Identification of Plant Vulnerabilities / Capabilities

Joint IAEA-ICTP Essential Knowladge Workshop on
Nuclear Power Plant Design Safety – Updated IAEA Safety Standards
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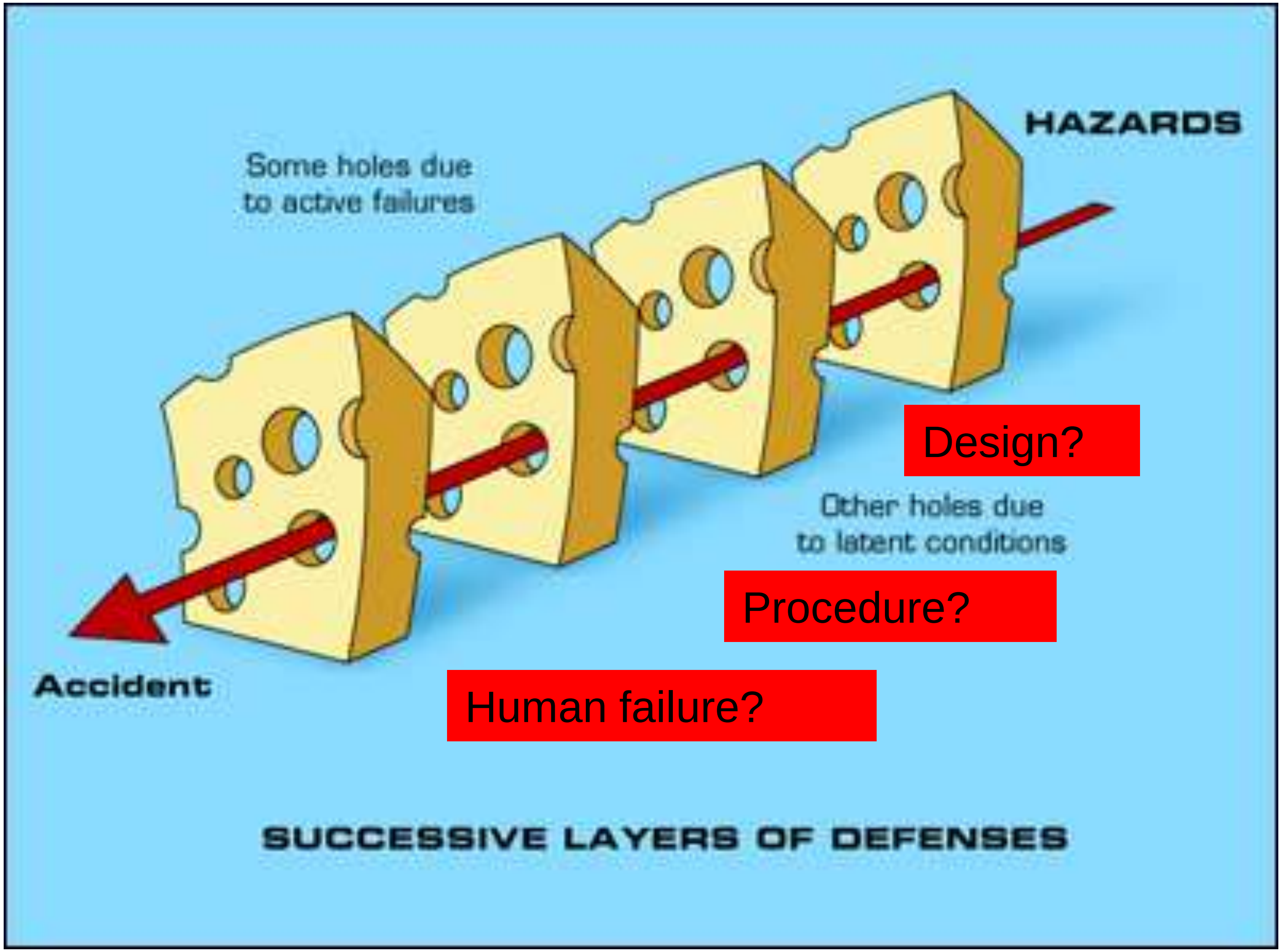
Overview

- Introduction
- PSA Level 1 and 2 for identification of Plant Damage States
- DSA for identification of key phenomena that may occur:
  - expected timing
  - expected severity
- SFP Vulerabilities

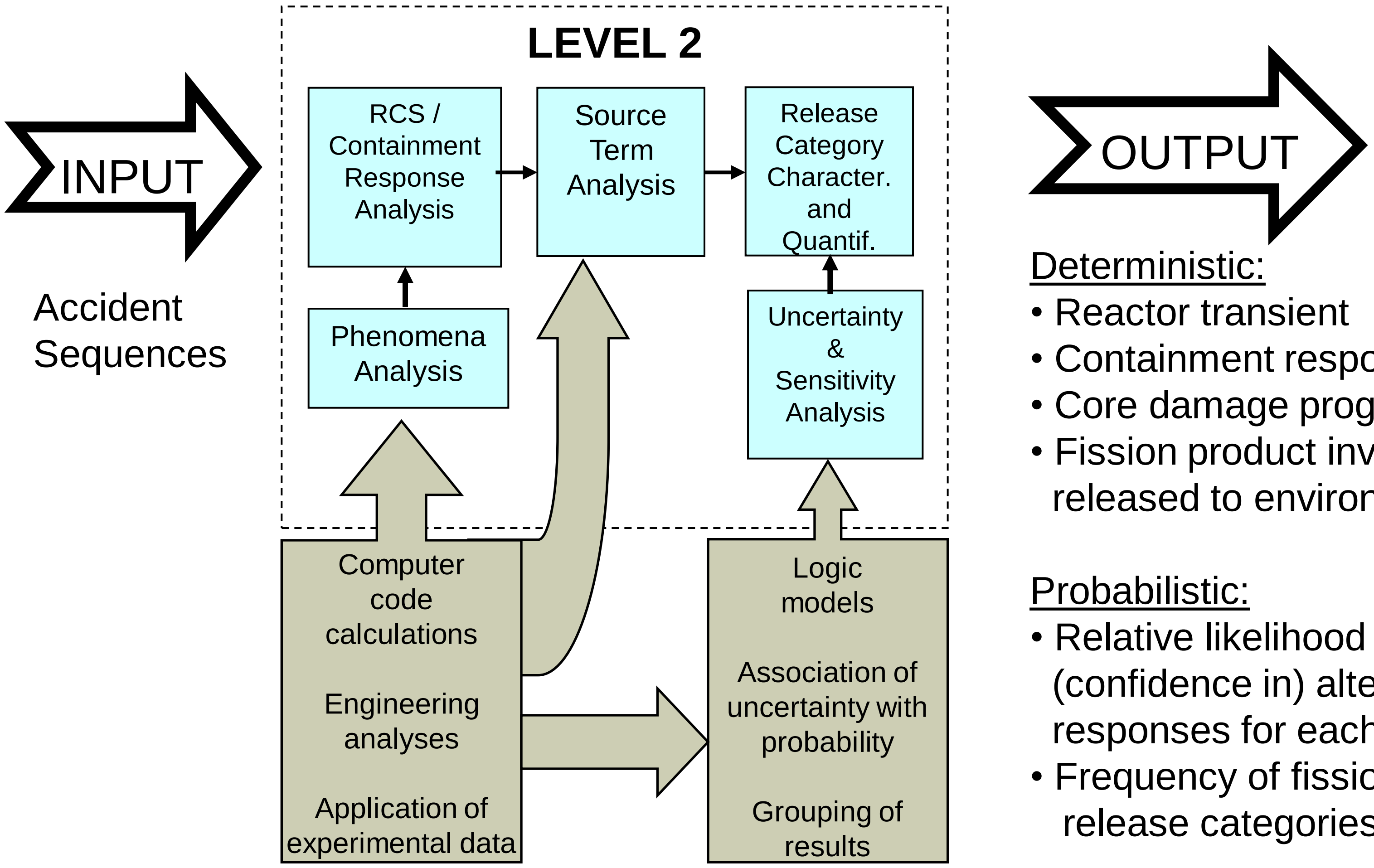
Vulnerabilities?



PSA Level 1 and 2

- Plant specific analysis (IPE – Individual Plant Examination) - plant response on Severe accident
  - PSA Level 1:
    - Event Trees and Fault Tree,
    - Core Damage State Evaluation
  - PSA Level 2
    - Containment Event Trees (PDS evaluation)
    - Deterministic analysis capability to simulate severe accidents (MAAP, MELCOR,..

Level 2 PSA is a Systematic Evaluation of Plant Response to Core Damage Sequences



Deterministic:
- Reactor transient
- Containment response
- Core damage progression
- Fission product inventory released to environment

Probabilistic:
- Relative likelihood of (confidence in) alternative responses for each sequence
- Frequency of fission product release categories

Relationship between IPE and SAMG

Plant-specific Severe Accident Management insights were developed based on the following:

IPE – Individual Plant Examination

Dominant core damage sequences from Level 1 study have been grouped and assessed following the criteria set out in NUMARC 91-04, Severe Accident Issue Closure Guideline

↓

For beyond 24 hour sequence (loss of SW, loss of CCW, station blackout), insights were developed based on the accident scenarios

↓

The Level 2 results have been grouped into release categories and insights have been derived based on these categories. Also, the phenomenological evaluations have been reviewed to gather additional insights.

Level 1 PSA

Sequences that lead to core damage after 24 hours

Level 2 PSA

Important severe accident phenomenology

PSA Level 2 investigates the severe accident phenomenology in two ways:

- "Phenomenological evaluations" (the current state of the art in severe accident research including experimental and analytical efforts)

- The analysis of the all dominant severe accident sequences identified in the level 1 PSA study (performed by MAAP or MELCOR)

PSA Background

- 1985: US NRC issued “Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants” - formulated an approach for systematic safety examination of existing plants
- To implement this approach, GL 88-20 issued, requesting that all licensees perform an IPE in order *“to identify plant-specific vulnerabilities to severe accidents”*
- Internal events + internal floods
- Submittal guidance: NUREG-1335

PSA Background (continued)

- 1991: US NRC issued Supplement 4 to GL 88-20 “IPEEE for Severe Accident Vulnerabilities”
- Each licensee to perform an IPE of external events to identify vulnerabilities, if any, to severe accidents
- The external events considered in IPEEE include:
  - seismic events
  - internal fires
  - high winds, floods and other (HFO) external events
- Procedural and submittal guidance: NUREG-1407

Similar to post Fukushima WENRA requirements for “stress tests”

NEK IPE / IPEEE Insights

- Internal events
  - CDF comparable to US plants
  - Risk profile - no outliers
  - Insights - generic for PWR plants (switchover to recirculation, heat sink - AWF / feed & bleed, SGTR - RCS cooldown & depressurization)
- Internal/External flood
  - Flood zones with dominant risk contribution identified
  - Contribution to Total CDF small

Post-IPE / IPEEE. Feedback & Applications

- NEK IPE / IPEEE performed : 1993 - 1997 (roughly)
- Largest risk contributor: fire-induced risk scenarios associated with several plant areas (CB-1, CB-3A, …)
  - Incorporate the insights into Fire Protection Action Plan (FPAP)
- Incorporation of IPE / IPEEE insights into other ongoing and developing plant programs and planned modifications
  - SAMG development
  - Wet Cavity
  - Passive Autocatalitic Hydrogen Recombiners (PARs)
  - Passive Containment Filtered Vent (PCFV)
  - Procedures: shutdown safety, seismic response

Phenomenology of Accidents

APoS

- Behavior up to core uncovery
- Cladding oxidation; transport, release and combustion of hydrogen
- Core uncovery, heatup, melt, relocation
- Core melt progression
- Hydrogen generation
- Natural circulation and creep failure phenomena
- Reactor vessel wall attack/melt-through

IN VESSEL

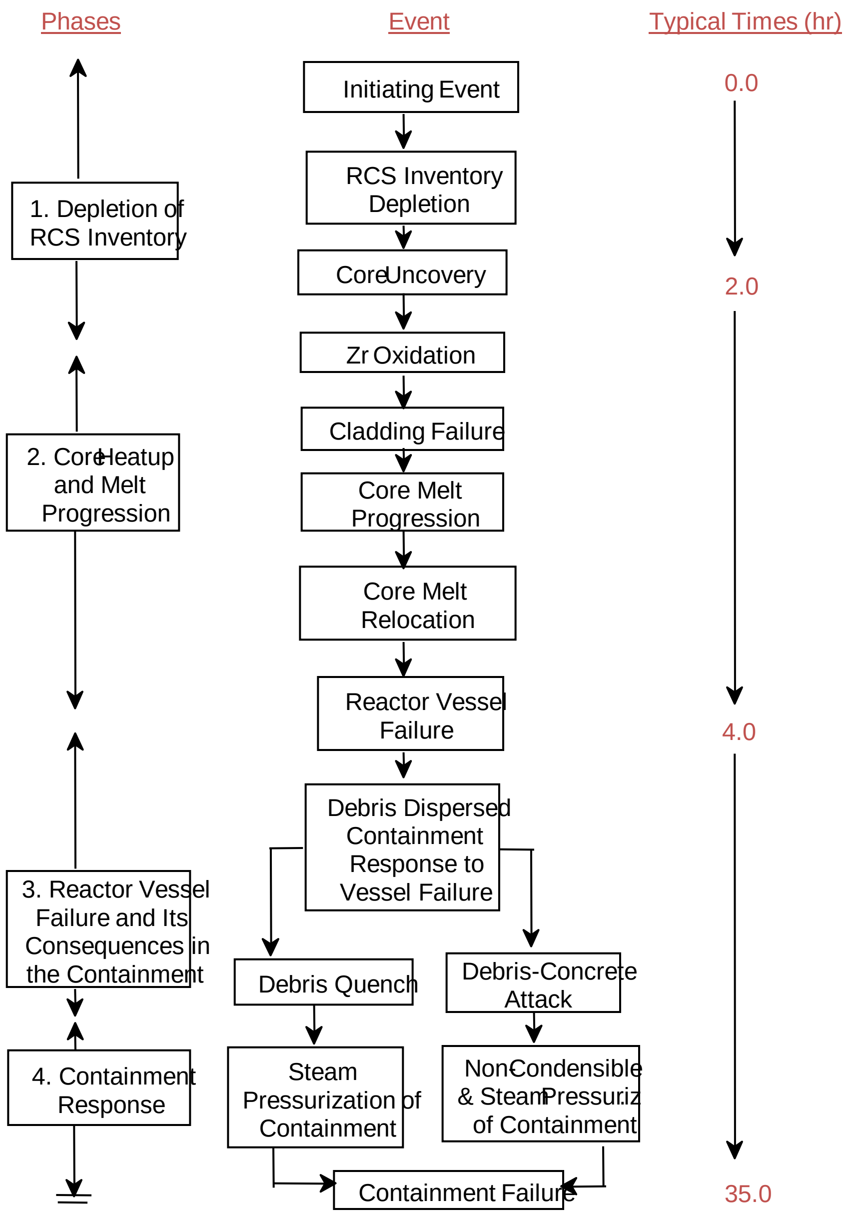
- Reactor vessel failure
- Effect of Operator actions on Accident progression
- High pressure vessel failures; code debris and coolant ejection
- Core debris dispersal - Direct Containment Heating (DCH)
- Core debris/water and debris/concrete interaction
- Cladding oxidation; transport, release and combustion of hydrogen

IN CONTAINMENT

- Fission product behavior
- Containment failure

RELEASES

Severe Accident Progression



Example for Vulnerability Evaluation

- Grouping of Core Damage Sequences
- Groups of Core Damage Sequences Not Involving Containment Bypass
- Core Damage Sequence Groups with Containment Bypass
- Beyond 24 Hours Insight
- Summary of High-Level Severe Accident Strategies and Insight

Example for Vulnerability Evaluation

- Grouping of Core Damage Sequences
  - The first step in development of insights from a plant specific PSA for the purpose of supporting the Severe Accident Management Guidelines development is the evaluation and grouping of PSA core damage sequences into core damage sequence group
  - Safety Guide NS-G-2.15 (http://www-pub.iaea.org/MTCD/publications/PDF/Pub1376_web.pdf) states that for this purpose, initially, all accident sequences will be chosen that, in the absence of preventive accident management measures, would lead to core damage.
  - As another example, the U.S. industry guideline NEI 91-04 (http://pbadupws.nrc.gov/docs/ML0728/ML072850981.pdf) provides the guidance for grouping Level 1 PSA core damage sequences based on the functions involved in the sequences (forming so-called functional accident sequences.

Example for Vulnerability Evaluation

- Grouping of Core Damage Sequences
  - NEI 91-04 starts from the fact that main objectives of a PSA included (in U.S., PSAs were originally performed by the utilities under the frame of so called Individual Plant Examination (IPE) programs):
    - Developing an appreciation for severe accident behavior;
    - Understanding the most likely severe accident sequences that could occur at nuclear power plants;
    - Gaining a more quantitative understanding of the overall probabilities of core damage and fission product releases; and
    - If necessary, reducing the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

Example for Vulnerability Evaluation

- As defined by NEI-91-04, each sequence group definition should designate PSA core damage sequences which are mutually exclusive of all others; that is, an individual PSA sequence should fall under only one of these group definitions. The schemes for grouping should include consideration of the following items:
  - Each category should be based on similarities in the plant response and plant system failures required to cause core damage (i.e., based on initiator grouping and the systems or functions which were required to prevent core damage, but failed);
  - Each category should be mutually exclusive of the others (i.e., the frequency of each PSA sequence should be counted in only one category); and
  - The categories should include all explicitly quantified core damage sequences analyzed in the PSA.

Example for Vulnerability Evaluation

Table 1: Functional Accident Sequences Definitions for PWR

Functional Accident Sequence	Definition
IA	Accident Sequences Involving Loss of Both Primary and Secondary Heat Removal in the Injection Phase
IB	Accident Sequences Involving Loss of Both Primary and Secondary Heat Removal in the Recirculation Phase
IIA	Accident Sequences Involving an Induced LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Injection Phase
IIB	Accident Sequences Involving an Induced LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Recirculation Phase
IIIA	Accident Sequences Initiated by a Small LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Injection Phase
IIIB	Accident Sequences Initiated by a Small LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Recirculation Phase
IIIC	Accident Sequences Initiated by a Medium or Large LOCA with Loss of Primary Coolant Makeup in the Injection Phase
IIID	Accident Sequences Initiated by a Medium or Large LOCA with Loss of Primary Coolant Makeup or Adequate Heat Removal in the Recirculation Phase
IV	Accident Sequences Involving Failure of Reactivity Control
VA	Systems LOCA Outside Containment with Loss of Effective Coolant Inventory Makeup
VB	Steam Generator Tube Rupture with loss of effective coolant inventory makeup

Example for Vulnerability Evaluation

- Basically, the functional accident sequences from Table 1 come from the following general categories:
  - Transient (IA and IB)
  - Induced LOCA (IIA and IIB)
  - Small LOCA (III A and IIIB)
  - Medium or Large LOCA (IIIC and IIID)
  - ATWS (IV)
  - Containment Bypass (VA and VB)

Example for Vulnerability Evaluation

- Core damage sequences from PSA model were reviewed and mapped to functional accident sequence categories which correspond to the above categories. This was facilitated by “functional grouping” of core damage sequences which was done in the PSA report.
- The mapping of core damage sequence from PSA model to functional accident sequence groups for the purpose of getting insights to support the severe accident analyses is described in Table 2.

Example for Vulnerability Evaluation

- The sum of contributions to the CDF from the 11 functional accident sequence groups basically matches the total CDF as calculated by the PSA model, which is because the functional accident sequence groups were defined as mutually exclusive.

- The largest contribution, by CDF frequency, comes from the category with failure of reactivity control (V) and from the RCP seal LOCA category (IIA). Following below is some further characterization of these two functional groups.

Functional Accident Sequence Group	Description	CDF (/yr)
IA	Transient with total loss of heat sink (from any cause, including Station Blackout or other failuers) with early core damage	1.4E-06
IB	Transient with total loss of heat sink (from any cause, including Station Blackout or other failuers) with later core damage	4.3E-07
IIA	Induced LOCA (RCP Seal LOCA) with loss of primary coolant makeup or adequate heat removal in the injection phase	4.9E-06
IIB	Induced LOCA with loss of primary coolant makeup or adequate heat removal in the recirculation phase	4.9E-08
IIIA	Accident sequence initiated by a Small LOCA with loss of primary coolant makeup or adequate heat removal in the injection phase	3.1E-07
IIIB	Accident sequence initiated by Small LOCA with loss of primary coolant makeup or adequate heat removal in the recirculation phase	4.6E-06
IIIC	Accident sequence initiated by Medium or Large LOCA with loss of primary coolant makeup in the injection phase	4.1E-06
IIID	Accident sequence initiated by Medium or Large LOCA with loss of primary coolant makeup in the recirculation phase	3.1E-06
IV	Accident sequences (transient or relevant LOCA) involving failure of reactivity control	5.8E-06
VA	Interfacing Systems LOCA outside containment with loss of effective coolant inventory makeup	3.0E-07
VB	Steam Generator Tube Rupture with loss of effective coolant inventory makeup	1.5E-06
	Total:	2.65E-05
	Reported CNPP-1 PSA CDF	2.64E-05

Example for Vulnerability Evaluation

- Based on the described grouping, the plant specific insights are presented in what follows on a functional accident sequence group basis.:
  - Groups of Core Damage Sequences Not Involving Containment Bypass (IA, IB, IIA, IIB, IIIB, IIIC, IIID, IV)
  - Core Damage Sequence Groups with Containment Bypass (VA, VB)
- Example: **Functional Accident Sequence Group IA**
- This group contains accident sequences involving **total loss of secondary heat sink** and **failure of primary feed and bleed in the injection phase**.
- The plant specific sequences result from the transient initiators and lead to **early core melt with reactor vessel failure at high** or low pressure (e.g., depending on the **occurrence of hot leg creep rupture**). For some sequences, the core may be recovered in-vessel. The total core damage frequency of this group is 1.4E-06 /yr. This represents 5.2% of the total CDF of the plant.
- The **symptoms** of this functional sequence group include **early high temperature indication on the core exit thermocouples** at high or low primary system pressures.
- Early action is required and the SAMG diagnostic flow chart should include:

Example for Vulnerability Evaluation

- Example: **Functional Accident Sequence Group IA**
- Early action (**Canditate High Level Action (CHLA)**) is required and the SAMG diagnostic flow chart should include:
  - Injection into SGs before possible SRC hot leg or SG U-tubes creep failure
  - Injection into the primary system;
  - Depressurization of the primary system;
  - Flooding the containment to cover debris in the reactor cavity and mitigate the molten core - concrete interaction;
  - Establishing the decay heat removal from the containment;
  - Hydrogen control in the containment.

Example for Vulnerability Evaluation

- Beyond 24 Hours Insights
  - In the PSA, like with many other PSAs, the accident sequences were quantified for a mission time of 24 hours.
  - The event sequences / scenarios of concern here are of two types:
    - Those PSA event tree sequences which are core damage sequences (by definition of success criteria for minimum of required important safety functions) for which core damage was demonstrated to occur later than 24 hours after the initiator and the sequences were, for this reason, not included in the core damage risk quantification;
    - Those PSA event tree sequences where at least minimum of important **safety functions have succeeded for 24 hours** and the sequences were declared as "success" (i.e. core damage avoided) in the PSA event trees. However, **some of those functions may fail in the time frame longer than 24 hours and thus convert the "successful" PSA sequences into the core damage sequences**, with core damage occurring at some time after 24 hours following the initiator. **It is important to recognize that some of the important safety functions from the mentioned "minimum" which succeeded might have been explicitly shown in the event trees while some other might have been implicitly assumed to succeed**.

Example for Vulnerability Evaluation

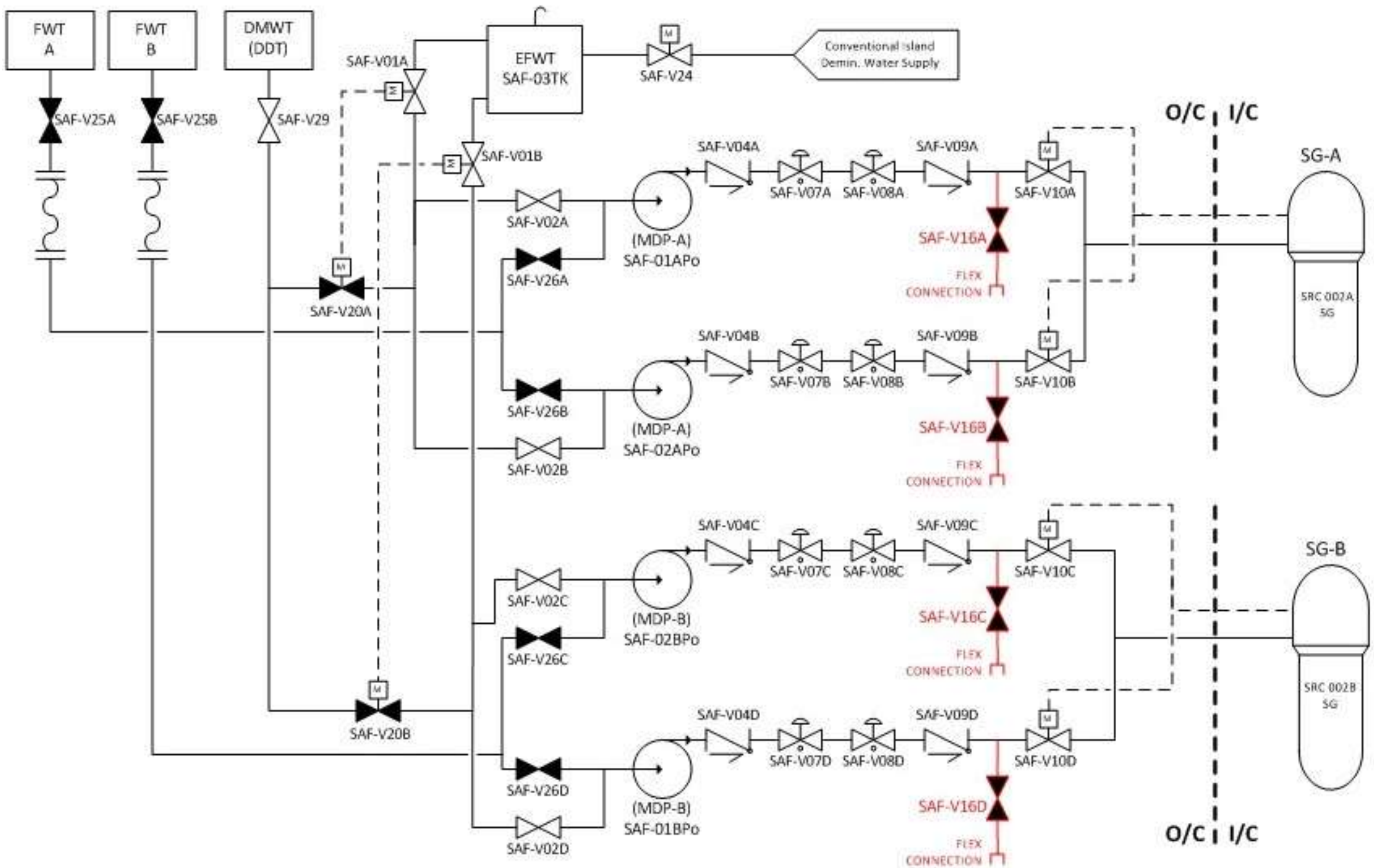
- Beyond 24 Hours Insights (cont)
  - The screening of the PSA model for the core damage sequences of the type 2 resulted in an identification of the following groups of sequences with potential to fall under type 2:
    - Sequences **with prolonged (beyond 24 hours) implementation of safety functions** including replenishment of tanks;
    - Sequences **where failure of recirculation causes core damage beyond 24 hours,** including failure of transfer to hot leg recirculation (as already mentioned above);
    - Sequences induced by a **total loss of ESW** which develop into core damage beyond 24 hours due to non-mitigated RCP seal LOCA;
    - Sequences induced by a **total loss of CCW** which develop into core damage beyond 24 hours due to non-mitigated RCP seal LOCA;
    - Sequences where loss of Main Control Room ventilation system causes core damage beyond 24 hours.

Example: Evaluation of Plant Specific System Capability

- For Each Detected CHLA the plant specific capability need to be evaluated and determined
  - Instrumentation for diagnostic and monitoring
  - Front line design SSCs
    - Alternative means
    - FLEX means if conections are available
  - Design Support systems
    - DC/AC, Cooling water,Instrument air
    - Alternate menas
    - FLEX means if connections are available

Example: Evaluation of Plant Specific System Capability

- Example: Inject to SGs



Reference: PC-1-17SAF-501-S030-45-1

(Sheet 1 of 2)

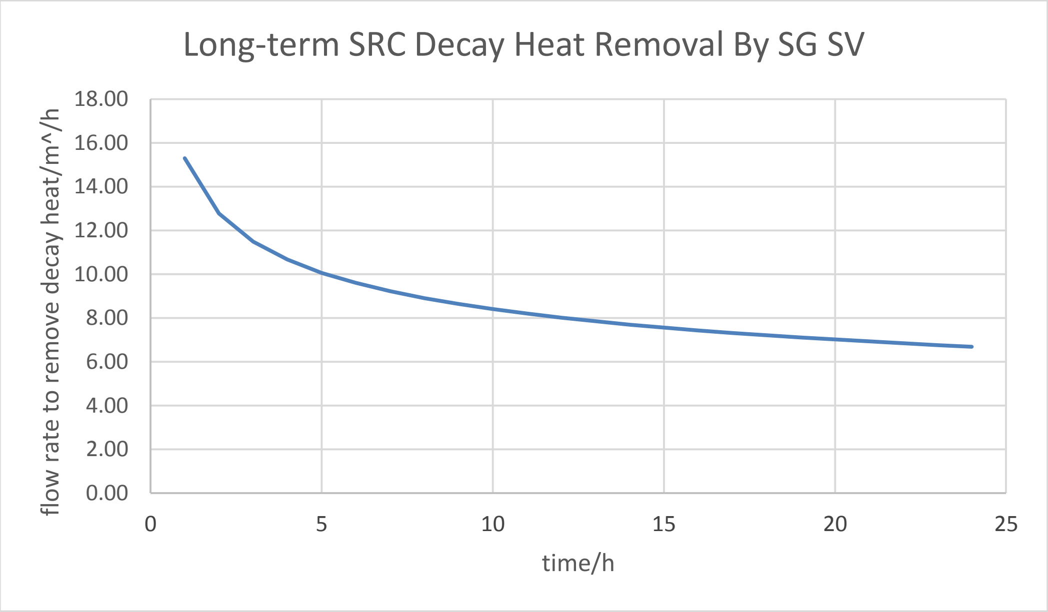
Example: Evaluation of Plant Specific System Capability

- Example: Inject to SGs

COMPONENT NAME	TAG NUMBER	Location	COMPONENT CHARASTERISTICS 1. Nominal flow-pressure, shutoff head, rated power, etc) 2. Limitations (NPSH for pumps, I&C interlocks, reset signals, etc.)	SUPPORT SYSTEMS			
				Instrument air (connection train/valve)/location	Cooling (train, valve, location)	AC BUS/MCC/BRKR (contact, location, voltage, power)	DC BUS/BRKR (contact, location, voltage, power)
Motor Driven SAF Pump	SAF-01APO	NA211	48.2m^3/h, 10Mpa, 10.1Mpa, 315kW, max. flowrate 71m^3/h, required NPSH <3.5mH_2O	-	SCW Header-A, SAF-27A and V28A, NA211	EMA130HP/13A/EX402 /35.7A/6kV/315kW	
	SAF-01BPO	NA219		-	SCW Header-B, SAF-27B and V28B, NA219	EMB130HP/13A/EX417 /35.7A/6kV/315kW	
Diesel Driven SAF Pump	SAF-02APO	NA209	48.2m^3/h, 10Mpa, 10.1Mpa, 450kW, max. flowrate 71m^3/h, required NPSH <3.5mH_2O	Starting air, SAF-11ATK	Reusing Water Summit Tank (Gravity Drain), SAF-V18A and V19A, NA209	-	
	SAF-02BPO	NA217		Starting air, SAF-11BTK	Reusing Water Summit Tank (Gravity Drain), SAF-V18B and V19B, NA209	-	
Startup and Shutdown Feedwater Pump	SSF-01PO	NA231	100m^3/h, max. Flowrate 130m^3/h, 10Mpa, 450kW, required NPSH <4.5mH_2O	-	WAB (service water), SSF-V13 and SSF-V14, NA231		
Emergency Feed water Tank Outlet Valves	SAF-V01A	ND-206	-	-	-	ECG1-1010AT/01T/ NA209/380V/0.55kW/ 2.4A	
	SAF-V01B	ND-206	-	-	-	ECG2-1010AT/01T/ NA217/380V/0.55kW/ 2.4A	

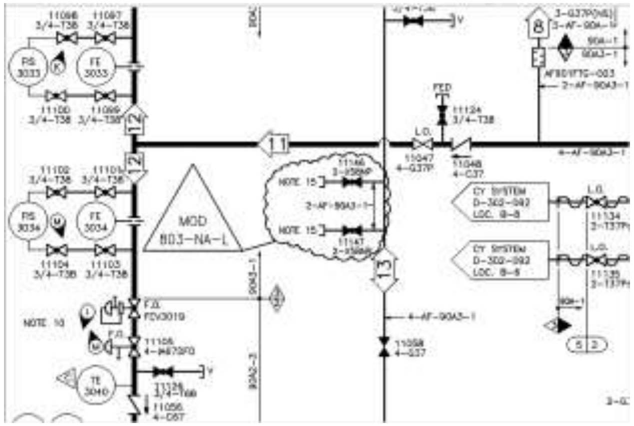
Example: Evaluation of Plant Specific System Capability

- Example: Inject to SGs



Alternative Means (FLEX)?

- Discussion about possible solutions...
  - Availability of FLEX fast conections?
  - Availability of people to do fast conections?
  - Available time window?
  - Expanded SEOPs and SAMG for FLEX



Example: Available time window from FSAR

Reference	19.2.4.3.1	19.2.4.3.2	19.2.4.3.3	19.2.4.3.4	19.2.4.3.5	19.2.4.3.6
Event Sequence / PIE	LLOCA w/o SI	LOCA w/o SIR	SLOCA w/o SI**	SLOCA w/o SIR**	LOMF w/o SAF and SI**	SGTR w/o SI and SAF
Top of core uncover	0.37	0.37	475.00	21062.00	8820.00	10100.00
Reactor trip signal	0.6	0.60	60.10	60.00	7.70	178.00
Main RCP trip	-	-	-	-	7.70	-
SG full depletion	-	-	-	-	2400.00	-
Safety Injection begins	-	0.60	-	102.00	-	-
1st Core uncovery completely (BOF	1.00	-	11960.00	37030.00	11770.10	-
Zr-water reaction begins	5.00	7450.00	-	-	11010.10	-
Accumulator A/B start inject	5.00	5.00	1730.00	690.00	54803.70	32200.00
Accumulator A/B Empty	60.00	60.00	5210.00	18090.00	-	32330.00
Safety Injection stop	-	-	-	17040.00	-	-
2nd Core uncovery completely (BO	150.00	-	-	-	-	-
Core begins to melt	-	-	12110.00	36910.00	-	15730.00
Core support plate failure	2893.00	14082.00	-	-	18127.3	-
RPV lower head failure	5724.00	18819.00	24337.00	54215.00	54803.70	32170.00
RPV lower head failure (h)	1.59	5.23	6.76	15.06	15.22	8.94

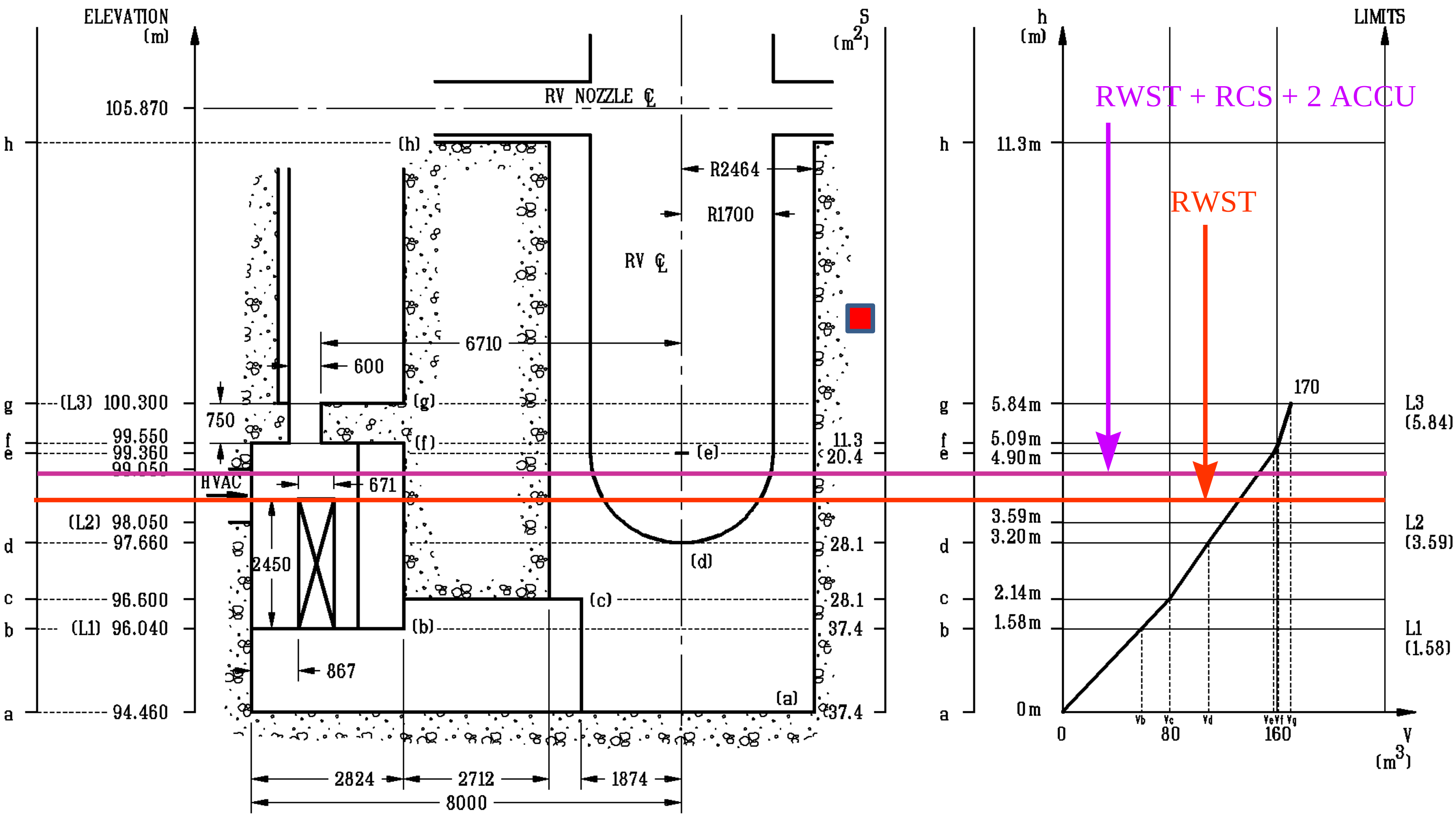
**Creep failure on SGTR or HL is not observed, time to RPV failure could be shorter

Long term response for flooding containment

Moderate time response for flooding containment

Critical time response for flooding containment, it should be checked with MELCOR calculation if quenching of corium in cavitz is possible to prevent M

Example: Containment Geometry/Wet Cavity ?



Example: Containment Geometry/Wet Cavity ?

(1) PDS	(2) PDS Frequency	(3) Sequence	(4) Sequence Frequency	(5) Time (sec) Reactor Scram	(6) Time (sec) Core Uncovery	(7) Time (sec) T > 650 K	(8) Time (sec) Onset of Core Melting	(9) Time (sec) Support Plate Failure	(10) Time (sec) RV Failure	(11) Time Win (sec) RV External Flooding (9)-(7)	(12) Time Win (sec) RV Cavity Flooding (10)-(7)
TEHAYN	1.282E-05	TRA12_1	3.80E-06	9	4171	4725	6126	6640	8440	**1915**	**3715**
TEHANN	7.217E-06	TRA12_33	5.40E-08	9	4175	4734	6136	6530	8330	**1796**	**3596**
ALLBYN	6.450E-06	LLO2_1	1.30E-06	0	2750	3301	4012	6698	14679	3397	11378
No CMT Heat Removal		LLO2_1A	N/A	0	4014	4635	5444	8909	15358	4274	10723
WUUUUB	4.424E-06	SGR9_1	3.00E-06	306	59661	62206	65039	68059	74381	5853	12175
SEHAYN	4.350E-06	SBO20_1	2.60E-06	0	2566	3148	4235	9636	11436	6488	8288
Increase Debris Mass Ejected		SBO20_1A	N/A	0	2566	3148	4235	9636	11436	6488	8288
SLLBYN	2.015E-06	SLO3_1	1.50E-06	558	61498	64944	122947	146458	181236	81514	116292
FR-C.1 Depressurization Fails		SLO3_1A	N/A	558	56027	60059	65320	69737	82423	9678	22364
VXXXXB	2.003E-06	ISL1_1	2.00E-06	9	20378	21327	22360	25621	32983	4294	11656
TEHNNN	1.819E-06	SBO63_37	3.90E-08	0	6004	6641	8217	10147	11947	3506	5306
Reduced Debris/Coolant CHF		SBO63_37A	N/A	0	6004	6641	8217	10147	11947	3506	5306
Reduced Spreadhout Area		SBO63_37B	N/A	0	6004	6641	8217	10147	11947	3506	5306
SLNNN	1.250E-06	INA3_1	6.80E-07	0	2638	3206	4307	15361	45670	12155	42464
One Fan Cooler Running		INA3_1A	N/A	0	2634	3212	4304	14710	39693	11498	36481
ALLBYI	9.330E-07	LLO2_2	1.60E-07	0	2630	3187	3890	6559	15390	3372	12203
TERAYN	4.177E-07	LSP6_1	4.10E-07	0	6200	6939	8594	25533	1.E+10	18594	1.E+10
N/A	N/A	CREEP6	N/A	9	4174	4732	6133	6467	1.E+10	**1735**	1.E+10

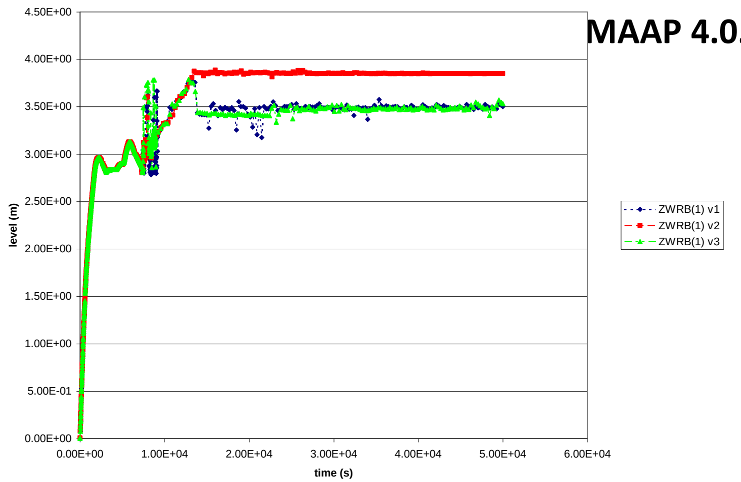
Minimum Time Window (sec) for RV External Flooding 1735
Minimum Time Window (min) for RV External Flooding 29

Minimum Time Window (sec) for RV Cavity Flooding 3596
Minimum Time Window (min) for RV Cavity Flooding 60

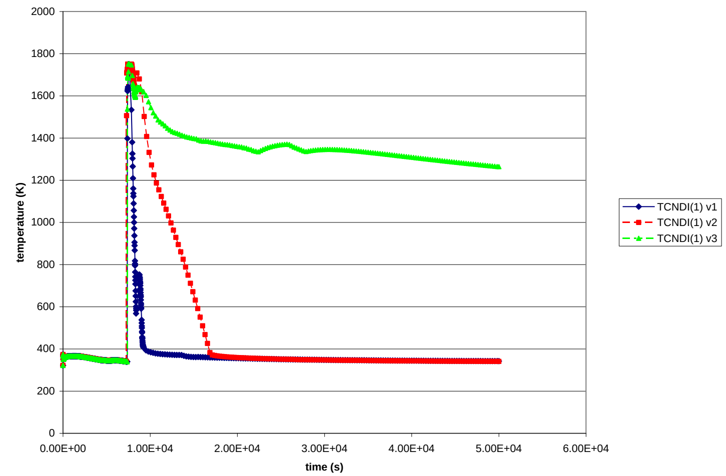
Note: The 1E+10 value is to represent a large time for the calculation since there is no RV failure.

Example: Containment Geometry/Wet Cavity ?

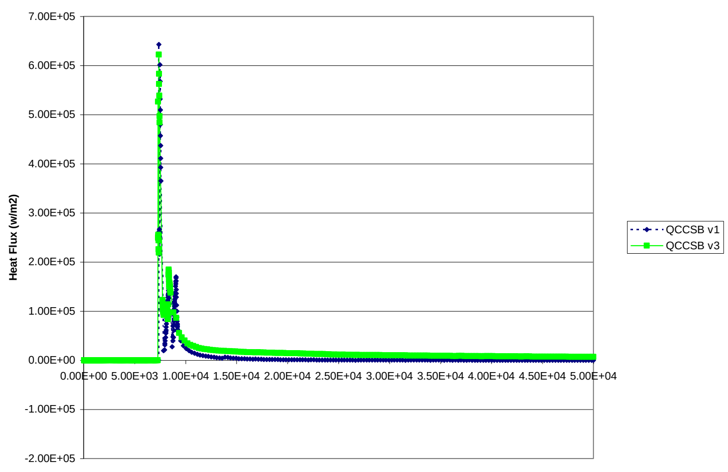
MAAP 4.0.5 Analysis



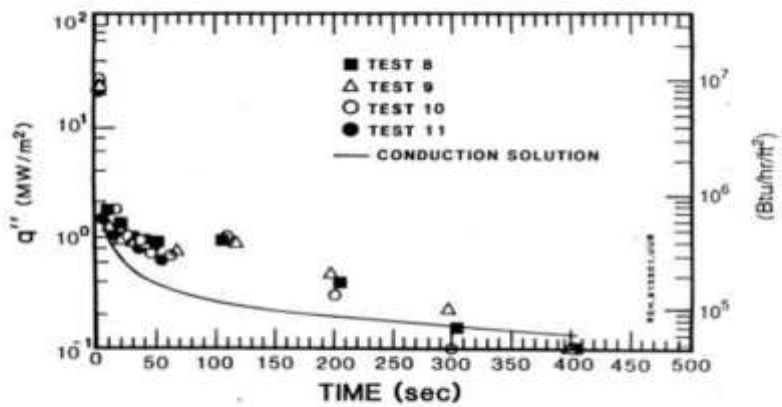
Water Level in Rx Cavity



Downward Corium-Concrete Interface Temperature



Heat Flux From Debris Pool IC to Concrete Floor



Comparison of the calculated heat fluxes for EPRI Tests 8, 9, 10 and 11 and which could be removed by conduction. (Source: Malinovic et al., 1989.)

DEC Earthquake Evaluation (Seismic PSA?)

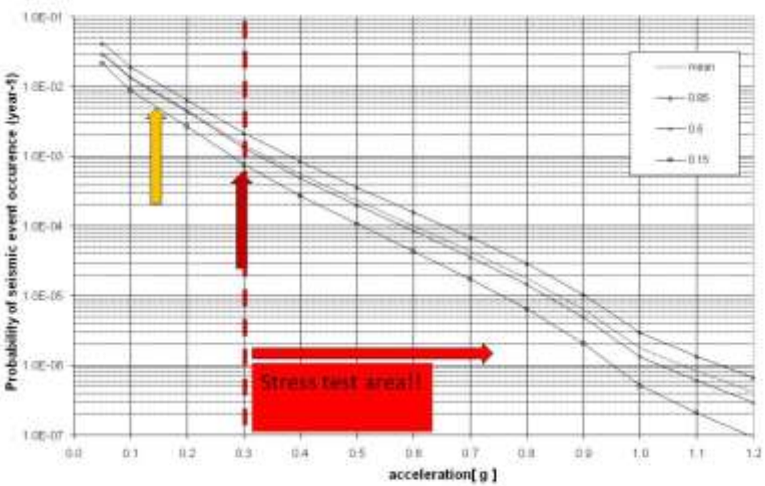
APoS

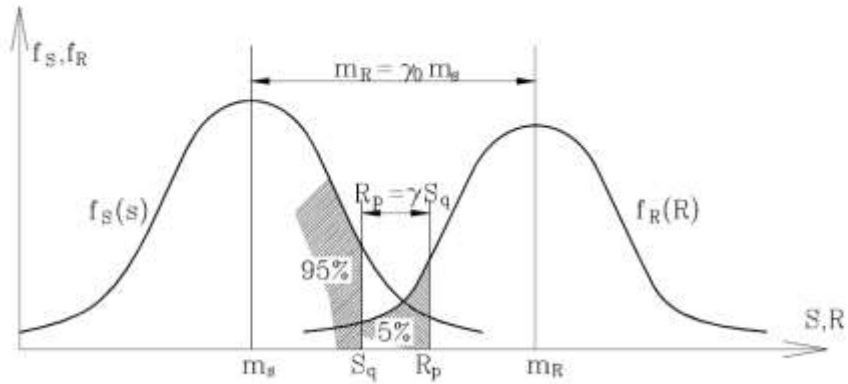
EPRI TBR Rev.1: External events.

The extreme nature of the external events at Fukushima Dai-ichi highlighted the degree to which such an event could severely alter the management of the accident. To further enhance the robustness of SAMGs in response to the accident at Fukushima Dai-ichi, this

report reflects the distinct challenges to accident management arising as a result of possible wide-scale damage to the infrastructure that supports mitigation systems. **Although they do not directly change the types of actions that can be taken, these challenges require specific consideration in ensuring the robustness of SAMG implementation for particular plants.**

Inclusion of these challenges is intended to support the enhancement of updates to specific site SAMG implementations.



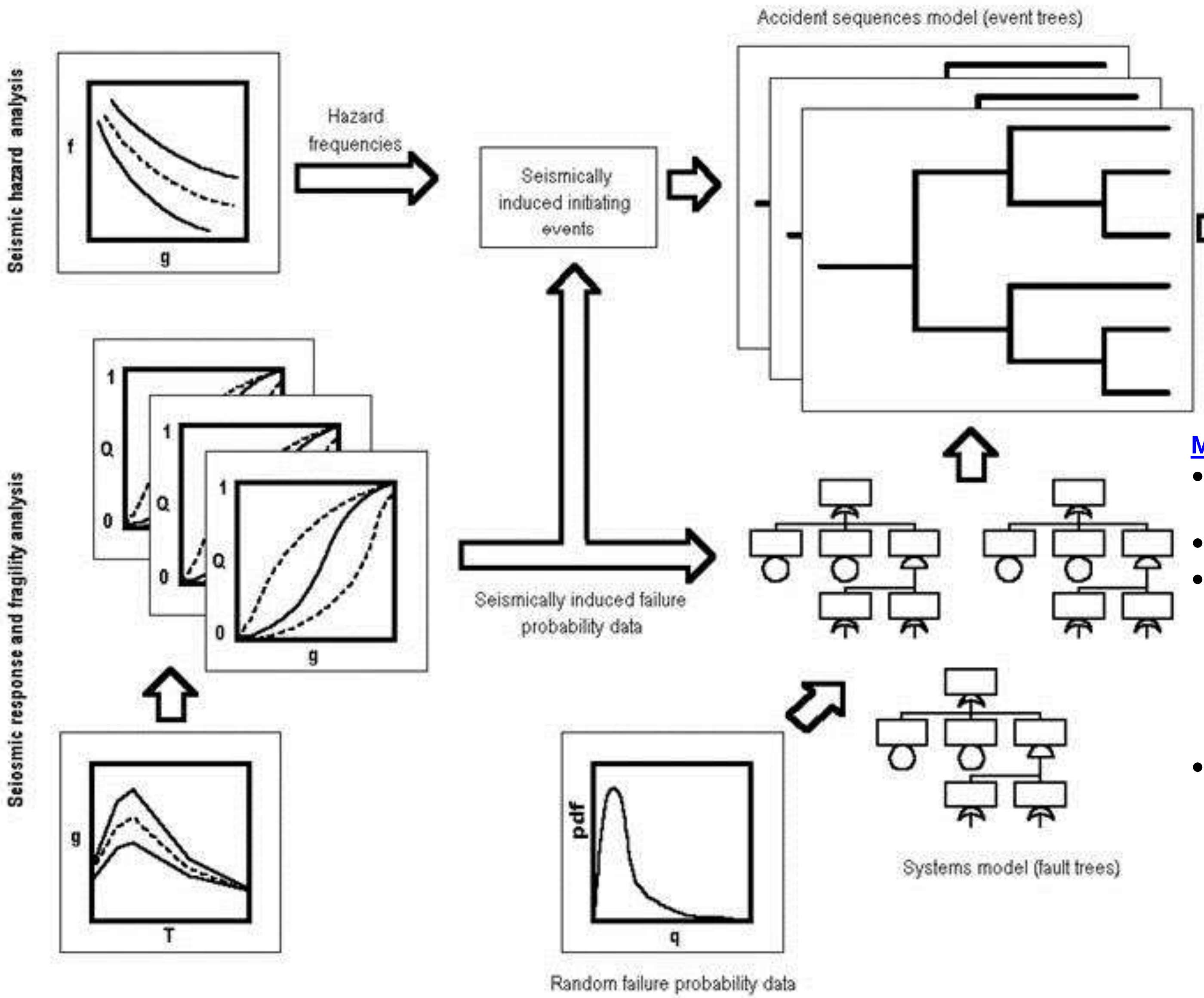


Probabilistic safety concept is based on the assumption that there is **no completely safe structures**.

Any structure or structural element has a probability of failure load.

The calculation takes all the variables which are statistically processed and uses them in the form of the distribution function of a certain probability.

DEC Earthquake Evaluation



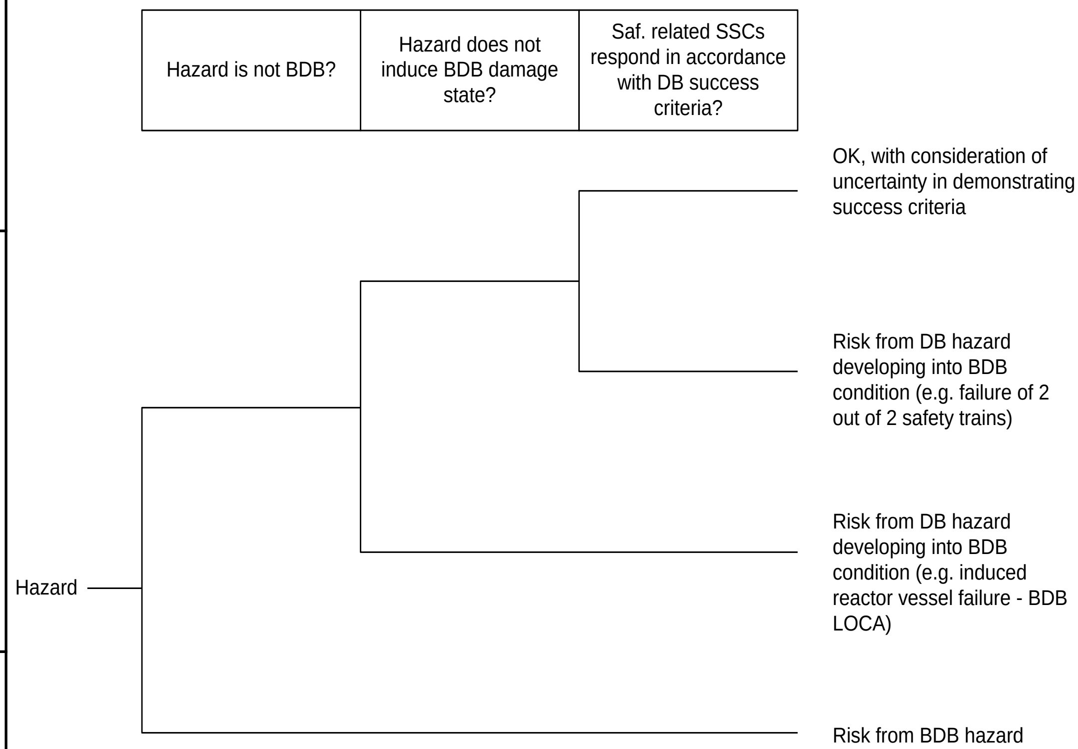
Methodology:

- **Identification of ETs Success Paths**
- **Mapping to critical SSCs**
- **Determination of failure mode with corresponding HCLPF identified for each critical safety function and initiator / damage state**
- **Evaluation of Plant Level Seismic Margin (PDS status, cliff edge, remaining success path)**

DEC Earthquake Evaluation

Table 1: Summarized Comparison of “Stress Testing” and PSA

Element	“Stress Testing”	PSA
Initiating Events	Return periods for exceeding DB events. Margins for inducing BDB states.	Hazard characterization. Hazard frequency curves. Hazard damage states.
Systems / Functions for Prevention of Core Damage	Regardless of the return periods and margins, postulate the loss of critical functions: SBO or / and Loss of UHS. Margins to core damage.	Accident sequences and functions. Logic models. CD risk quantification.
Containment Status / Severe Accident Management	Regardless of the possibilities for the avoidance, postulate core damage. Margins to containment integrity failure.	Containment systems models. Phenomena models. Release risk quantification.



Spent Fuel Pool Vulerability

Example: SFP States for Risk Significance Evaluation, Time Window to Recover SFP cooling

State	Description	SFP Decay Heat (MW)	SFP Water Inventory	Time to Boil (hr) (1)	Time to Evaporate to FA+1m (hr) (2)	Duration (days)	Duration (%)
SFP1	Complete core from the previous cycle in the SFP (3)	6.40 – 4.39	C1	11.0 – 20.0	111.3 – 162.6	15.2	2.8%
SFP2	Partially burnt FAs from previous cycle returned to the core. Decay heat level higher than 1.5 MW.	2.37 – 1.50	C1	44.8 – 74.9	303.3 – 474.7	71.2	13.0%
			C2 (C3)	32.0 – 53.5 (32.0 – 53.5)	224.7 – 351.7 (174.1 – 272.6)		
SFP3	Decay heat level lower than 1.5 MW.	< 1.50	C2 (C3)	> 53.5 (> 53.5)	> 351.7 (> 272.6)	461.5	84.2%
					Total:	547.9	100%

Fukushima accident – SANDIA Evaluation

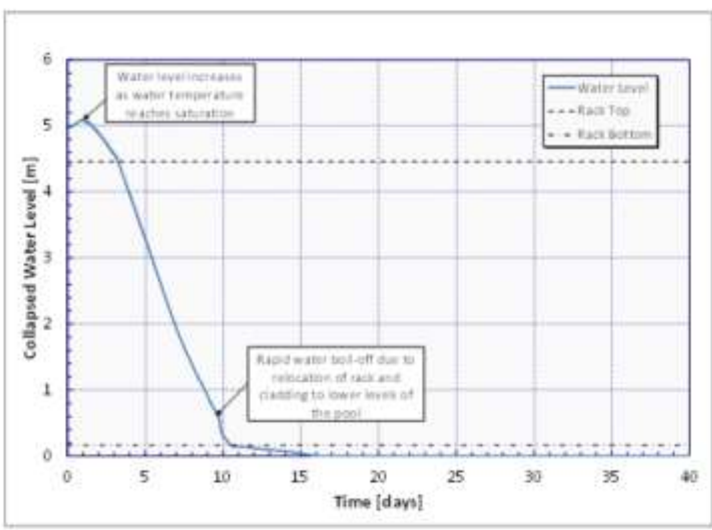


Figure 122. MELCOR Predicted Spent Fuel Pool Collapsed Water Level (0.5 m above Top of Racks Case).

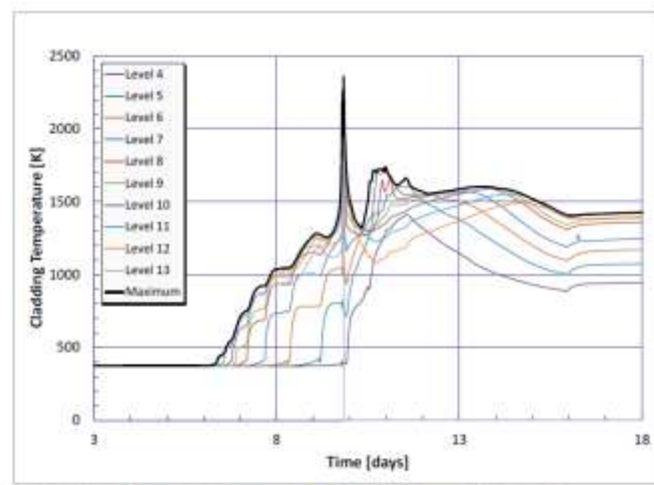


Figure 124. MELCOR Predicted Spent Fuel Pool Maximum Cladding Temperatures (0.5 m above Top of Racks Case, Detailed View).

Questions?

Comments?

Thanks for your attention!