A Review of U.S. Sodium Fast Reactor PRA Experience

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Abstract: The U.S. has a long history of sodium fast reactor (SFR) development. From the almost 30 years of successful EBR-II operation to the competing designs of the Advanced Liquid Metal Reactor (ALMR) project of the early 1990s, much work has been conducted related to SFR safety analysis. Part of this work has involved the creation of PRAs for both operational reactors and those in the conceptual design phase. A review of four of the past U.S. SFR PRAs was conducted, and their strengths and weaknesses were assessed. As part of this review, the past SFR PRAs were compared to the newly issued ASME/ANS Advanced Non-LWR PRA standard, which for the first time offers guidance on the criteria needed for a "complete" advanced reactor PRA. The results of this comparison offer direction for future analyses concerning what methods can be used from the past SFR PRAs, and what new techniques will need to be developed.

Keywords: PRA, Sodium Fast Reactor, ASME/ANS PRA Standard.

1. INTRODUCTION

From the development of the second Experimental Breeder Reactor (EBR-II) in the early 1960s to the Advanced Liquid Metal Reactor project of the 1990s, the U.S. has heavily invested in sodium fast reactor research (SFR) and development. One element of this work has been the development of probabilistic risk assessments (PRAs) for both conceptual designs and operational reactors. In an effort to preserve the knowledge that was gained through these endeavors, this work seeks to review past U.S. SFR PRAs and to assess their capabilities and deficiencies. This includes comparing the past SFR PRAs to the newly issued ASME/ANS PRA Standard for Advanced Non-LWR Nuclear Power Plants [1]. The results of this comparison offer direction for future analyses concerning what methods can be used from the past SFR PRAs, and what new techniques will need to be developed.

The following four PRAs were reviewed for this work:

1) Clinch River Breeder Reactor PRA – CRBRP-4 (1984) [2]
2) Sodium Advanced Fast Reactor (SAFR) PRA (part of PSID[*]) – AI-DOE-13527 (1985) [3]
3) Experimental Breeder Reactor II Level 1 PRA – EBR-II PRA (1991) [4]
4) Power Reactor Innovative Small Module (PRISM) PRA (part of PSID[*]) – GEFR-00793 (1987) [5]

It should be noted that there are many PRA versions and revisions for several of these SFR designs. The versions listed above were chosen since they are openly distributed and widely available, and in the case of SAFR and PRISM, form the basis of a NRC review. The only exception is the SAFR PRA, which is part of the export-controlled Preliminary Safety Information Document (PSID). The SAFR PRA information referenced here is derived from the NRC review in NUREG-1369 [6], which is open access.

2. ASME/ANS PRA Standard for Advanced Non-LWR Nuclear Power Plants

The American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) Joint Committee on Nuclear Risk Management recently approved for trial use the new PRA Standard for Advanced Non-LWR Nuclear Power Plants [1]. The source material for the standard was the current ASME/ANS LWR Level-1 PRA standard [7], as well as the draft standards under development for low power and shutdown [8], Level-2 PRA [9], and Level-3 PRA [10]. These documents were modified in order to encompass a technology-neutral field of non-LWR designs. The ASME/ANS

[*] PSID – Preliminary Safety and Information Document

non-LWR writing group also worked closely with the Advanced Light Water Reactor (ALWR) writing group [11] to ensure consistency between the two standards. In general, the ASME/ANS PRA standards state what must be done in order to provide an adequate PRA, but not how to do it.

The scope of the advanced Non-LWR PRA standard includes the following [1]:

a) Sources of radioactive material within and outside the reactor core
b) Different plant operating states and shutdown modes
c) Internal and external initiating events (excluding sabotage/terrorism)
d) Different sequence end states, plant damage states, or release categories
e) Evaluation of risk metrics based on sequence end states
f) Quantification of event sequence frequencies, mechanistic source terms, offsite radiological consequences, risk metrics, and associated uncertainties

Unlike the LWR PRA standards, the Non-LWR PRA standard does not use the level-1, level-2, and level-3 terminology. Instead, a single standard encompasses the complete PRA analysis, from initiating event to offsite consequence calculation. This was done since the three-level PRA designation may not be appropriate for some advanced reactor designs, such as those that have liquid fuel. The Non-LWR PRA standard includes the 18 PRA elements seen in Table 1. Like other PRA standards, the Non-LWR PRA standard includes general high level requirements (HLRs) for each PRA element. In order to meet the HLRs, a series of supporting requirement (SRs) must be satisfied. Many of the SRs have different needs depending on the capability category. The capability category is a three-level designation that is determined based on the maturity of the design and the intended application of the PRA.

Table 1: Non-LWR PRA Standard: PRA Elements [1]

A	Plant Operating State Analysis (POS)
B	Initiating Event Analysis (IE)
C	Event Sequence Analysis (ES)
D	Success Criteria (SC)
E	Systems Analysis (SY)
F	Human Reliability Analysis (HR)
G	Data Analysis (DA)
H	Internal Flood PRA (FL)
I	Internal Fire PRA (FI)
J	Seismic PRA (S)
K	Other Hazards Screening Analysis (EXT)
L	High Winds PRA (W)
M	External Flooding PRA (XF)
N	Other Hazards PRA (X)
O	Event Sequence Quantification (ESQ)
P	Mechanistic Source Term Analysis (MS)
Q	Radiological Consequence Analysis (RC)
R	Risk Integration (RI)

There are several HLRs and SRs that are particularly important when reviewing past SFR PRAs. The first topic is the treatment of uncertainty. While the Non-LWR PRA standard does not contain a separate uncertainty PRA element, it embeds the requirements for addressing uncertainty within other elements. For example, the event sequences (ES) PRA element mandates that the sources and assumptions associated with the ES uncertainty must be identified, and the evaluation of those uncertainties must be documented [1]. How this process is carried out is left to the analyst performing the PRA. As with previous PRA standards, capability category I mandates interval estimates for uncertainty, while capability categories II and III require more detailed uncertainty characterization depending on the significance of the uncertainty. Once again, the standard dictates what must be done, but not how to do it.

Next, the mechanistic source term analysis (MS) element describes what must be done to properly model and characterize the source term. The use of a mechanistic source term is a drastic change for advanced Non-LWR designs compared to the current LWR fleet, which uses a postulated source term. As the next section will show, the NRC has criticized past attempts to characterize the source term for SFRs. The Non-LWR standard mechanistic source term HLRs include topics like defining the release categories, establishing the mechanistic analysis method for each category, and the quantitative evaluation of the characteristics of each category. As mentioned above, this includes uncertainty and sensitivity analyses of both parameter and model uncertainty.

For common cause failure analysis, capability category I of the standard mandates the use of the beta-factor approach [12], or equivalent. Capability categories II and III permit the use of the alpha factor model [13], and more sophisticated approaches, like the multiple greek letter model [13]. For human reliability analysis, the standard mandates the use of a systematic method like THERP [14] or ASEP [14] regardless of the capability category.

Lastly, most SFR designs have included passive safety systems, or those systems that require little to no human action and electrical power. How to properly assess the reliability of such systems has been debated for over two decades. The Non-LWR PRA standard does include several references to this issue, including HLR-SC-B5, which mandates the use of mechanistic models supported by empirical data and the characterization of uncertainties associated with safety functions performed by passive means, including those using natural physical processes [1].

3. PAST U.S. SFR PROBABILISTIC RISK ASSESSMENTS

For each of the four past U.S. SFR PRAs, the basic characteristics of the analysis are provided, including the application, scope, and level of detail, which are important inputs when determining the requirements of the new Non-LWR PRA standard. This is followed by any comments from the NRC, if the PRA was reviewed as part of the licensing process.

3.1. Clinch River Breeder Reactor

An overview of the Clinch River Breeder Reactor (CRBR) PRA can be found in Table 2. While the CRBR project was cancelled before the completion of the PRA, there was considerable time and effort dedicated to constructing the analysis. The CRBR was planned as a single 975 MWt, 350 MWe loop-type SFR that was to be built next to Oak Ridge along the Clinch River in Tennessee. CRBR was designed as an oxide-fueled breeder reactor. The project began in 1970, and PRA development started in 1981, two years before the project's termination in 1983.

The development of the CRBR PRA was divided into two phases. Phase I centered on the creation of a tentative list of initiating events and the construction of system fault trees and core damage, core-response, and containment-response event trees. Phase II was to focus on the radionuclide release, health and risk analysis, uncertainty analysis, and common cause failure analysis. However, only the independent common cause failure analysis element of phase II was conducted before the program's termination.

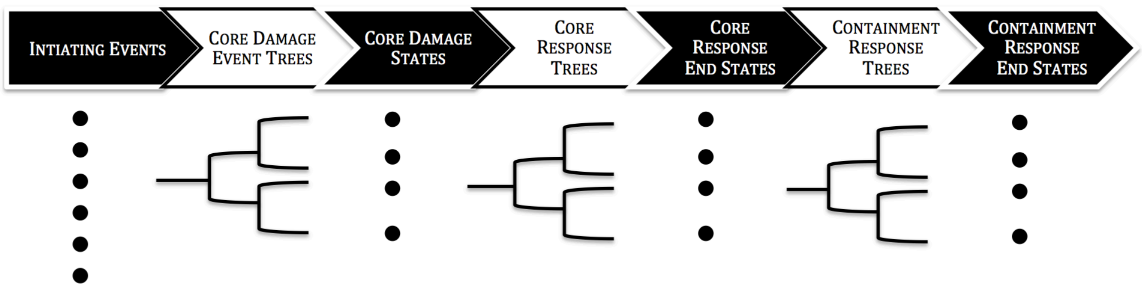
Table 2: CRBR PRA Overview

Metric	CRBR PRA Characteristic
Application	PRA prepared as a supplement for licensing and to aid in design process. Original Preliminary Safety Analysis Report (PSAR) did not include a PRA. Project cancelled before completion of PRA.
Siting	Site specific: Tennessee Valley Authority site on the Clinch River, adjacent to Oak Ridge, Tennessee.
Scope	Power Levels – Full power only Initiating events – Internal initiators (including fire, liquid-metal interactions, internal flood, and missiles) and external events (seismic, tornadoes, wind, and aircraft impact)
End States	Core damage states, core response end states, and containment response end states
Level of Detail	Conceptual design Detailed fault trees for all front-line and essential support systems (except RSS[1]), common cause failure not explicitly modeled in fault trees Accident sequences not modeled mechanistically Human reliability analysis (Combination of THERP [14], OATS [15], and SLIM [16])
Consequence	No radiological consequence analysis or source term calculation
Uncertainty	No detailed uncertainty analysis

[1] RSS – Reactor Shutdown System

The CRBR PRA contained three separate event trees, as shown in Figure 1. The first in the series of event trees, the core-damage tree, consisted of seven top events, including reactor shutdown and short and long-term cooling. However, it is important to note that the accident sequences described in the event trees were not modeled mechanistically, meaning they are postulated sequences that were not derived from a system analysis code or model. Detailed fault trees were created for the systems depicted by top events in the core-damage event tree that had the potential for a significant impact on the frequency of core damage. This included front-line and supporting systems of safety and non-safety-related systems. This analysis also incorporated a detailed human reliability assessment using a combination of THERP [14], OATS [15], and SLIM [16]. As noted earlier, common cause failure was not addressed until phase II of the analysis.

Figure 1: CRBR PRA Structure



The core damage states were inputs to the core response event trees, which describe core energetics, reactor vessel integrity, and fuel debris coolability. If energetics did occur, or if there was a loss of sodium within the reactor vessel, the plant would be in one of six core response end states where significant core damage occurred with a breach of the primary system boundary. These events were then passed to the containment response event trees, which determined the magnitude of the radiological consequences, and represented such events as containment isolation and liner integrity. In total, there were 180 sequences modeled in the containment-response event tree, which were grouped into 13 bins based on similar release characteristics. As mentioned at the start of this section, the source term for the containment-response tree end states and offsite radiological consequences were not calculated before the termination of the CRBR project. This is also true of the uncertainty analysis.

3.2. Sodium Advanced Fast Reactor (SAFR)

An overview of the SAFR PRA is shown in Table 3. It was provided as part of the Preliminary Safety Information Document (PSID) [3]. SAFR was a planned 900 MWt, 350 MWe pool-type SFR designed by the Rocketdyne Division of Rockwell International. The complete plant site would consist of four independent reactor modules. SAFR was designed to use metallic fuel (although the PRA considers both metallic and oxide fuel) with HT-9 cladding, and would have a breeding ratio greater than one. The SAFR conceptual design PRA was relatively simple compared to other PRAs in this section, since it was only partially completed before the project's termination in 1988. However, it did include an offsite consequence analysis and an attempt at uncertainty quantification. Also, the NRC reviewed the PSID, including the PRA, before the project was cancelled, and documented their findings in NUREG-1369 [6].

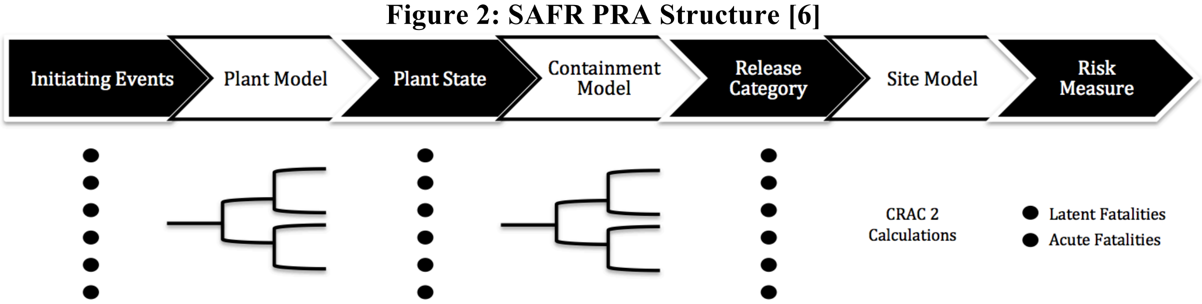
Table 3: SAFR PRA Overview

Metric	SAFR PRA Characteristic
Application	PRA provided as appendix to Preliminary Safety Information Document (PSID) for licensing.
Siting	Generic site on Northeastern seaboard of the United States. Population distribution based on licensability for 75% of existing United States LWR sites.
Scope	Power Levels – Full power only Initiating events – Internal initiators (only TOP[1], LOF[2], (U)LOHS[3]), external initiators (limited seismic only)
End States	Plant damage states, release categories, risk measure (latent and acute fatalities)
Level of Detail	Conceptual design No detailed fault trees (some failure state diagrams) Accident sequences not modeled mechanistically No human reliability assessment
Consequence	Source term includes timing of release and core inventory fractions, but based on study of oxide fuel, then scaled to metal fuel. Offsite dispersion calculated using CRAC-2 [17].
Uncertainty	Lognormal distribution used for all input parameter uncertainties based on error factor of 10. Uncertainty propagated by method of moments and discrete probabilistic arithmetic method. Passive reactivity feedback based on engineering judgment and FFTF and TREAT tests.

[1] TOP – Transient overpower
[2] LOF – Loss of flow
[3] (U)LOHS – (Unprotected) Loss of heat sink

As shown in Figure 2, the SAFR PRA consisted of two event trees and an offsite dispersion calculation. However, this description is a bit misleading. Unlike the CRBR PRA, the initiating events include the initial plant response, such as the plant protection system. There were no fault trees developed for any SAFR systems (although some were modeled with failure state diagrams). The SAFR PRA only considered four initiating events, TOP, LOF, LOHS, and ULOHS, but did investigate both metallic and oxide fuel. A very limited examination of seismic initiators was also conducted.

Figure 2: SAFR PRA Structure [6]



The plant model event tree (also called the core-vessel response) represented events such as energetics, vessel integrity, and in-vessel debris coolability. The containment model event tree included containment isolation, ex-vessel coolability, and transport. However, few containment event tree branches were actually quantified in the analysis before the project's cancellation. The offsite dispersion analysis reviewed 20 release categories, but the source term was not mechanistically derived, and oxide fuel data was scaled to reperesent metal fuel releases. A preliminary uncertainty analysis was also conducted, where a lognormal distribution with an error factor of 10 was assumed for all uncertainties. Then the method of moments and discrete probability arithmetic techniques were used to propagate uncertainties.

In its review of the SAFR PSID, the NRC criticized the simplistic treatment of systems faults, human error, common-cause failures, and passive feedback uncertainty, although it was understood that the fidelity of some of these analyses would have increased without the project's termination. The use of scaled oxide fuel data for the creation of the metal fuel source term was also deemed insufficient by the NRC [6].

3.3. Experimental Breeder Reactor II (EBR-II)

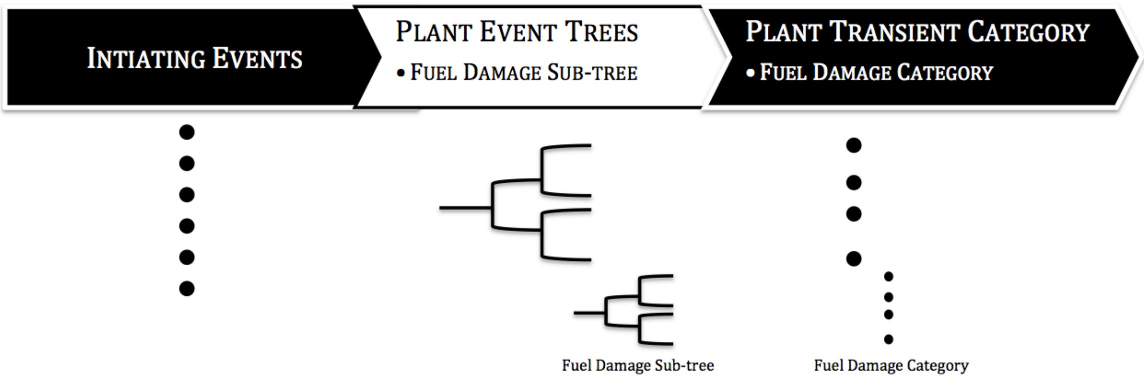
An overview of the EBR-II PRA can be found in Table 4. It is the only reactor of those reviewed in this document that was built and operated. EBR-II is a 62.5 MWt, 20 MWe pool-type SFR that was built in 1965 in Bingham County, Idaho. EBR-II used metal fuel with various clad types, and as the name suggests, was capable of breeding. In 1994, the reactor was permanently shut down after almost 30 years of operation. In 1988, a National Academy of Science study recommended that a PRA be performed for all U.S. Department of Energy (DOE) Class-A reactors, which included EBR-II. A final version of the PRA was released in 1991. The PRA only evaluates to the point of core damage (considered a level-1 PRA in LWR-space).

Table 4: EBR-II PRA Overview

Metric	EBR-II PRA Characteristic
Application	PRA created over 25 years after initial plant operation as part of a DOE study to assess reactor safety.
Siting	Specific site: actual location of reactor in Bingham County, Idaho.
Scope	Power Levels – Full power only (refueling treated independently) Initiating events – Internal initiators (including liquid-metal interactions), external initiators (fires, Na fires, tornadoes, high winds, volcanism, floods, lightning, aircraft impact, industrial accidents, missiles, internal flood), seismic not included in final PRA
End States	Plant transient category, fuel damage category
Level of Detail	As-built plant design Detailed fault trees (beta factor for common cause failure) Accident sequences modeled mechanistically using SAS4A/SASSYS-1 [18] Human reliability analysis included (THERP [14])
Consequence	No radiological consequence analysis or source term calculation (PRA ends at core damage)
Uncertainty	Included data (parameter), model, and completeness uncertainties. Lognormal distribution used for all parameter uncertainties other than reactivity feedbacks, which used screening and a response surface. Quantification involved 5000 Monte Carlo samples. Sensitivity and importance analyses also conducted.

As shown in Figure 3, the EBR-II PRA structure is fairly simple, but overall the EBR-II PRA is the most detailed of those reviewed here, including a vast range of internal and external initiating events. The plant event trees cover such events as scram and decay heat removal. For each system depicted in the plant event trees, a detailed fault tree was constructed, which included both the frontline system and many supporting systems, such as instrument air. Common-cause failure was modeled explicitly in the fault trees through the use of the beta factor technique. Much of the reliability data used to quantify the fault trees was based on the 20+ years of EBR-II operation prior to the PRA development. If fuel damage was encountered, then the event was passed to the fuel damage sub-tree. This tree calculated the time at temperature for the fuel pins and the likelihood of sodium boiling. This information was found by mechanistic modeling using SAS4A/SASSYS-1.

Figure 3: EBR-II PRA Structure



The EBR-II PRA contained a relatively detailed human reliability analysis that centered on human action leading to failure to scram and failure to recover decay heat removal. Since EBR-II was operational at the time of the PRA development, interviews with plant staff were used during the human reliability assessment. The training, procedures, maintenance, control system, and operations information were analyzed using THERP [14]. Human reliability event trees were constructed for actions that could lead to possible system failure.

Several techniques were used to assess the uncertainty within the PRA. First, the uncertainties associated with inherent reactivity feedbacks were addressed through the multistep process seen in Table 5. Due to limited computational resources at the time, parameter screening and response surface methods were used to analyze the reactivity feedback uncertainty. For each unprotected accident sequence, a quasi-static reactivity balance, which included several uncertain parameters, was used to screen initiators. For each uncertain parameter in the reactivity balance, a normal distribution was assumed, and the unprotected accident sequences were investigated. If the uncertainty had negligible impact on the peak cladding temperature of the sequence, it was dropped. If not, a detailed analysis of the sequence was necessary. Next, a SAS4A/SASSYS-1 model was used to screen important uncertain parameters in the reactivity balance for each ATWS sequence. Then, more detailed probability density functions (PDFs) were assigned to the important uncertain parameters. The PDFs of uncertain parameters were used as inputs for a select number of SAS4A/SASSYS-1 calculations, which were used to create a response surface. Finally, roughly one million Monte Carlo sampling calculations were conducted to propagate the important uncertainties using the response surface.

Table 5: EBR-II Reactivity Feedback Uncertainty Analysis Procedure

Step	Description
1	Screen Accident Initiators
2	Calibrate EBR-II Model in SAS4A/SASSYS-1 Transient Code
3	Screen Parameters
4	Quantify Parameter Uncertainties
5	Develop Experimental Constraints
6	Compute Response Surfaces
7	Propagate Parameter Uncertainties Subject to Experimental Constraints
8	Assess Accuracy of Failure Probabilities

For data (parameter), model, and completeness uncertainties, different approaches were used. For data uncertainty, such as the reliability data used in the fault trees, lognormal distributions were assigned to the parameters and 5000 Monte Carlo simulations were used to propagate the uncertainties through the fault and event trees. This process allowed distributions to be found for all output metrics. It should be noted that the uncertainties associated with natural circulation within the reactor vessel were ignored, and natural circulation was assumed to occur. As for completeness uncertainty, the EBR-II PRA states that the PRA is "complete," regarding the scope of the project, as a result of an extensive search of plant records and system models [4]. Model uncertainty was handled through conservative, simplifying assumptions, and separate human error and common mode failure analyses. As part of these analyses, a sensitivity analysis was conducted to gauge the impact of these and other factors. Lastly, an importance calculation was conducted in order to rank the basic events by their contribution to damage frequency.

3.4. Power Reactor Innovative Small Module (PRISM)

An overview of the PRISM can be found in Table 6. PRISM (Mod-A) was designed as a 470 MWt, 180 MWe pool-type SFR by General Electric Co. The conceptual design could include up to nine metal-fueled reactors per site. The PRISM project began in 1984, and the PSID was first submitted in 1986 (with several revisions following). The NRC reviewed the PSID, including the PRA, and responded with NUREG-1368 in 1989 [19], with continued revisions until 1994. However, DOE funding for the program was terminated in the early 1990s.

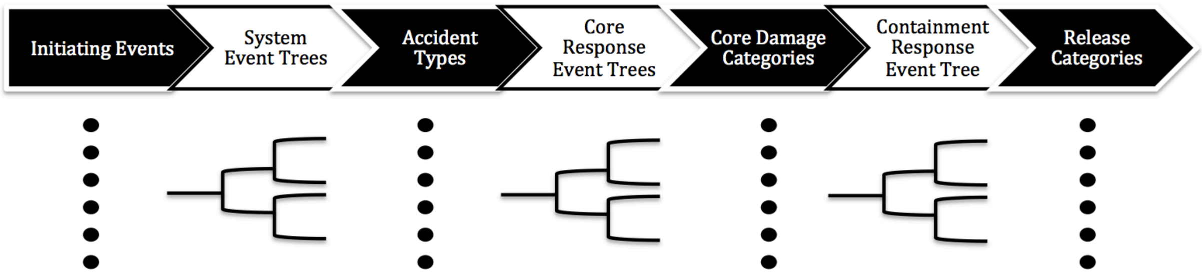
Table 6: PRISM PRA Overview

Metric	PRISM PRA Characteristic
Application	PRA provided as appendix to Preliminary Safety Information Document (PSID) for licensing.
Siting	Siting envelope using GESSAR II[1], which encompassed the majority of potential reactor sites.
Scope	Power Levels – Full power only Initiating events – Internal initiators (reactivity insertions, heat removal faults, liquid-metal interactions), external events (limited seismic only)
End States	Accident types, core damage categories, release categories
Level of Detail	Conceptual design Simplified fault trees for only three systems (RPS, RSS, and EM pump coastdown), common cause failure modeled using beta factor method Limited or no mechanistic analysis of accident sequences Very limited human reliability analysis
Consequence	Source term includes timing of release and core inventory fractions, but based on study of oxide fuel, then scaled to metal fuel. Offsite dispersion calculated using MACCS [20].
Uncertainty	No uncertainty analysis, best-estimate values only. Several sensitivity studies performed.

[1] The GESSAR (General Electric Standard Safety Analysis Report) II PRA evaluates a BWR plant design, however the siting characteristics are considered applicable to many U.S. locations.

Like the CRBR PRA, the PRISM PRA structure, shown in Figure 4, included three event trees. The system event tree, which was first of the three, included events such as reactor SCRAM, pump coastdown (due to the use of electromagnetic pumps), and decay heat removal. Only three of the top events in system event trees had an associated fault tree: the reactor shutdown system, the reactor protection system, and the electromagnetic pump coastdown. The fault trees are relatively simple and do not include supporting systems or human error, but do use the beta factor approach for common-cause failure. Much of the data used to quantify the trees were derived through judgment, and the NRC believed that many of the system failure probabilities had been understated [19].

Figure 4: PRISM PRA Structure



The core response event trees included events such as sodium boiling, clad failure, and energetic release. These sequences were not modeled mechanistically, and as with SAFR, oxide fuel experiments were used as the basis for the analysis, then scaled for metal fuel. The containment response event trees included events such as vessel failure, debris coolability, and late energetics. The output of this analysis was used in the offsite dispersion code MACCS [20], which calculated prompt and latent fatalities.

No comprehensive uncertainty analysis was conducted for the PRISM PRA, although sensitivity studies were performed to asses the impact of initiating event frequencies and the scaling factors used to modify oxide fuel data for metal fuel. More comprehensive uncertainty analyses were planned before the project was cancelled. The NRC found several major sources of uncertainty that would need further analysis in future PRISM PRA revisions. These included the lack of design detail for plant systems (including supporting systems, common mode failures, and human factors), limited test data and experience (related to the electromagnetic pump coastdown, seismic isolators, natural convection decay heat removal systems, and passive feedbacks), human reliability, mechanistic fuel modeling, and metal fuel source term data [19].

4. DISCUSSION

Of the four past SFR PRAs reviewed in this paper, three projects were cancelled before the design reached completion, and one PRA was performed after 25 years of reactor service. Even with those drawbacks, there are lessons that can be learned from these analyses. Table 7 has an overview of three deficiencies seen in past SFR PRAs.

Table 7: Past SFR PRA Deficiencies

Deficiency	Description
Mechanistic scenario modeling	Other than the EBR-II PRA, very little mechanistic modeling (*i.e.*, computer model simulation) was conducted for scenarios depicted in PRA, including the risk dominant sequences. The NRC criticized this deficiency. There was also a lack of mechanistic modeling of passive system performance and reliability during transients, which is now required as part of the ASME/ANS PRA standard.
Mechanistic source term analysis	Of those PRAs that conducted a source term analysis, scaled oxide fuel source terms were used in place of mechanistic metal fuel source terms. The NRC criticized this deficiency, and believed this method may underestimate or mischaracterize the metal fuel source terms.
Uncertainty analysis	Other than the EBR-II PRA, little effort was directed to uncertainty analyses. While this deficiency can be partially accredited to project cancellations, uncertainties were not addressed from the initial stages of the analysis, which is required by the ASME/ANS PRA standard. This was a point of criticism from the NRC. It is also true for the metal fuel performance and passive system reliability analyses.

While all of the SFR designs reviewed here used mechanistic scenario modeling for parts of the safety analysis, there was limited mechanistic modeling of many of the scenarios depicted in the PRAs, including the risk-dominant scenarios, for all of the designs except EBR-II. The NRC highlighted this weakness for both the SAFR and PRISM PRAs. In NUREG-1368 and NUREG-1369, the preapplication reviews of PRISM and SAFR, the NRC stated that due to the lack of mechanistic sequence modeling, the generic assumptions made in the PRA do not accurately represent some of the more important accident sequences, or do not appear to properly represent expected plant responses under specific conditions. In its review of the SAFR PRA, the NRC [6] stated that during fuel damage "mechanistic analyses have not been performed that could otherwise support the generic sequences in this portion of the PRA." A similar criticism was made by the NRC in regard to the accuracy of the metal source term developed for both analyses, with multiple comments directed at the use of scaled oxide fuel data.

The NRC was also critical of both the PRISM and SAFR PRAs for their minimal treatment of uncertainties related to metal fuel behavior, passive system operation, and source term calculations. Even though these projects were cancelled before the PRA and detailed uncertainty analyses could be completed, it does reflect on how uncertainties were to be treated in the analyses. Only point estimate values were used in the PRISM PRA, without identifying the uncertainty interval for input parameters. This would not satisfy even the lowest capability category of the Non-LWR PRA standard, which states that uncertainty intervals must be estimated and the results should include the point estimate and uncertainty bound [1]. The SAFR PRA attempted a simple uncertainty quantification process using a lognormal distribution with an error factor of 10 for all uncertainties, but the level of detail in the PRA was very limited. The NRC questioned the completeness of both the SAFR and PRISM PRAs, and repeatedly stated that the modeling of certain phenomena was not mature enough to forgo a detailed uncertainty analysis. What level of maturity will be adequate for regulatory modelling is an important issue that must be addressed by any future SFR, but it is outside the scope of this work. Even the EBR-II PRA, which is the most detailed of those reviewed here, did not include an uncertainty assessment of the performance of passive decay heat removal, such as the development of natural circulation within the reactor vessel.

Outside of those major deficiencies, there are also shortcomings related to common cause failure and human reliability analysis. For common cause failure analysis, only the EBR-II and PRISM PRAs could meet the capability category I requirement of the Non-LWR standard by using the beta-factor approach. However, the PRISM PRA had simplified fault trees for only three plant systems. The SAFR and CRBR PRAs did not explicitly model common cause failure within the PRA. Both the EBR-II and CRBR PRAs used THERP for their human reliability analysis, which is an acceptable approach in the Non-LWR standard. SAFR and PRISM did not conduct detailed human reliability analyses in the PRA versions reviewed here.

5. CONCLUSION

While the SFR PRAs reviewed here have certain limitations, as described in the previous section, they provide an excellent starting point for future SFR analyses. This is especially true of the EBR-II PRA, which meets many of the Non-LWR standard requirements outlined in Section 2. All of the past SFR PRAs provide information about the types of initiating events that analysts believed were possible for SFRs, and also information regarding how they foresaw the plant response for such transients. The CRBR and EBR-II PRAs give insight into the development of fault trees for sodium specific components and systems, and preliminary data on human reliability analyses. The EBR-II PRA in particular provides very detailed fault trees that included both common cause failure and human actions explicitly.

There are other lessons to be learned from the deficiencies found in these PRAs too. First, the treatment of uncertainty should begin from the initial stages of the analysis. Not only is this now required by the Non-LWR standard, but also is important for the identification of influential uncertainties. Second, demonstrating the reliability of passive safety systems for risk dominant accident sequences is an important task for SFRs, and one which should be completed using mechanistic analyses and taking into account uncertainties. Lastly, the creation of a mechanistic source term is largely an open question for advanced reactors. While none of the reviewed PRAs attempted a mechanistic source term, the strategy used by EBR-II to determine core damage may provide a good starting point. In their analysis, mechanistic simulation was used as a basis to determine time-at-temperature values for regions of the core, and could lead into more detailed fuel damage calculations, if sufficient models exist.

Acknowledgements

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