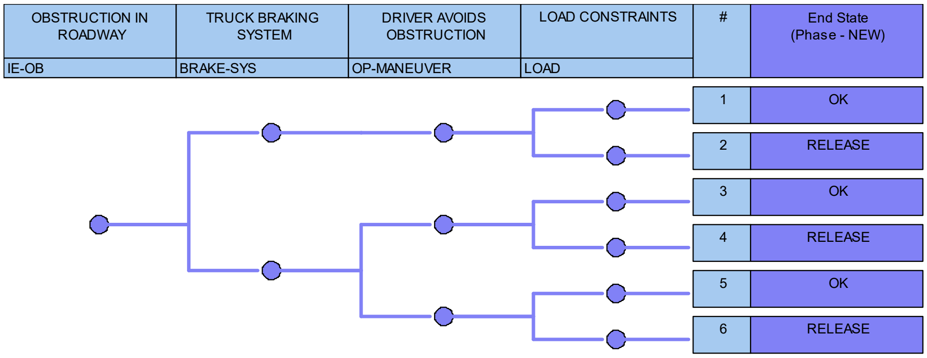

Figure 2. Event tree example.

In a typical event tree, these events generally represent binary events. However, more complicated events can also be represented.

Each top event in the event tree corresponds to either a system or operator function. Putting together the accident scenario is a matter of stringing together the applicable top events (either success or failure) along with the initiating event. From this process, the event tree sequence will define the probability of seeing a certain outcome related to that particular sequence of events.

Since top events are part of a string of other top events, the associated probabilities for each top event must be conditional upon previous top events. Consequently, if differences in prior events—say flooding in the previous example—result in changes to conditional probability on subsequent events, then the model would be further decomposed into specific scenarios. In the case of different flood situations (e.g., volume of water, duration), one may decompose the scenarios into individual event trees, each representing the unique aspects of the flood types. This type of decomposition represents the same line of thinking in nuclear power plant PSA where LOCAs sized are modeled using different event trees.

Fault Tree Models

The previous section demonstrated how event tree analysis is used in a PSA to identify accident sequences which can result in core damage, containment failure, and sometimes other outcomes of interest. The next step in the PSA is to identify the specific combinations of component failures, human errors, and other plant conditions that can result in the system failures which comprise the accident sequences. The PRA Procedures Guide (NUREG/CR-2300) states that the system modeling techniques which are used in PSA should possess the following characteristics:

1. The technique should be capable of predicting the unavailability of complex systems in a manner that can be employed by a variety of practitioners.
2. The technique should be proceduralized to the extent that it can be used for a wide variety of systems in a manner that is traceable, repeatable, and verifiable.
3. The technique should provide reasonable assurance of completeness.
4. The technique should enhance understanding, communication, and the use of results.
5. The technique should produce a model that promotes understanding of the principal ways in which the system can fail and the ways in which failures can be prevented or their impacts reduced.

The fault tree analysis technique fully satisfies these criteria. Fault tree analysis provides a disciplined, rigorous approach to the identification and quantification of system failures.

Fault tree analysis has been defined as:

> *"An analytical technique, whereby an undesired state of the system is specified (usually a state that is critical from a safety standpoint), and the system is then analyzed in the context of its environment and operation to find all credible ways in which the undesired event can occur" (NUREG-0492).*

There are several aspects of this definition which are important for a basic understanding of fault tree analysis and require further explanation. These important aspects are presented below.

The technique specifies an undesired state of the system. The starting point of the fault tree analysis is a definition of the undesired operability state of the system. The description of this undesired state of the system constitutes the top event of the fault tree model. The fault tree analysis technique systematically identifies the combinations of events which can result in the occurrence of the top event.

Fault tree analysis evaluates the system in the context of its environment and operation. In assessing how the undesired system operability state might occur, the system's operating environment and potential effects on system performance are considered. In this way, fault tree analysis considers that compartment flooding may produce pump failures which in turn fail cooling or, that high levels of operator stress during some accident sequence might increase the probability of operator errors which could contribute to system failures. The degree to which these environmental considerations can be incorporated into the fault tree is limited only by the analyst's understanding of what those environmental considerations might be.

The technique is directed at finding all the credible ways in which the undesired event can occur. A fault tree model cannot be used to identify all the ways a system can fail, only those which are credible as assessed by the analyst. The analyst's skill in identifying credible system faults is derived in part from a thorough knowledge of how the system functions, a familiarity with component failure rate data, experience with similar systems, and sometimes, a subjective combination of logic and intuition.

A key feature of fault tree analysis is the fact that it results in re-definition of the undesired top event in terms of combinations of very basic level failure events. This depiction of fault combinations serves as a kind of a “roadmap” of system fault paths. This pictorial representation of fault logic may be presented at any of various levels of analytical resolution depending on the specific goals of the analysis and on the level of detail associated with available data. One very important point which must be made in describing the nature of fault tree analysis is that the technique allows the examination of multiple failures. Many other system modeling techniques (Failure Modes and Effects Analysis, for example) consider the occurrence and effects of only single failures. Fault tree analysis allows the postulation of multiple faults which might occur or exist simultaneously or sequentially. This capability to systematically identify fault combinations which would result in system failure is one of the features that make fault tree analysis such a powerful tool.

The PRA Procedures Guide (NUREG/CR-2300) identifies and describes five essential tasks which comprise the fault tree analysis process. Although this paradigm tends to oversimplify what can be a highly complex and iterative process, it does describe the essential elements of fault tree analysis.

1.2.1.2 Dynamic Methods

Classical PSA tools such as fault and event trees are adequate to represent situations where time or phenomena are not strongly involved in a scenario. However, even for simple cases where time is involved in a scenario, dynamic PSA approaches have been developed to represent the timing directly. As an illustration of a simple case, let us assume we have two coolant pumps, where one is in standby as a backup (see Figure 3). In this example, the failure rate has a value of 0.01/hour and the mission time (t) is 24 hours (we need at least one pump to function for 24 hours).

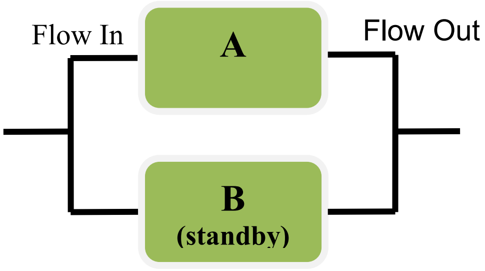

Figure 3. Two-component parallel system with one component required for success.

We can simulate this scenario (for example, in Excel or the INL developed tool Event Modeling Risk Assessment using Linked Diagrams [EMRALD]) and find:

Simulation results: Probability system fails in 24 hours = 0.024

However, if we created a fault tree for this case we would see the cut sets:

Cut set result = Pump A Fails AND Pump B Fails = [1-exp(-0.01*24)]* [1-exp(-0.01*24)] = 0.046

In this case the cut set calculation that is used in classical PSA is too large by a factor of almost two. Note that some PSA append a convolution-based “factor” (which would be the ratio of the correct to approximate results, or 0.54 in this example) to the specific minimal cut sets containing the failure combination described above. Figure 4 illustrates the nature of the time-dependence of this example.

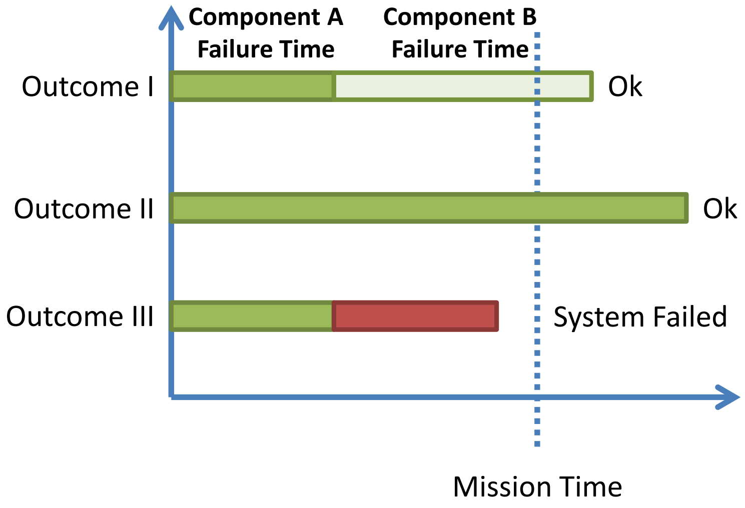

Figure 4. Failure or success-time concept for two components in a switching-type of system.

To address PSA scenarios that include time or phenomena directly, the INL has created the dynamic PSA analysis platform called EMRALD. EMRALD is a software tool that performs dynamic PSA and focuses on the following key aspects:

- Simplifying the modeling process by providing a structure that corresponds to traditional PSA modeling methods.
- Providing a user interface (UI) that makes it easy for the user to model and visualize complex interactions.
- Allowing the user to couple with other analysis applications such as physics-based simulations. This includes one-way communication for most applications and two-way loose coupling for customizable applications.
- Providing the sequence and timing of events that lead to the specified outcomes when calculating results.

Traditional aspects of components with basic events, fault trees, and event trees are all captured in a dynamic framework of state diagrams, which are displayed in a user-friendly modeling manner. Each component is represented by a compact state diagram with basic events driving the current state of that component. A logic tree using components corresponding to a fault tree can be evaluated dynamically during the simulations. Finally, event trees are captured in a plant response diagram, with events (including those from the dynamic logic evaluation) driving an end state result. This approach allows the user to implement dynamic methods with only needing to learn the dynamic state aspects of the model.

After running the EMRALD model, the user can not only obtain probabilistic results, but also able to see dynamic benefits such as timing and event sequences for specified simulation results. Additionally, an open standard for communication is used which allows for coupling to other simulation-based or physics-based analysis. The open standard allows the user to include complex phenomena simulation capabilities such as flood or fire analysis directly in the PRA model.

As an example of EMRALD, we represented the two-pump example (https://emraldapp.inl.gov/) for a mission time of eight hours. The model for the two-pump switching case is constructed in two parts:

- A plant-level diagram (see Figure 5) representing the "high-level" operation of the facility (i.e., it starts and needs to run for eight hours).
- A system-level diagram (see Figure 6) representing the two-pump system with switching (i.e., the system switches to Pump 2 only after Pump 1 is failed).

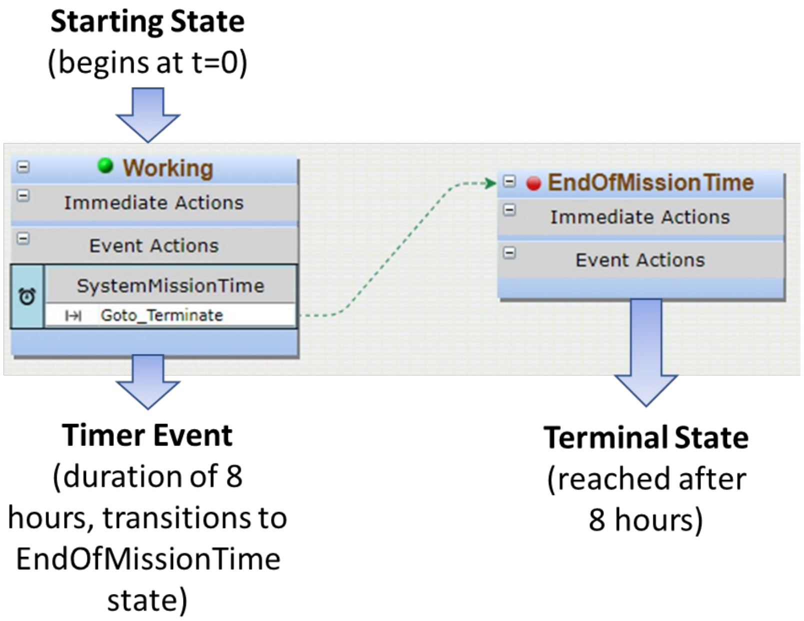

Figure 5. Plant-level diagram for the two-pump example.

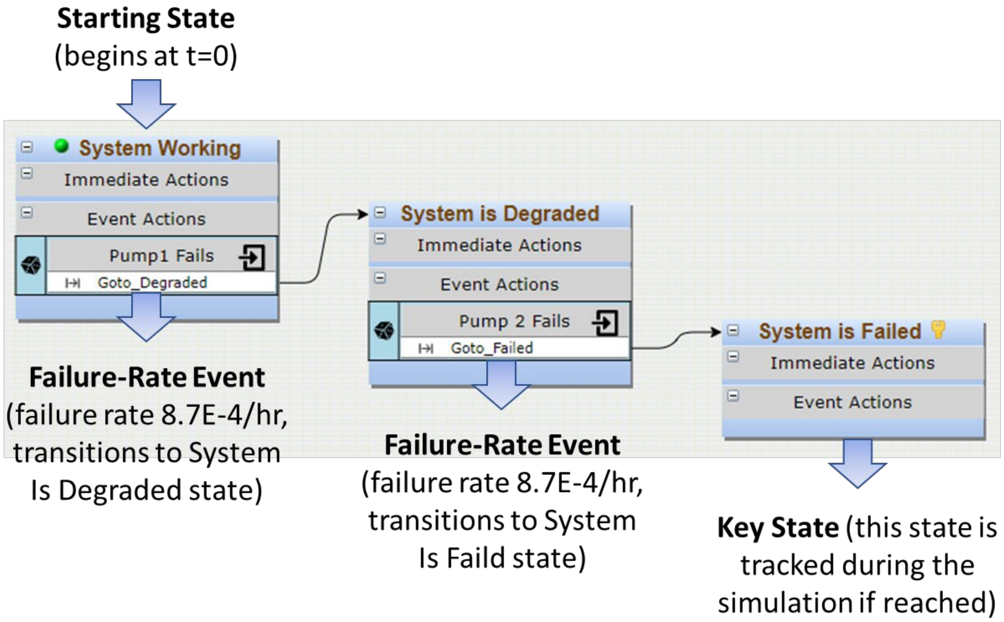

Figure 6. System-level diagram for the two-pump example.

The Key State that is tracked is "System Is Failed." If this state is reached prior to the plant operating for eight hours, then a failure is recorded, otherwise a success is recorded. We ran 100,000,000 iterations. The results of this calculation gave the system failure probability as 2.4×10^{-5} with a mean failure time of the system of 5 hours and 17 minutes.

1.2.2 Probabilistic Safety Criteria

While the design ensures that specific consequence targets are met for accidents that are within the design basis, the PSA is performed to study the range of possible events from likely transients to accidents so infrequent that they are not expected to occur even in a large fleet of plants.

The comprehensive PSA calculates a quantitative value for risk considering the frequency and consequence of all postulated events for the ARC-100. This includes an exhaustive study of all internal failure events (i.e., equipment failures, operator errors) and external events.

There are four main components for the criteria by which the PSA will be judged.

1. The "derived" risk goals established by the NRC which translates into a core damage frequency 10-5 per reactor year .
2. Societal risk goals, from which the above derive are established to limit risk associated with latent cancer effects of ionizing radiation.
3. Individual risk goals, which limit accidental fatality risk imposed on members of the public.
4. Frequency/consequence risk thresholds as defined by Figure 7.

The individual risk goal for the ARC-100 is set such that there is no increase beyond the daily risks to which citizens are exposed. Without a specific site, it is not possible to establish a numerical value so the value consistent with work in the U.S. (NUREG-1860, Reference 7-6 and7-9) that has justified an individual risk target of 5 × 10-7 per plant-year is adopted as the goal.

Similarly, the societal risk has been established in the U.S. as 2 × 10-6/per reactor year.

The Level 1 PSA traditionally calculates the probability of the plant experiencing core damage within a given year. Requirements for Level 1 PSA are given in IAEA SSG-3 (Reference 7-11). The frequency of core damage is chosen because it has long been justified as a surrogate for latent cancer risks. The historical context for this is given in Appendix D.3 of NUREG-1860 (Reference 7-9).

The Level 2 PSA traditionally calculates the probability of a plant experiencing an accident that involves a large early release within a given year. The Level 2 starts from the Level 1 results of core damage and continues events sequences to track the propagation of radionuclides from the core to the containment and beyond. Requirements for Level 2 PSA are given in IAEA SSG-4. Like with core damage, experience with water-cooled reactor technology led to the development of a large early release frequency. The calculation in Appendix D.2 of NUREG-1860 shows how an individual risk goal of 5 × 10-7 per reactor year is met by setting a large early release frequency target of 10-5 per reactor year. This, again, is specific to the core composition and source term behavior of water-cooled reactors.

Finally, the Level 3 PSA traditionally calculates the health or economic consequences of the releases described in the Level 2 results. The Level 3 assessment directly calculates the performance of the plant against the risk goals. In other words, the Level 3 assessment quantifies the metrics that the risk surrogates used in Level 1 (core damage) and Level 2 (large release) were developed to estimate.

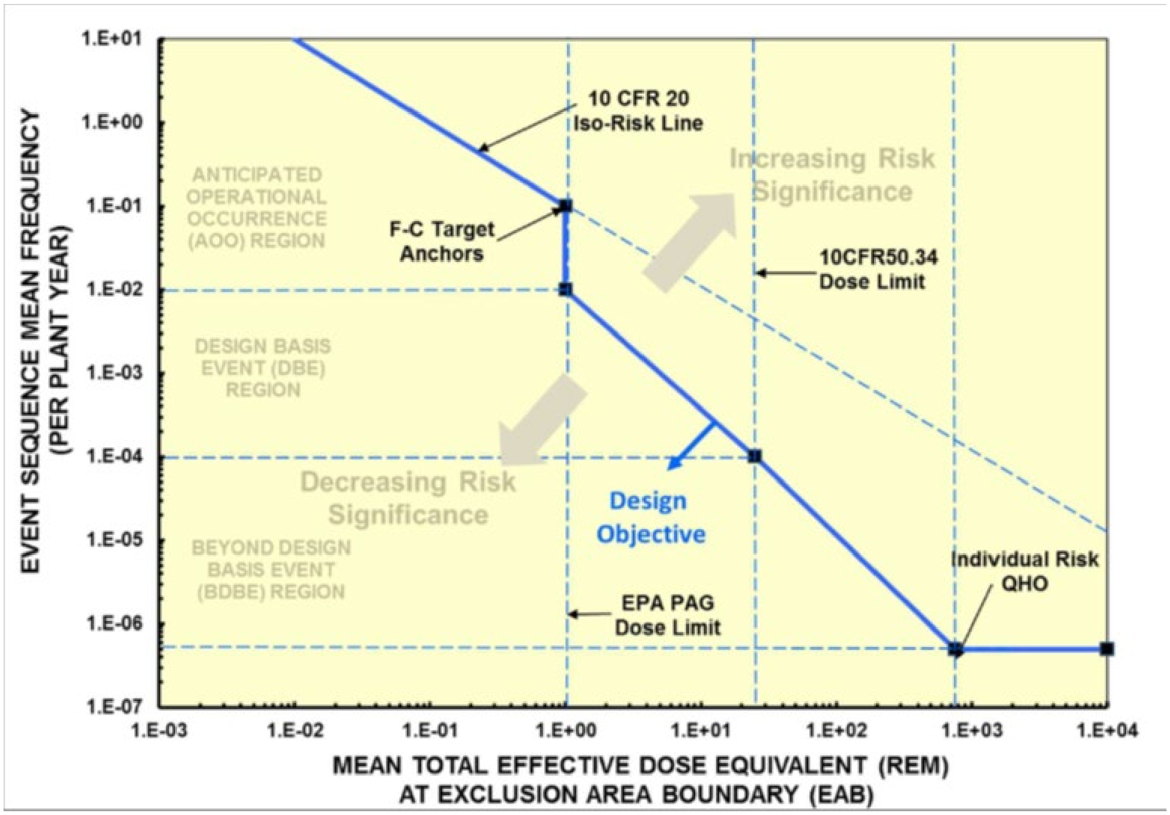

Figure 7. Frequency consequence risk curve from NEI 18-04.

The ARC-100 PSA will calculate a core damage frequency, but as supported by the ASME/ANS standard on non-water-cooled reactor PSA (ASME/ANS RA-S-1.4-2013), technology-neutral risk measures associated with quantitative health objectives are more appropriate for the assessments of advanced non-water-cooled reactor reactors. As discussed above, core degradation was used as the risk surrogate for water-cooled reactors. The surrogate approach may not produce equivalent risk results when used for the ARC-100, for the following reasons.

- For a pool-type reactor such as ARC-100, there are no credible mechanistic events where the fuel is uncovered. Events with damage to fuel are therefore protected by the extra physical barrier provided by the sodium coolant pool.
- For this plant, core damage does not directly relate to releases because the chemical compatibility between the fission products that escape from the fuel and the sodium coolant traps many of the radionuclides rendering them unavailable for release.
- The reactor operates at atmospheric pressure, dramatically reducing accident-related loads on the containment. Following an accident involving damage to the fuel, the lack of pressurization minimizes driving forces leading to the release of radionuclides from the vessel and then further release from containment.

2. ARC-100 CONCEPTUAL PLANT DESIGN

2.1 Pre-licensing Design Characteristics

The following design goals are utilized in the conceptual design of the ARC-100:

- Reactor Safety: The reactor and all of its supporting systems incorporate inherent safety features and utilize passive safety systems. For operational flexibility, some of the passive safety systems (e.g. the Direct Reactor Auxiliary Cooling System [DRACS]) are designed to also operate in an active mode, but that mode is not necessary for safety.
- Economics: The design is suitable for a large variety of sites and applications. It emphasizes simplicity, reliability, and long life for all its SSCs. Furthermore, the design maximizes modularization to enable those components and systems to be factory-made and easily transportable to the deployment sites. This approach minimizes costs without compromising the safety mission.
- Utilize the results of Operations and Testing of Fuels and Materials in a Fast Neutron Spectrum: The fuel of the ARC 100 and ARC 200) utilizes the design and operational experience from the large investment made by the Department of Energy in sodium-cooled fast reactor, notably Experimental Breeder Reactor II (EBR-II) and Fast-Flux Test Facility (FFTF), as well as its design of advanced fast sodium-cooled reactors.
- Transuranic Transmutation: The ARC-100 core design and time between refueling. Twenty years life to promote fission of the transuranics created during operation. In this way, the used fuel in the core (with remaining transuranics) could be recycled to create additional cores.
- Non-Proliferation: The ARC-100 is designed with a virtually inaccessible core that prevents unauthorized movement of fuel and could introduce the required refueling equipment only when the core is replaced at the end of its 20-year life.

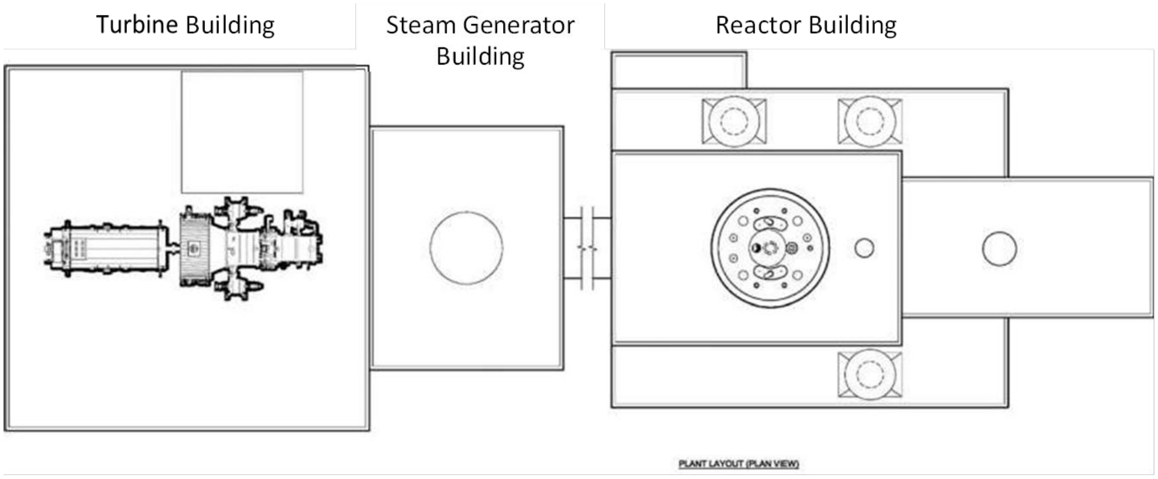

Figure 8. Overview of general layout for the ARC 100 reactor facility.

2.1.1 Reactor Overview

The ARC-100 is a 260 MWt - 100 MWe sodium-cooled fast reactor [Wade and Walters 2010]. It has an 80 MWtd/kg fuel average burnup. Although its initial fuel is enriched uranium (less than 20%), its fuel is recycled at the end of its 20-year refueling interval and only requires depleted uranium makeup to reload the core. After several recycling periods, the core composition shifts to an equilibrium transuranic fuel composition. Heat from the reactor is transferred through a forced-circulation sodium intermediate loop, delivering 500°C to drive a supercritical Brayton Cycle heat converter with about 40% conversion efficiency. The reactor is capable of passive decay heat removal and a passive load follow feature.

2.1.2 Core Overview

The core layout of ARC-100 design is shown in Figure 9. It has fourteen 7-assembly clusters comprising 92 fuel assemblies and six primary control rod assemblies. These 92 fuel assemblies are arranged in three radial enrichment zones. The 28 inner core assemblies have the lowest enrichment of 10.1%. The 28 middle core assemblies are of 12.1% enrichment, and the 28 outer core assemblies are enriched to 17.2%. The average enrichment of these 92 assemblies is 13.5% with a total natural uranium mass of 664.2 tonnes. The average specific power is 12.5 kwt/kg HM which is quite low relative to the traditional fast reactor designs of 50 kwt/kg HM and above. The derating of fuel specific power implies a reduction in decay heat levels per unit fuel mass, allowing the fuel reload handling to be conducted on a 7-assembly cluster basis with a minimal cooling time after shutdown.

In addition, it has two safety rod assemblies and a central core clamping assembly. There are 42 assemblies of steel reflectors and 48 outer assemblies of shield assemblies. These assemblies fit in a core barrel of 3 meters in diameter. This size constraint is made to promote shipability of factory-fabricated modules for rapid assembly at the site. The active fuel height is 1.5 meters.

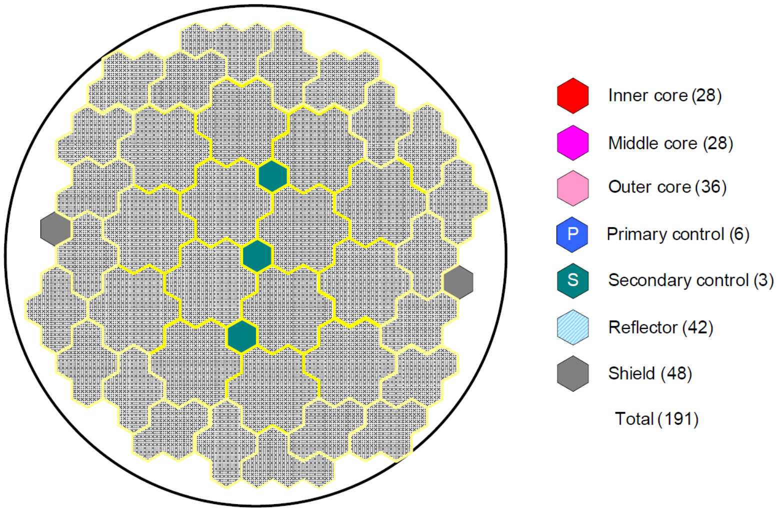

Figure 9. Core configuration of ARC-100.

The reactor core's thermal hydraulics parameters are listed in Table 1 below.

Table 1. ARC 100 reactor design parameters.

Parameter	Value
Inlet coolant temperature	355^{o}C
Outlet coolant temperature	510^{o}C
Average coolant flow rate per pin	0.112 kg/s
Hot channel coolant flow rate per pin	0.148 kg/s
Average coolant velocity in lattice	4.6 m/s
Average linear heat rate	13.9 kW/m
Peak linear heat rate	25.5 kW/m
Average fuel temperature (fuel centerline/inner/outer)	546 / 442 / 435 ^{o}C
Peak fuel temperature (fuel centerline/inner/outer)	686 / 556 / 553 ^{o}C
Margin to design limits: Fuel centerline temperature (margin / limit)	395 / 1081 ^{o}C
Fuel/cladding eutectic temperature (margin / limit)	94 / 650 ^{o}C
Linear heat rate (margin / limit)	11 / 37 kW

3. HAZARD ANALYSIS

3.1 Screening Hazards

Effective screening of hazards in any type of PSA promotes an efficient modeling practice for risk assessment. However, current practice indicates a wide variety of criteria being used to screen hazards in both the design and licensing phase and during the risk analysis. An example of the potential variation in screen can be seen in the Nuclear Energy Agency (NEA) report "Examination of Approaches for Screening External Hazards" which summarized different external hazard screening processes from a variety of different countries including the U.S. (NEA, 2018).

A risk analysis contains a set of scenarios, frequencies, and associated consequences, developed to inform decisions in design or operation. A scenario contains an initiating event (IE) and (usually) one or more events leading to an undesired outcome of concern. Since it is not practical to model all possible initiating events and outcomes, we use a screening process to exclude scenarios of low concern. In general, there are three types of screening processes used in practice:

1. Deterministic screening. This screening type is usually dictated by standard practice, judgment, or a regulatory requirement. A typical example of this type of screening is for cases where hazards are excluded when they are not applicable to a specific reactor design.
2. Absolute frequency (or probability) screening. In this approach a specific frequency (or probability) is selected as a cutoff value, items lower than the specified cutoff can be excluded from consideration.
3. Relative probabilistic considerations conditional upon the specifics of a particular plant design. For example, conditional core damage probabilities (CCDPs) could be calculated for potential initiating events and compared to predetermined thresholds.
4. Consequence screening.

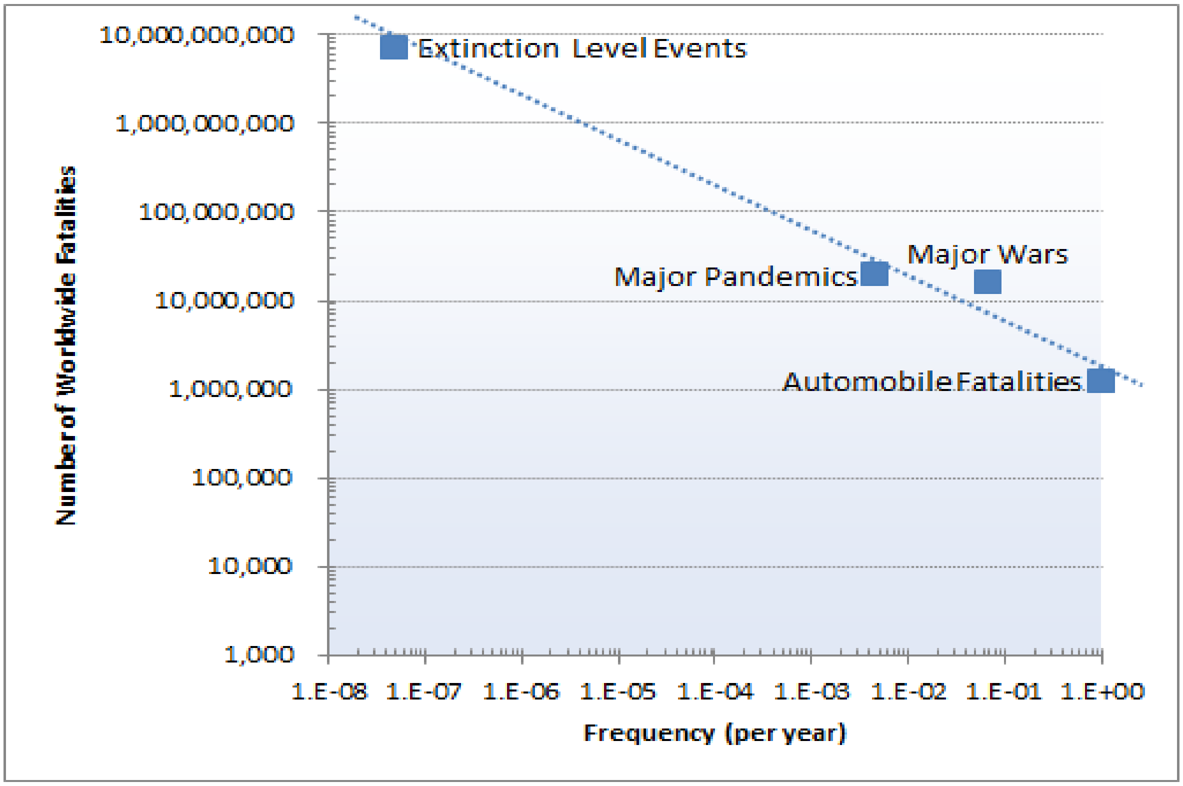

Figure 10. Risk curve for societal events.

The ARC-100 reactor system is a Sodium-cooled Fast Reactor (SFR) reactor model. It is the same reactor model with the EBR-II, therefore it will benefit from the safety analysis already done for EBR-II.

In general, the generic classes of sodium-fast reactor initiating events are:

- Loss of flow
- Reactivity insertion
- Loss of primary heat sink
- Overcooling
- Loss of decay heat removal systems
- Transients (including shutdown)
  - Spurious reactor trips
  - Anticipatory shutdown
  - Normal shutdown
  - Core characterization transients
- Loss-of-coolant (sodium)
- Core support or other structural failure
  - Blockage

- Support system failures
- External events
- Leak between primary/secondary system.

Argonne National Laboratory [Hill, Ragland, and Ribas, 1991] has developed a comprehensive list of initiating events for each of these classes as shown in Table 2 below.

Table 2. Initiating events under consideration.

Generic event class	Specific initiating event	Relevance to ARC-100
Loss of flow	Loss of clutch power	None, because ARC-100 uses electromagnetic pumps.
	Loss of primary pump power	Relevant. Either causing loss of flow to one pump or all four primary pumps. Loss of all pump power may be caused by the loss of 13.8 kV, 2400 V and 480 normal AC supplies.
	Single pump seizure or coast-down	Minimal. ARC-100 uses Annular Linear Induction Electromagnetic pumps. Reliable flow-coast-down systems have been proposed for this pump type [Aizawa et. al. 2012].
	Single pump seizure and coast-down of other pumps	Minimal. EBR-II used two centrifugal pumps in the primary sodium system, each pumping 4500 gpm. In the EBR-II secondary sodium system, one 6500 gpm Annular Linear Induction Pump (ALIP) provided the flow. There was one auxiliary 500 gpm permanent magnet Electromagnetic (EM) pump in the outlet pipe to ensure flow in case both primary centrifugal pumps are lost. The ARC-100 uses 4 ALIPs, thus eliminating the need of shafts which can cause pump seizure if it fails catastrophically. Reliable flow-coast-down systems exist for this pump type. Furthermore, parallel configuration of modular EM pumps reduces the flow rate of each pump and consequently the effect on sodium flow if a pump undergoes coast-down.
	Total blockage of core inlet	May be possible in a severe earthquake
	High Pressure Plenum (HPP) inlet pipe break	
	Low Pressure Plenum (LPP) inlet pipe break	
	Inlet pipe break (all inlet pipes)	
	High Pressure Plenum (HPP) to Low Pressure Plenum (LPP) leakage excessive	
	Missing S/A’s	

Table 2. (continued).

Generic event class	Specific initiating event	Relevance to ARC-100
	Leakage past S/A's	
Reactivity insertion	Automatic Control Rod Drive System (ACRDS) insertion @ 12c/sec (Fast Speed)	
	ACRDS insertion @ 1c/sec (Slow Speed)	
	Control rod insertion (one or many)	
	Core support failure	
	Safety rod insertion	
	Voiding or gas bubble in core	
	Subassembly motion due to hydraulic force	
	Sudden core radial movement (step insertion)	
Loss of primary heat sink	Group 1: Condenser cooling water low flow Loss of condenser vacuum	
	Group 2: Loss of condenser system flow Loss of feedwater flow	
	Group 3: Major system leak	
	Group 4: Secondary system water-to-sodium leak Secondary system pump leak Sodium leak in the secondary sodium system Outlet pipe rupture Inadvertent drain of secondary sodium	
	Group 5: Loss of secondary sodium pump	
Overcooling	Primary pump faults	

Table 2. (continued).

Generic event class	Specific initiating event	Relevance to ARC-100
	Balance of Plant (BoP) transients (system leak)	
	Secondary pump faults	
Loss of decay heat removal systems	Loss of shutdown cooler	
	Loss of shield cooling	
Transients	Scram, spurious	
	Anticipatory shutdown	
	Normal shutdown	
	Core characterization transients	
Loss of sodium	Siphon of Na (defeating siphon break system) through sodium purification system	
	Primary and guard tank failure and leaks	
	Pumpout Na via Fuel Element Rupture Detection (FERD)	
	Pumpout Na via Breached Fuel Test Facility (BFTFs)	
	Intermediate heat exchanger failure	
	Shutdown cooler failure	
Core support failure	Core distortion to more-reactive/less-coolable configuration	
Support system failures	Loss of normal power (13.8 kV)	
	Loss of instrument air	
	Loss of water systems	
	Loss of constant power	
	Loss of 125V DC	
	Loss of 480V AC	
Local faults	Driver fuel failure	
	Experiment failure	
	Local blockage (driver or experiment)	
	Others	

Table 2. (continued).

Generic event class	Specific initiating event	Relevance to ARC-100
External events	Earthquake	
	Fire (sodium and non-sodium)	
	Aircraft impact	May be minimized due to below-grade design configuration.
	Tornado/High winds	May be minimized due to below-grade design configuration.
	Lightning	
	Flood	Site-specific. If flood occurs, and there are openings / crackings to the reactor core, the effect will be significant due to the below-grade design configuration.
	Volcanism	
	Extreme snow	Site-specific. Extreme snow may lead to component failures due to ventilation clogging [Yamano, Nishino, and Kurisaka 2016]

Research has been done on the Level 1 PRA of EBR-II. The research screened external events which include sodium fire events, tornado, high wind, volcanism, lightning, aircraft impact, turbine missiles and internal floods. It identified that among these external events, only two sodium fire sequences and a turbine missile sequence having frequencies exceeding the PRA cutoff threshold of 1E-10.

Table 3. External events under consideration.

External event	IE Frequency	Sequence	Fuel Damage	Frequency
Fires				
BFTF	7.2E-4	SSFR-3	CSD[a]	<1E-10
FERD	9.1E-4	SSFR-6	CSD	**1.5E-6**
Primary sodium purification out of cell	2.2E-3	SSFR-7	CSD	
Secondary sodium within containment	1.5E-3	SSFT-8	CSD	
Shutdown coolers	4.5E-3	SDFR-3	CSD	5.8E-9
		SDFR-6	CSD	**2.1E-6**
		SDFR-7	CSD	1.7E-8
		SDFR-8	CSD	4.3E-9
Tornado	5E-7	TNDO-2	CSD	<1E-10
		TNDO-3	CSD	<1E-10
High wind	1E-4			
Volcanism				
Any	1E-5			
Affecting EBR-II	<1E-8			
External floods	<1E-8			
Lightning				

Table 3. (continued).

External event	IE Frequency	Sequence	Fuel Damage	Frequency
Aircraft impact	8.2E-9			
Turbine missiles	3.1E-4 – 3.5E-5			**3.1E**-7
Internal floods	<1E-10			<1E-10

a. CSD = Core/Structural Damage

EBR-II safety study results can serve as an initial starting point to evaluate the safety of ARC100 reactor. However, since the ARC100 reactor is below-grade (underground) as shown in the figure below, it is hypothesized that the turbine missile external event may be screened out, and that external and internal flooding events need to be reconsidered. For example, a seismic event may possibly cause concrete and pipe breaks that may flood the reactor and cause exothermic sodium-water reaction.

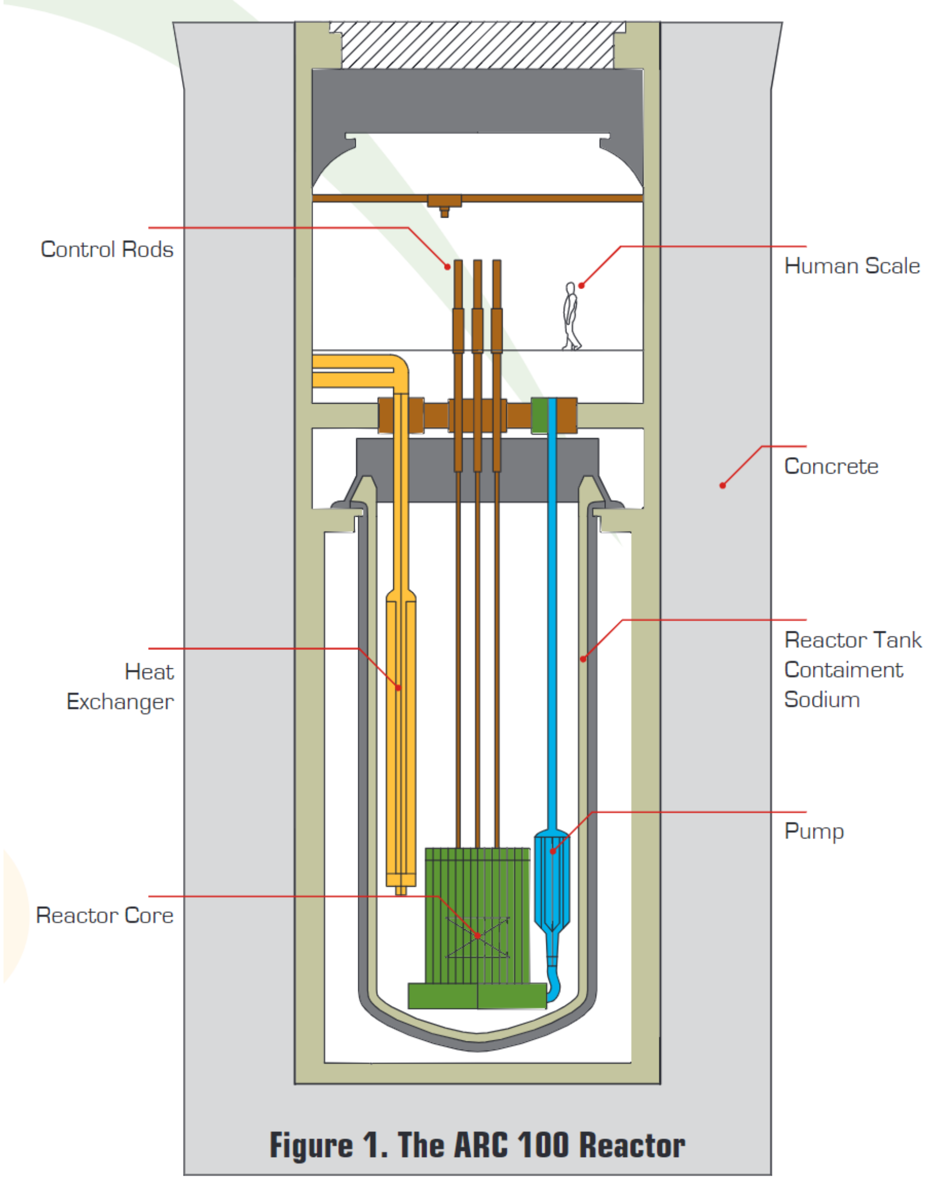

Figure 11. Notional ARC 100 reactor.

3.2 Evaluated Hazards and Licensing Basis Events

Possible risks at nuclear power plants may come from several different types of fundamental hazards that exist. An illustration of these potential fundamental hazards is shown in Figure 12.

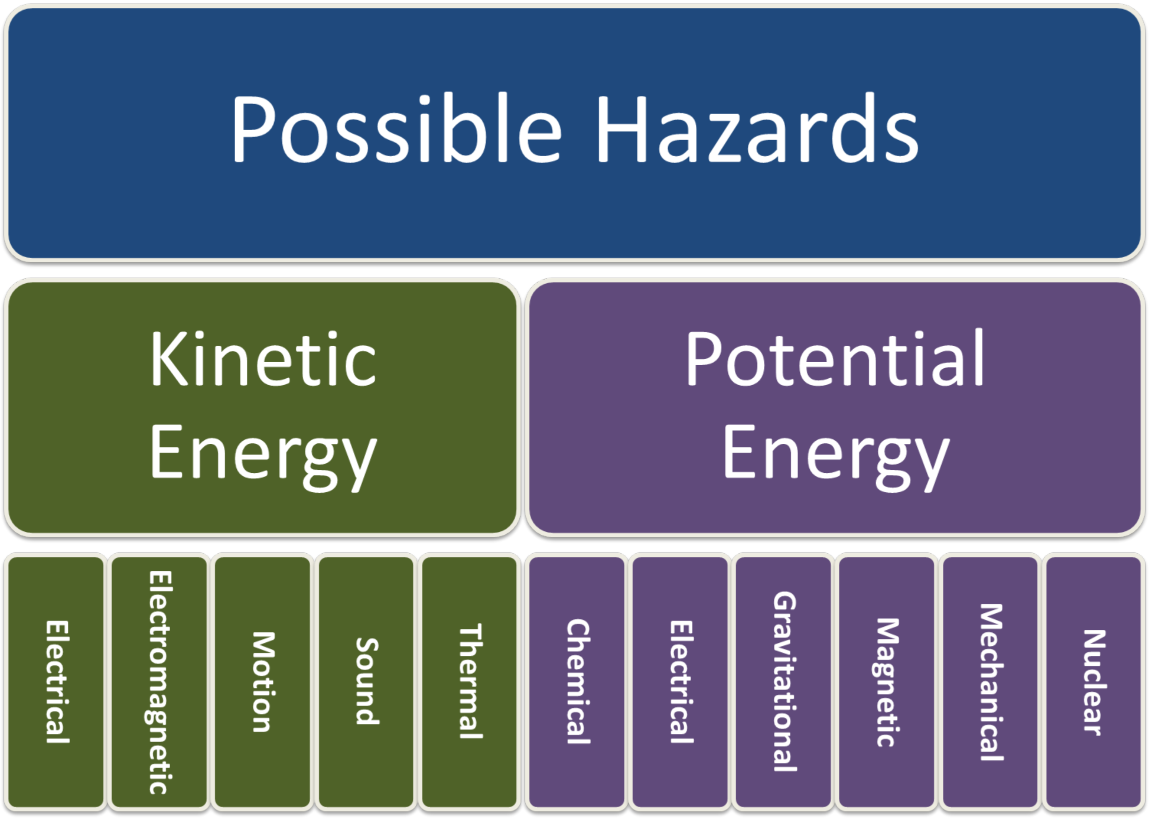

Figure 12. Fundamental hazards that exist in nature.

3.3 Initiating Events

One of the initial modeling issues that must be resolved in the evolution of a PSA is the identification of accident scenarios. This modeling, the "what can go wrong?" follows the systematic identification of accident initial causes, called initiating events, grouping of individual causes into like categories, and subsequent quantification of its likelihood. Off-normal scenarios are the result of an upset condition (the initiating event) and the consequential outcome following the upset. It is the responsibility of the PSA analysts to determine which initiator are of interest for the PSA modeling effort.

Operational events (also called precursors) may directly or indirectly indicate the types and frequencies of applicable upset situations. Conversely, analysts may deduce initiating events through techniques such as failure modes and effects analysis (McCormick, 1981) and master logic diagrams (Modarres, 1993). Also, a second deductive method useful for determining initiating events utilizes fault tree models or simulation to find situations where localized failures can cause general plant upsets. For nuclear power plants, the initiating events we evaluate are those that potentially may result in an off-normal condition such as plant disruptions, component damage, or potential release of radioactive material.

3.3.1 Plant Transients Related to PRA Analysis

Shutdowns and general transients are separated based on the plant response necessary to preserve reactor safety. Included are those events that do not require successful performance of reactor scram to ensure reactor safety but require decay heat removal (DHR) once shutdown conditions are achieved. Examples of these events include:

- Manual shutdowns: Purposeful, controlled descents from critical power, including both planned shutdowns (such as for refueling) and unplanned shutdowns (such as those that may arise from experimental issues or component performance concerns).
- Testing scrams: Purposeful reactor scrams initiated to demonstrate reactor protection system actuation or plant response to prescribed conditions.
- Spurious scrams: Incidental scrams caused by spurious RPS signals.

General transients include events or component failures that have no impact or an indirect impact on safety systems; however, they require immediate action (typically scram) to prevent further degradation or challenges to plant systems. With these initiating events, both reactivity control (scram) and decay heat removal are required. Examples of these events include:

- Minor core faults (e.g., flow disruption between neighboring pins)
- Small reactivity changes
- Full or partial loss of a sodium CO2 heat exchanger
- Instrument line breaks
- Secondary Sodium Purification System failure or leak .

3.3.1.1 Loss of Heat Sink Events

There are two types of loss of heat sink (LOHS) initiators: loss of heat removals system active cooling, which results in the loss of operating power heat removal but allows the heat removal system to operate in passive mode for decay heat removal, and total loss of both active and passive heat removal system capability.

The **loss of active heat rejection** system cooling considers events such as loss of the Brayton CO2 cycle operation, secondary EM pump trips, and other support system failures. During these conditions, active heat removal through the heat rejection system is lost, but the ability for the heat rejection system loop to passively remove heat remains. The LOHS of both active and passive cooling represents events where heat removal through the HRS is no longer possible. Such events include:

- Opening of the sodium loop drain valves, including spurious opening
- Overcooling causing freezing of the secondary sodium (or low temperature resulting in draining the sodium to avoid freezing).

The **loss of active heat rejection from Loss of Off-site Power** (LOOP) considers interruptions of normal power to the electrical buses, which will result in loss of the primary and intermediate EM pumps and reactor scram. **Loss of Flow events** considered are:

- Loss of 1 or 2 Primary EM pumps, with successful coast-down
- Loss of 1 or 2 Primary EM pumps, without successful coast-down
- Loss of 4 Primary EM pumps

3.3.1.2 Transient Overpower Events

Transient Overpower (TOP) events are the result of positive reactivity insertions during operation, which can be introduced by means of control rod withdrawal, sodium voiding (such as gas entrainment), overcooling of the primary system, or a change in core configuration due to bowing, melting, or slumping of fuel. Due to the diverse set of potential causes, the TOP initiators were segregated into two main categories to aid in quantification:

- Core events (e.g., spurious control rod withdrawal)
- Overcooling (e.g., primary/secondary EM pump overspeed).

Core faults include stochastic fuel clad failures and core flow reduction events, such as those caused by loose parts, foreign material, and assembly bowing or deformation. Stochastic fuel clad failures are not a challenge to reactor safety, given the compatibility of metal fuel and the sodium coolant, and are therefore not considered. Core faults that can lead to core flow reduction were grouped into three categories: minor, moderate, and major blockages, based on the necessary plant response.

Minor core blockage is a blockage of minimal size such that any potential fuel damage is expected to be extremely localized and not a challenge to core safety. Examples include small blockages from loose material or pin defects, such as fuel pin wire wrap separation. If detected, these conditions could warrant a scram or plant shutdown.

Moderate core blockages can result in a LOF to a small portion of the core, such as a group of fuel pins or section of a fuel assembly. Past analyses for similar metal fuel SFRs typically show that such events are not a challenge to reactor safety as propagation of damage is unlikely.

Due to core blockage incidents at early sodium reactors, significant effort has been taken to preclude the possibility of **core-wide blockage** through the design and configuration of the core flow paths. Breaks or flow diversion in the inlet plenum are also exceedingly unlikely given the pool design of the system. Therefore, major core blockages have been practically eliminated (i.e., the frequency is below the screening criteria of the PRA).

3.3.1.3 Loss of Containment Boundaries

The **Cover gas cleanup system** is designed to continuously purify the cover gas within the reactor vessel. The system contains equipment both within and outside the containment system (the reactor room). A breach of this system could result in the release of radionuclides contained within the reactor cover gas, such as activated sodium, activated argon, and fission products from fuel pin failures (including both stochastic failures and experiments).

Leaks in the Intermediate Heat Exchanger can be separated into two categories: tube leakage and shell leakage. Shell leakage results in primary sodium leaking from the IHX back into the sodium pool. While this may impact heat removal capabilities through the IHX, it does not represent a primary coolant boundary breach or radionuclide barrier bypass and is therefore not considered a separate IE. An IHX tube leak of sufficient size may limit the ability of that loop to remove heat and presents an opportunity for radionuclides located within the primary sodium to bypass the reactor vessel and containment system barriers. (Radionuclide bypass can likely only occur if the secondary side pumps are off and the loop is drained). The primary concern with an IHX tube leak is the movement of secondary sodium into the primary system (due to the pressure differential between the systems, as required by the SFR GDC), which has the potential for vessel over pressurization or primary sodium entrainment . The frequency of the IHX tube leakage IE was based on SFR component reliability data.

A **vessel leak** considers a breach of the reactor vessel and may result in a loss of primary sodium inventory to the gap between the reactor vessel and guard vessel. The ARC-100 containment system provides a guard vessel which surrounds the reactor vessel and contains the sodium coolant at a level to ensure submersion of the core and heat removal pathways in the event of a reactor vessel leak. Given the lack of vessel penetrations and the high reliability of the reactor vessel and guard vessel, vessel leaks are not quantitatively modeled in the conceptual design PRA.

3.3.1.4 Design Basis Accident (DBA) Categories

Each postulated DBA is assigned to one or more of the following overall categories:

- Transient overpower (TOP)
- Loss of heat sink (LOHS)
- Loss of flow / station blackout
- Loss of off-site power
- External events (Section 4.2.6)
- Mishandling or malfunction of equipment (Section 4.2.7)
- Experiment malfunction (Section 4.2.8)
- Mishandling or malfunction of fuel (Section 4.2.9)

The initiating events under consideration include:

- Reactor transients, 1.5 per reactor year
- LOHS with scram, 1.6 per reactor year
- LOHS without scram, 1.6E-04 per reactor year
- Loss of 1 or 2 Primary EM Pumps with Functional Coast-down, 0.127/pump per reactor year
- Loss of 1 or 2 Primary EM Pumps without Functional Coast-down, 0.140/pump per reactor year
- Loss of Off-Site Power (LOOP), 6.14E-02 per reactor year
- Transient Overpower (total): 3.42E-1 per reactor year
    - From Overcooling, 3.3E-01 per reactor year
    - From Core Events, 1.0E-04 per reactor year
- IHX Bypass Leak, 4.6E-04 per reactor year
- Cover Gas Cleanup System Leak:
    - Outside Containment, 7E-05 per reactor year
    - Inside Containment, 3E—05 per reactor year

The frequencies were developed based on a review of available data from past SFRs, the Light-Water Reactor operating fleet, and other U.S. test reactors. Details of the quantification process can be found in TEV-3823. The frequency of the LOF initiators was derived from SFR operating experience, a preliminary analysis of the EM pump and coast-down mechanism structure, and a preliminary analysis of local AC/DC power supply routing. The frequency of cover gas cleanup system breach was determined based on a preliminary analysis of applicable component failure data, as described in TEV-3823.

It is interesting that the only event to result in radiological release and exposure to operating personnel was breach of the cover gas cleanup system, an extremely unlikely event.

4. DATA ANALYSIS

4.1 Component Reliability

In a typical PSA, component failures are typically provided via an aleatory-type of model. These generally fall into one of three categories:

- A binomial model (demand failures)
- A Poisson model (failures in time)
- An exponential model (time to an event)

4.1.1 The Binomial Model

The binomial distribution is often used as an aleatory model when a component must change state in response to a demand. For example, a motor-operated valve may need to open to provide flow during an accident; the binomial model may be used to represent failure to open upon demand of the valve. The following assumptions underlie the binomial distribution:

- There are two possible outcomes of each demand, typically denoted by success and failure.
- There is a constant probability (p) of failure (or success) on each demand.
- The outcomes of each demand are independent, that is, earlier demands do not influence the outcomes of later demands (i.e., the order of failures/successes is irrelevant).

The unknown parameter in this model is p, and the observed data are the number of failures, denoted by x, in a specified (i.e., known) number of demands, denoted by n.

The probability of obtaining exactly x failures in n total demands is given by the binomial(p, n) distribution as

$$p(\ n, p) = f(n, p) = (n\ x\ )p^x q^{n-x} \tag{1}$$

where $0 \le x \le n$, $q=1-p$, and $\binom{n}{x}$ is the binomial coefficient [n!/(x!(n-x)!)]. The binomial coefficient gives the number of ways that x failures can occur in n demands (i.e., the number of combinations of n demands selected x at a time).

Note that the binomial distribution describes the aleatory uncertainty in the observed number of failures, x.

4.1.2 The Poisson Model

The Poisson model is often used in PSA for failures of normally operating components, failures of standby components that occur at some point in time prior to a demand for the component to change state, and for initiating events. The following assumptions underlie the Poisson distribution:

- The probability of an event (e.g., a failure) during a small time interval is approximately proportional to the length of the interval. The constant of proportionality is denoted by lambda (λ).
- The probability of simultaneous events during a short interval of time is approximately zero.
- The occurrence of an event during one time interval does not affect the probability of occurrence during another, non-overlapping time interval.

Note that λ represents a rate and has units of inverse time. Also note that λ is not a function of time (in this model), so the simple Poisson distribution cannot be used for reliability growth or aging.

The unknown parameter in this model is λ, and the observed data are the number of events, denoted by x, in a specified time period, denoted by t. The form of the data for the Poisson model is shown graphically in Figure 13.

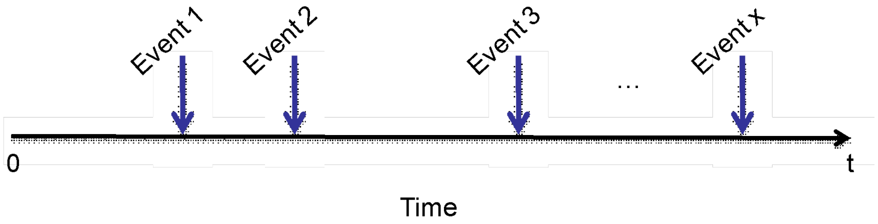

Figure 13. Representation of Poisson-distributed events in time.

The probability of observing exactly x events in a total time t is given by the Poisson(λt) distribution, which is a discrete distribution as a function of x:

$$p(\lambda) = f(\lambda) = \frac{(\lambda t)^x e^{-\lambda t}}{x!} \tag{2}$$

where $x = 0, 1, 2, \ldots$ and λ is the rate of the Poisson process.

The Poisson model is used to describe the aleatory uncertainty in the number of events, x. In other words, we cannot predict exactly how many Poisson events will be seen over some period of time. Bayesian inference is then used to describe the epistemic uncertainty in λ.

4.1.3 The Exponential Model

There are cases where we observe the times at which random events occur instead of the number of such events in a specified period of time (e.g., times to failures of components, times to suppress a fire). If the assumptions for the Poisson distribution listed earlier in this chapter (restated below) are met, then the times between events will be exponentially distributed with unknown parameter λ; this is the same λ that appears as the unknown parameter in the Poisson distribution. Thus, if the *times* at which Poisson-distributed events occur are observed, then the likelihood function is now based on the exponential distribution. The following assumptions underlie the exponential distribution:

The probability of an event (e.g., a failure) in a small time interval is approximately proportional to the length of the interval. The constant of proportionality is denoted by λ.

- The probability of simultaneous events in a short interval of time is approximately zero.
- The occurrence of an event in one time interval does not affect the probability of occurrence in another, non-overlapping time interval.
- The random event that is observed is the time to an event.

The concept of exponential data is illustrated in Figure 14.

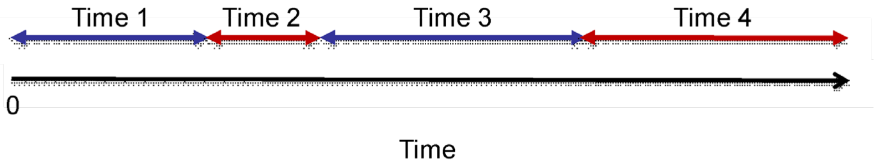

Figure 14. Representation of exponentially distributed events in time.

The probability density function (pdf) for the Exponential(λt) distribution is a continuous distribution and is given as:

$$f(\lambda) = \lambda e^{-\lambda t}. \qquad (3)$$

Note that the pdf does not by itself represent a probability for continuous distributions (such as the exponential). As we will see in the next chapter, probabilities may be calculated from a pdf by integrating the pdf, which produces what is called a cumulative distribution function (cdf). Integrating the exponential density function from zero to the mission time T to determine a probability yields:

$$p(\lambda) = 1 - e^{-\lambda t}. \qquad (4)$$

These three probabilistic models, the binomial, Poisson, and exponential, provide the backbone to the uncertainty analysis in most PSA and reliability modeling activities. These, coupled with deterministic models such as fault trees and event trees, provide the general framework of traditional PSA.

5. EVENT SEQUENCE SCENARIO ANALYSIS

As part of this analysis, we should point out where defense-in-depth (DID) is captured in the PSA.

5.1 Modeling Scenarios using EMRALD

Sandia National Laboratory has published a technical report on the safety and licensing research plan for SFRs [SNL 2012]. The report proposes potential research priorities to support the licensing of a SFR reactor. It identifies several high-level research needs, among which are the sodium fire modeling capability and the improved abilities to model accident phenomena in a post-Fukushima world. These recommendations motivate the research to model dynamic sodium fire scenarios using a recent fire modeling tool and dynamic risk assessment tool.

There are different types of sodium fires [Polidoro and Manzini 2010]. The fire resulting from a spray-type sodium leak is considered as one of the most dangerous fire scenarios. Spray leakage in general creates a large surface area for sodium droplets to react with oxygen in the air. As a result, the Heat Release Rate (HRR) of spray sodium fires is relatively higher than pool fires, in the range of several MW to hundreds of MW. Sodium fires also generate sodium oxide aerosols that can reduce visibility and damage operators/responders breathing. The dynamics of leakage, sodium spray, combustion, smoke propagation and fire suppression create the need to model fire scenarios dynamically.

This capability may be needed to be coupled with EMRALD if sodium fires are to be considered.

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