Severe Accident Assessment of NRU Reactor

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Ninth International Probabilistic Safety Assessment and Management Conference
May 2008, Hong Kong, China
Introduction

• National Research Universal (NRU) reactor
  – heterogeneous reactor in operation since 1957
  – power up to 135 MW thermal
  – cooled and moderated by D$_2$O
  – 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)

**VENTED CONFINEMENT SYSTEM**

- Remote Stack at high elevation
- Buoyancy head alone can maintain effluent flow & negative $\Delta P$ in confinement envelope
- Condensable & reactive fission products are removed from effluent
- Fans maintain negative $\Delta P$ in all rooms into which activity could be released

**PRIMARY COOLANT AND MODERATOR ARE ONE AND SAME SYSTEM**

- 8 SYMMETRICAL CIRCUITS
- Heat Exchanger
- Check Valve
- Flywheel
- AC motor
- Pump
- DC motor (on 4 circuits)
- Boiler Sections (top shield)
- Irradiation site
- Fuel site
- Side Thermal Shield
- Reflector
- Upper Bottom Shield (water cooled)
- Bottom Header (inlet)
- Bottom Header (inlet)
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

<table>
<thead>
<tr>
<th>Criterion</th>
<th>Acronym</th>
<th>Value ($a^{-1}$)</th>
<th>Explanatory Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>Severe Core Damage Frequency</td>
<td>SCDF</td>
<td>$1 \times 10^{-4}$</td>
<td>Risk surrogate. Consensus value for existing older nuclear power plants, used in past CANDU PSAs.</td>
</tr>
<tr>
<td>Large Release Frequency</td>
<td>LRF</td>
<td>$1 \times 10^{-6}$</td>
<td>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).</td>
</tr>
<tr>
<td>Individual Early Fatality Risk</td>
<td>IEFR</td>
<td>$4 \times 10^{-6}$</td>
<td>Value is 1% of the risk of prompt death from all other accidents in Canada; similar to value used in past CANDU PSAs.</td>
</tr>
<tr>
<td>Individual Delayed Fatality Risk</td>
<td>IDFR</td>
<td>$4 \times 10^{-5}$</td>
<td>Value is 1% of the risk of delayed fatalities from all types of cancer in Canada; similar to value used in past CANDU PSAs.</td>
</tr>
<tr>
<td>Large Release Risk</td>
<td>LRR</td>
<td>$7 \times 10^{-4}$</td>
<td>Addresses the same issue as the surrogate LRF (risk that large numbers of nearby residents would have to be relocated).</td>
</tr>
</tbody>
</table>
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

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

- SCDF criterion is met

![Diagrams showing fuel release scenarios](image)
NRU reactor employs low enriched uranium (LEU) fuel cycle based on U₃Si dispersed in Al

- Aluminium fuels liquefy at low temperatures (~ 660°C)
- U₃Si 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
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
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

Diagram: Intermittent releases of melt into vessel

Legend:
- Top of fuel meat uncovered
- 120 MW & minimum storage capacity
- 135 MW & minimum storage capacity
- Onset of severe core damage
- Bottom of fuel meat uncovered
- Time when level starts dropping
- Effect of initial power
- Uncertainty due to heat storage capacity
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

best estimate  no long term re-melting

best estimate  re-melting in ~10 hours
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$_2$O

Subcritical even with neutron absorbers (control rods) assumed not to be present in reconfigured core
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 $\rightarrow$ 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)
Probabilistic Assessments

Release Source Terms (RSTs)

<table>
<thead>
<tr>
<th>Magnitude &amp; Composition (% core inventory)</th>
<th>Duration (hr)</th>
<th>Location</th>
<th>Frequency Magnitude (per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Noble Gases</td>
<td>Cesium &amp; Iodine</td>
<td>Tellurium</td>
<td>Other</td>
</tr>
<tr>
<td>100</td>
<td>100</td>
<td>100</td>
<td>100</td>
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<tr>
<td>100</td>
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</table>

- 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
Active or passive heat sinks in voided reactor vessel

- Relocation into bottom header
- Relocation onto heat baffle plate

- Passive heat storage in massive stainless steel structures at vessel bottom
- Limited number of ‘conduction bridges’

- Nominal corium thickness ~ 2 cm

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