Severe Accident Assessment of NRU Reactor

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Ninth International Probabilistic Safety Assessment and Management Conference

May 2008, Hong Kong, China

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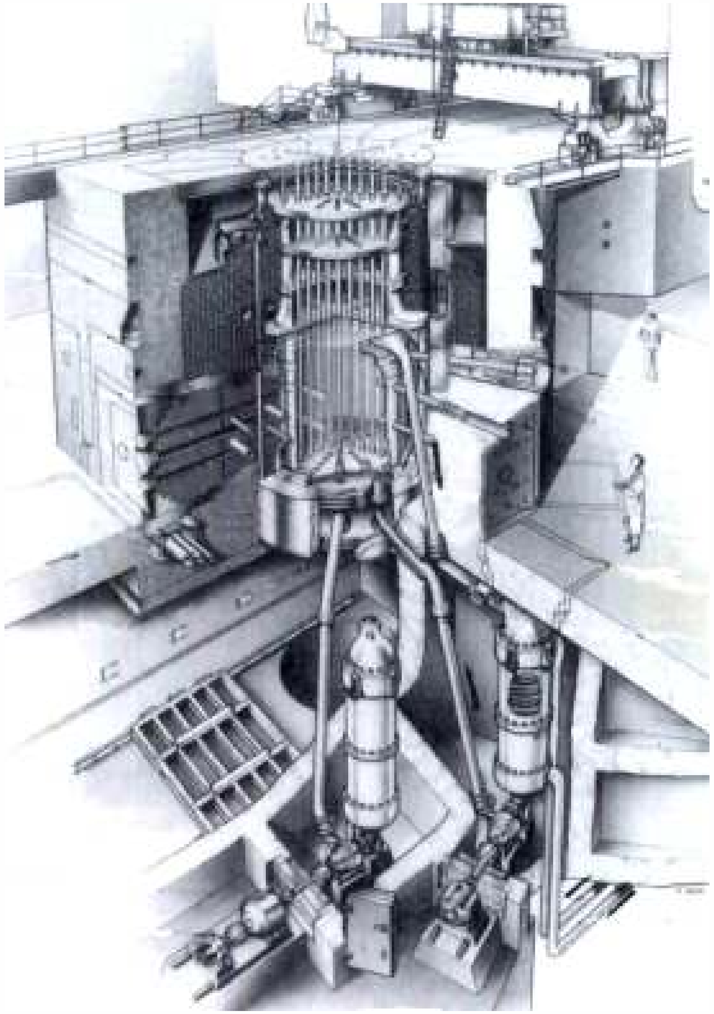
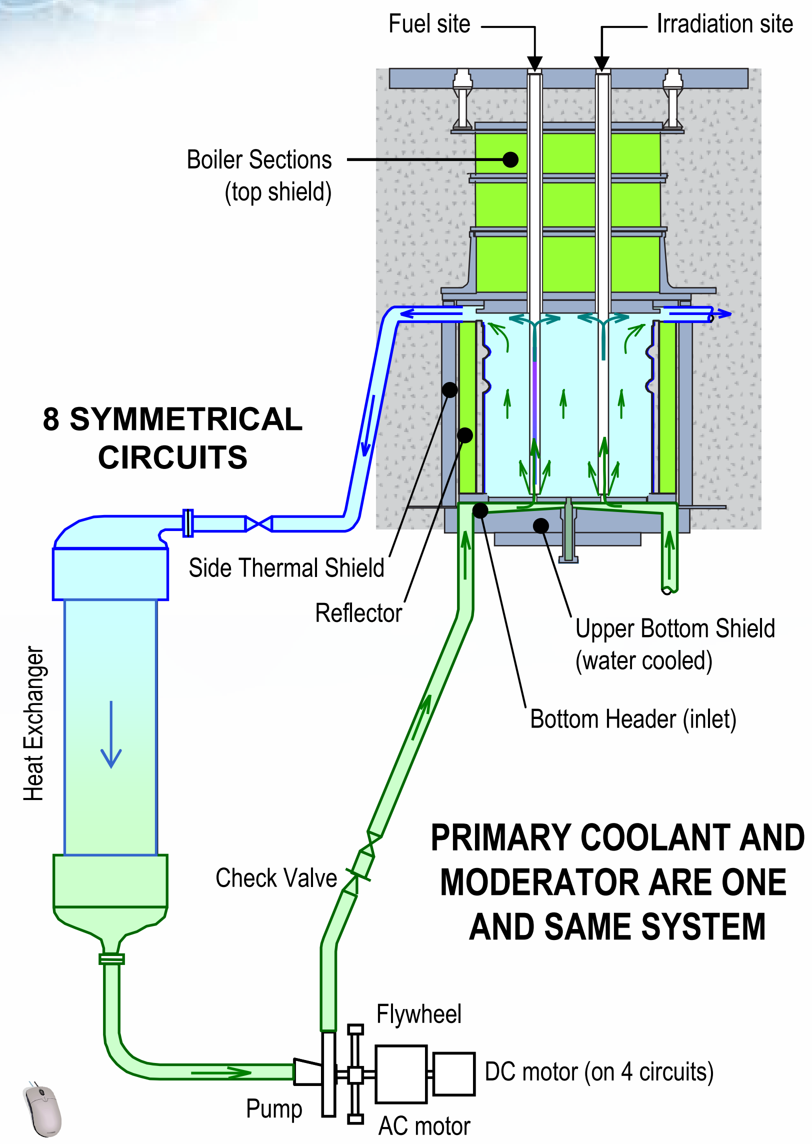
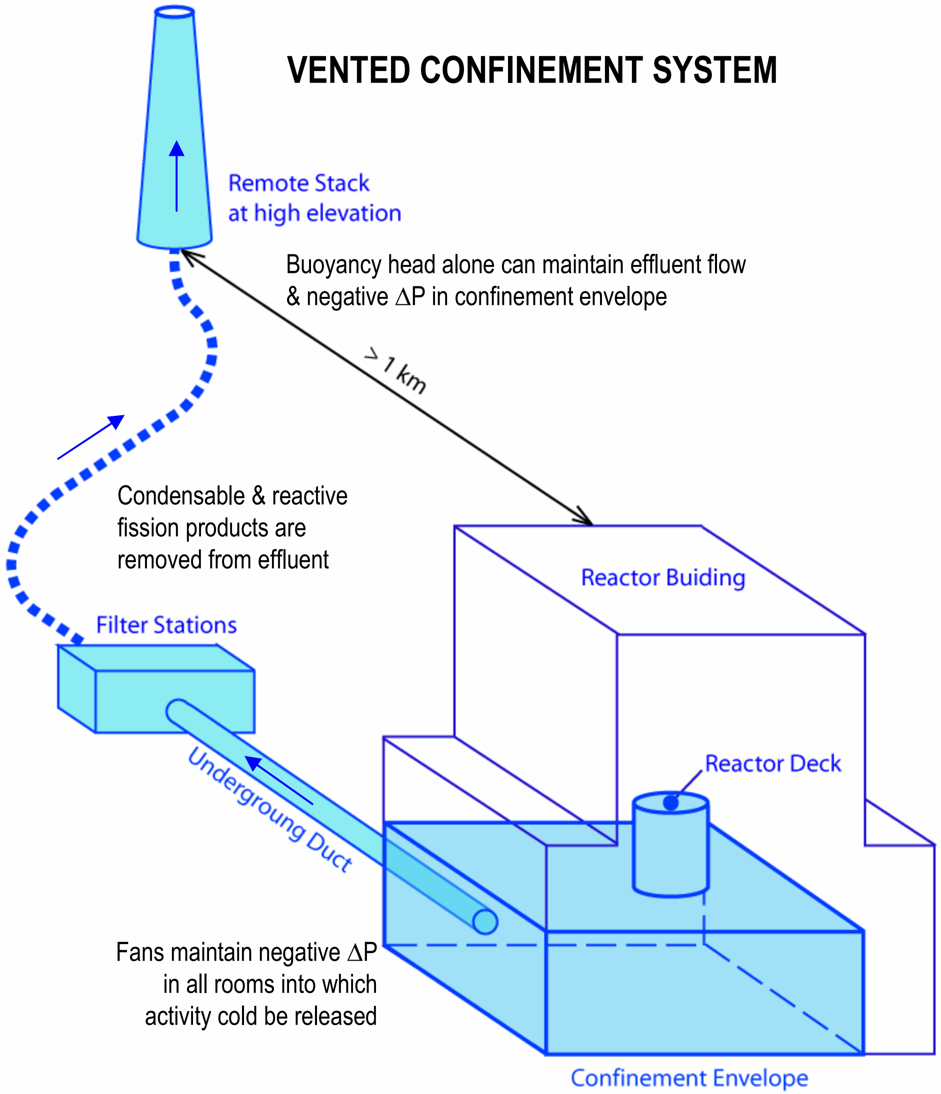
Introduction

- **National Research Universal (NRU) reactor**
  - heterogeneous reactor in operation since 1957
  - power up to 135 MW thermal
  - cooled and moderated by D_2O
  - 227 vertical lattice sites arranged in hexagonal array
    - control and fuel rods in ~ ½ sites
    - remaining sites for experiments and isotope irradiations
  - three reactor test sections supplied from two separate high-pressure, high-temperature loops
  - two independent trip systems that activate shutdown system



Introduction (continued)





Introduction (continued)

- **Severe Accident Assessment (SAA) performed to extend internal event analyses of upgraded NRU facility into severe accident realm**
- **Regulatory bodies or IAEA do not define risk goals for non-power reactors**

Acceptance criteria defined for and applied in NRU SAA

Criterion	Acronym	Value (a^{-1})	Explanatory Notes
Severe Core Damage Frequency	**SCDF**	**$1 \cdot 10^{-4}$**	Risk surrogate. Consensus value for existing older nuclear power plants, used in past CANDU PSAs.
Large Release Frequency	**LRF**	**$1 \cdot 10^{-6}$**	Risk surrogate. Large release is 10^{15} Bq of Cs-137. Proposed for new nuclear power plants in Canada (subsequently reduced to 10^{14} Bq).
Individual Early Fatality Risk	**IEFR**	**$4 \cdot 10^{-6}$**	Value is 1% of the risk of prompt death from all other accidents in Canada; similar to value used in past CANDU PSAs.
Individual Delayed Fatality Risk	**IDFR**	**$4 \cdot 10^{-5}$**	Value is 1% of the risk of delayed fatalities from all types of cancer in Canada; similar to value used in past CANDU PSAs.
Large Release Risk	**LRR**	**$7 \cdot 10^{-4}$**	Addresses the same issue as the surrogate LRF (risk that large numbers of nearby residents would have to be relocated).

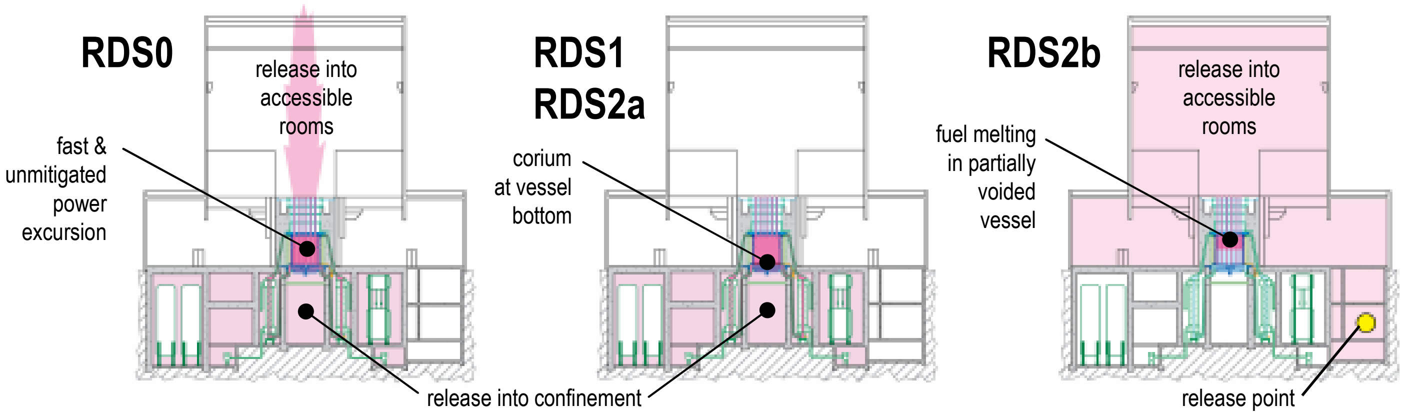
Level 1 PSA Insights

- **Reactor Damage States (RDS) are derived from Level 1 event tree and fault tree analyses using feedbacks from Level 2 deterministic analyses**

fuel release ≤ 1%

<table>
<tr><th>State</th><th colspan="3">Description</th><th>Frequency Magnitude</th></tr>
<tr><td>SR</td><td colspan="3">Stable fuel state – limited cladding damage.</td><td>E-02/year</td></tr>
<tr><td>RDS4</td><td colspan="3">Local fuel damage due to a power cooling mismatch in an experimental loop</td><td>E-03/year</td></tr>
<tr><td>RDS3</td><td colspan="3">Local fuel damage due to a power cooling mismatch in a single flow tube</td><td>E-03/year</td></tr>
<tr><td rowspan="2">RDS2</td><td rowspan="2">Widespread fuel damage due to a power cooling mismatch at decay power</td><td>RDS2A</td><td>without confinement bypass</td><td>E-05/year</td></tr>
<tr><td>RDS2B</td><td>with confinement bypass</td><td>E-05/year</td></tr>
<tr><td>RDS1</td><td colspan="3">Widespread fuel damage due to a power cooling mismatch at high power</td><td>E-07/year</td></tr>
<tr><td>RDS0</td><td colspan="3">Widespread fuel damage due to a failure to shut down after a power excursion</td><td>E-08/year</td></tr>
</table>

SCDF criterion is met



Level 2 PSA Insights

NRU reactor employs low enriched uranium (LEU) fuel cycle based on U_3Si dispersed in Al

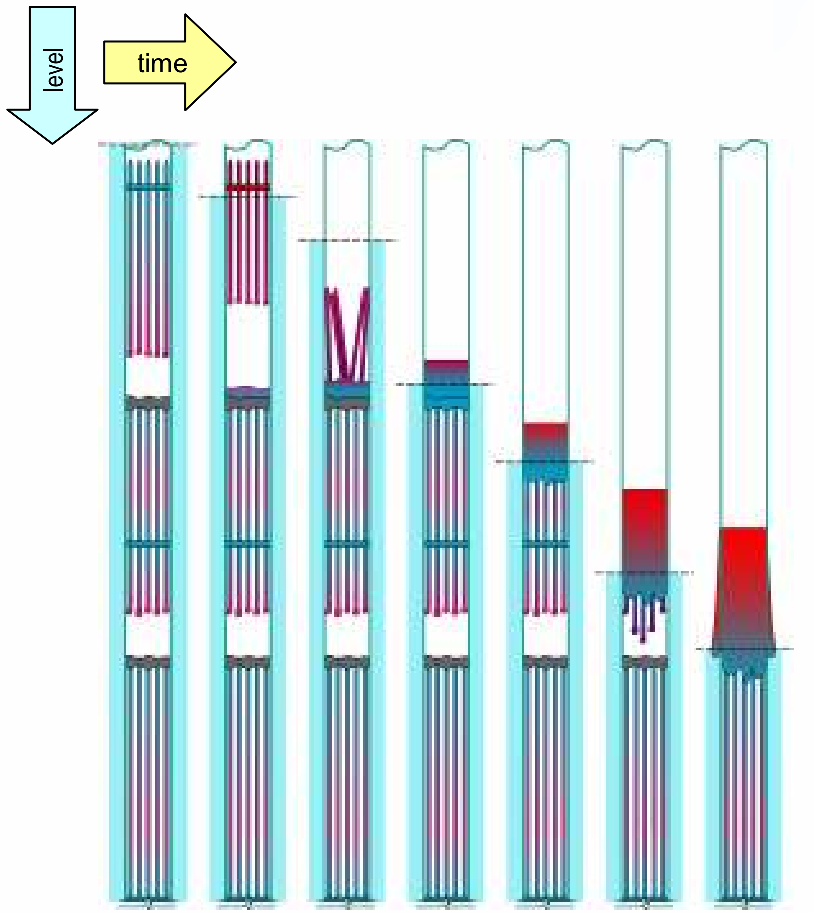
- Aluminium fuels liquefy at low temperatures (~ 660°C)
- U_3Si exothermically reacts with aluminium at ≥ 500°C
  - reaction heat is small relative to decay heat → small shift in timing of events
- Molten fuel releases only few fission products (noble gases, iodine and cesium) unless significantly superheated
- Interactions of molten fuel with steam or water do not produce appreciable hydrogen and do not readily trigger steam explosions unless melt is superheated to ≥ 1000°C
  - solid silicide and aluminide particles suspended in liquid aluminium tend to inhibit fragmentation of melt that is necessary for steam explosion

Level 2 PSA Insights (continued)

Deterministic Assessments

Liquefied fuel is mobile

- **Melt flows (relocates) into contact with various passive heat sinks and solidifies**
  - conduction heat transfer is important
- **Melt superheating during relocation is only possible in fast and large power excursions**
  - catastrophic steam explosions are not plausible during core meltdown at steady or decay power levels

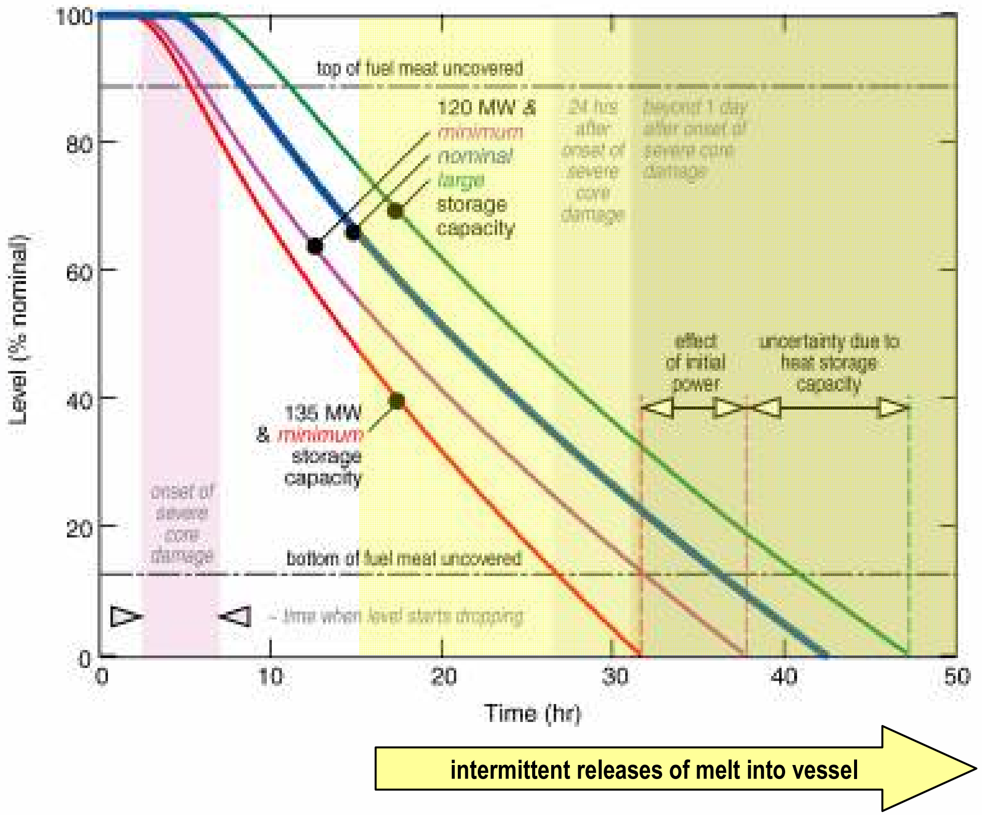


Example of relocation at decay power slowly voiding vessel

Level 2 PSA Insights (continued)

Deterministic Assessments

Key passive heat sink in intact reactor vessel is residual water inventory



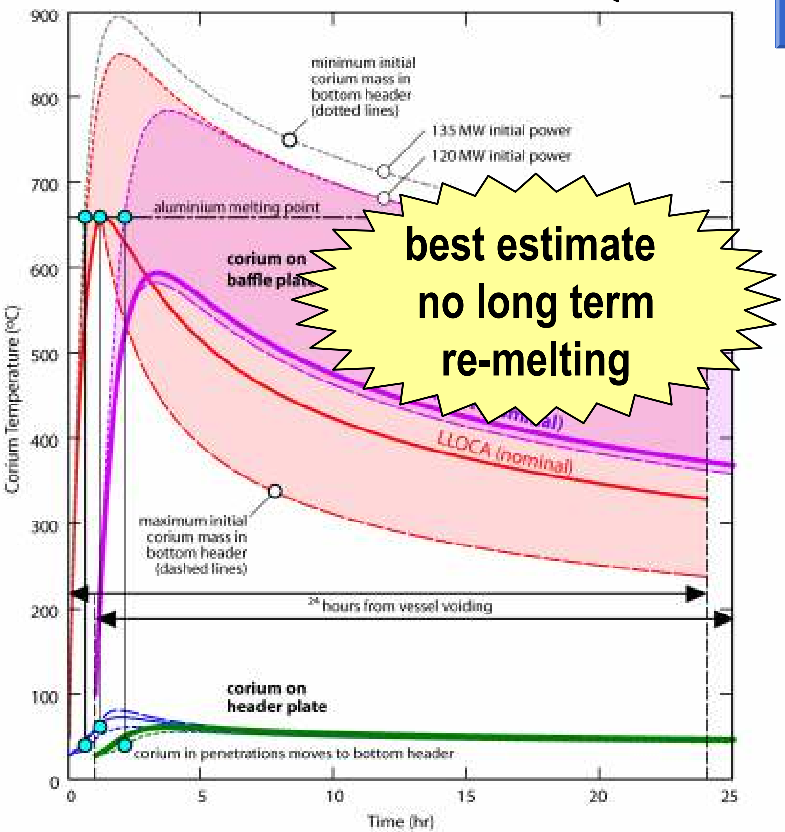
- Available for > 1 day after onset of Severe Core Damage (SCD)
- Residual water does not prevent fuel melting and relocation within vessel but precludes debris from re-heating and re-melting on vessel bottom

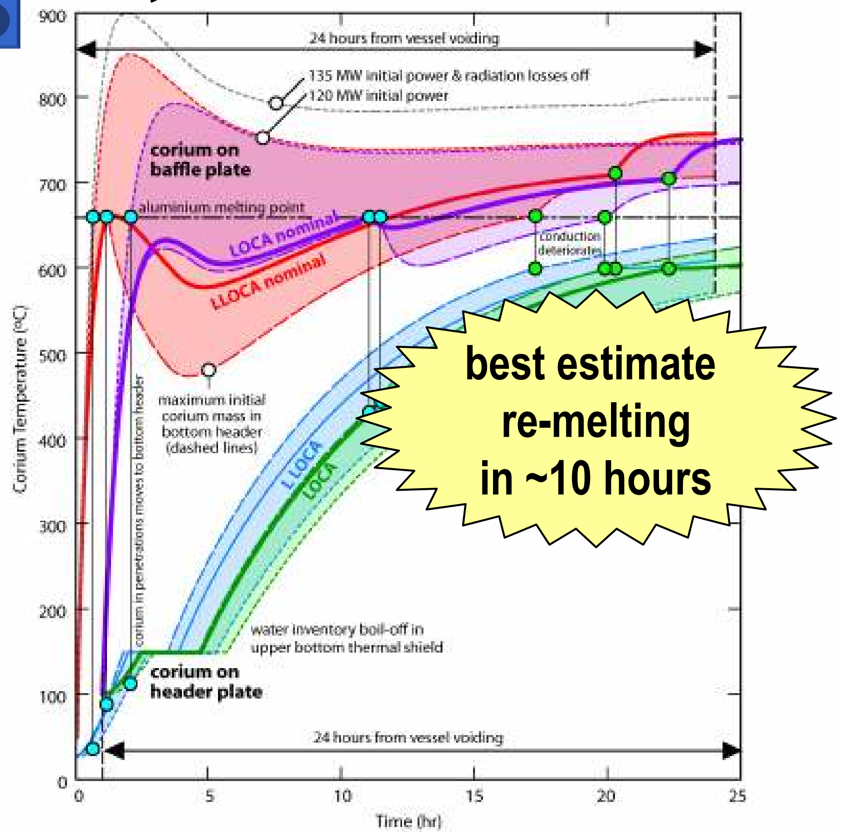
Level 2 PSA Insights (continued)

Deterministic Assessments

Active or passive heat sinks come into play in voided reactor vessel

available ⟵ active Thermal Shield Cooling ⟶ **unavailable**





Level 2 PSA Insights (continued)

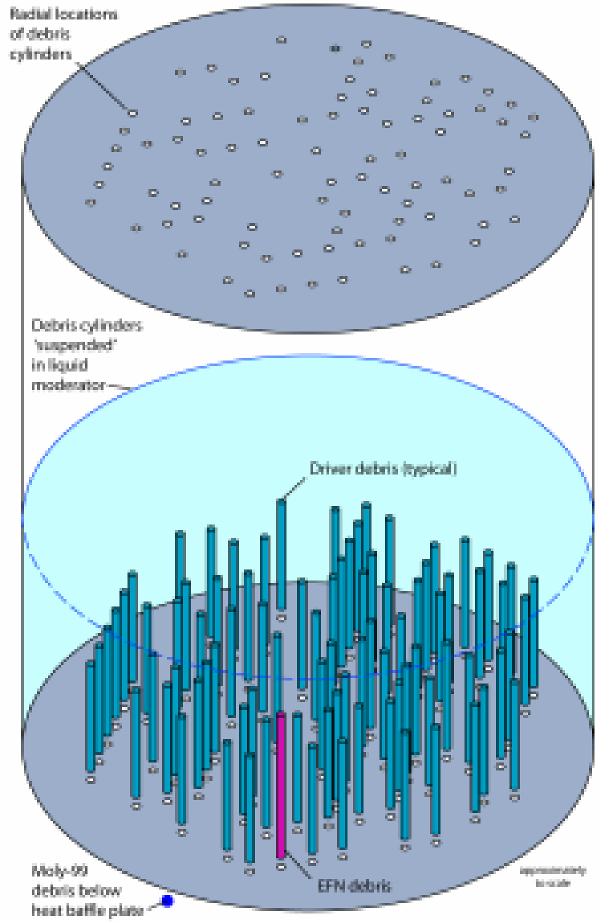
Deterministic Assessments

Reconfigured Reactor Core

- Broad range of debris/moderator geometries = potential for re-criticality
- All conceivable configurations evaluated by supplementary analysis

Configuration most prone to re-criticality in NRU is array of debris cylinders submerged in D_2O

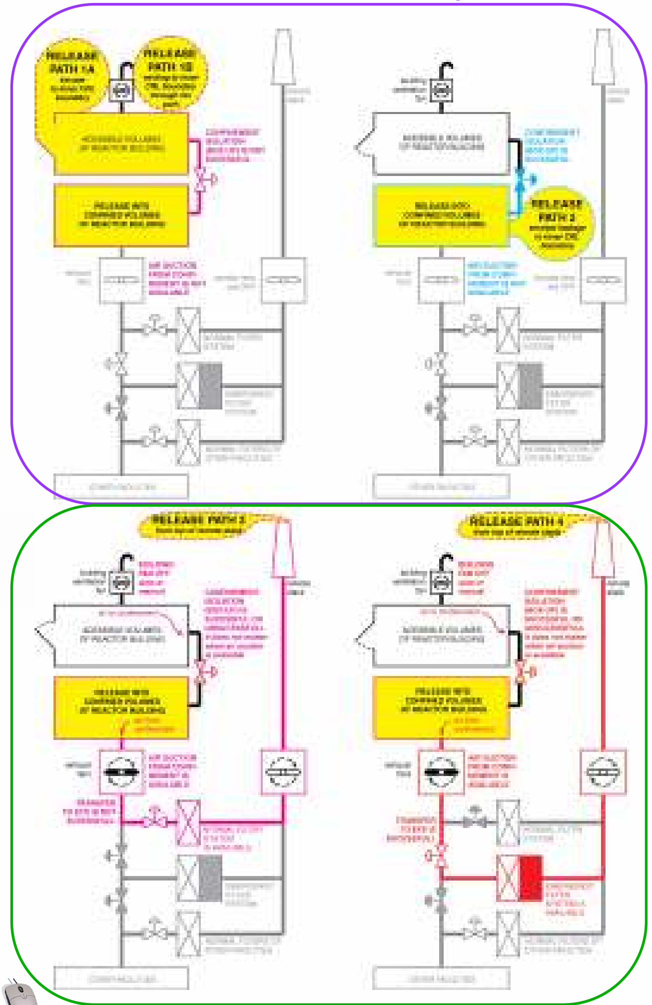
Subcritical even with neutron absorbers (control rods) assumed not to be present in reconfigured core



Level 2 PSA Insights (continued)

Deterministic Assessments

no retention in transport
credited for confinement bypass



retention during transport evaluated

Release of Fission Products

- Volatile fission products are released from molten core continuously
- Gases and vapours remain largely confined within breached reactor vessel until accident termination
  - Operator actions to quench debris purge volatile fission products from vessel
- All potential release routes from reactor vessel into environment are identified and categorized

Level 2 PSA Insights (continued)

Probabilistic Assessments

- Destructive severe accident phenomena are plausible in SCD accidents due to fast and large power excursions
  - steam explosion and hydrogen deflagration
  - power excursions have very low frequency of occurrence in NRU
- Modest water coolant interactions capable of reactor vessel damage are possible in other SCD sequences but are not imminent
  - pressure surges in vessel → consequential draining of vessel
- There are no other challenges to integrity of reactor vessel or confinement for at least one day after SCD onset

NRU Accident Progression Event Trees (APETs) are relatively simple. Branches deal mainly with unavailability of confinement subsystems.

Level 2 PSA Insights (continued)

Release Source Terms (RSTs)

Magnitude & Composition (% core inventory)				**Duration**	**Location**	**Frequency Magnitude**
Noble Gases	**Cesium & Iodine**	**Tellurium**	**Other**	(hr)		(per year)
100	100	100	100	1	building	E-08
100	100	10	0	1	building	E-07
100	10	1	0	24	building	E-08
100	10	1	0	24	stack	E-08
100	1	0.1	0	24	building	E-05
100	1	0.1	0	24	stack	E-05

- **RSTs are stylized categories**
  - **pre-defined on basis of fission product volatility before deterministic transport assessments were performed**
  - **transport analyses show that, for the last four RST categories in above table, fractional release of cesium is exaggerated by ~ factor of ten**

Level 3 PSA Insights

- **Individual Early Fatality Risk is of the order of 10^{-8} per year**
- **Individual Delayed Fatality Risk is of the order of 10^{-7} per year**
  - **With these values, Large Release Risk need not be enumerated → LRR criterion is met when IDFR criterion is met**

~ 2 orders of magnitude margin to acceptance criteria

Conclusions

- SAA provides information on severe accidents
  - previously missing in NRU safety assessments
- Acceptance criteria met with large margin to spare
  - Uncertainties are being addressed by supplementary assessments and experiments
  - It is evident that operation of upgraded NRU reactor does not present public with any additional risks that are significant in comparison with risks to which public is normally exposed
- Insights applied in Severe Accident Management, emergency planning and risk-informed decision-making

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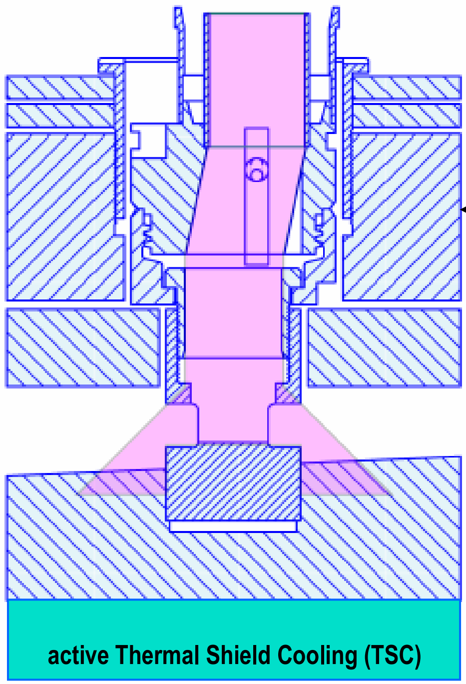
Level 2 PSA Insights (continued)

Deterministic Assessments

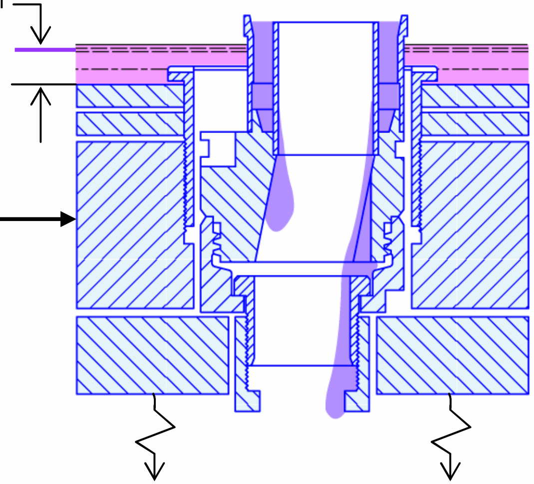
Active or passive heat sinks in voided reactor vessel

Relocation into bottom header



passive heat storage in massive stainless steel structures at vessel bottom

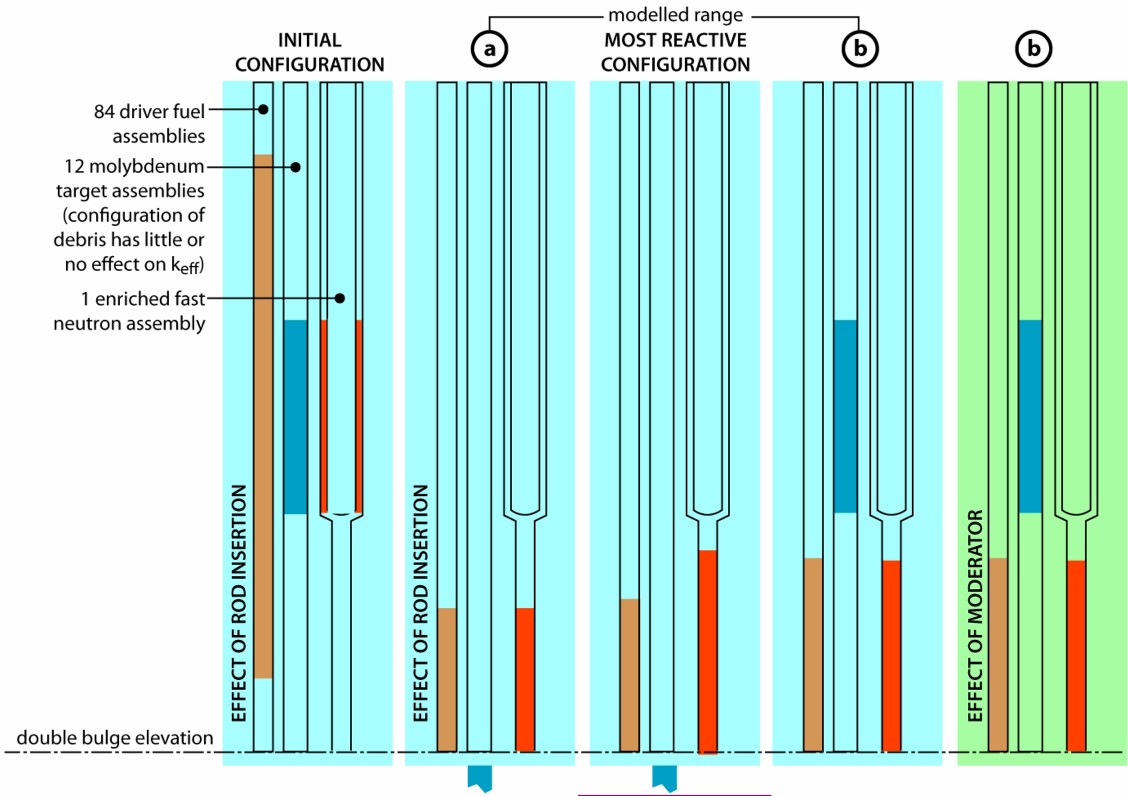
Relocation onto heat baffle plate



Approximately to scale

Level 2 PSA Insights (continued)

Deterministic Assessments



Moderator	D_2O	D_2O	D_2O	D_2O	H_2O
Control rods up	$k_{eff} = 1.00$	$k_{eff} = 0.96$	N/A	N/A	N/A
Control rods down	$k_{eff} = 0.84$	$k_{eff} = 0.91$	**$k_{eff} = 0.92$**	$k_{eff} = 0.93$	$k_{eff} = 0.31$

Debris re-criticality is very low probability phenomenon in NRU reactor