Source Terms Issues and Implications on the Nuclear Reactor Safety

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Technical Meeting on Source Term Evaluation for Severe Accidents, Vienna International Centre, Austria, from 21 to 23 October 2013.
Contents

I. Severe Accident Acceptance Criteria
II. Source Term and Mitigation Measures
III. Balanced Prevention and Mitigation
I. Severe Accident Acceptance Criteria

- Countries have different forms of ‘acceptance criteria’
- Two major approaches:
  - Safety goals:
    USA, Japan, Korea: Qualitative Health Objective
  - Prescribed limits:
    Finland, Sweden: 0.1% of the core inventory of the caesium isotopes 134 and 137
    France, Germany: Major challenges to FP boundaries should be designed out (‘practically eliminated’)

- Criteria can be different for existing and new reactors
- Most criteria address health effects; few address property damage, land or sea contamination. However, it has to be changed to consider Post-Fukushima Socio-economic Impact
Highly Populated NPPs in East Asia

Kori site:
3 M people within 30 km

On average 1 out of 10 people in Korea live within 30 km
Nuclear Power Plants in Korea

- 23 operable nuclear power reactors
  - 19 PWRs and 4 PHWR-CANDUs
  - Installed capacity of 20.7 Gwe
    - 30~35% share of electricity supply
- 5 PWR units under construction
  - 1 OPR1000’s and 4 APR1400’s
- The most economical source of electricity
- Highly populated city near a NPP Site:
  - 3 M people within 30 km
- New energy policy proposed 2013. 10
  Maintain ~ 20% of Installed Capacity until 2035, Electricity bill 13-21% increase/year
II. Source Term and Mitigation Measures

Mitigation of Severe Accidents with a goal of practically eliminating significant radiological releases to the public and environment
II. Source Term and Mitigation Measures

- **Reactor Vessel Integrity** – Release to the containment, Release to the environment during the recovery
- **Containment Integrity** – Release to the environment
- **Bypass Scenarios** – Direct release path
- **Mitigation of Fission Product Release**
(1) RV – Release to the Containment

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No Closed Loop Cooling – Release to the Sea

Water Inventory Control

Situation of unit 2

Central Radioactive Waste Processing Building
30,000 m³

Discharge into sea
9,000 m³

Transportation from Apr. 19

S/P Surge Tank

Trench Water (Mar. 30)
I-131: 6.9 x 10⁶ Bq/cm³
Cs-137: 2.0 x 10⁶ Bq/cm³

T/B Basement Water (Mar. 27)
Dose Rate: > 1000 mSv/h
I-131: 1.3 x 10⁷ Bq/cm³
Cs-137: 3.0 x 10⁷ Bq/cm³
25,000 m³

Shaft
Depressurization and Alternate Water Injection

Dedicated rapid depressurization valves or existing PORVs can be used.
- Power should be timely secured under SBO. It will take time to open them and secure powers.
- The valves should be qualified under high temperature steam flow. Since valves were qualified under DBA condition, there is a chance of abnormal function.

Taken from MELCOR Best Practices - An Accident Sequence Walkthrough (L. Humphries et. al.)
### Recent analysis result of Fukushima Unit 1

<table>
<thead>
<tr>
<th>March</th>
<th>Marc</th>
</tr>
</thead>
<tbody>
<tr>
<td>14</td>
<td>16</td>
</tr>
<tr>
<td>18</td>
<td>20</td>
</tr>
<tr>
<td>22</td>
<td>24</td>
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<td>02</td>
<td>04</td>
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<td>06</td>
<td>08</td>
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<td>10</td>
<td>12</td>
</tr>
<tr>
<td>14</td>
<td>16</td>
</tr>
<tr>
<td>18</td>
<td></td>
</tr>
</tbody>
</table>

- **March 11**: CV Vent
- **March 12**: Water injection

**Earthquake** (14:46)

**Tsunami SBO** (15:42)

**Water injection** (19:04)

**H₂ Explosion** (15:36)

**RPV failure** (5:46)

**Melt start** (18:46)

**Melt start** (about 18:00)

**CV Vent** (14:30)

**Analysis by JNES at June**

**Analysis by IAE at May**

**Delay of AM about 20 hours**

Taken from, Naitoh, Prepared for KNS Fall Conference, October, 2011
Analysis for APR1400 Severe Accident Progression using MELCOR 1.8.6. SBLOCA initiated and SBO initiated severe accidents are analyzed. Sensitivity studies on the operator recovery actions are performed.

<table>
<thead>
<tr>
<th>Events</th>
<th>SBLOCA w/o SIP</th>
<th>SBLOCA + 1 SIP (4 hrs)</th>
<th>SBLOCA + 1 SIP (4.5 hrs)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Time (hr)</td>
<td>Time (hr)</td>
<td>Time (hr)</td>
<td></td>
</tr>
<tr>
<td>SBO, RX trip, MFW trip</td>
<td>0</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Core uncovery</td>
<td>0.10</td>
<td>0.10</td>
<td>0.10</td>
</tr>
<tr>
<td>Coe Exit Temp at 650 °C</td>
<td>0.17</td>
<td>0.17</td>
<td>0.17</td>
</tr>
<tr>
<td>SIP actuation</td>
<td>0.53</td>
<td>0.53</td>
<td>0.53</td>
</tr>
<tr>
<td>Actuation of SITs</td>
<td>6.47</td>
<td>4.00</td>
<td>4.50</td>
</tr>
<tr>
<td>Molten fuel relocation</td>
<td>6.94</td>
<td>8.39</td>
<td>8.31</td>
</tr>
<tr>
<td>Reactor vessel failure</td>
<td>7.59</td>
<td>N/A</td>
<td>8.87</td>
</tr>
</tbody>
</table>
In case of Station Black Out, sensitivity studies on the operator recovery action of depressurization and alternate injection were performed. Operator has less than 30 minutes after an indication of core exit temperature to prevent reactor pressure vessel. Sensitive to the capacity of depressurization and water injection.

<table>
<thead>
<tr>
<th>Events</th>
<th>SBO</th>
<th>SDS (2.7hrs)</th>
<th>SDS (2.7 hrs) + 1 SIP</th>
<th>SDS (3 hrs) + 1 SIP</th>
</tr>
</thead>
<tbody>
<tr>
<td>SBO , RX trip, MFW trip</td>
<td>0</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Core uncovery</td>
<td>2.13</td>
<td>2.13</td>
<td>2.13</td>
<td>2.13</td>
</tr>
<tr>
<td>Coe Exit Temp at 650 °C</td>
<td>2.53</td>
<td>2.53</td>
<td>2.53</td>
<td>2.89</td>
</tr>
<tr>
<td>Core Dry out</td>
<td>2.89</td>
<td>2.98</td>
<td>N/A</td>
<td>2.53</td>
</tr>
<tr>
<td>SIP actuation</td>
<td>N/A</td>
<td>N/A</td>
<td>2.79</td>
<td>3.02</td>
</tr>
<tr>
<td>Actuation of SITs</td>
<td>N/A</td>
<td>3.15</td>
<td>3.35</td>
<td>3.60</td>
</tr>
<tr>
<td>Molten fuel relocation</td>
<td>3.23</td>
<td>3.08</td>
<td>N/A</td>
<td>3.45</td>
</tr>
<tr>
<td>Reactor vessel failure</td>
<td>4.89</td>
<td>8.53</td>
<td>N/A</td>
<td>7.15</td>
</tr>
</tbody>
</table>
(2) Containment – Release to the Environment
Cooling Performance/Ablation Depth highly dependent on concrete composition

Molten Core Concrete Interaction

Corium LCS concrete content at cavity flooding

- 0.0 mass %
- 5.0 mass %
- 10.0 mass %
- 15.0 mass %
- 20.0 mass %

Corium SIL concrete content at cavity flooding

- 0.0 mass %
- 5.0 mass %
- 10.0 mass %
- 15.0 mass %
- 20.0 mass %
Ex-Vessel Corium Coolibility

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Containment Filtered Venting System

Reduce Risk of Late Containment Failure: Installation of the filtered ventilation system can reduce the amount of radioactive material released significantly, and minimize the long-term evacuation area.

- Taken as one of Post-Fukushima Actions in Korea
- Need to set up requirement; up to 72 hours, should be passive,…
- Verification of performance required
- Behavior of Ruthenium Oxide, Organic Iodine….
Status of implementation of FCVS

- One unit of PHWR (Wolseong-1) has completed an installation of FCVS (AREVA wet type) at the end of 2012 as a first unit in Korea, and under preparation to get the approval for the continued operation after 30 years operation.

- The remaining 22 NPP units (PWR-19, PHWR-3) in Korea are planned to install either FCVS or emergency back up spray until the end of 2015 to reflect the Post-Fukushima action item in Korea.

- Four units of APR-1400 NPPS under construction have original designs including emergency containment spray backup system (ECSBS) which are dedicated to the depressurization of containment atmosphere during severe accidents.

- Initiated as voluntary actions from the operator and it will be discussed more.
Containment/Calandria Vessel

Fig. 2. Multiple volumes of water surrounding the fuel in the CANDU

1. Calandria
2. Calandria End Shield
3. Shift-off and Control P.
4. Poison Injection
5. Fuel Channel Assembly
6. Feeder Pipes
7. Vault

470 Mg $\text{H}_2\text{O}$ in Calandria Vault / Shield Tank

120 Mg $\text{D}_2\text{O}$ in Heat Transport System (including pressurizer)

230 Mg $\text{D}_2\text{O}$ in Calandria Vessel / Moderator

Fuel heatup

Onset of loss of core integrity

Debris remains within calandria

T. Nguyen et al. / Nuclear Engineering and Design 238 (2008) 1093–1099
### Comparison of PWR and PHWR

<table>
<thead>
<tr>
<th>Sequence of Event</th>
<th>Wolsung Unit 2</th>
<th>OPR-1000</th>
</tr>
</thead>
<tbody>
<tr>
<td>SG dry out</td>
<td>2.2</td>
<td>1.1</td>
</tr>
<tr>
<td>LRV (PSV Open)</td>
<td>2.3</td>
<td>0.9</td>
</tr>
<tr>
<td>Core Uncovery</td>
<td>3.0</td>
<td>1.8</td>
</tr>
<tr>
<td>Core Melt</td>
<td></td>
<td>2.8</td>
</tr>
<tr>
<td>Corium Relocation</td>
<td>4.1</td>
<td>4.2</td>
</tr>
<tr>
<td>RV Dry out /Calandria Dry out</td>
<td>9.0</td>
<td>4.5</td>
</tr>
<tr>
<td>RV Failure /Calandria Failure</td>
<td>37.4</td>
<td>4.5</td>
</tr>
<tr>
<td>Containment Failure</td>
<td><strong>24.8</strong></td>
<td>~60</td>
</tr>
</tbody>
</table>
Containment/Calandria Vessel

Containment failure time (sec)  Calandria vessel failure time (sec)

IAEA CRP (2009-2012): Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Application

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Description of FCVS installed

- **FCVS in Wolseong-1(PHWR)**
  - Includes venturi scrubbers and metal fiber filter.
  - Installed inside the concrete structure just outside the containment.
  - Cylindrical pressure vessel dimension are approximately 6m height and 4m diameter.
  - Flow rate and containment pressure can be monitored in the FCVS control room.
  - The general performance of filtering efficiency would be as follows;
    - aerosol : 99.99%
    - elemental iodine : 99.5%
    - organic Iodine : 80~90 %
  - As the Wolseong-1 FCVS has installed recently and has not been in operation (due to delay for a permit for the continued operation), no data for maintenance operations is available at present.
  - There are no additional filtering performance tests which were performed specifically for Wolseong-1 FCVS. However, a full scale in-plant air-water hydraulic test was performed to check the system flow rate and pressure drop.
Development of CFVS in Korea

Development of CFVS
1st stage: 2013. 6.~ 2014. 5,  2nd stage: 2014. 6 ~ 2017.5

1st stage [1 year]
- Establishment of strategy for the securing of inherent technology
- Conceptual design of containment filtered venting system
- Design of integral test facility, Determination of test program

2nd stage [3 years]
- Detail design of containment filtered venting system
- Development of component
- Construction of integral test facility
- Integral test
- Separate effect test
- Evaluation of safety and application to Korean NPP
## Technical Goal

<table>
<thead>
<tr>
<th>Item</th>
<th>Unit</th>
<th>Best Performance level</th>
<th>Country / Company</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Aerosol</td>
<td>DF</td>
<td>10,000</td>
<td>(France/AREVA)</td>
</tr>
<tr>
<td>2. Elemental Iodine</td>
<td>DF</td>
<td>1,000</td>
<td>(USA/WH)</td>
</tr>
<tr>
<td>3. Organic Iodine</td>
<td>DF</td>
<td>1,000</td>
<td>(Swiss/IMI)</td>
</tr>
<tr>
<td>4. Passive operation</td>
<td>day</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>5. Separate effect test</td>
<td></td>
<td>TH test, re-suspension test, re-volatilization test</td>
<td></td>
</tr>
<tr>
<td>6. Integral test</td>
<td></td>
<td>10 bars, 200°C</td>
<td></td>
</tr>
</tbody>
</table>
(3) Bypass Scenarios – direct release

Typical Fraction of Containment Failure Frequency

- NO CF: 48.6%
- BMT: 30.8%
- CFBRB: 3.5%
- LCF: 1.7%
- ECF: 7.1%
- BYPASS: 8.3%
- SO.FAL: 0.02%

Taken from Ref. 2-1, 2-2
Temperature Induced SGTR

Failures are likely to occur in the hot-leg or the surge-line leading to depressurization, which prevents steam generator tube failure.

They compete. Occurrence of TI SGTR highly depends on the mixing at the SG inlet plenum and pre-existing flaws in the SG tubes.

Taken from Ref. 4
• Provide Means of Flooding Ruptured SG

- Turbine Driven Auxiliary Feed Water Pump
- Injection by Fire Pump
- Pressure can be reduced by opening ADVs
- Need SG level measurement

• Provide a means of filtering at the exit of ADVs – how?
III. Balanced Prevention and Mitigation

• Severe Accident Prevention and Mitigation
  With a safety goal of practically eliminating the significant release of radioactive material, an innovative approach for preventions and mitigation strategy need to be developed.

  A new severe accident management strategy with dedicated mitigation measures and qualified essential instrumentations to take timely recovery action is recommended.

• Inclusion of Severe Accident in a Plant Design Basis
## Accident Management for the Worst Scenario

<table>
<thead>
<tr>
<th>Prevention and Mitigation Strategy</th>
<th>Dedicated Mitigation Measures</th>
</tr>
</thead>
<tbody>
<tr>
<td>Prevent Reactor Vessel Failure:</td>
<td>Feed and Bleed Operation by Depressurization System and Low Pressure Safety Injection System</td>
</tr>
<tr>
<td>Decay Heat Removal and Inventory Makeup</td>
<td>Essential Instrumentations</td>
</tr>
<tr>
<td>Prevent High Pressure Melt Ejection:</td>
<td><em>(Timely recovery action very necessary, Secure power supply, such as mobile DG and/or Battery, Qualification of Equipments and Instrumentation under Severe Accident Condition required)</em></td>
</tr>
<tr>
<td>Prevent Containment Failure:</td>
<td><strong>PARs:</strong> Core Catcher or Cavity Flooding Containment filtered venting system</td>
</tr>
<tr>
<td>Hydrogen Mitigation</td>
<td>Essential Instrumentations</td>
</tr>
<tr>
<td>Molten Core Cooling</td>
<td><em>(Passive system relying on natural force Qualification of Equipments and Instrumentation under Severe Accident Condition required)</em></td>
</tr>
</tbody>
</table>
Depressurization System Qualification needed

SFP Level Measurement

WGAMA CAPS on CFVS, PASAM, Long Term Operation?

Core Catcher

AP1000 Passive Safety System

OECD-THAI, PAR Performance
Summary and Conclusion

Acceptance Criteria for the Severe Accident have to be formulated to protect people, minimize the property damage and contamination of environment to reflect the lessons learned from the Fukushima Accident.

Instead of an optimistic view on progression of severe accident and available resources, the worst case scenario has to be considered to be prepared for the unexpected situation.

With a safety goal of no significant release of radioactive material, an innovative approach for a mitigation is needed to be investigated. It should consist of a new severe accident management with dedicated mitigation measures and qualified essential instrumentations.

Most of technologies are already proposed while further R&Ds are needed to improve the performance.
For the Clean and Safe Nuclear Technology