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

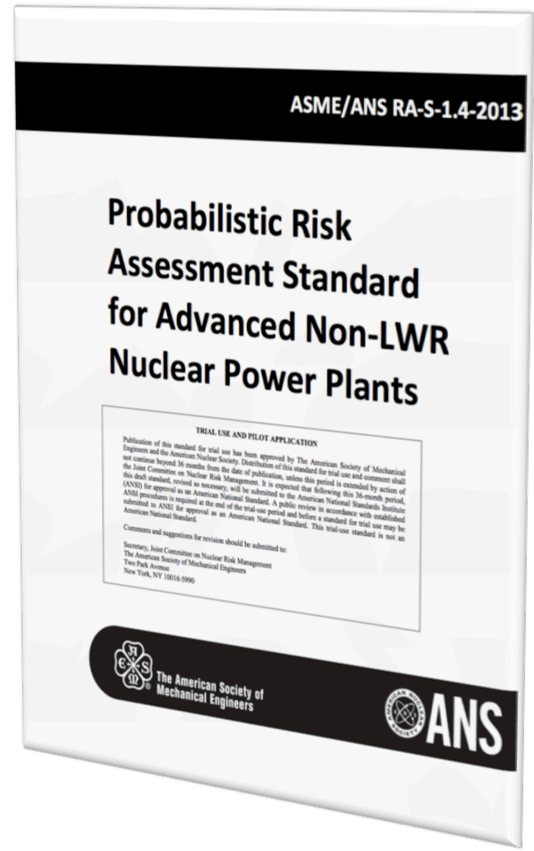
Dave Grabaskas
Argonne National Laboratory

EXPERIMENTAL BREEDER REACTOR II (EBR-II)
LEVEL 1 PROBABILISTIC RISK ASSESSMENT
VOLUME I
FINAL DRAFT
JUNE 30, 1991

December 1987
PRISM
Preliminary Safety Information Document
Prepared for U.S. Department of Energy
Under Contract No. DE-AC03-85NE37937
Volume I
Chapters 1-4
GENERAL ELECTRIC
ADVANCE NUCLEAR TECHNOLOGY
SAN JOSE, CALIFORNIA

Motivation

In 2013, the Probabilistic Risk Assessment Standard for Advanced Non-Light Water Nuclear Power Plants (ASME/ANS RA-S-1.4-2013) was released for trial use.



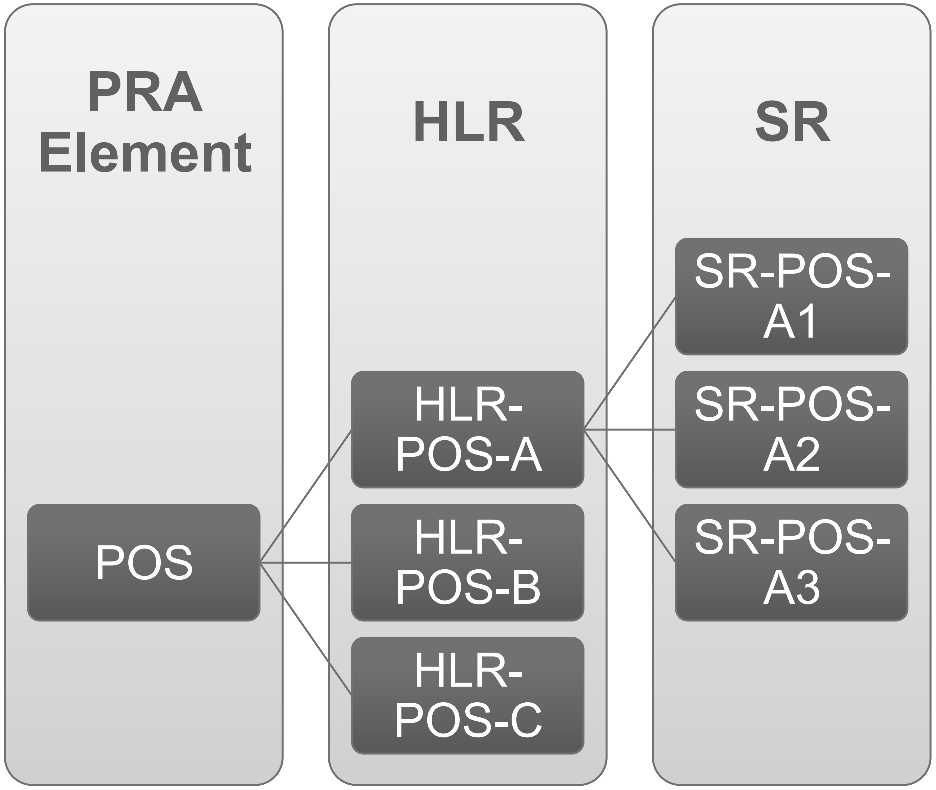
Non-LWR PRA Standard

- No PRA-Level I, II, III distinction (not suitable for some advanced designs)
- The standard consists of 18 PRA Elements

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)

Non-LWR PRA Standard

- Each PRA element consists of a series of requirements
  - High Level Requirements (HLR) – Over 200 total
  - Supporting Requirements (SR) – Over 1000 total

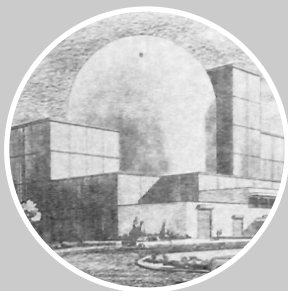


Non-LWR PRA Standard

- Each Supporting Requirement is separated into 3 *Capability Categories*, depending on the application of the PRA

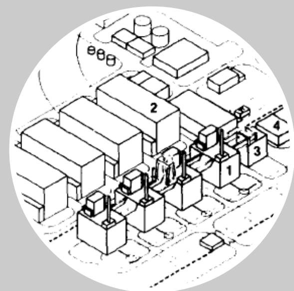
	Capability Category I	Capability Category II	Capability Category III
SR-HR-E1	REVIEW applicable generic analyses for all POSs for similar plants to assess whether the list of initiating events caused by at-initiator HFEs included in the model accounts for industry experience, and to track cases where at-initiator human failure events impact later human responses.	REVIEW applicable generic analyses AND OPERATING EXPERIENCE for all POSs for similar plants to assess whether the list of initiating events caused by at-initiator HFEs included in the model accounts for industry experience, and to track cases where at-initiator human failure events impact later human responses.	
SR-HR-E2	For equipment and POSs modeled in the PRA, INCLUDE in the modeling of support system initiating fault trees (see SR IE-C9) the contribution of operator error leading to the initiating event, or provide the basis for exclusion of such errors.		
SR-HR-E3	INCLUDE the identified at-initiator human failure events as separate initiators from the associated hardware failures if dependencies between the at-initiator human failure event and post-initiator human failures are identified (see HLR-K8). AVOID double-counting events as human failure events and initiating events identified in addressing HLR-IE-C and its supporting requirements.		Always INCLUDE the identified at-initiator human failure events as separate initiators from the associated hardware failures. AVOID double-counting events as human failure events and initiating events identified in addressing HLR-IE-C and its supporting requirements.

U.S. Sodium Fast Reactor PRAs



Clinch River Breeder Reactor (CRBR)

1983



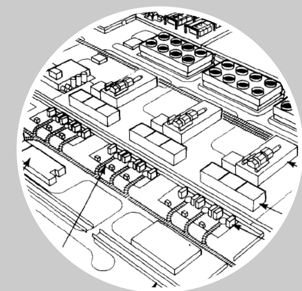
Sodium Advanced Fast Reactor (SAFR)

1988

Experimental Breeder Reactor II (EBR-II)

1991

Operated 1965-1994



Power Reactor Innovative Small Module (PRISM)

1988-1994

General Deficiencies

- No mechanistic scenario modeling
  - CRBR, SAFR, and PRISM
  - EBR-II did, but only to point of core damage
- Source term analysis
  - SAFR, PRISM scaled oxide fuel data
  - CRBR, EBR-II – no source term analysis
- Uncertainty analysis
  - CRBR, PRISM – cancelled before uncertainty analysis
  - SAFR – simplistic treatment
  - EBR-II – fairly detailed uncertainty analysis

- A mechanistic analysis of the accident sequences has not been performed. Generic assumptions made in the PRA may not accurately represent some of the more important accident sequences.

NUREG-1368

- Source-term estimates may be low for some scenarios as a result of extrapolating from oxide fuel to metal fuel.

The PRISM PRA employed standard event-tree, fault-tree, and plant-system models to assess accident sequence frequencies. This methodology is well accepted by the PRA community. Best-estimate values (no uncertainty distribution) were used throughout the quantification process. LWR experience (Refs. A.5 and A.6) and

The SAFR PRA employed standard LWR matrix formalities to estimate the risk from internal and seismically initiated events. LWR experience and the Clinch River PRA were used to identify and estimate frequencies of initiating events and probabilities of component failures. Five generic internal initiating events and eleven seismic initiating events were used to represent and bound the event spectrum. These initiators pass through three types of event trees that model the plant, core, and vessel response. The CRAC-2 code was used to perform the final dose calculations. All branch probabilities were termed "mean best estimate." Each had an arbitrarily chosen log normal distribution (error factor of 10) to account for uncertainty. The "discrete probabilistic arithmetic"

Comparison to PRA Standards

	CRBR	SAFR	EBR-II	PRISM
Human Reliability	✔ THERP/SLIM/ OATS	✖	✔ THERP	✖
Common Cause Failure	✖	✖	✔ Beta (CC-I)	✔ Beta (CC-I)
Uncertainty Analysis	✖	✔ (CC-I)	✔ (>CC-I)	✖

Conclusions

- EBR-II PRA closest to meeting the ASME/ANS standard
  - The best starting point for a future SFR PRA

Pros

- Mechanistic modeling of scenarios using SAS4A/SASSYS-1
- Detailed uncertainty analysis (including parameter, model, and completeness)
- Detailed fault trees with beta factor method for common cause failure
- Detailed human reliability analysis using THERP

Cons

- Scope limited – only to the onset of core damage
  - Core melting and mechanistic source term development is one of the largest open issues
- Developed after almost 25 years of operation
  - Much of the reliability data based on plant experience

Questions?