State of the Art and Challenges in Level-2 PSA for New and Channel Type Reactors in India

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Outline

Introduction

PART 1: Uncertainties Associated with PHWR SA Progression
  – Introduction
  – Severe Accident Progression
  – Uncertainties
  – Conclusion

PART 2: Studies Carried out

PART 3: VVER

PART 4: Issues in Advanced Reactors
PSA

Level 1 PSA
core damage frequency

Level 2 PSA
Large early release frequency

Level 3 PSA
probability and level of radiological consequences on humans and on environment

Thermal-hydraulics
EOPs

severe accident management guidelines

Radioactivity releases to the environment
Emergency planning
Level-2 PSA

Level 1 Event Tree + Interface Characteristics (Level 1+)

Plant Damage States

Accident Progression Event Tree (Level 2 Event Tree)

Release Categories

- CD states are binned based on the similarities of the accident progression and demand on the engineered safety features to arrive at the plant damage states.
Specific plant features that can influence the progression of a severe accident and Level 2 PSA should be identified.

Binning of Core Damage States

Core degradation behaviour, containment behaviour etc.

Release of radioactive fission products outside the containment
Level-2 PSA Team

**Operators and operational analysts:** Specialists in the design and operation of the plant and key containment systems, the emergency operating procedures and the severe accident management guidelines.

**Specialists in phenomena:** Specialists in severe accident phenomena, containment performance, uncertainties associated with severe accidents, chemical and physical processes governing accident progression, containment loads, releases of radio-nuclides and computer codes for the analysis of severe accidents.

**Structural specialists:** Specialists in the structural design, the pressure capacity and the failure modes of the containment.

**PSA specialists:** Specialists in event tree analysis, fault tree analysis, human reliability analysis, uncertainty analysis, statistical methods, processes for expert elicitation and judgement, PSA computer codes and Level 1 PSA.
Objectives of Level-2 PSA

- To gain insights into the progression of severe accidents and the performance of the containment.
- To identify plant specific challenges and vulnerabilities of the containment to severe accidents.
- To identify major containment failure modes and their frequencies and to estimate the associated frequencies and magnitudes of radionuclide releases.
- To provide an input into the development of strategies for off-site emergency planning.
- To evaluate the impacts of various uncertainties, including uncertainties in assumptions relating to phenomena, systems and modelling.
- To provide an input into the development of plant specific accident management guidance and strategies.
- To provide an input into determining plant specific options for risk reduction.
- To provide an input into the prioritization of research activities for the minimization of risk significant uncertainties.
PART-1: Introduction – PHWR SA

- Development of computer codes, SAMG, Level-2 PSA, implementation of design provisions for accident mitigation etc. needs thorough understanding of the severe accident progression.

**Limited Core Damage Accidents**
- Involve single channel e.g. flow blockage, feeder break, PT and/or CT failure
- Full core events e.g. LOCA + LOECCS
- Core geometry is preserved
- Availability of moderator prevents gross melting

**Severe Core Damage Accidents**
- SCDA can initiate from LCDA
- Moderator is not available
- Involve multiple channels
- LOCA + LOECCS + LOM
- Extended SBO with multiple failures
• SA progression in PHWRs is different from LWRs till the corium debris bed is formed at the bottom of the calandria vessel. Thereafter, it is similar to LWRs.
• Presence of moderator slows down the core disassembly in PHWRs.
• Differences in corium composition and corium geometry.
Sequence of Events – PHWR SA

- **Intact Circuit Faults (Non-LOCAs)**
  - e.g. Transients, Flow blockage
- **Feed water (Main & Auxiliary)/ shutdown cooling not available**
- **SG crash cooling and Fire water injection to SG not available**
- **PHT heat-up, pressure rise etc. may lead to PT/CT failure**
- **Hydrogen Generation & combustion, detonation may occur**
- **ECCS not available**
- **Fire water injection to PHT not available**
- **Moderator cooling not available**
- **Moderator inventory boil-off, core damage**
- **Loss of Coolant Accidents**
  - **Core Disassembly**
  - **Calandria vault water boil-off if replenishment is not available**
  - **Calandria vessel failure, debris relocation to vault water**
  - **Steam explosion may occur, MCCI**
  - **Containment failure**

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Severe Accident Progression

• Stage 1:
  – Loss of cooling, fuel channels dry out and heat up.
  – Oxidation of fuel cladding.
  – Pressure tubes balloon/sag and contact with the calandria tubes.
  – Heat removal from the moderator.
Stage 1: Uncertainties

- Failure of PT&CT in a high pressure scenario affect the integrity of the neighbouring channels and calandria vessel?
- Effect of Pipe whip & Jet impingement.

Jet & hot material impingement
Expanding steam bubble
Shattered and dispersed fuel
Pipe whip
Vessel pressurization
Stage 1: Uncertainties

- Non-uniform circumferential temperature distributions lead to pressure tube rupture? Presently temperature based criteria is used in the MAAP4-CANDU and ISSAC codes.
- Requires detailed models to simulate local melt-through of PT and CT or sag which may cause failure of fuel channels. Presently temperature based criteria are used in the codes.
- Stratification of steam and water in the coolant channels (single/multiple channels) may occur during certain accident scenarios (Stagnation channel break/critical break). This may lead to failure of the PT&CT even before channel uncovering from the outside.
- Uncertainties in heatup and hydrogen generation during reflooding of core at high temperatures. – Applicability to PHWRs
Severe Accident Progression

• Stage 2:
  – Moderator level drops down and exposes several upper channels
  – Exposed channels heat up, sag, oxidise and break apart
  – Collapse on to lower submerged channels/to bottom of the calandria vessel

Uncertainties:
• Heat distribution in the PHT to quantify the risk of PHT failure and possible containment bypass. SG tubes may rupture.
Stage 2: Uncertainties

- Core disassembly model – Experimental/theoretical studies are required.
  - MAAP4-CANDU: If debris mass exceeds certain amount (25 tons), fuel channel failure is assumed. Can it fail with less debris?
- Deformation of the PT and CT
- Material properties at high temperatures
- Hydrogen production – availability of steam in the interior of debris?
- Effect of oxidising environment on core disassembly?
Severe Accident Progression

- **Stage 3:**
  - Loss of moderator
  - All channels heatup, sag, oxidise and break apart
  - Corium at the bottom of the calandria vessel
  - Vault water boil-off

- Steam supply into the interior of debris is a source of uncertainty.
  - Affect the heat and hydrogen generation
  - Affect the accident progression
- Possibility of occurrence of steam explosion in the calandria vessel?
- Failure of calandria vessel at high heat fluxes?
- Location and size of opening of the calandria and its ablation?
- Hydrogen generation during relocation of the melt?
- Melt slumping to water and particulate formation
- Sudden drop of a molten material to the bottom portion of the calandria may lead to strong steam surges.
  - Pressurisation of calandria can take place and challenge the integrity of the vessel.
Severe Accident Progression

• Stage 4:
  – Melt through of calandria vessel and corium is on the calandria vault floor.

Uncertainties:
• Steam explosion may occur while the relocation of debris to the calandria vault. Whether it leads to failure of containment?
• Efficiency (conversion of thermal energy to mechanical energy) of the explosion.
• Debris coolability from the overlaying pool of water?
  ✓ Coolability depends on the composition, porosity etc.
• Formation of debris layers (oxide layers/metallic layers)?
• Formation of crusts and rupture?
• Melt spreading?
• Hydrogen generation during relocation of the melt.
Severe Accident Progression

**Stage 5:**
- MCCI begins
- Ablation of concrete
- Generation of non-condensables

**Uncertainties:**
- Generation of non-condensables (H$_2$, CO etc.), steam etc. during the MCCI.
- Oxidation of Zr by air. Produces lot of heat.
- Material configuration (layers)
- Rate of concrete ablation?
- Hydrogen generation and distribution in the containment. Hydrogen combustion/detonation?

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Part 1 - Conclusion

• Severe accident progression till the corium bed formed at the bottom of the calandria vessel is unique to PHWRs.

• Experimental/theoretical studies required for better understanding and to resolve these issues, which would help in the development of:
  – Severe Accident Management Guidelines
  – Computer codes and their validation
  – Level-2 PSA studies
  – Design provisions
PART-2: Studies carried out in NSAD on Severe Accidents

– Severe Accident Analysis
  • VVER using RELAP5/SCDAP
  • VVER using ASTEC (IRSN)
  • TMI-2 base case analysis using RELAP5/SCDAP and ASTEC (NRC)
  • CANDU 6 severe accident analysis

– Containment Thermal Hydraulics/Hydrogen Distribution Studies
  • Hydrogen distribution analyses for TAPP3&4 containment using ASTEC
  • Development of wall condensation model (IIT-B)
  • Simulation of MISTRA test facility (OECD/SETH-II) (IIT-B)

– Conclusion
Severe Accident Analyses - VVER

- In-vessel scenario
- The following cases were analysed for a typical VVER
  - Simultaneous rupture of all main steamlines (MSLB ALL)
  - SBO
  - LB LOCA + SBO
Nodalisation of Vessel – SCDAP/RELAP

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Nodalisation of Primary Loop - SCDAP/RELAP
ASTEC Capabilities

- **CESAR** for thermal hydraulics in the RCS
- **ICARE/DIVA** core degradation till vessel failure
- **ELSA** for FP release from fuel rods
- **SOPHAEROS** for FP vapour and aerosol transport in the RCS
- **RUPUICUV** for Direct Containment Heating
- **CORIUM** for heat transfer between containment atmosphere and corium entrained out of the cavity by DCH
- **WEX** for MCCI in the cavity
- **CPA** Containment thermal hydraulics, aerosol and FP behaviour inside the containment
- **IODE** for iodine behaviour in the containment
- **ISODOP** for calculation of activity and decay heat in the reactor zones
- **SYSINT** for management of safety systems (fan, cooler, spray, recombiners etc.)
Accident Scenario:
- Simultaneous double-ended rupture of all four main steamlines.
- Main steam isolation valves (MSIVs) are assumed to remain open.
- Steam from the steamlines flows through the breaks. The flow rate increases due to the large pressure difference across the break.
- Resulting in more heat removal from the primary side through secondary circuit initially and results in pressure and temperature reduction of the primary circuit.
- Secondary inventory decreases and SGs dry out. Primary pressure keeps rising due to loss of heat sink.
- When the primary pressure reaches the set points of control pulse safety device (PSD), the valve will open and close.
- The primary coolant flows out through PSD and results in core uncovery.
- The fuel temperature rises and cladding oxidation takes place due to Zircaloy and steam interaction. This causes further rise in temperatures.
Pressure decreases due to large heat removal from the secondary side initially and subsequently increases due to loss of secondary inventory.
Initially the temperature increases at a slower rate due to decay heat. Once it reaches the oxidation threshold temperatures, the zircaloy oxidation takes place and it results in larger amount of heat generation. This causes rapid increase of temperatures. Initially the temperature increases at slower rate up to about 8000 s. Later on the temperature increases rapidly due to oxidation heat generation in the core.
**Accident Progression:**

- The initiating event leads to switch-off of RCP pumps, make-up pumps and main feed water pumps and BRU-K. The shut-off valves of the turbine are closed. Reactor scrams on RCPs trip.
- RCPs trip leads to increase in primary coolant heat-up and primary pressure.
- Closure of the turbo-generator stop valve leads to rise in secondary pressure. Failure of BRU-A is assumed which leads to steam generator safety valves (SG-PSDs) opening. The periodical opening and closing of SG safety valves leads in fluctuation of secondary as well as primary pressure.
- PHRS is assumed to be unavailable. Steam generator dry-out leads to further growth in primary pressure, which results in opening of pressurizer safety valves.
- Loss of primary and secondary inventory through PSDs results in fall in water level of reactor pressure vessel and steam generator.
- Core heat up begins.
SBO - Pressure

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Accident Progression:

- The double-ended rupture in the cold leg and station black out are assumed to occur simultaneously.
- RCPs trip, reactor scrams, main feed water and auxiliary feed water pumps trip and pressuriser electrical heaters also trip due to SBO.
- The turbine shut off valves close due to SBO.
- The flow reversal takes place in the broken cold leg and flow rate increases due to large pressure difference across the break.
- Active high-pressure and low-pressure emergency core cooling systems are not available.
- The passive hydro accumulators actuate at primary pressure less than 5.89 MPa.
- The credit of second stage hydro accumulators and passive heat removal system designed for mitigation of severe accidents are not taken into account in this analysis.
- Due to rapid uncover of core and mismatch between heat production and removal, the heating up of the core takes place.
Oxidation causes rapid increase of temperatures due to large amount of oxidation heat. The clad temperature rises to a peak value of 3250 K at about 610 s. Later on it decreases to about 2893 K due to heat transfer to surroundings structures and decrease in oxidation heat generation.
LOCA + SBO

Core Total Hydrogen Generation Rate (kg/s)

-0.1
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0
1000
2000
3000
4000
5000
6000
Time (s)

Hydrogen Generation Rate (kg/s)

0
0.02
0.04
0.06
0.08
0.10
0.12
0.14
0.16
0
2000
4000
6000
8000
10000
Time (s)

Hydrogen Generation (kg)

0
5
10
15
20
25
0
2000
4000
6000
8000
10000
Time (s)

150 kg approx

SCDAP

ASTEC V1.3

ASTEC V1.3

H2 production in the core

1 Fe
2 B4C
3 UO2-Zr
4 Total

Time (s)
• SPE on TMI-2 was launched in bi-lateral technical co-operation with USNRC
• TMI-2 accident scenario was analysed using severe accident codes SCDAP and ASTEC - Base case
• Uncertainty and sensitivity analyses were carried out using sampling based techniques.
<table>
<thead>
<tr>
<th>Phase 1: 0-100 minutes</th>
<th>Phase 2: 100-174 minutes</th>
</tr>
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<tbody>
<tr>
<td>Loss of Feedwater (MFWA and B Trip)</td>
<td>Core degradation</td>
</tr>
<tr>
<td>Pressuriser PORV Opens on high pressure</td>
<td>Pressuriser PORV closes</td>
</tr>
<tr>
<td>Reactor Trip on high primary pressure</td>
<td>Phase 3: 174-200 minutes</td>
</tr>
<tr>
<td>PORV Fails to re-close on low pressure</td>
<td>MCP 2B switched on</td>
</tr>
<tr>
<td>EFW for SG A and B actuation signal occurs on low SG level</td>
<td>Pressuriser heaters are shutoff</td>
</tr>
<tr>
<td>Trains A and B of HPI are automatically actuated as a result of low reactor coolant pressure</td>
<td>Pressuriser PORV opens</td>
</tr>
<tr>
<td>HPI stopped on PRZ level rise</td>
<td>MCP 2B switched off</td>
</tr>
<tr>
<td>EFW block valves opened</td>
<td>Phase 4: 200 - 300 minutes</td>
</tr>
<tr>
<td>MCP 2B switched off because of low pressure, high vibration, low flow and concern about pump seal failure</td>
<td>HPI started</td>
</tr>
<tr>
<td>MCP 1B switched off</td>
<td>PORV closes</td>
</tr>
<tr>
<td>MCP 2A switched off</td>
<td>PORV opens</td>
</tr>
<tr>
<td>MCP 1A switched off</td>
<td></td>
</tr>
</tbody>
</table>
TMI-2 Primary Pressure Response

- Pressure increases initially because of loss of heat sink from the secondary and reactor trips on high pressure. The trip resulted in rapid fall in pressure and is constant up to 100 minutes due to sufficient make-up. Subsequently pressure started falling due to increase in FW flow. Pressure increased due to block valve closure.
Estimated total \( \text{H}_2 \) production in TMI-2 is about 460 kg

Estimated \( \text{H}_2 \) production in TMI-2 at 174 min is about 300 kg
Hydrogen Distribution Analysis - Introduction

- Containment thermal hydraulic analysis was carried out to predict the hydrogen distribution and to check the flammability limits in various compartments of the containment.
- Analysis was carried out for LOCA + LOECCS
- ASTEC CPA module is used.
- Analysed in two parts:
  - Part 1: with the existing features only
  - Part 2: with the proposed scheme
- Proposed scheme:
  - Forced mixing system with a large capacity of blowers to mix hydrogen
  - Coolers will initially be switched off (for 1 hour)
Hydrogen Distribution Analysis

Containment Building

Dome Region (C9)

Pump Room (C6)

Pump Room (C7)

C5
F/M Vault (North)

C4

C2

C3

Calandria Vault (C1)

(C8)
F/M Vault (South)

V2 Volume (C11)

Suppression Pool (C10)

Nodalisation

Hydrogen Distribution Analysis- Modelling

• The containment walls, floors, ceilings and metal internals (pipelines, vessels, equipment etc.) are modeled with 50 heat structures.

• The heat structures, steel, concrete and paint are split into 3, 14 and 2 temperature nodes.

• The blowout panels are simulated using valve components in the code. These valves are kept closed initially and are opened on certain pressure difference between Volume V1 and V2.

• **Assumptions:**
  – The break location is conservatively assumed to occur in the north FM vault of the containment (C4) as the pump room volume is several times larger than the FM vault room which reduces the local hydrogen concentration once it mix-up with the pump room atmosphere.
  – Pump room and FM vault coolers are not modeled in the analysis.
Hydrogen Distribution Analysis - Results

Hydrogen Concentration

Part 1 - ASTEC

Part 2 - ASTEC

Part 2 - PACSR
Hydrogen Distribution Analysis - Results

Ternary Diagram for the Break Compartment

Part 1 - ASTEC

Part 2 - ASTEC

Part 2 - PACSR

Coolers effect: mixing, non-inerting

Detonation Limits

Deflagration Limits

3600 s
Development of Wall Condensation Model

• Hydrogen distribution study is an important safety issue
  – Needs accurate prediction and CFD codes seem necessary
  – Steam condensation model developed for CFD codes
  – Validation against the idealised version of the CONAN test facility
  – Validation against the TOSQAN experiments
Simulation of TOSQAN Facility – Condensation Model Validation

TOSQAN Test Facility

UDFs developed for simulating the wall condensation through sinks of mass, momentum, energy, specie, k and ε.

2-D Axisymmetric Model
Grid Sensitivity - TOSQAN Test Facility

Grid Sensitivity Model Sensitivity

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MISTRA Facility - Introduction

- Stratification of hydrogen is possible in certain compartments of the containment during the accident progression.
- INITIALA test of MISTRA simulated to study the characterization of stratification and its erosion by the diffusion.
- Validation of the CFD codes OECD/SETH-II.
- SS containment of 97.6 m$^3$
- Inner compartment equipped with an annular floor called ring plate.
- Injection of helium radially at (6.559 m) to create a stable stratified layer of helium. From four injection points (22.6 mm dia each)
- Vent (200 mm dia) is located in the lower part.

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MISTRA Facility – Results

Helium Mole Fraction \((r=1.54 \text{ m, } \theta=7.5^\circ, Z=z)\)

Contours of Helium Mole Fraction

Just end of the injection

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He Mole Fraction after 90 minutes of Helium Injection
Grid Sensitivity on He Mole Fraction
Part 2: Conclusion

- Development of the capability of analysing severe accidents for light water reactors.
- PHWR severe accident analysis is under progress.
- Hydrogen distribution analysis using lumped parameter codes.
- Development of capability for applying CFD codes to containment hydrogen distribution studies.
PART3: VVERs Level2 PSA

• Preventing the core damage;

• Preventing the primary circuit boundary failure;

• Ensuring containment integrity;

• Reduction of fission products release.
Prevention of Core Damage

- ECCS of high and low pressure;
- Primary circuit make-up system (for small leaks);
- Heat removal via second circuit.
Severe Accident Mitigation

- Prevent scenario with high-pressure core melt ejection from the primary circuit;
- Core melt localization (in-vessel localization for VVER of medium capacity or ex-vessel localization for VVER of high capacity);
- Hydrogen safety;
- Prevent long-term containment failure;
- Control of volatile iodine compounds.
Prevention of HPME

Preventing high-pressure core melt ejection

Primary depressurization system operates when the core exit temperature exceeds a certain value which depressurize the primary system and the failure of reactor pressure vessel during the accident progression would be at lower primary pressure, thus this feature practically eliminate the high pressure melt ejection and direct containment which would in-turn eliminates the large early during the accident progression.
Prevention of Detonation

- Hydrogen safety
  - Passive catalytic hydrogen recombiners;
  - Use of natural inertization of the gas medium by steam.

The PARs combine the hydrogen with the oxygen and in turn the hydrogen concentration is reduced in the containment and decrease the probability of containment failure because of hydrogen detonation. This eliminates the early failure of the containment during the accident progression due to hydrogen detonation. This in turn reduces the early failure of the containment.
Core catcher would collect the molten material and cool it. This would eliminate the molten corium concrete interaction (MCCI), hydrogen generation during the MCCI, other combustible and non-condensable gases generation. Containment basemat melt through etc. This would further reduce the failure probability of the containment and release of radioactive fission products.
Features of VVERs that Reduce Source Term

- Core catcher would collect the molten material and cool it. This would eliminate the molten corium concrete interaction (MCCI), hydrogen generation during the MCCI, other combustible and non-condensable gases generation. Containment basemat melt through etc. This would further reduce the failure probability of the containment and release of radioactive fission products.

- Passive heat removal system and second stage hydro-accumulators together would be able to remove the decay heat during the loss of coolant accident with station blackout conditions for reasonable amount of time. After which, recovery actions are possible to remove the decay heat. (Hook-ups)

- The quick boron injection system injects boron into the primary passively with the help of differential pressure across the pump. This system is designed for anticipated transients without scram. This would further reduce the probability of core damage and containment failure in the events in which reactor power excursions takes place.

- The above features practically eliminate the large early releases and also reduce the likelihood of containment failure in the long term during the various accident scenarios. These features would substantially reduce the large early release frequency for VVERs.
L3: The initiating event ‘Large LOCA with equivalent diameter of 135 mm to 279 mm on hydro accumulator pipeline” is considered for this analysis.

E: Offsite power supply (NPS)

F: High-pressure emergency core cooling system (HP ECCS). The success criterion considered in the analysis is the availability of one train of HP ECCS. Dependent failure of one train due to fire is considered in this analysis.

M: First stage hydro accumulators. The success criterion considered in the analysis is the availability of two hydro accumulators. Dependent failure of one hydro accumulator due to break in the hydro accumulator pipeline was considered.

G: Low pressure emergency core cooling system (LP ECCS). The success criterion is the availability one LP ECCS train. Dependent failure of one LP ECCS train due to break in the hydro accumulator pipeline was assumed.
Event Tree for LBLOCA

- **AS1:** NPS and HP ECCS are successful in this sequence and leads to successful end state of cold shutdown state (CSS). The core cooling is achieved by HP ECCS.
- **AS2:** In this sequence the NPS is available and HP ECCS is unavailable. The core cooling is achieved by HAs and LP ECCS and leads to successful end state of CSS.
- **AS3:** This sequence follows similar to the sequence AS2 except the LP ECCS failure. Hence this leads to core damage (CD), as the function core cooling is not achieved.
- **AS4:** The NPS is successful, whereas HP ECCS and HAs are unavailable. Due to the unavailability of HP ECCS and HAs, this will also lead to the end state of CD.
- **AS5-AS8:** The accident sequences AS5 to AS8 are similar to the sequences AS1 to AS4 respectively except that the non-availability of offsite power supply failure.

Accident Sequences AS1 to AS4 was quantified under the condition of availability of offsite power supply and accident sequences AS5 to AS8 were quantified with the unavailability of offsite power supply failure.
# Level-1 PSA for VVER Independent Verification

<table>
<thead>
<tr>
<th>S. No.</th>
<th>System Code</th>
<th>System Name</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>JNA10-40</td>
<td>Residual heat removal system</td>
</tr>
<tr>
<td>2.</td>
<td>JND10-40</td>
<td>High pressure emergency boron injection system</td>
</tr>
<tr>
<td>3.</td>
<td>JNG10-40</td>
<td>First stage hydro accumulators</td>
</tr>
<tr>
<td>4.</td>
<td>KAA</td>
<td>Component cooling system</td>
</tr>
<tr>
<td>5.</td>
<td>PE</td>
<td>Essential service water system</td>
</tr>
<tr>
<td>6.</td>
<td></td>
<td>Emergency power supply system</td>
</tr>
<tr>
<td>7.</td>
<td>KLC-01</td>
<td>Air cooling system of pumps and heat exchanger rooms of reactor coolant system</td>
</tr>
<tr>
<td>8.</td>
<td>KLC-02</td>
<td>Air cooling system of pumps and heat exchanger rooms of component cooling system</td>
</tr>
<tr>
<td>9.</td>
<td>SAQ20, 40</td>
<td>Ventilation system in rooms of essential service water system pumps</td>
</tr>
<tr>
<td>10.</td>
<td>SAC23, 43</td>
<td>Plenum exhaust ventilation system in storage battery rooms</td>
</tr>
<tr>
<td>11.</td>
<td>SAC25, 45</td>
<td>Air cooling system in switch gear and uninterruptible power supply unit rooms</td>
</tr>
<tr>
<td>12.</td>
<td>SAC27, 47, 01</td>
<td>Ventilation system in chilling plant rooms</td>
</tr>
<tr>
<td>13.</td>
<td>QKB</td>
<td>Essential chilled water supply system</td>
</tr>
<tr>
<td>14.</td>
<td>SAD01</td>
<td>Diesel generator machine room air cooling system</td>
</tr>
</tbody>
</table>
Core Total Hydrogen Generation Rate

Hydrogen Generation

150 kg approx

SCDAP

ASTEC V1.3

ASTEC V1.3

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Severe accident analysis for these categories are being carried out.

Nodal questions/phenomenological questions during the accident progression.
PART 4: Issues in Advanced Reactors

1. Decay Heat Removal
   - active/passive System & Actuation
   - Passive System Reliability
   - Natural circulation instability
   - Fukushima like extended SBO
   - Database a) joint probability b) distributions c) methodology for knowledge gap
d) target number/cost efficiency
   - SBO UHS – water, air, tanks, hook up
   - SBO power supply
   - Energy converters active/passive

2. Code used RELAP/SCDAP, ASTEC Model Uncertainty
   - Design & Safety Codes Mass, momentum, energy eqn. soln. 3 6 eqn. CFD 3D
     kinetics and point kinetics, multiple point kinetics
   - Best estimate uncertainty safety margins
   - Conservative
Issues in Advanced Reactors

3. ATWS poison injection
   - active/passive System & Actuation
   - Passive System Reliability
4. Power density – large core Xe oscillations
5. High burn-up, Fission product inventory, fuel integrity
6. No Experience with New material & design & concepts
7. DSR: Operating aspects complex or control system, advanced but not reliable
8. No Database of New innovative component failure
9. New Clad material
10. Reactor independent safety criteria not evolved?
11. Refueling accidents & issues Cold Pressurization
12. ECCS injection active/passive actuation
13. Seismic BDBA
14. SBO cooling hampered by reactivity sub criticality margins
15. Hydrogen distribution recombiners performance cooling passive
16. MHT configuration ECCS connection with PHT and interloop connection without scram, LOCA with ECCS, LOCA depressurization
17. Scaling Uncertainties and reactor power with respect to various primary SG system, neutronics.
18. Fuel fabrication CHF margin & Fuel reprocessing Th.-232
19. Boron dilution and corrosion
21. Sever Accidents
   - Metal water reaction and water addition in SAMG
   - SAMG and debris bed cooling
   - Avoiding core damage and severe accident
   - Core catcher performance

22. SAMG, EOP, human reliability combined with uncertainties in scenarios observed with respect to anticipated scenarios, analysed and provided for operator training

23. Cold pressurization (Pressure RV open)

24. Water steam hammer

25. Ageing of structure & effect on CHF

26. 40-90 years design life channel replacement

27. Simplification in design and operation

28. Flood PSA, seismic, DGs availability flood level