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Severe Accident Analysis and Probabilistic Safety Assessment for an LFR

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Severe Accident Analysis

Severe Accident Definition

Severe Accident in US Regulatory History

- Origin of Severe Accident Analysis: **WASH-740** ("*Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants*", US Atomic Energy Commission, 1957):
  - ✓ Study of **possible consequences** in terms of injury to persons and damage to property, if certain **hypothetical major accidents** should occur in a typical large nuclear power reactor (LWR).
  - ✓ Goal: understand possible reactor hazards to expand atomic energy use and site power reactors near populated areas.
- Introduction of '**Maximum Credible Scenario**' concept: **hypothetical core-melt scenario** based on **extremely conservative assumptions**:
  - ✓ Unlikely (though conceivable) combinations of failure and error and weather conditions
  - ✓ No containment
  - ✓ ½ core inventory released to atmosphere
  - ✓ Use of atomic bomb fallout data to estimate radiological consequences
- **Source term**† defined for **bounding** purposes **irrespective** of **reactor design**

*† **Source term**: the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release.*

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Severe Accident in US Regulatory History

- **Radiological consequences** in **WASH-740** were **grossly overestimated**
- This changed with advent of **Probabilistic Safety Assessments** (WASH-1400, 1975) and the concept of '**Mechanistic Source Term**' (NUREG-1150, 1991 and NUREG-1465, 1995), which introduced **realistic** (vs. hypothetical) **accident sequences** and **reactor-specific**, **mechanistically** derived **source terms** based on **accident progression modelling**
- The 'Major Accident' (maximum credible scenario) becomes the '**Severe Accident**' (realistic accident sequence of greatest severity)
- For a LWR, a **Severe Accident** is defined as a **beyond design basis event** involving **significant core damage** (core melt), irrespective of the off-site consequences
- The Severe Accident is used to **size** the **last confinement barrier** and **design** appropriate Level 4 **Defence-in-Depth provisions** (mitigation)

⚠ **Severe Accident as 'realistic accident sequence of greatest severity', not a reactor-specific plant damage state (e.g. core melt)**

IAEA Event Categories (Water-Cooled Reactors)

<table>
<tr><th colspan="6">Defence-in-Depth Levels</th></tr>
<tr><th>Level 1</th><th>Level 2</th><th>Level 3</th><th colspan="2">Level 4</th><th>Level 5</th></tr>
<tr><td colspan="5">Plant states (considered in design)</td><td rowspan="5">Off-site emergency response (out of the design)</td></tr>
<tr><td colspan="2">Operational States</td><td colspan="3">Accident Conditions</td></tr>
<tr><td rowspan="3">Normal Operation</td><td rowspan="3">Anticipated Operational Occurrences</td><td rowspan="3">Design Basis Accidents</td><td colspan="2">Design Extension Conditions</td></tr>
<tr><td>DEC-A</td><td>DEC-B</td></tr>
<tr><td>Without significant fuel degradation</td><td>With core melting</td></tr>
</table>

Severe Accident

- **Accident Conditions**: deviations from normal operation that are less frequent and more severe than anticipated operational occurrences.
- **Design Basis Accident**: a postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits.
- **Design extension conditions**: postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.
  - ✓ **DEC-A**: conditions in events <u>without significant fuel degradation</u>
  - ✓ **DEC-B**: conditions in events <u>with core melting</u>
- **Severe Accident**: *accident **more severe** than a **design basis accident** and involving **<u>significant core degradation</u>**.*

GIF Event Categories for an LFR

Defence-in-Depth Levels					
Level 1	**Level 2**	**Level 3**	**Level 4**		**Level 5**
Plant states (considered in design)					Off-site emergency response (out of the design)
Operational States		Accident Conditions			
Normal Operation	Anticipated Operational Occurrences	Design Basis Accidents	Design Extension Conditions		
			DEC-A	**DEC-B**	
			Without significant fuel degradation	**With limited core damage**	

Severe Accident

- The Generation IV International Forum (GIF) defined **event categories applicable** to a **Lead-cooled Fast Reactor**
- GIF safety design criteria **encourage LFR designers** to **seek** the **elimination by design** of **accident situations** with **large core melting**
- This implies a **redefinition** of **DEC-B situations** compared to IAEA standards: from '**core melt**' to '**limited core damage**'
- But what are 'fuel degradation' and 'core damage' for an LFR?

What is Core Degradation in an LFR?

- **Nuclear fuel** — **fissionable nuclear material** in the form of **fabricated elements** for **loading** into the **reactor core** of a civil nuclear power plant or research reactor

- **Reactor core** — the **central part** of a **nuclear reactor** where **nuclear fission occurs** and which contains the **fuel assemblies** and the **structures** and **systems necessary** to **sustain** and **control** the **fission chain reaction**, notably:
  - ✓ Fuel (including fuel rods and the fuel assembly structure)
  - ✓ Coolant
  - ✓ Moderator
  - ✓ Control rods

- The **reactor core** therefore **ensures** all **three fundamental safety functions**: reactivity control, cooling and confinement (first physical barrier)

  ⇒ **Core degradation implies loss** of **multiple safety functions** (fuel confinement, but also reactivity control and/or core cooling) — is this true for an LFR?

Core Degradation in an LFR

What is Core Degradation in an LFR?

- In an LFR, **generalised core melt** can be made **physically impossible by design** by ensuring e.g.:
  - ✓ Negative global reactivity feedback
  - ✓ Practical elimination of prompt-criticality scenarios
  - ✓ Extreme improbability of loss-of-coolant and total loss-of-cooling scenarios

- Dominant **fuel pin degradation mechanisms** in an LFR with stainless steel fuel cladding:
  - ✓ Corrosion
  - ✓ Creep
  - ✓ Swelling
  - ✓ Pellet-clad interactions

  ➔ **Cumulative effects** leading to risk of **fuel pin rupture** (by creep or plastic deformation)

- The realistic accident sequences of greatest severity in an LFR are thus associated to **loss** of **first confinement barrier**, but *what is the effect on the other fundamental safety functions*?

Fuel Pin Rupture Types in a Fast Reactor

01

Crack

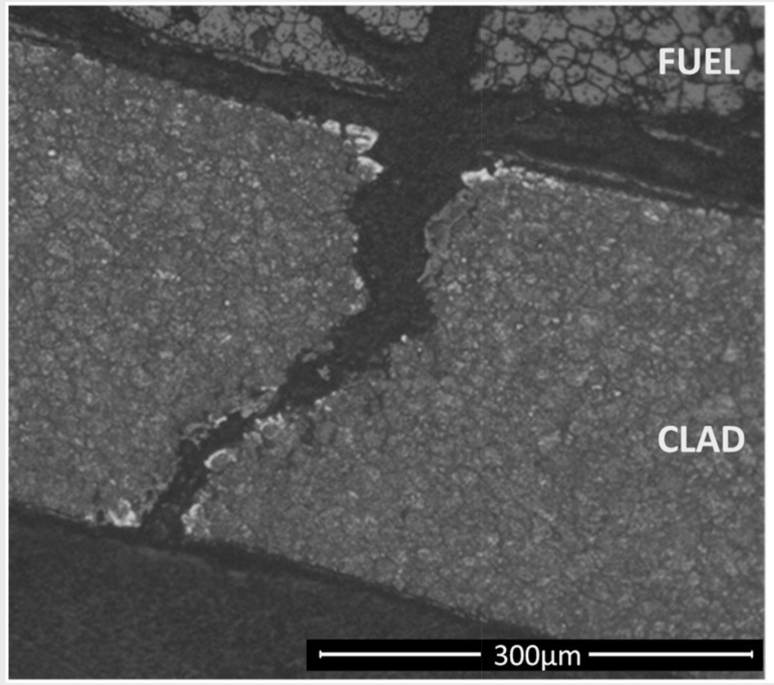
Leakage of gaseous FP into coolant

02

Narrow rupture

Release of gaseous and volatile FP

Limited release of liquid & solid FP



Narrow rupture fuel pin failure in an AGR

Source: *M. A. Barker, Fuel pin failure modes in the UK's advanced gas cooled reactors, Progress in Nuclear Energy 175 (2024) 105328*

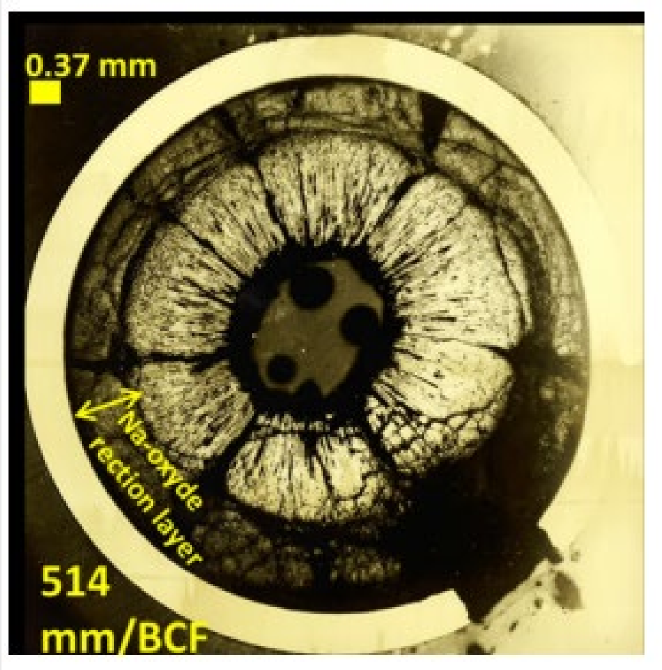
03

Wide rupture

Significant lead-fuel contact

Large release of FP

Potential release of fissile material



Wide rupture fuel pin failure in PHENIX reactor

Source: *C. Ding (2022). Multiphysical study of cladding failure detection in Sodium cooled Fast Reactors*

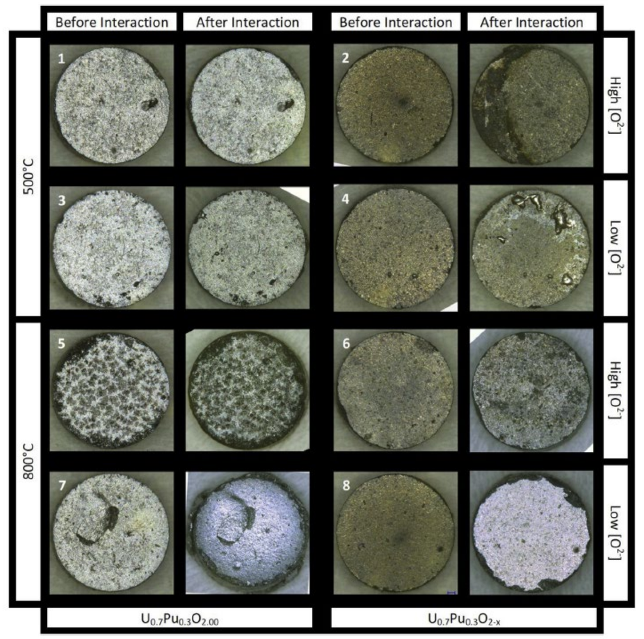
Fuel-Coolant Interactions in an LFR

- In Sodium Fast Reactors (SFR), the **MOX-Na interaction** leads to the **formation** of **sodium orthouranate/orthoplutonate**, whose **density** and **thermal conductivity** are ~½ that of MOX

$$\mathrm{(U,Pu)O_2 + Na_2O + \frac{1}{2}O_2 \rightarrow Na_2(U,Pu)O_4}$$

➔Severe **pellet-cladding mechanical interaction** leading to **wide rupture** and risk of fuel dispersion

- Experimental data in **LBE** (up to 800°C)†:
  - ✓ **Pellet integrity preserved**, no MOX–Pb interaction compounds
  - ✓ **No Pb diffusion** into surface regions of MOX pellets
  - ✓ **Limited actinide dissolution** (@ 800°C and low [O_2])



- **No** known **MOX–Pb chemical reaction** ➔ no propagation of rupture following Pb ingress into fuel pin

⇒ **<u>Narrow rupture approximation</u> = fission product release + limited actinide dissolution + no macroscopic fuel dispersion**
[*N.B.: To be verified experimentally above 800°C*]

† J.-F. Vigier *et al*., Interaction study between MOX fuel and eutectic lead–bismuth coolant, Journal of Nuclear Materials 467 (2015) 840–847

*new*cleo's Safety Design Criteria & Severe Accident Definition

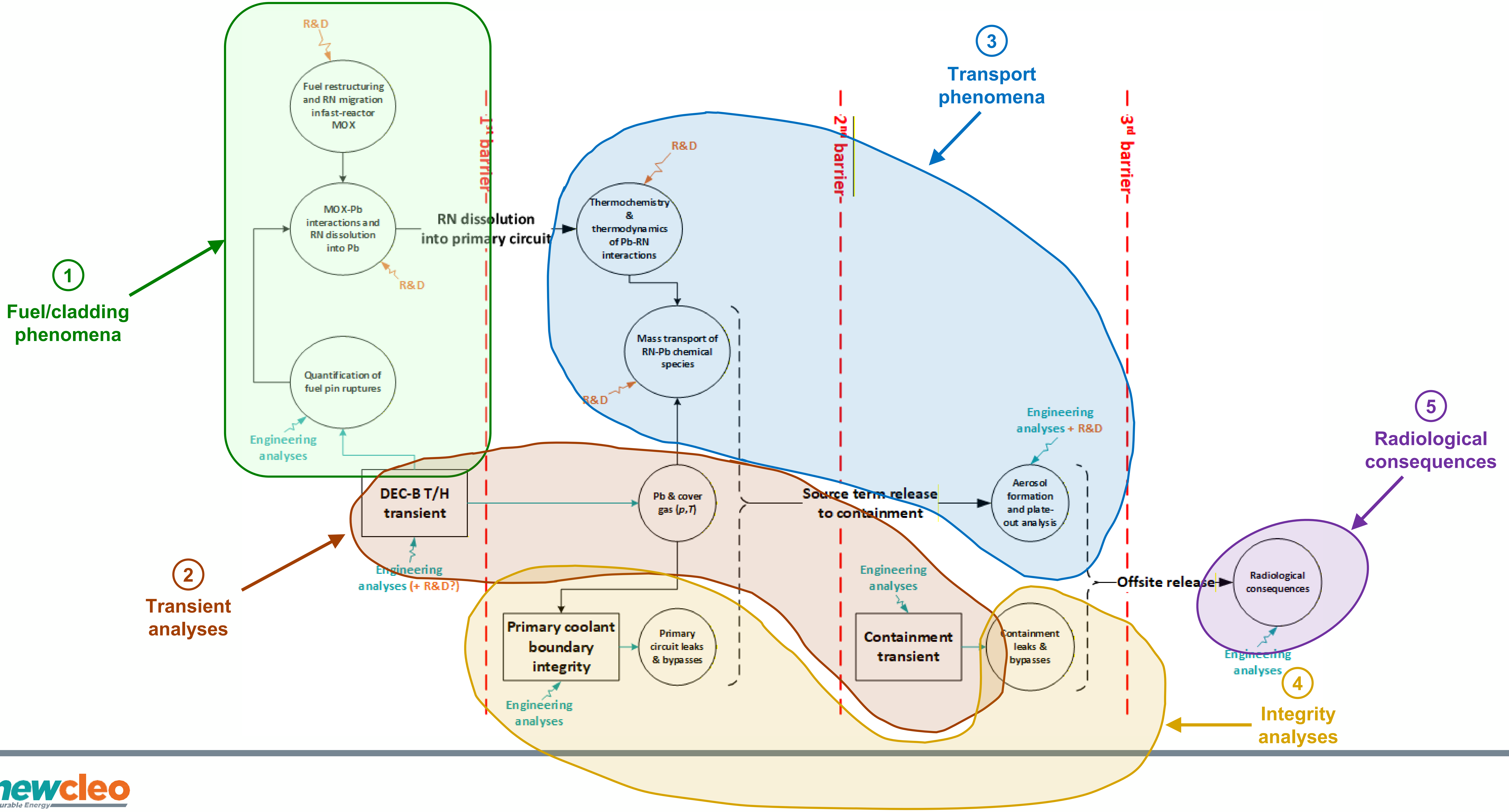
- *new*cleo's **safety design criteria** include:
  - ❑ Achieve by design the **physically impossibility** of **generalised core melt** by ensuring:
    - ✓ **Practical elimination** of **prompt-criticality** scenarios
    - ✓ **Practical elimination** of **total loss** of **core cooling** scenarios
    - ✓ **Total instantaneous blockage** scenarios are made **extremely improbable** and lead at worst to localised fuel/cladding melt
  - ❑ Demonstrate **Steam Generator Tube Rupture** scenarios do **not** lead to **significant reactivity insertion** (or to other conditions causing a simultaneous threat to the 1st and 2nd barrier)
  - ❑ Ensure sufficient **margins** exist to **localised fuel melting** to **prevent significant fuel dispersion** into primary circuit

- Under such conditions, **core cooling** and **reactivity control** are **preserved** $\Rightarrow$ **fuel degradation** and **core damage** are **<u>not</u> realistic accident sequence end states**

➔ **Severe accident**: *a situation* ***more severe*** *than a* ***design-basis event****, involving* ***significant <u>cladding</u> degradation*** *that, if* ***unmitigated****, could lead to radiological* ***releases*** *exceeding* ***off-site limits***

Severe Accident Analysis in an LFR

Focus points for developing an LFR-specific SAA methodology



Focus points for developing an LFR-specific SAA methodology

1. **Fuel/cladding phenomena**
   - ✓ Cladding degradation phenomena and **quantification** of **fuel pin ruptures** in a DEC-B scenario
   - ✓ **Burnup effects**: fuel restructuring and fission product migration in fast-reactor MOX
   - ✓ MOX–Pb interactions and **actinide/fission product dissolution** into Pb

     ⇒ **Source term released into primary circuit**
2. **Transient analyses**
   - ✓ Coupled **thermal-hydraulic/pin thermomechanical** transient in a DEC-B scenario
   - ✓ **Containment** transient in DEC-B scenario
3. **Transport phenomena**
   - ✓ **Thermochemistry** and **thermodynamics** of **interaction compounds** in **MOX–cladding–Pb system**
   - ✓ **Mass transport** of chemical species in source term
   - ✓ Aerosol formation and **plate-out analysis**
4. **Integrity analyses**
   - ✓ Primary coolant boundary integrity analysis → evaluation of **primary circuit leaks** and **bypasses**
   - ✓ Containment integrity analysis → evaluation of **containment leaks** and **bypasses**

     ⇒ **Off-site release (source term released into containment then to the environment)**
5. **Radiological consequences**
   - ✓ Evaluation of **off-site radiological consequences** to the public

Severe Accident Analysis in an LFR

| Differences Compared to Light Water Reactors and Sodium Fast Reactors

- **No core melt $\Rightarrow$ no corium formation $\Rightarrow$ no risk of hydrogen explosion**

- **Integrity** of second confinement barrier (**primary coolant boundary**) **not** deterministically **compromised** by the accident scenario

- **Coolant @ atmospheric pressure** + **no energetic reactions** involving coolant and fuel $\Rightarrow$ **no** appreciable **containment pressurisation** $\Rightarrow$ much **lower leak rate** to the **atmosphere**

- **Chemical affinity** of **Pb** with dominant NO-volatile **fission products** (Cs, I, Ba, Mo, etc.) $\Rightarrow$ **high retention factor** of **radiological inventory**

$\Rightarrow$ **Severe accident off-site release in an LFR much more <u>benign</u> than in an LWR/SFR**

R&D Needs for the Severe Accident Analysis in an LFR

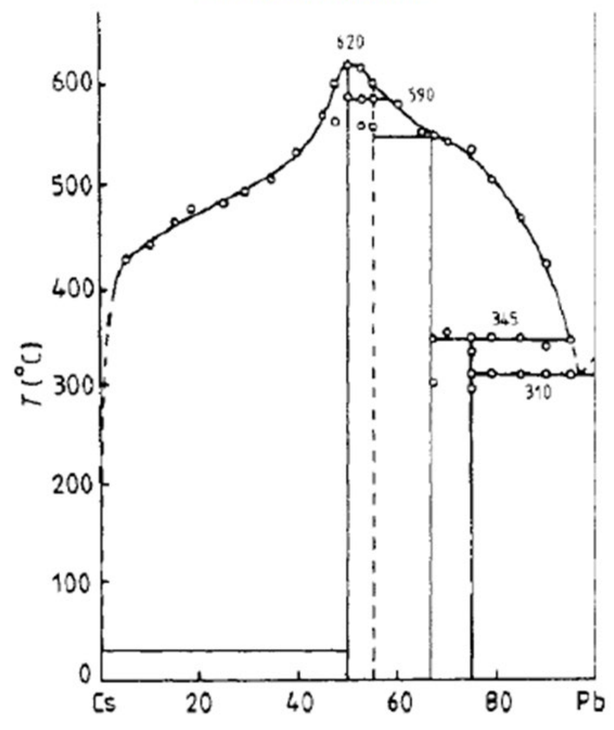
- **Claim** on **preservation** of **MOX pellet integrity** upon **interaction** with **Pb** must be **experimentally confirmed** at **T** representative of a **SA situation**

- Partial **experimental substantiation** of safety claim on "**in-Pb retention**"
  - ➔ Further **R&D** needed to **underpin** calculation of **retention factors**
    - ✓ Full **thermochemical** and **thermodynamic** analysis of **MOX–cladding–Pb interaction compounds**
    - ✓ **Burnup** effect (fuel restructuring) on **FP** migration and **dissolution** in **Pb**
    - ✓ Measurement of **chemical activity** of **interactions compounds** in **Pb** $\Rightarrow$ **radionuclide vapour pressure** in **cover gas** (source term)

- **Aerosol species** formed in **Pb** and **cover gas** may **differ** from those in **LWRs**
  - ➔ Experimental **characterisation** of **LFR-specific aerosols** required to quantify **plate-out effect**

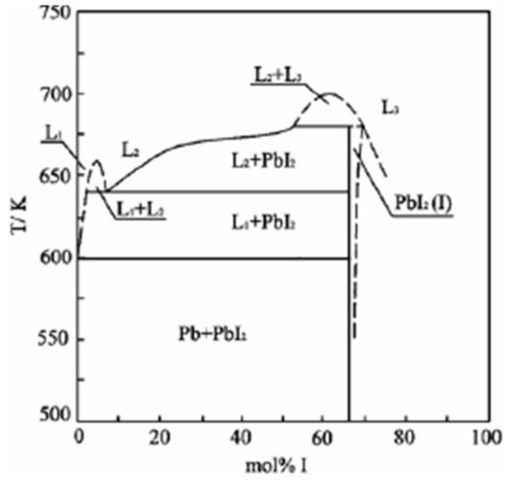
- **R&D programme** will allow **relaxing requirements** on **containment**

Figure 5.2.31: Phase diagram of the Cs-Pb system



Source: Reproduced with permission of IOP Publishing from Meijer (1986).

Figure 5.2.23: Phase diagrams of the Pb-I system



Source: Revised version (Zhu, 2006) of the phase diagram on the left published in Konings (1995), reproduced with permission of Wiley-VCH. Phase diagram on the right reprinted with permission from Rosztoczy (1965), © 1965 American Chemical Society.

Probabilistic Safety Assessment

Probabilistic Safety Assessment – Historical Context

- **Probabilistic Safety Assessment** (PSA) is a methodological approach to identifying **accident sequences** that can follow from **broad range** of **initiating events**, and includes a **systematic** and **realistic determination** of **damage** and **radioactive releases** and their **frequencies**

- Formally introduced in **1975** by the US NRC's **WASH-1400** report, it forms the **second pillar** of a reactor's **safety analysis** along with the **Deterministic Safety Assessment**, with a goal to:
  - ✓ **Balanced design**: ensure no single feature or initiating event dominates the risk profile and verify independence of defence-in-depth levels
  - ✓ **Cliff edge prevention**: prevent scenarios in which small deviations in plant parameters could give rise to large variations in plant conditions
  - ✓ **Compliance**: validate design against acceptance criteria for societal risk (where applicable)

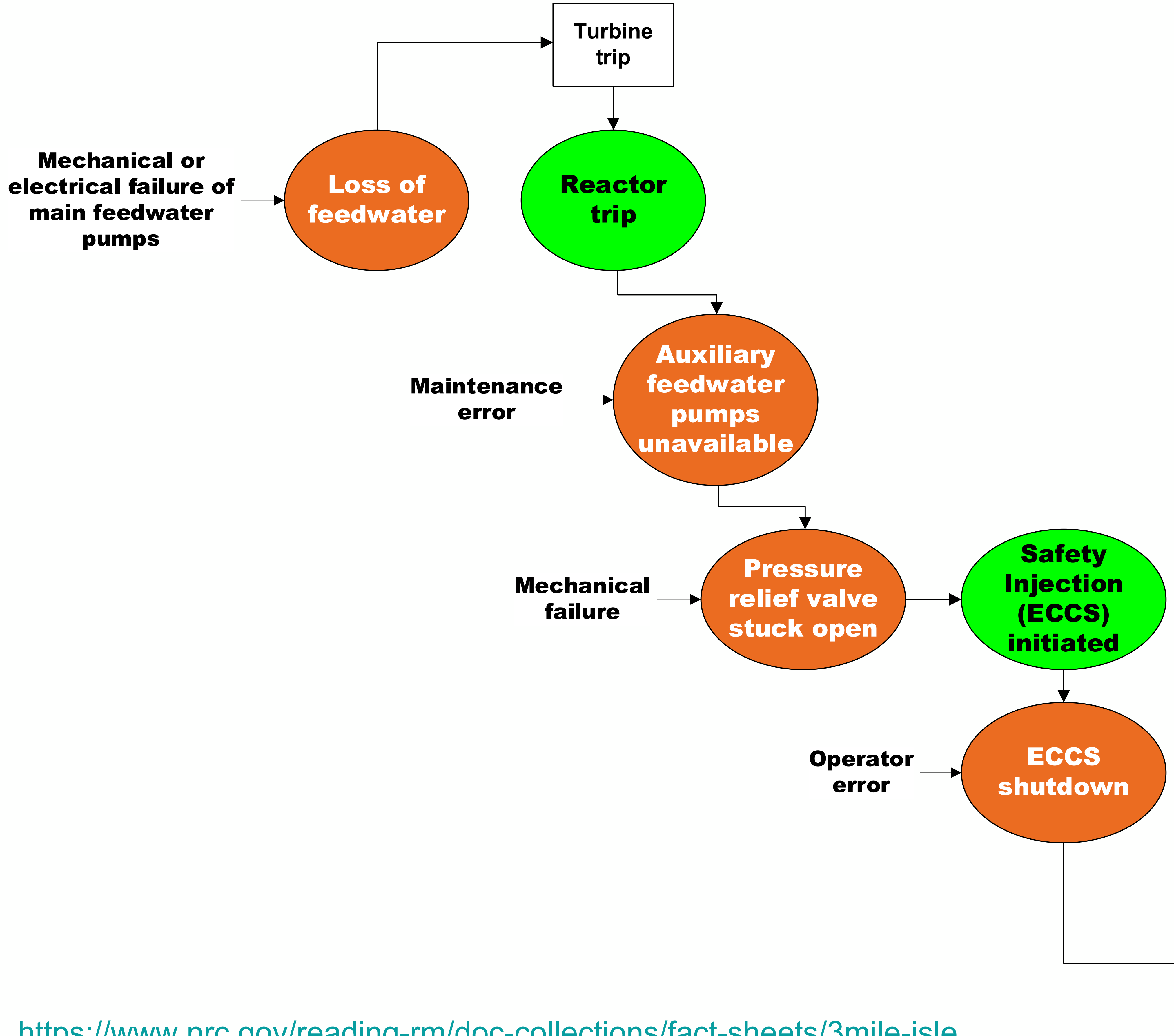
- In international practice, three levels of PSA are historically defined:
  - ✓ **Level 1** – Identifies event sequences leading to core and/or fuel damage and their frequency
  - ✓ **Level 2** – Analyse accident progression & barrier integrity to quantify large early release frequency
  - ✓ **Level 3** – Quantify off-site radiological consequences to assess public health and societal risk

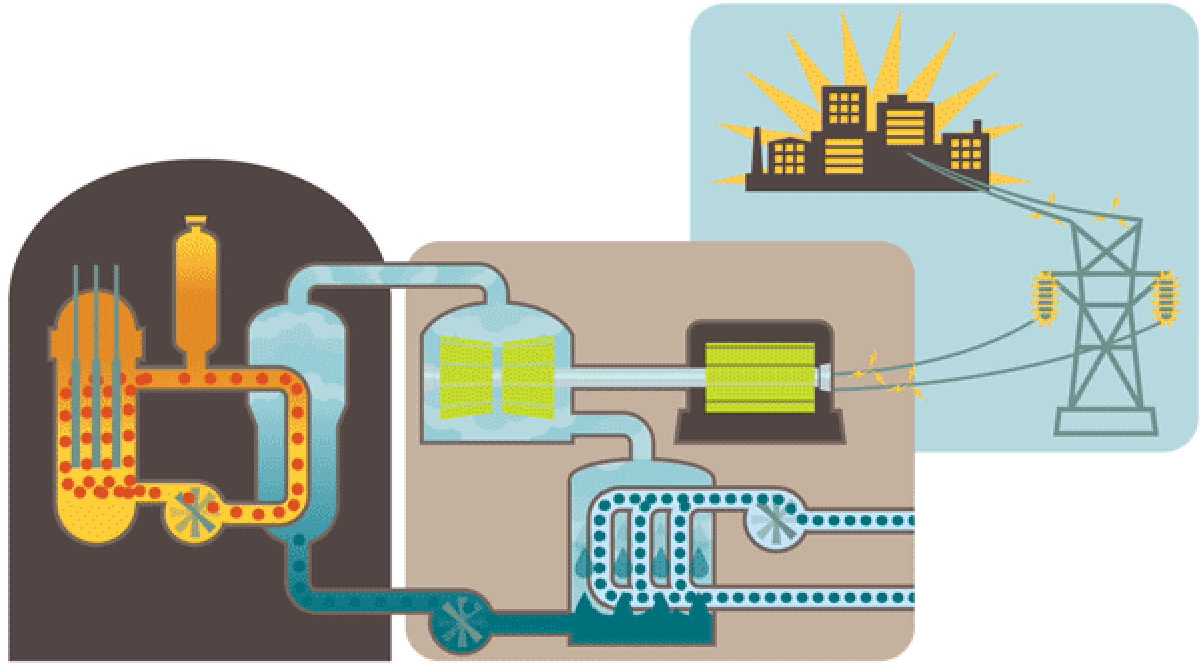
Design Basis – Deterministic Safety Assessment

- Historically, the **design** of a reactor's **safety architecture** is based on **deterministic safety analyses**
  - ↪ While probabilistic safety is about what is *likely* to happen, **deterministic safety** is about **proving** that the **plant** can "**survive**" a given list of a few **accident scenarios**

- **Deterministic safety** relies on a fixed set of **predefined accident scenarios**, comprising:
  - ✓ **Postulated initiating event**: a hypothesised event identified in design as capable of leading to abnormal plant conditions (anticipated operational occurrences) or accident situations
  - ✓ **Aggravating conditions**: specific hypothetical failures or environmental conditions added to a postulated initiating event to make it more challenging for the safety systems to handle, such as:
    - **Single Failure Criterion** (e.g., most reactive control rod stuck)
    - **Loss of Offsite Power**
    - Most **unfavourable** external **environmental conditions**
  - ✓ **Conservatism**: the practice of deliberately choosing input values that are more pessimistic than what is realistically expected to ensure adequate safety margins.

Three Mile Island Accident — March 28, 1979





https://www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle

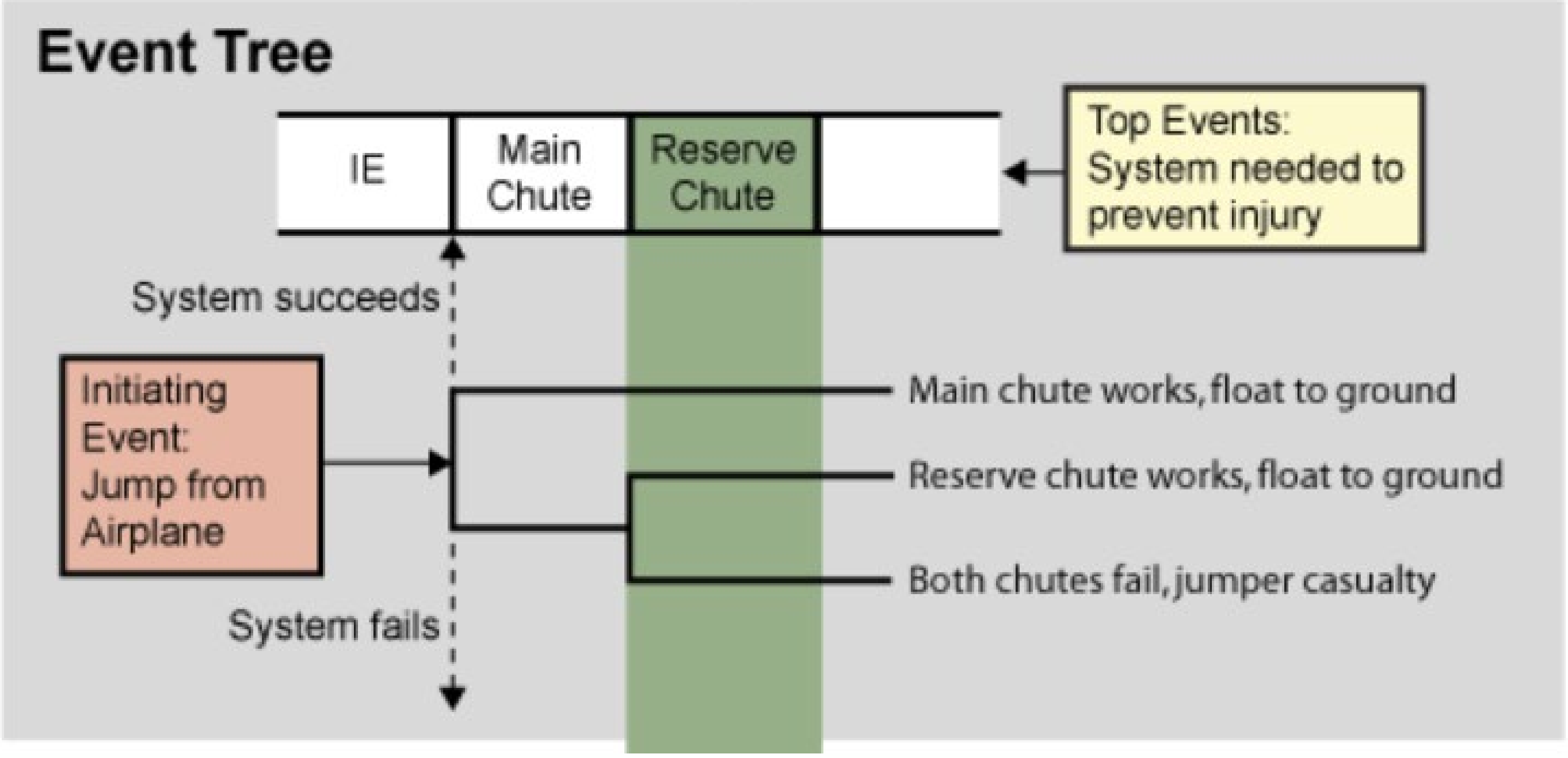
Probabilistic Safety Assessment – Accident Sequences

Risk = frequency of undesired event × severity of consequences

- ✓ **What** can go **wrong**?
- ✓ **How likely** it is?
- ✓ What might its **consequences** be?

- **Event Tree Analysis**
  Graphical mapping of potential **accident sequences** following an **initiating event**. It groups sequences by their consequence and calculates the **frequency** of each **consequence**.

Sample PRA

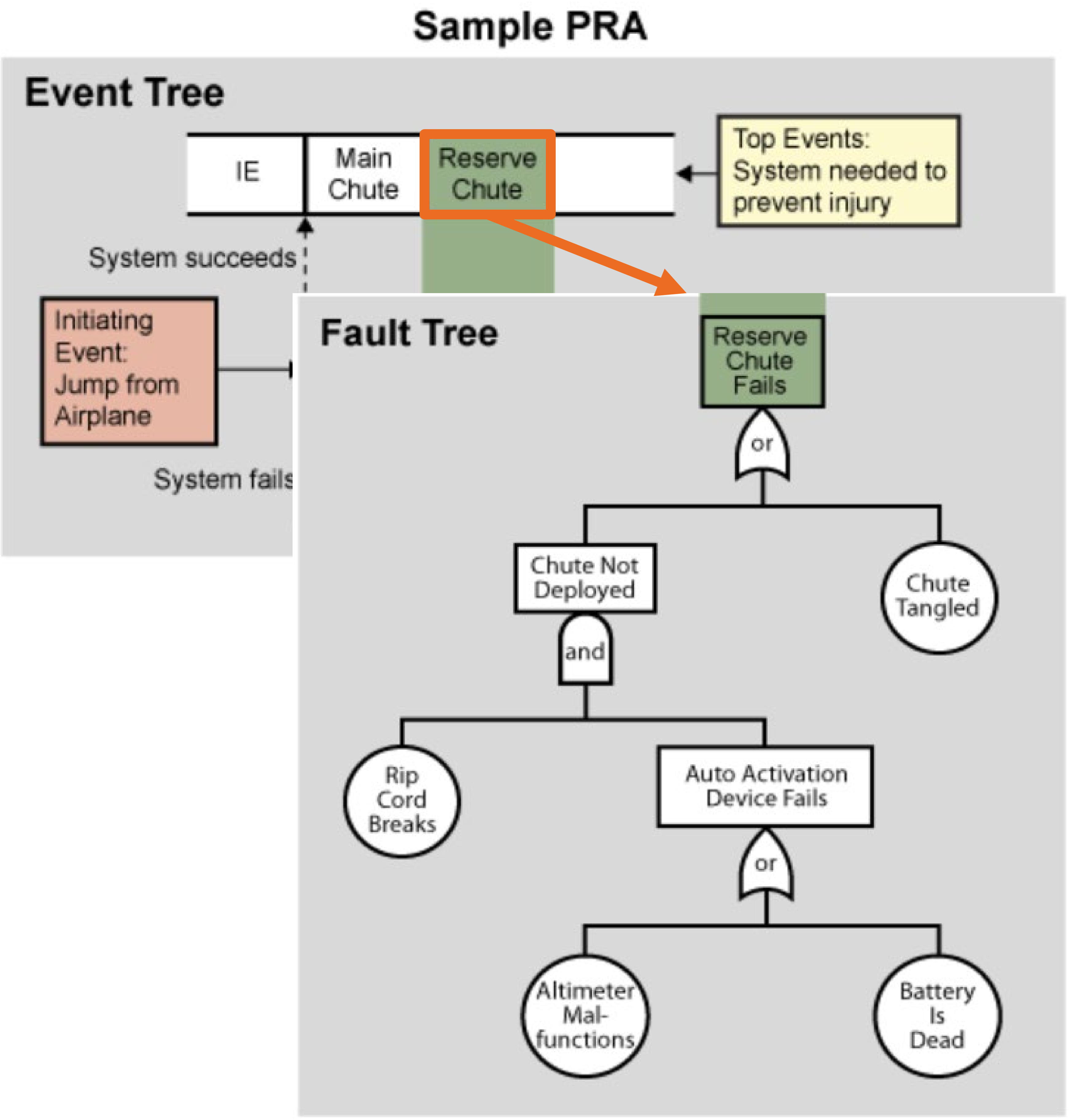


Probabilistic Safety Assessment – Accident Sequences

Risk = frequency of undesired event × severity of consequences

- ✓ **What** can go **wrong**?
- ✓ **How likely** it is?
- ✓ What might its **consequences** be?

- **Event Tree Analysis**
  Graphical mapping of potential **accident sequences** following an **initiating event**. It groups sequences by their consequence and calculates the **frequency** of each **consequence**.
- **Fault Tree Analysis**
  Top-down logical framework that uses **Boolean algebra** to identify the **root causes** of a **system failure** and **quantify** their **probability**.
- **Human Reliability Analysis**
  Collection of tools used to **predict human errors** and calculate **Human Error Probabilities**, to be then integrated into the Fault Trees.

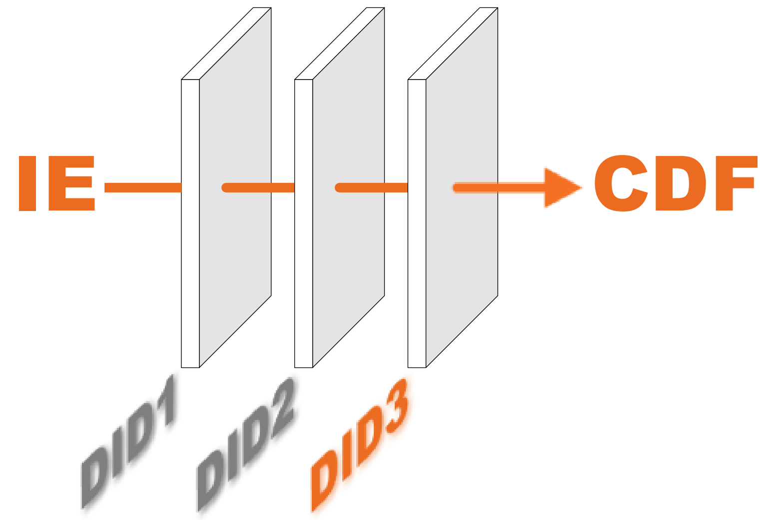


Probabilistic Safety Assessment – Defence-in-Depth

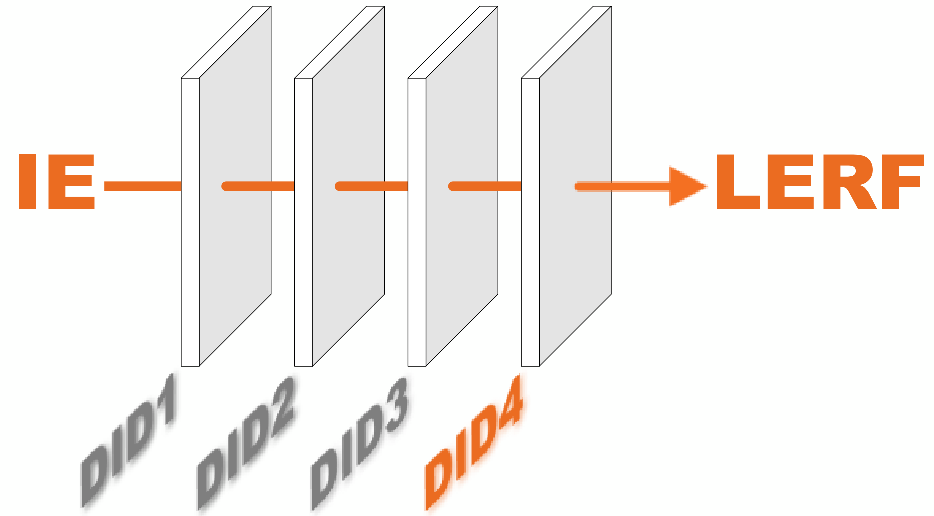
PSA Level	Risk metric
1	Core Damage Frequency
2	Large Early Release Frequency
3	Health Effects Frequency

DID Level	Objective
1	Prevention
2	Control
3	Protection
4	Mitigation of SA consequences
5	Off-site response

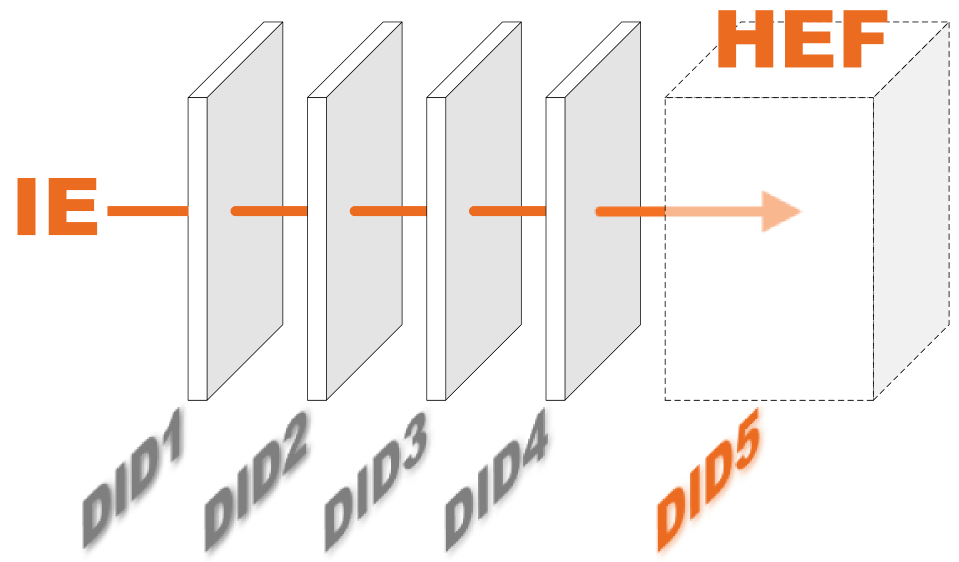
Level 1 PSA



Level 2 PSA



Level 3 PSA



Probabilistic Safety Assessment – Risk Metrics for an LFR

- **Severe accident**: *a situation* ***more severe*** *than a* ***design-basis event****, involving* ***significant cladding degradation*** *that, if* ***unmitigated****, could lead to radiological* ***releases*** *exceeding* ***off-site limits***
  - ➥ **Level 1 PSA risk metric = Cladding Damage Frequency**
  - ➥ Level 2 PSA risk metric = Large Early Release Frequency
  - ➥ Level 3 PSA risk metric = Dose Band Frequency

- **Accident sequence *S_i* ➔ probability** $P_{\mathrm{rupt},i}$ of a **fuel pin rupture** (function of transient):

$$P_{\mathrm{rupt},i} = F[T_i, \sigma_i, t_i, \varphi_i, \ldots]$$

- **Expected number** of **ruptured pins** in **sequence *S_i***:

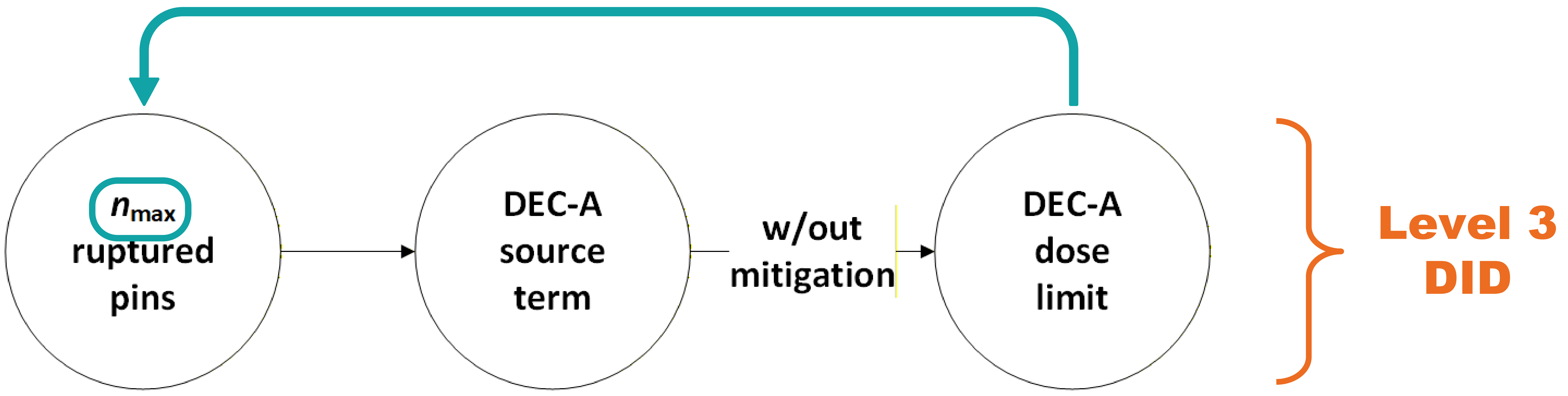
$$n_i \cong n_{\mathrm{tot}} \bar{P}_{\mathrm{rupt},i}$$

- In **DEC-A** scenarios, **dose limits** impose a **deterministic upper bound** on **acceptable number of fuel pin ruptures** ($n_{\max}$) **for a given accident** sequence (w/out reliance on DID 4)

$\Rightarrow$ **Quantitative definition of significant cladding degradation**: $n_i > n_{\max}$

PSA Level 1 Risk Metric – Cladding Damage

| Thermal-hydraulic calculations in support of the Severe Accident Analysis



PSA Level 1 Risk Metric – Cladding Damage

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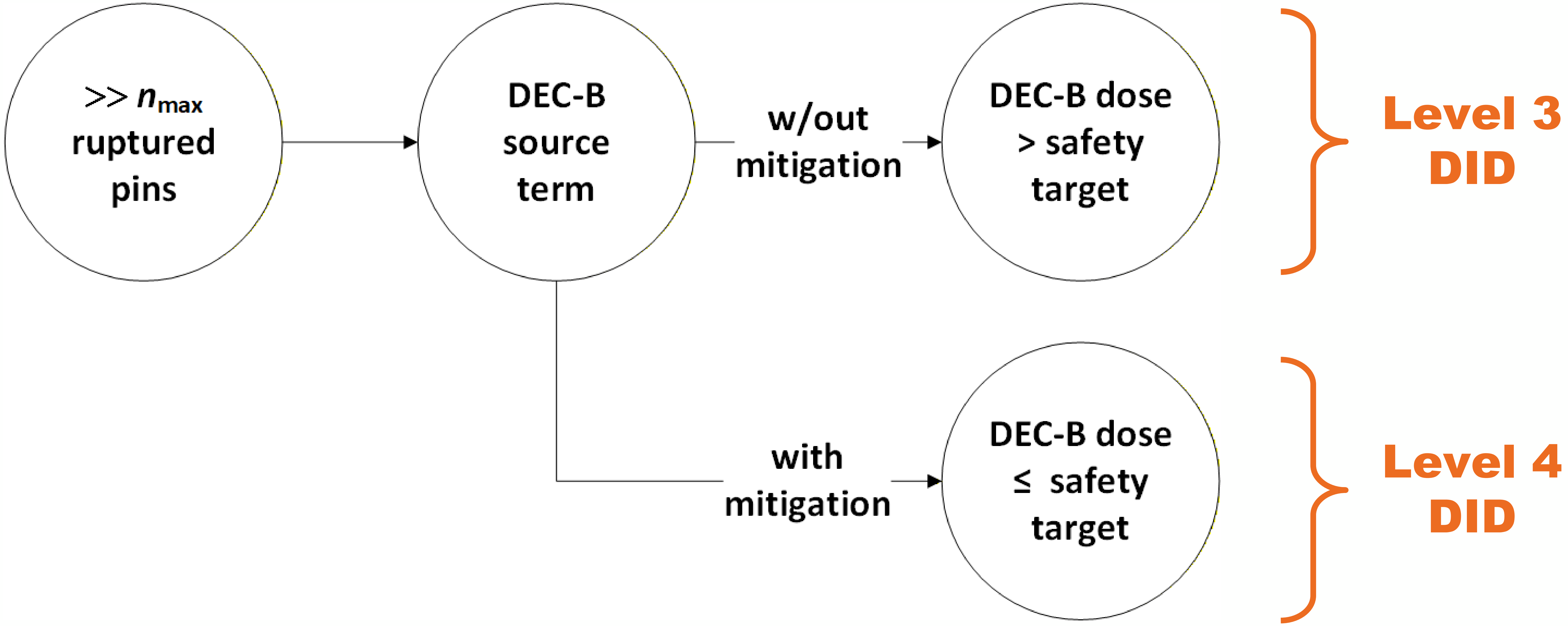
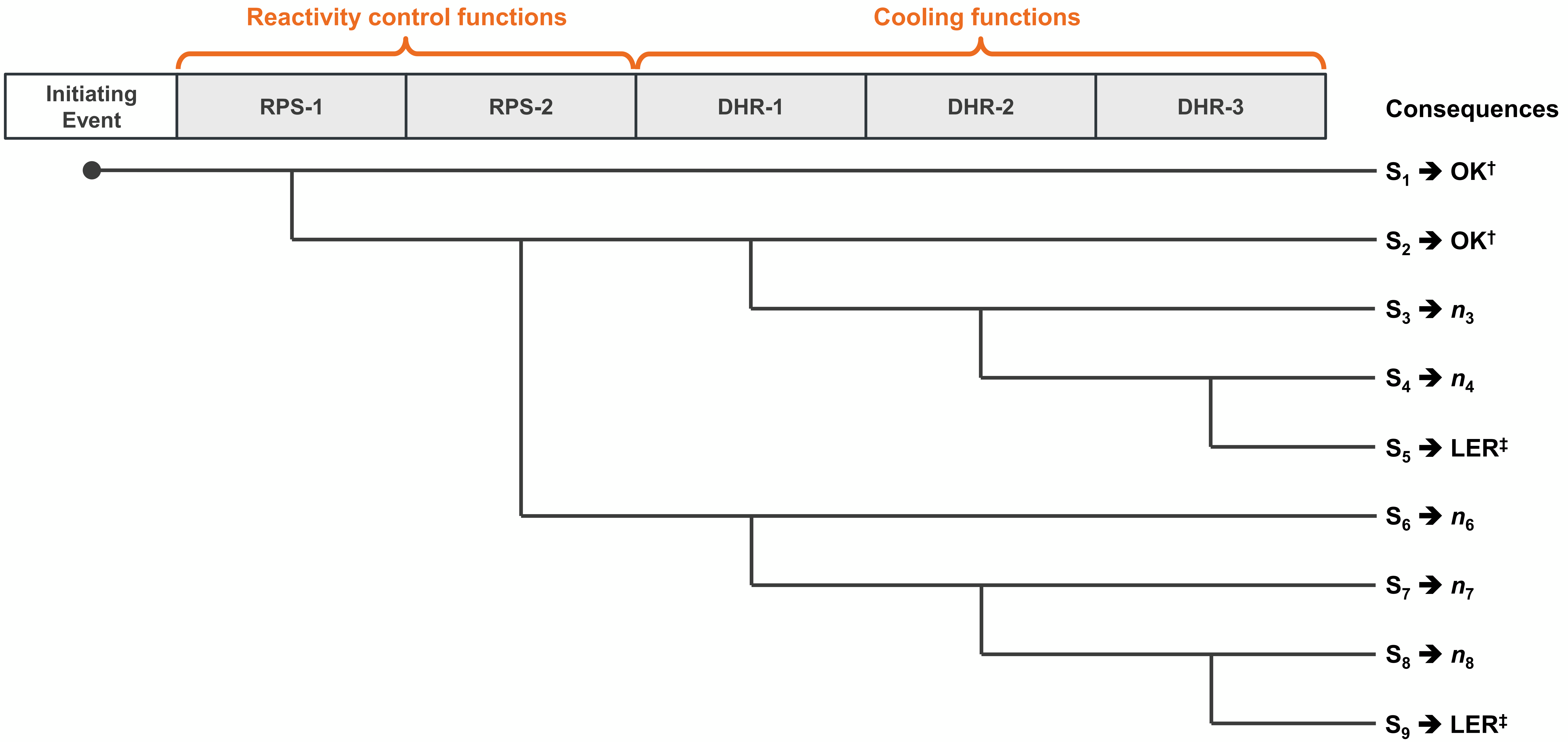


Level 1 PSA



(†) *OK (acceptable consequences) = fuel pin rupture probability* $\ll 1$

(‡) *LER = Large and Early Release*

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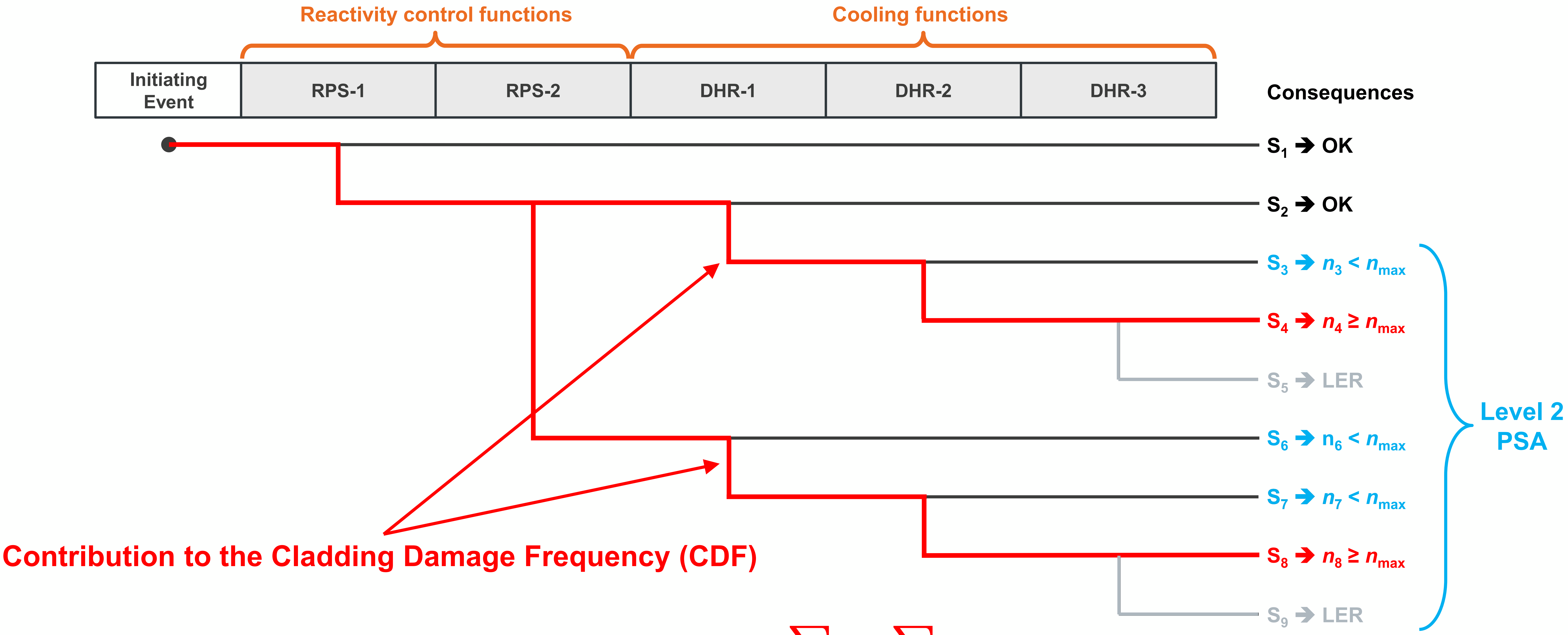
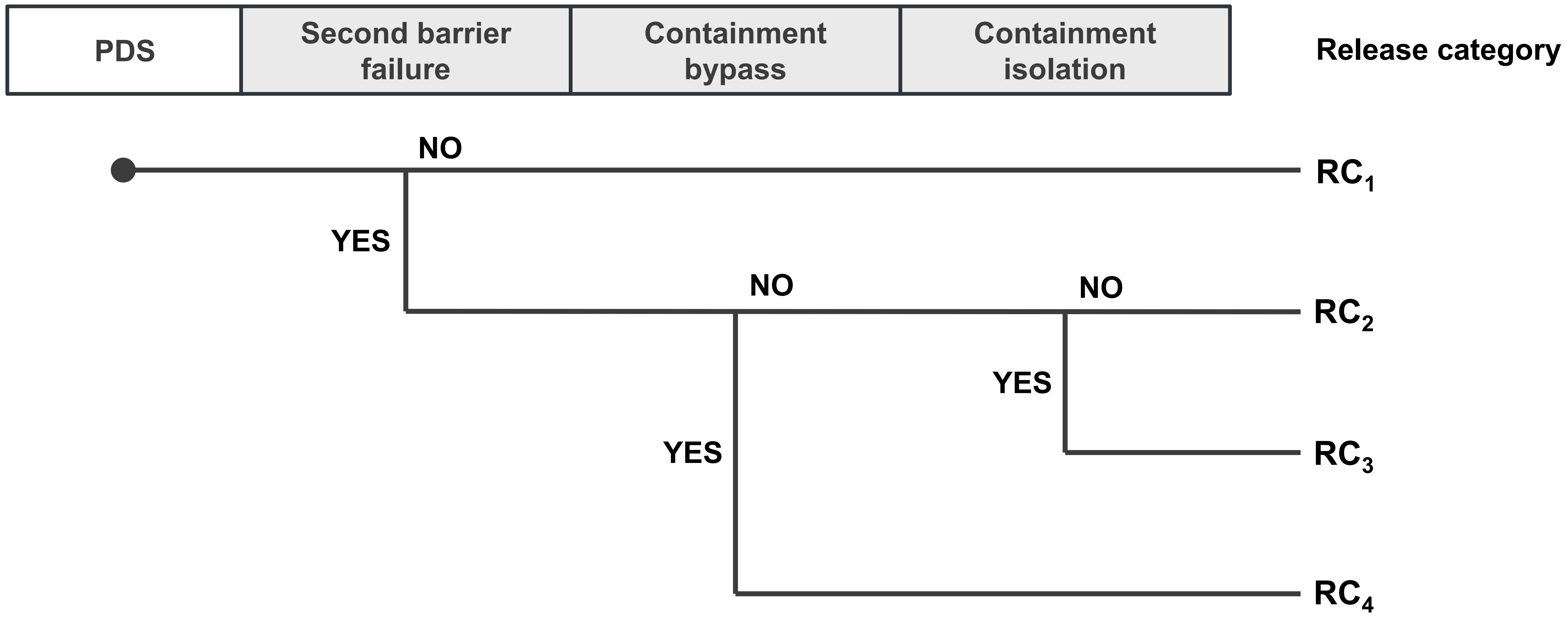
Level 1 PSA



$$\mathrm{CDF} = \sum_{\mathrm{IE}} \sum_{S_i \mid n_i \geq n_{max}} f_{S_i}$$

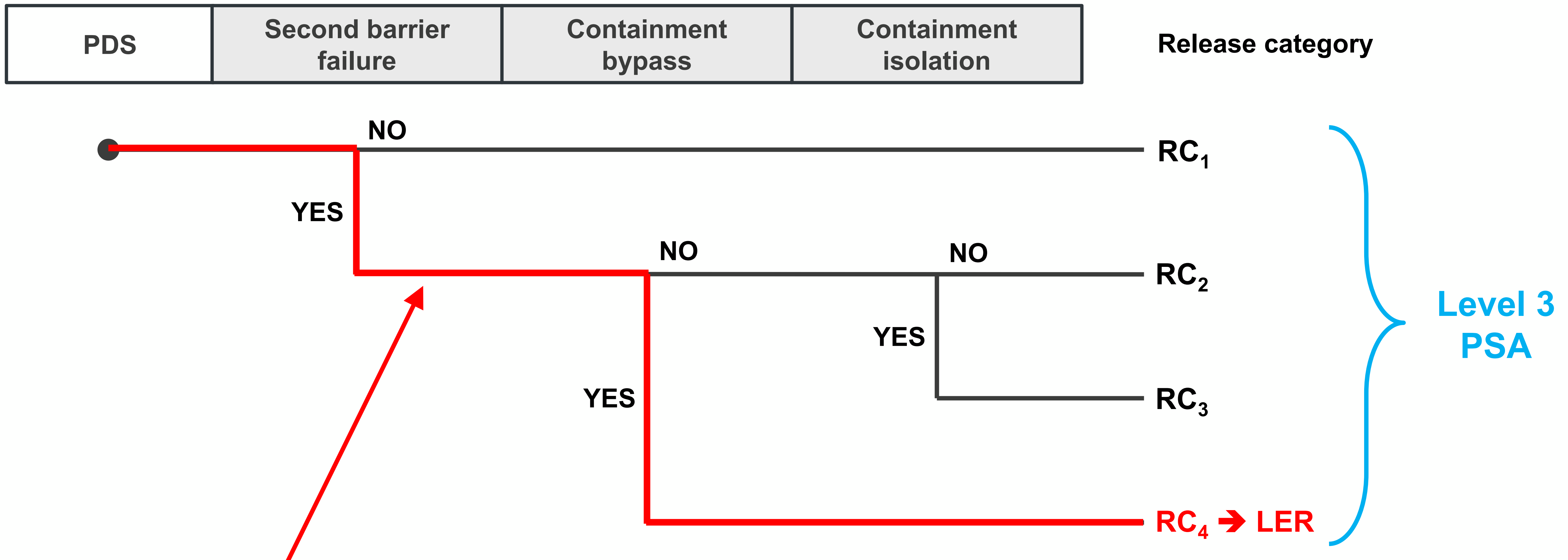
Level 2 PSA

| Plant Damage States (PDS)



Level 2 PSA

| Plant Damage States (PDS)



Contribution to the Large Early Release Frequency (LERF)

$$\mathrm{LERF} = \sum_{\mathrm{PDS}} \sum_{S_i \,|\, \mathrm{LER}} f_{S_i} + \sum_{\mathrm{IE}} \sum_{S_j \,|\, \mathrm{LER}} f_{S_j}$$

Thank you