Severe Core Damage Accidents and MAAP4 CANDU

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Presentation Outline

- Introduction
- Key Safety Features of ACR Design
- MAAP4 CANDU Code and Accident Progression
- MAAP4 CANDU Validation Activities
- Severe Accident Management Activities
- Summary

Introduction

- Presentation addresses Severe Core Damage Accident Analysis using MAAP4 CANDU

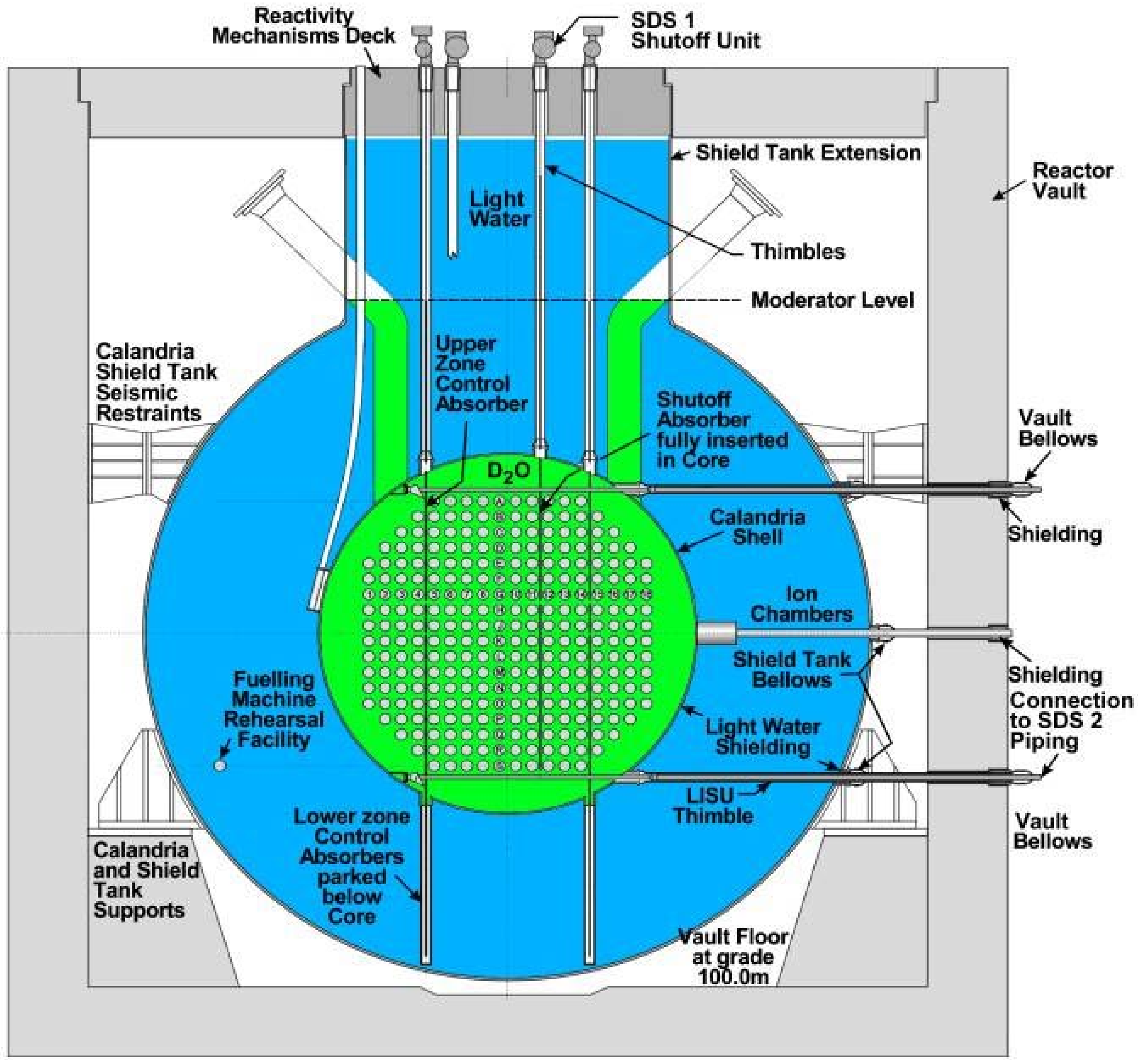
- Severe Core Damage Accident
  - Accident in which substantial damage is done to the reactor core structure whether or not there are serious off-site consequences
    - Reactor Cooling System and Moderator back-up heat sinks are unavailable. In ACR-700, RWS must also fail (very unlikely scenario)

"Significant quantity of heat sinks surrounding core, therefore CANDU Severe Core Damage Progression is slow. Operator will have sufficient time to arrest accident progression so that corium can be contained in calandria vessel. Corium-concrete interaction is unlikely"

Key Safety Design Features of ACR

- The moderator system provides a backup heat sink capable to maintain core coolability for loss-of-coolant accidents combined with the unavailability of the emergency core cooling
- The moderator and the shield water systems have sufficient thermal capacity to slow down a severe core damage progression so that the operator will have sufficient time to implement severe accident management measures
- Passive makeup systems from Reserve Water Tank are provided for the moderator and the shield water to extend the duration of their heat removal capability for Severe Core Damage Accidents

ACR-700 Reactor Core



284 channels

H2O in RCS: 118 Mg

H2O in ST: 456 Mg

D2O in CV: 102 Mg

RWT: 2500 Mg

*) Compared to a CANDU 6 ACR-700 has no basement, area below ST can be flooded

Severe Core Damage progression in CANDU is slow

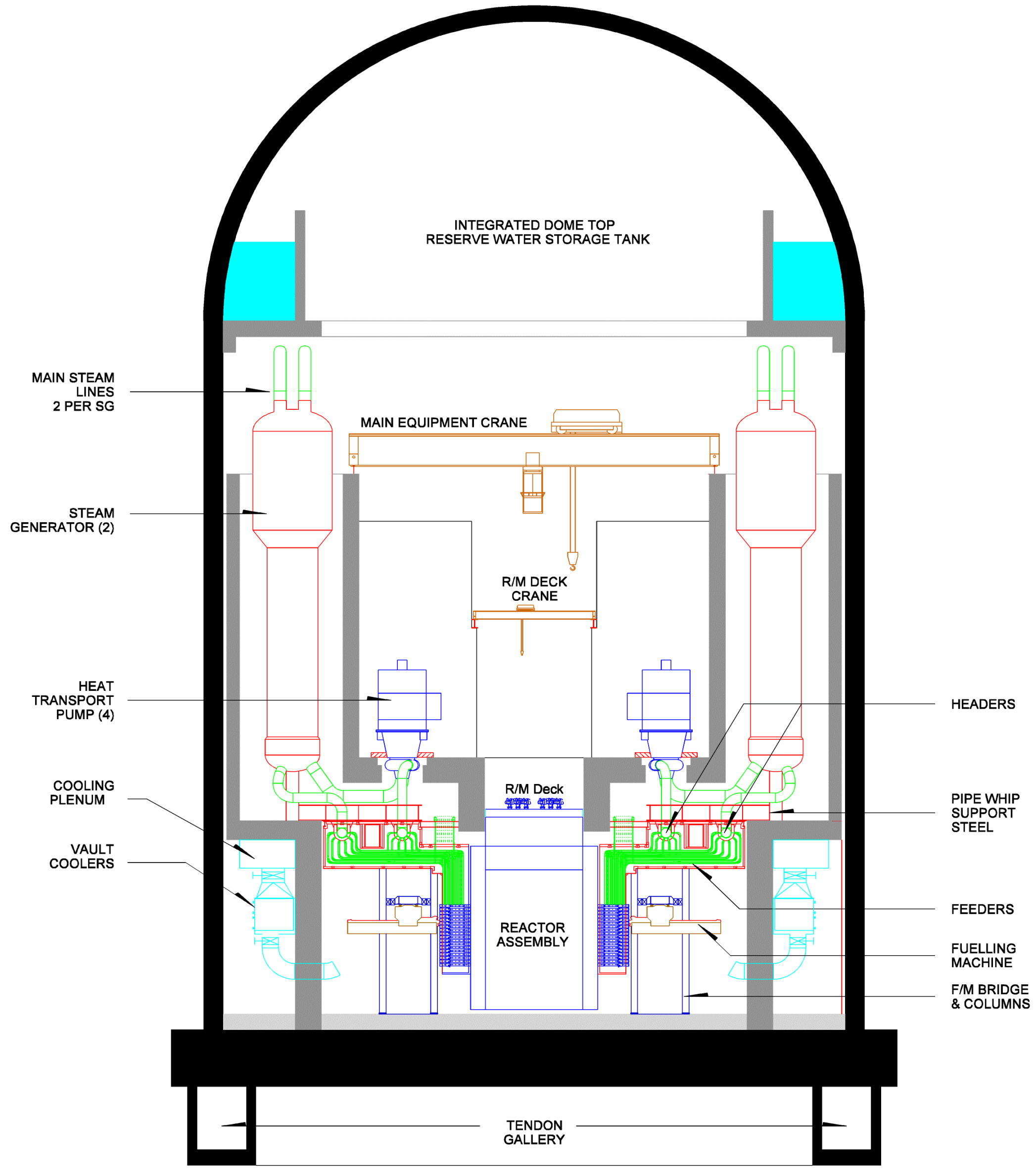
ACR Safety Systems

- **Safety systems are incorporated to mitigate the consequences of process failures requiring shutdown, to remove decay heat and/or to retain radioactive releases**
  - **Shutdown Systems (1&2): not discussed here**
    - **Accident progression after reactor shutdown addressed**
  - **Emergency Core Cooling System: carried out by two systems**
    - **Emergency Coolant Injection System (ECI) for high-pressure coolant injection after a Loss-of-Coolant Accident (LOCA)**
    - **Long Term Cooling (LTC) System for fuel cooling**
      - **in the long term (recovery stage) of a LOCA**
      - **removes decay heat for all conditions with the heat transport system (HTS) pressure boundary intact.**
      - **serves the function of shutdown cooling for cooldown after a normal shutdown.**
  - **Containment System**
    - **Retains radioactive releases**

Containment

- Steel-lined, pre-stressed concrete containment designed for a low leakage rate
- Containment isolation system automatically closes penetrations open to the containment atmosphere when high pressure or high radioactivity in containment
- Containment cooling system with local air coolers suitably distributed inside containment for heat removal from containment atmosphere
- Passive Autocatalytic Recombiners for Hydrogen Control

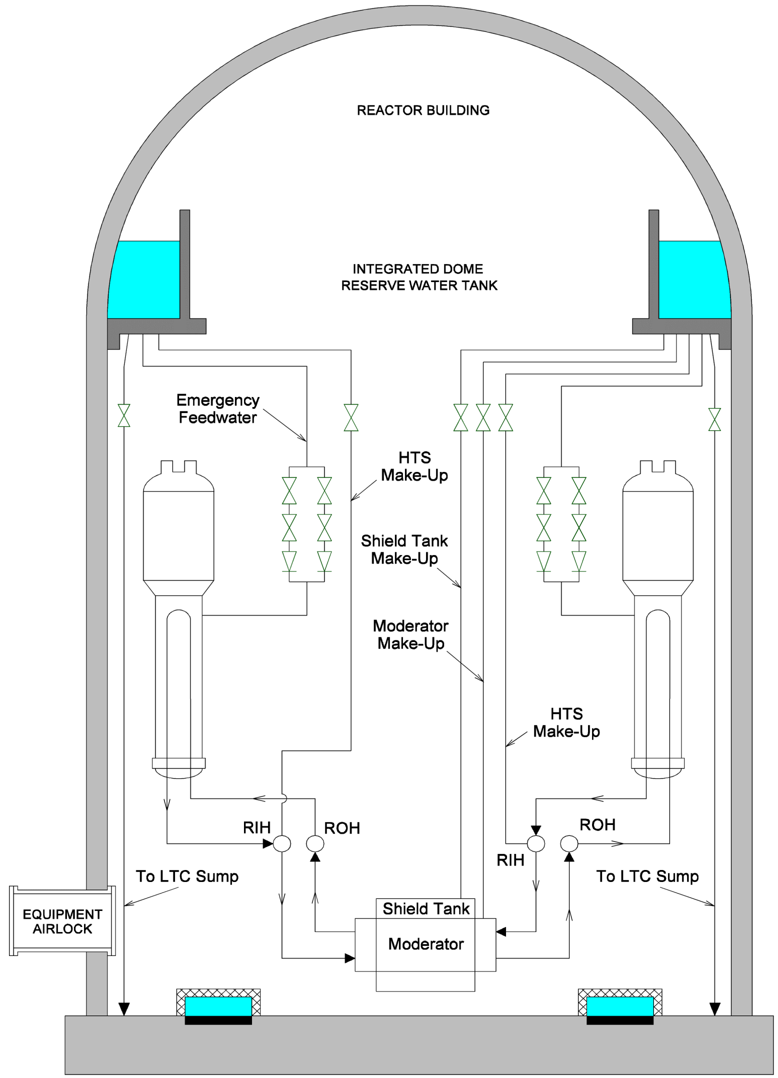
Containment



Safety Support System
Reserve Water System

- **Large water tank (~2500 m^3) at high elevation within containment**
- **Passive supply of water by gravity to**
  - Containment sumps, supply for LTC pumps
  - Emergency feedwater to steam generators
  - Emergency make-up to the RCS
  - Make-up to moderator and shield tank for enhanced mitigation of severe accidents

Reserve Water System



Severe Core Damage Accident Resistance

- Large separate volumes of water in and around the core
- Moderator backup heat sink maintains core coolability for LOCA combined with the unavailability of the emergency core cooling
- Even when moderator cooling is unavailable, the large quantities of moderator inside calandria and light water in shield tank slow down the progression of severe core damage; therefore, challenge to the containment boundary will be benign with more time for recovery actions
- *In ACR provision of passive water makeup to moderator and shield tank from the reserve water system to extend their passive thermal capacities*

Severe Core Damage Accident Phenomena

Severe Accident Progression

<u>Inside calandria</u> (In-vessel)

1. Channel heat-up and thermal hydraulics
2. Moderator heat-up and boil-off
3. Shield tank heat-up and boil-off
4. Core disassembly
5. Debris behavior

<u>Outside calandria</u> (Ex-vessel)

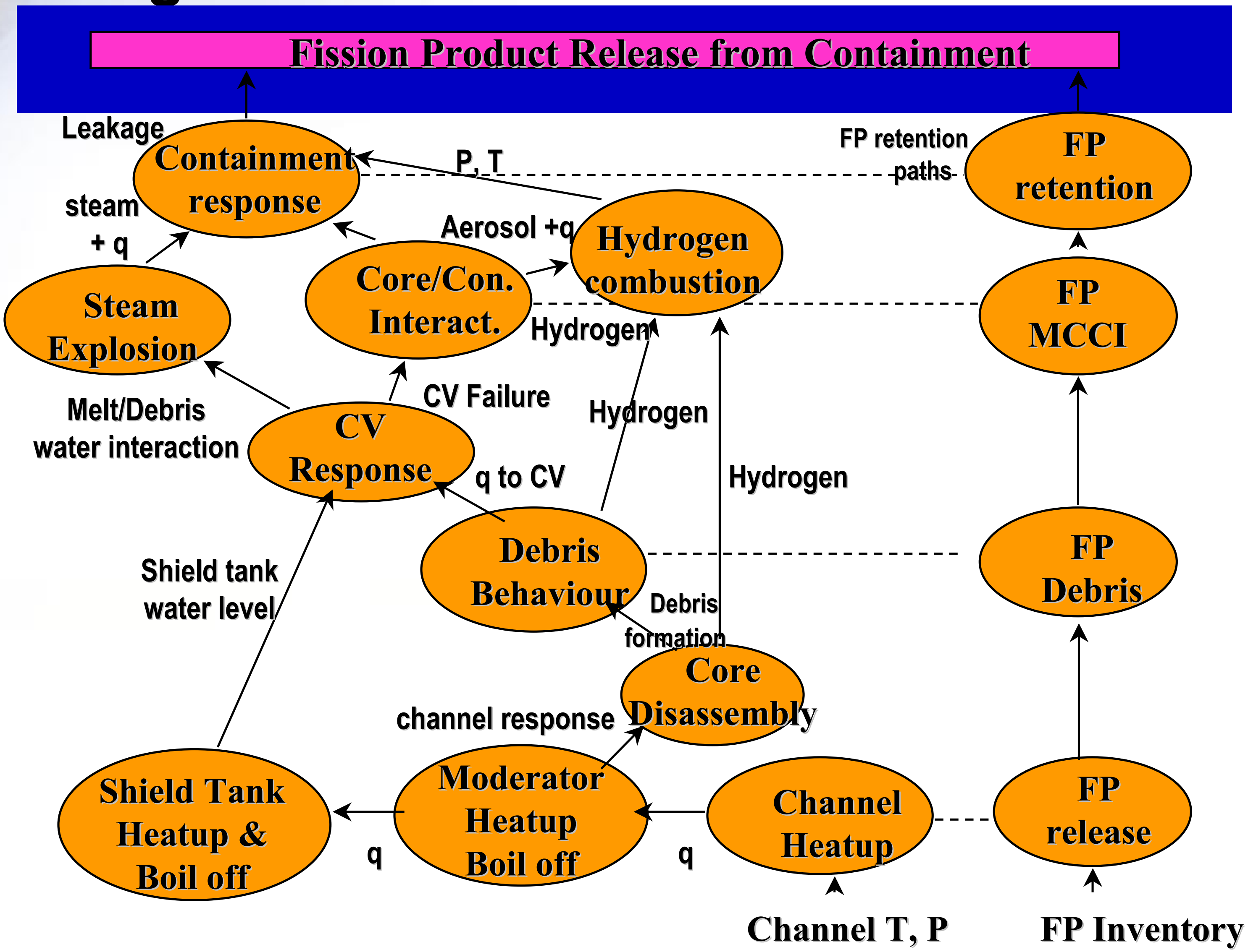
6. Core/concrete interactions (prevented by make up to shield tank in ACR)

Parallel processes*

A. Hydrogen combustion
B. Steam explosion
C. Containment thermal hydraulic response
D. Fission product release

(*) Separate system codes are available for parallel processes, but integrated codes like MAAP4 CANDU do integrated calculations

Integrated Model of Severe Accidents



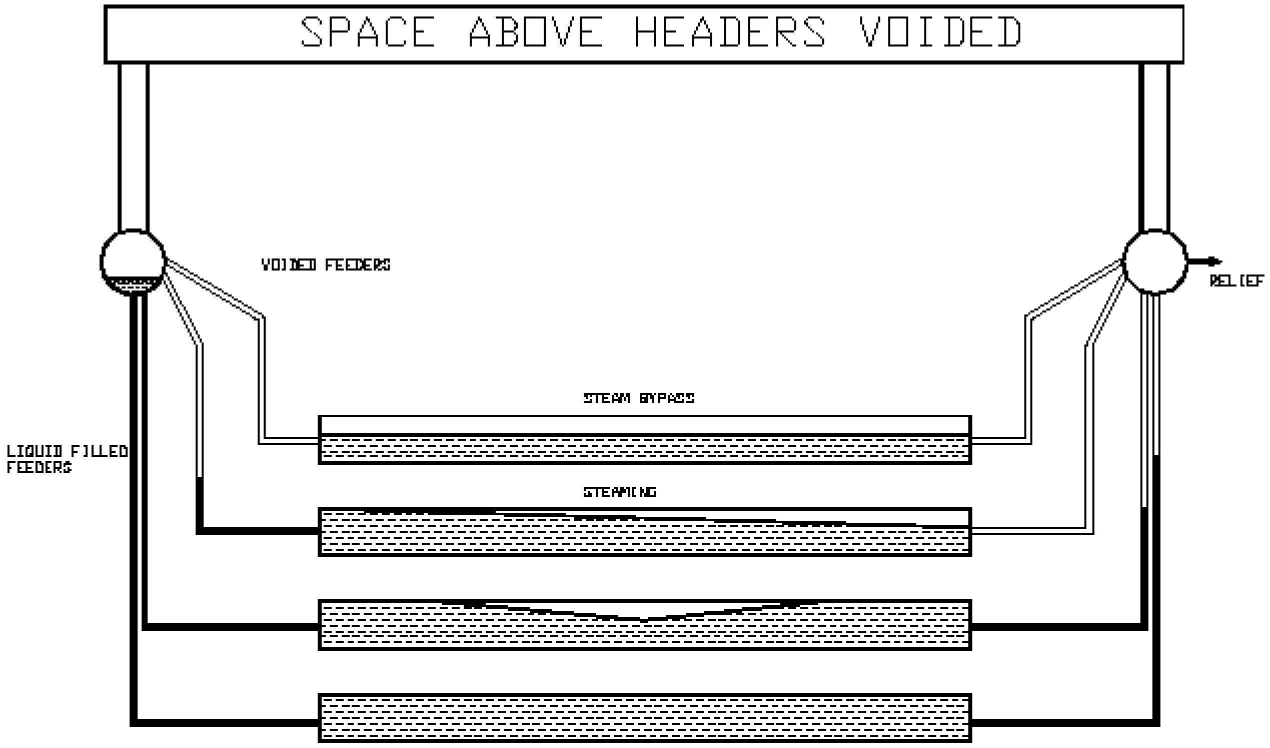
Key Generic Phenomena Specific to CANDU

- Fuel Channel Behavior
  - High Pressure Failure
  - Low Pressure Failure
- Channel Disassembly
- Calandria Vessel Behavior
- Shield Tank Behavior

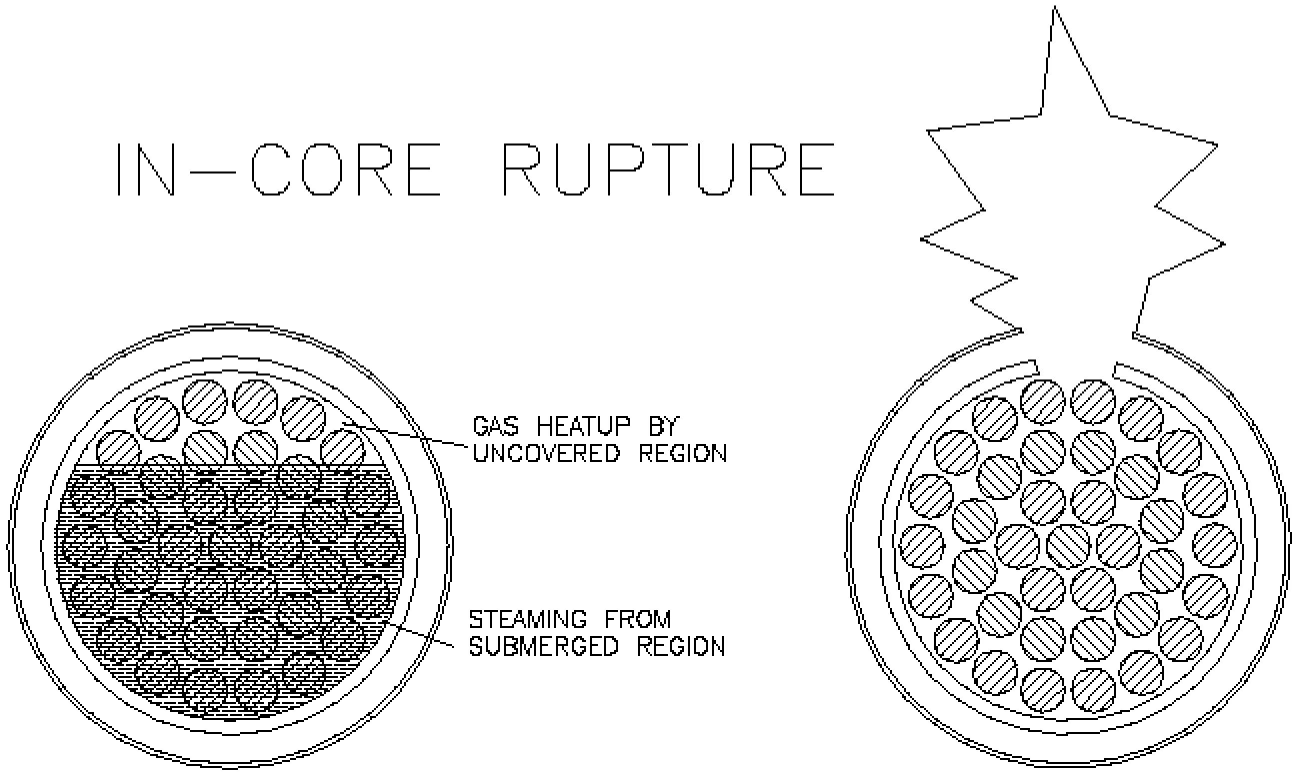
Fuel Channel Failure at High HTS Pressures

- RCS + Moderator heatsink + RWS make-up lost (ACR-700)
- Reactor shut down and RCS is intact (eg. Station Blackout)
- SG inventory become depleted, RCS pressurizes to set point
- Coolant lost through Liquid Relief Valves and deterioration of decay heat removal
- Pressure tube of hottest channel to uncover first (at high elevation with smallest liquid inventory below headers) balloons
  - non-uniform circumferential temperature distribution under high pressure causes localized strain of PT, failure of PT and subsequent failure of CT
- Rapid depressurization of RCS introducing forced flow in channels and convective heat removal from remaining channels
- Combination of low pressure and convective cooling avoids ballooning of remaining channels; therefore only very few channels are expected to fail at high pressures
- The remaining channels fail and disassemble at low pressures

Fuel Channel Failure at High Pressures



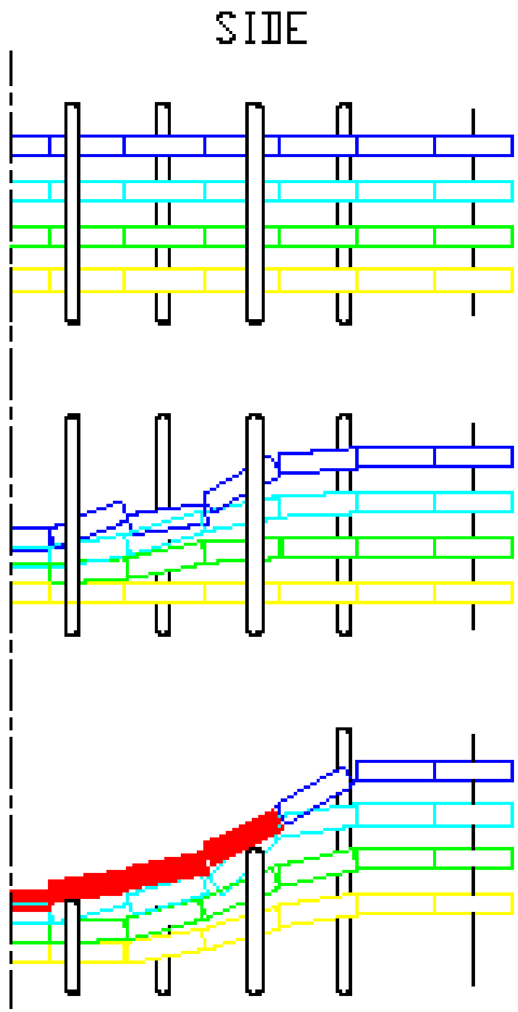
Channel Failure at High Pressures

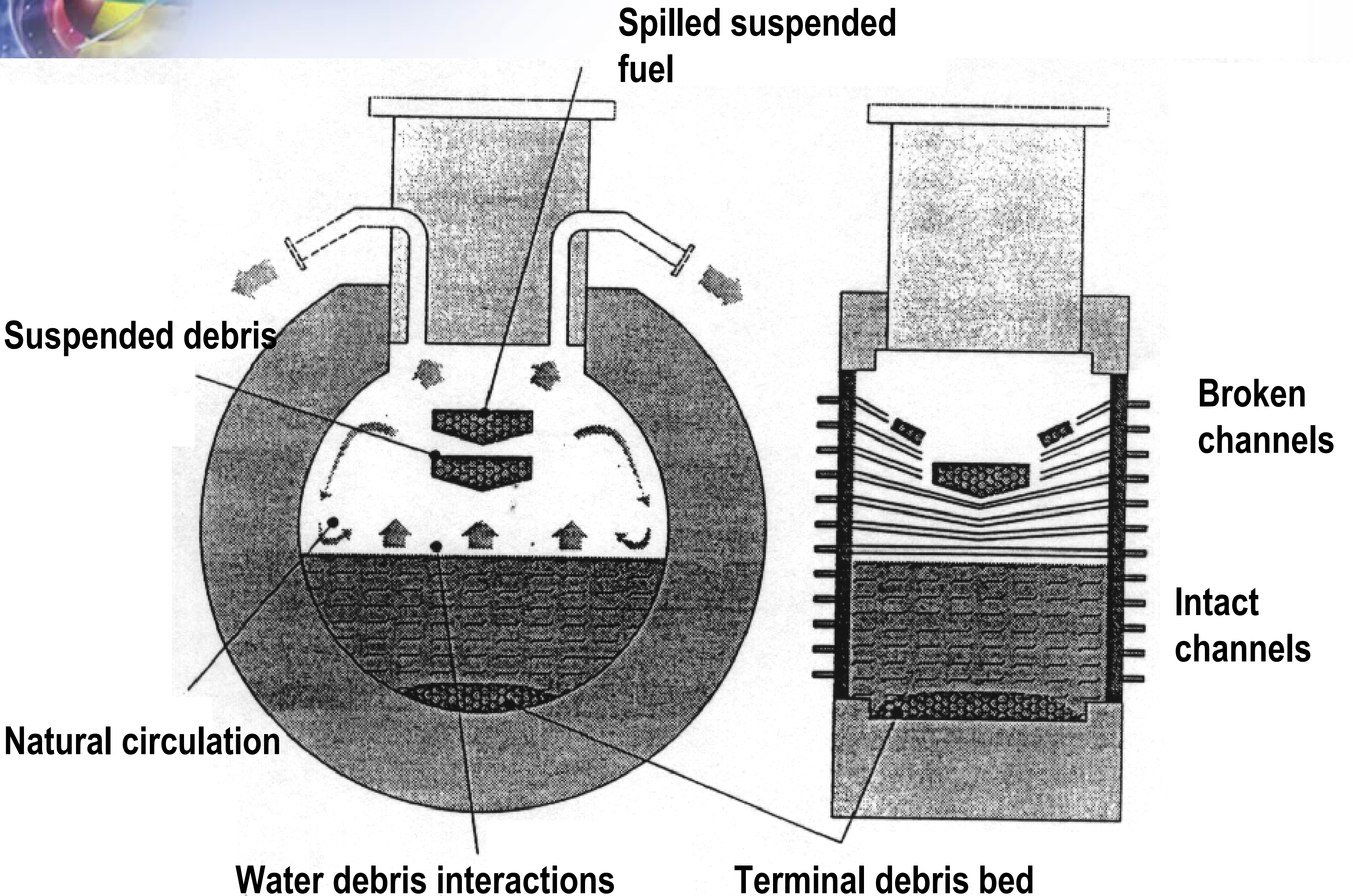


Channel Failure at Low Pressures

- When cooling of both PT interior and CT exterior are impaired simultaneously potential for channel failure at low RCS pressure since local channel temperatures rise
- Channel failure by sagging and local deformation or by melt-through of tube walls, but no fuel melting
- When failures occur in many channels, core disassembly phenomenon
- Core Disassembly:
  - Hot channels at high elevations sag, localized strain concentrations at bundle junctions, channels fail and debris are formed. Mostly solid debris supported by channels (Suspended Debris Bed), which are immersed in moderator. When load on supporting channel exceeds pull-out strength of its rolled joints the core Collapses. Terminal coarse debris bed submerged in water inside calandria vessel.

Channel Disassembly at Low Pressures




Spilled suspended fuel
Suspended debris
Broken channels
Intact channels
Natural circulation
Water debris interactions
Terminal debris bed

Calandria Vessel Failure

- Calandria vessel cooled on the outside by water in shield tank
- As long as calandria vessel wall is cooled by water in the shield tank, calandria vessel integrity can be maintained (ex-vessel cooling)
- In current generation CANDUs without passive makeup to the shield tank:
  - water in shield tank boils off as a result of heat transfer from calandria vessel
  - When water in shield tank reaches close to the bottom of the calandria vessel, creep failure of calandria vessel bottom
  - When shield tank fails, corium relocates onto the concrete floor, where it will be partially quenched by water on the floor
  - Subsequent erosion of reactor floor from corium-concrete interaction producing large amounts of hydrogen and fission products

Requirements for Integrated Modeling

- **Feedback between interfacing systems and phenomena**
  - Systems
    - Core
    - Reactor Structures
    - Reactor Cooling System
    - Containment
    - Safety Systems
    - Process Systems
  - Phenomena
    - Release and Transfer of Thermal Energy
    - Release and Transfer of Fission Products
    - Thermo-mechanical Deformations
    - Thermo-chemical Reactions
    - Chemical Reactions
  - Operator Actions

MAAP4 CANDU Code

(Analytical Tool to Calculate Accident Progression)

MAAP4 CANDU Code: Background

- MAAP (Modular Accident Analysis Program) is an integrated computer code designed for Severe Accident Analysis in nuclear plants, used by more than 40 international Utilities
- MAAP4 CANDU (M4C) developed by Fauske & Associates (FAI), based on MAAP widely used by PWR/BWR.
- M4C contains CANDU core heat-up module developed by Ontario Power Generation (OPG)
- AECL/OPG together with FAI developed the models for CANDU 6 stations. Current program to develop ACR 700 models together with FAI.

MAAP4 CANDU

- MAAP4 CANDU has models of horizontal CANDU-type fuel channels and CANDU-specific systems such as Calandria Vessel, Reactor Vault, Reactor Cooling System, Containment Systems such as Dousing, LACS, etc.
- Can assess influence of Severe Accident Management strategies to mitigate and recover from an accident state

MAAP4 CANDU

- **MAAP4 CANDU calculates severe accident progression starting from normal operating conditions for a set of plant system faults and initiating events leading to:**
  - **Reactor Cooling System (or PHTS) inventory blow-down or/and boil-off**
  - **Core heatup and melting**
  - **Fuel channel failure**
  - **Core disassembly**
  - **Calandria vessel failure**
  - **Shield tank / Reactor vault failure**
  - **Containment failure**

MAAP4 CANDU CAPABILITIES

Physical Processes Modeled in M4C:

- Thermal hydraulics processes in RCS, calandria vessel, shield vault/shield tank, end-shield, containment compartments
- Core heat-up, melting and disassembly
- Zr oxidation by steam and hydrogen generation
- Material creep and possible rupture of PHTS components, calandria vessel and shield tank walls
- Ignition of combustible gases
- Energetic corium-coolant interactions
- Molten corium-concrete interaction
- Fission product release, transport and deposition

Activities using MAAP4 CANDU

- Completed:
  - Assembled parameter file for a generic CANDU 6 plant
  - Produced preliminary results for a generic CANDU 6 plant for a Station Blackout and a Large LOCA Scenario
- Current Program:
  - Develop ACR 700 plant models
  - Assemble ACR 700 parameter file

Generic CANDU 6 Analysis Results using MAAP4 CANDU

***) Sample results are presented here to demonstrate CANDU-specific accident response and MAAP4 CANDU capability for CANDU applications. Shield tank failure is not expected in ACR due to passive make up from the Reserve Water System**

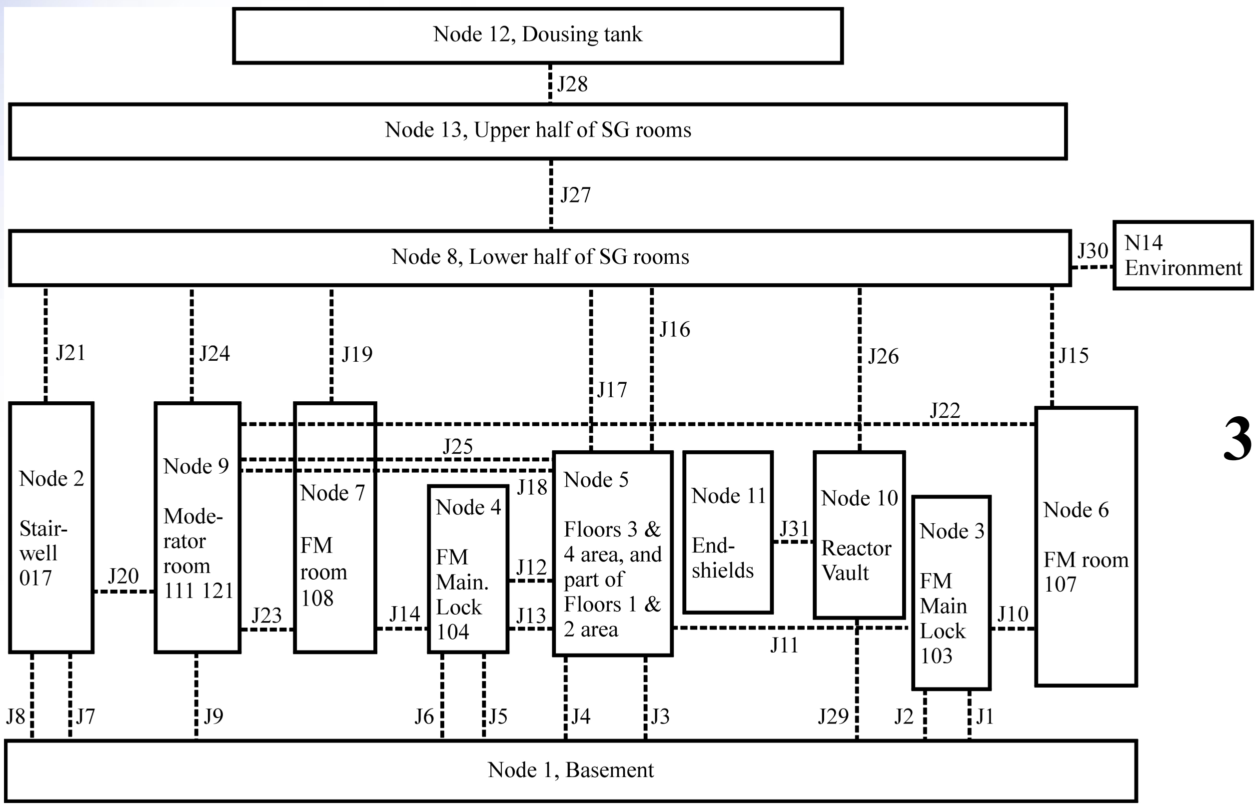
Severe Core Damage accident Progression depends on the initiating event and the unavailability of safety systems.

Severe Core Damage Accident progression is slow, allowing sufficient time for operator intervention to arrest accident progression.

Nodalization of CANDU 6 Station

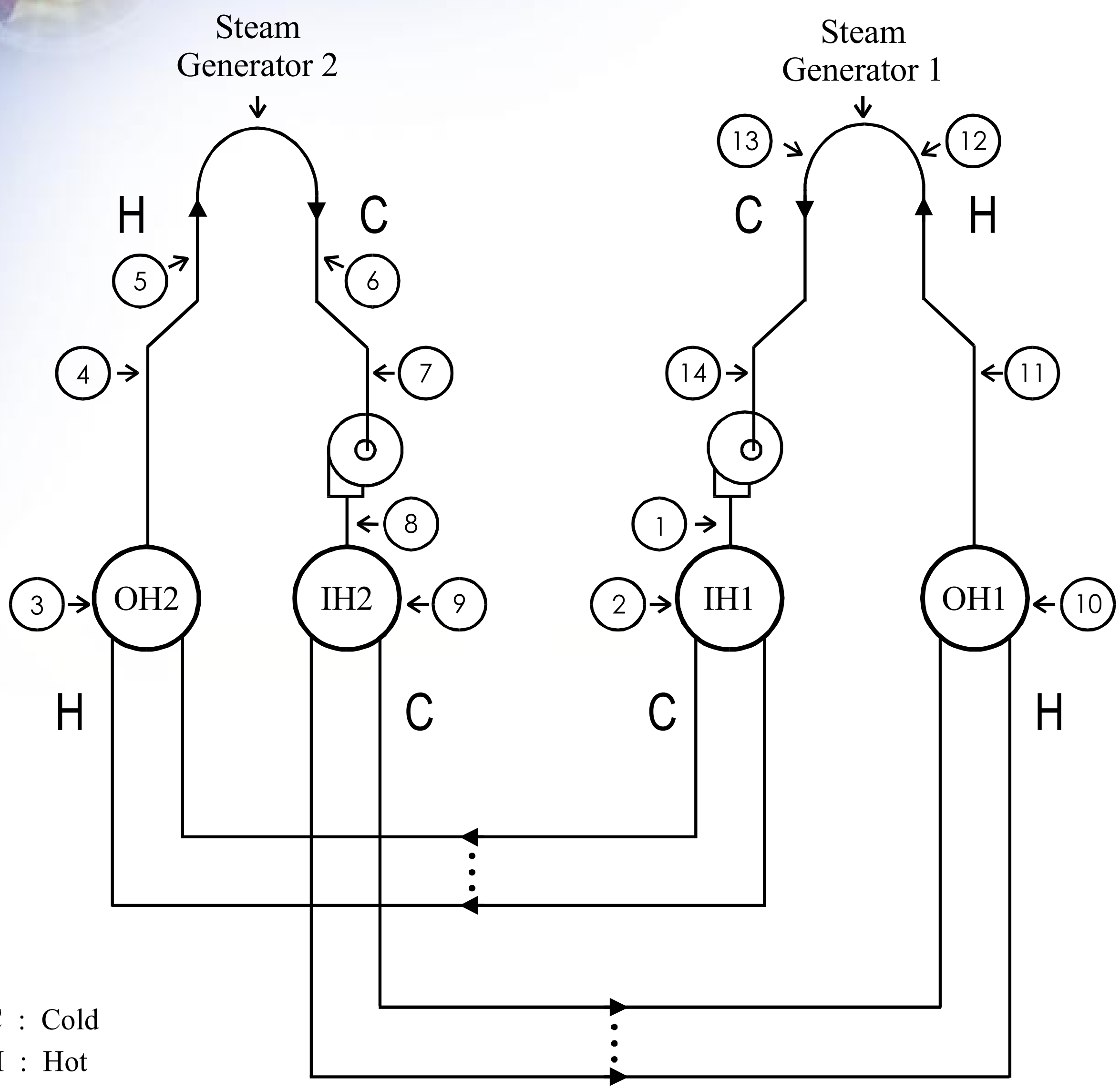
- **Generalized Containment Model**
  - Compartments represented by 13 nodes connected by 31 flow junctions
  - Containment walls modeled as 90 "heat sinks"
- **PHTS**
  - Two symmetric loops, each following "figure of 8"; 14 nodes in each loop: ROH, RIH, SG inlet piping, etc.
- **Core**
  - 380 fuel channels arranged in 22 rows and 22 columns, represented by 18 characteristic channels/loop, positioned in 6 vertical nodes
  - 3 power groups of channels in each vertical core node
  - 12 bundles represented by 12 axial nodes
  - 37 fuel elements, pressure and calandria tubes modeled as 9 concentric rings
- **Steam Generator**
  - Primary side modeled as 2 nodes ("hot" and "cold"); secondary side - as 1 node

Nodalization of CANDU 6 Containment



13 Nodes
31 Flow Junctions
90 Heat sinks

Nodalization for CANDU 6 RCS



14 Nodes/Loop

C : Cold
H : Hot

Nodalization of CANDU 6 Core

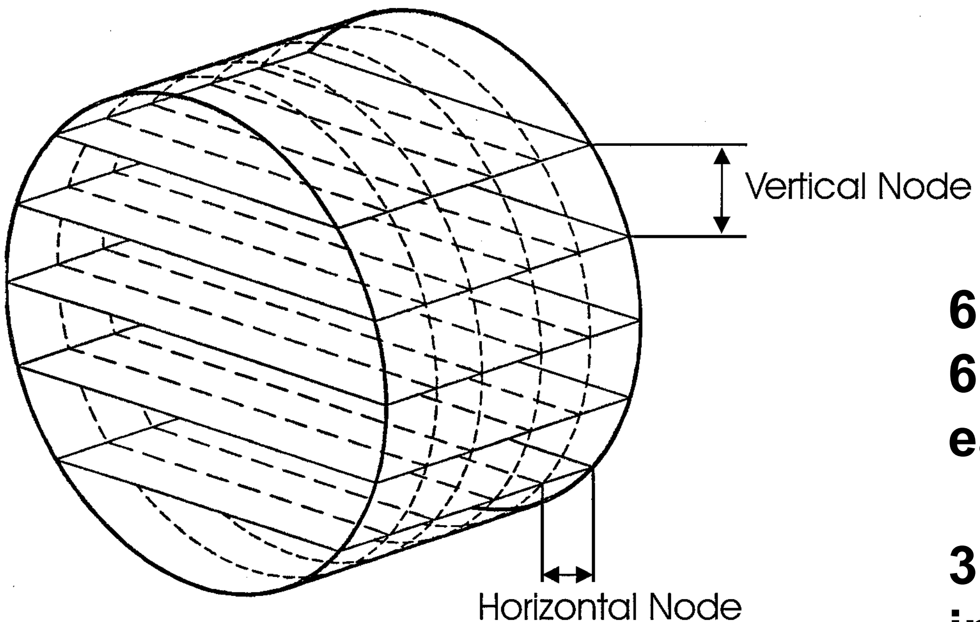
Channel Power Distribution

The following Time-Average Channel Power Distribution Map is for CANDU6

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	Number of Channel H	M	L
A									3187	3299	3378	3378	3298	3187									0	1	2
B						2862	3421	3919	4206	4394	4437	4437	4394	4205	3917	3419	2860						0	5	1
C					3194	3808	4437	4908	5215	5363	5333	5333	5362	5213	4906	4434	3805	3191					0	6	1
D				3319	4014	4716	5316	5727	5971	6059	5970	5969	6058	5969	5725	5313	4713	4010	3314				3	4	1
E			3188	4012	4748	5390	5901	6216	6356	6387	6278	6278	6386	6355	6213	5899	5386	4744	4006	3179			3	5	1
F			3868	4689	5320	5857	6241	6437	6369	6350	6285	6284	6350	6367	6435	6239	5853	5315	4683	3859			4	4	1
G		3497	4410	5224	5693	6111	6362	6479	6415	6397	6400	6400	6396	6414	6478	6360	6108	5689	5219	4404	3491		5	4	1
H		3983	4925	5665	6002	6303	6469	6536	6460	6441	6456	6456	6440	6459	6535	6467	6301	5999	5662	4921	3984		6	4	0
J	3213	4317	5302	5977	6192	6329	6447	6485	6450	6425	6414	6414	6425	6449	6484	6447	6328	6191	5977	5303	4322	3207	7	3	1
K	3434	4611	5584	6207	6301	6362	6463	6475	6432	6383	6320	6320	6383	6432	6475	6463	6362	6301	6209	5587	4620	3432	8	2	1
L	3577	4763	5739	6365	6493	6515	6526	6495	6427	6346	6228	6228	6346	6427	6495	6527	6516	6495	6368	5744	4775	3578	8	2	1
M	3564	4768	5762	6413	6572	6593	6575	6521	6440	6350	6221	6220	6350	6441	6522	6576	6595	6574	6417	5768	4782	3568	8	2	1
N	3394	4610	5627	6323	6549	6608	6592	6542	6467	6395	6295	6296	6395	6468	6543	6593	6609	6552	6327	5364	4624	3399	8	2	1
O	3184	4334	5365	6113	6467	6596	6585	6557	6495	6458	6444	6444	6549	6495	6558	6586	6598	6469	6116	5369	4345	3186	7	3	1
P		3975	4937	5713	6100	6409	6539	6584	6499	6485	6513	6514	6485	6500	6584	6540	6410	6102	5715	4939	3981		6	4	0
Q		3470	4368	5173	5639	6070	6343	6482	6439	6447	6481	6480	6447	6439	6483	6343	6071	5640	5173	4367	3471		5	4	1
R			3762	4534	5096	5656	6129	6390	6372	6402	6392	6393	6402	6273	6390	6130	5656	5096	4534	3761			4	4	1
S			3045	3818	4479	5156	5784	6182	6385	6459	6372	6372	6459	6385	6182	5784	5156	4478	3817	3044			3	5	1
T				3073	3756	4485	5155	5640	5944	6076	6003	6003	6076	5944	5640	5155	4485	3756	3072				3	4	1
U					3073	3756	4485	5155	5640	5403	5384	5384	5403	5206	4832	4288	3596	2926					0	6	1
V						2548	3187	3761	4132	4384	4463	4463	4384	4132	3761	3187	2548						0	5	1
W									2961	3174	3300	3300	3174	2961									0	1	2

- 380 channels represented by 36 characteristic channels
- 3 regions represent High, Medium & Low Power channels

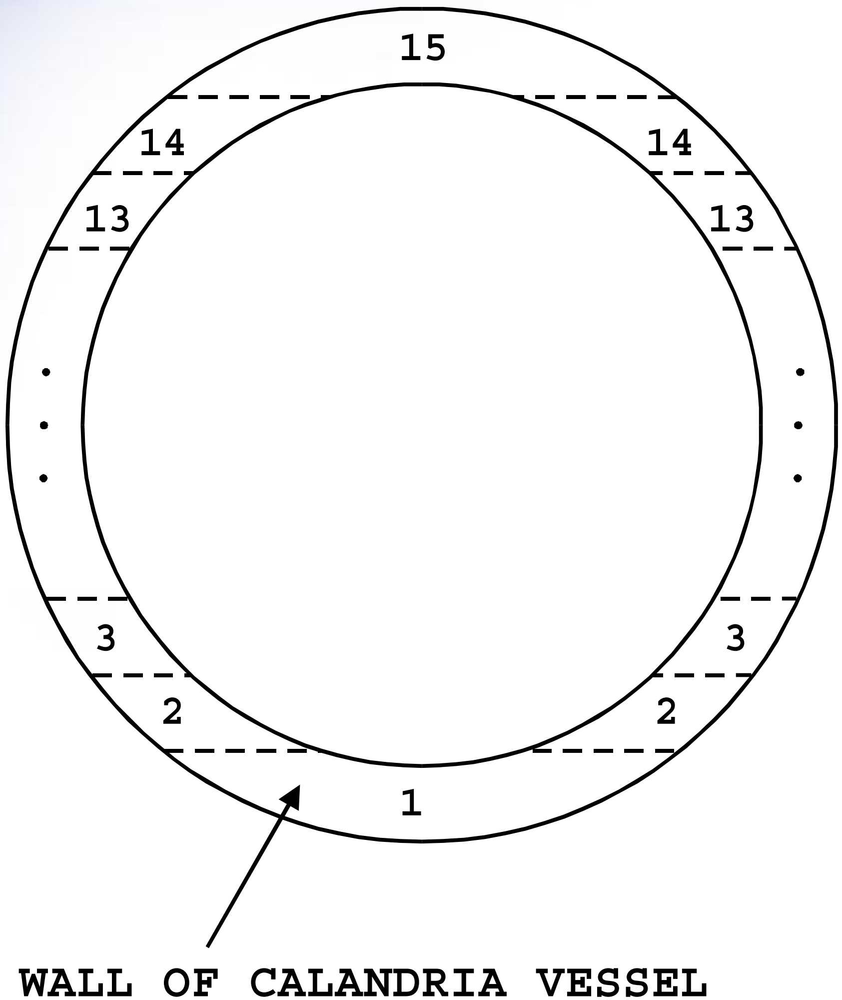
Nodalization of CANDU 6 Core



6 vertical core nodes with 6 characteristic channels in each node

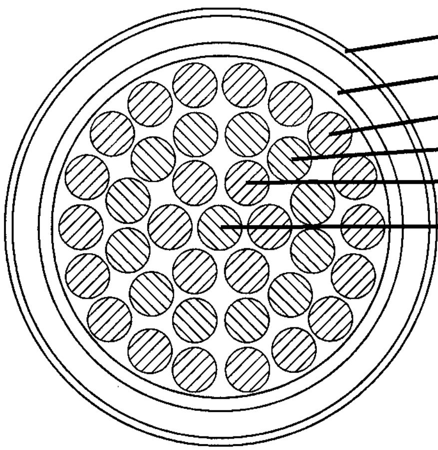
3 power groups of channels in each vertical core node

Nodalization of Calandria Vessel



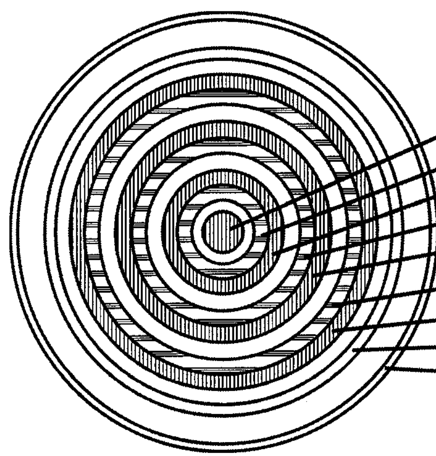
15 horizontal nodes

Nodalization of CANDU 6 Fuel Channel



CANDU 6 Channel with 37 Element Fuel Bundle, Pressure Tube and Calandria Tube

CANDU 6 Channel Nodalization Scheme 9 rings, 12 axial nodes



Data for Generic CANDU 6 Parameter File

- **C6 Plant Data collected from:**
  - Safety Analysis Reports
  - Design Manuals
  - Safety Analysis Data List
  - Other code (e.g., CATHENA, PRESCON 2) Input Listings
  - Engineering Drawings
  - Technical Specification Manuals
  - COG* Reports

*) COG = CANDU Owners Group

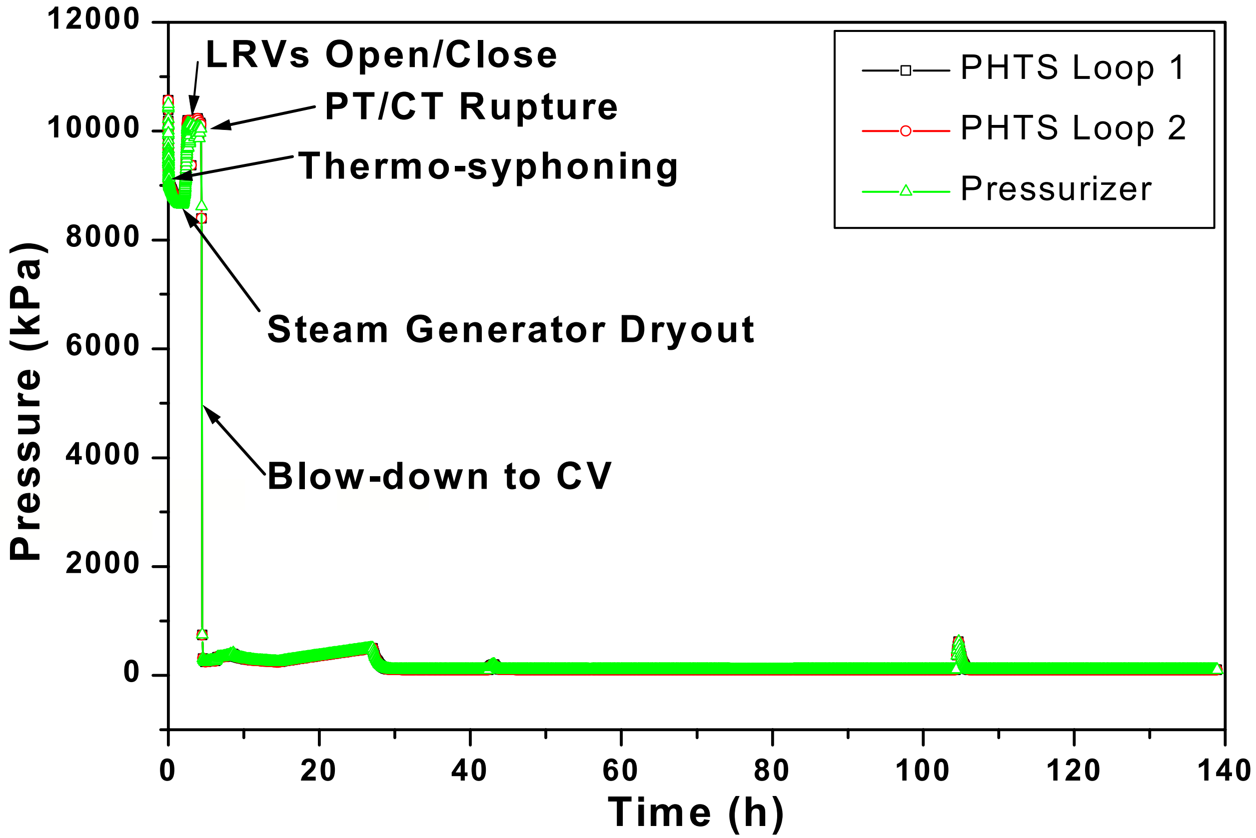
Generic CANDU 6 SBO Analysis Assumptions

Unlikely and most severe scenario

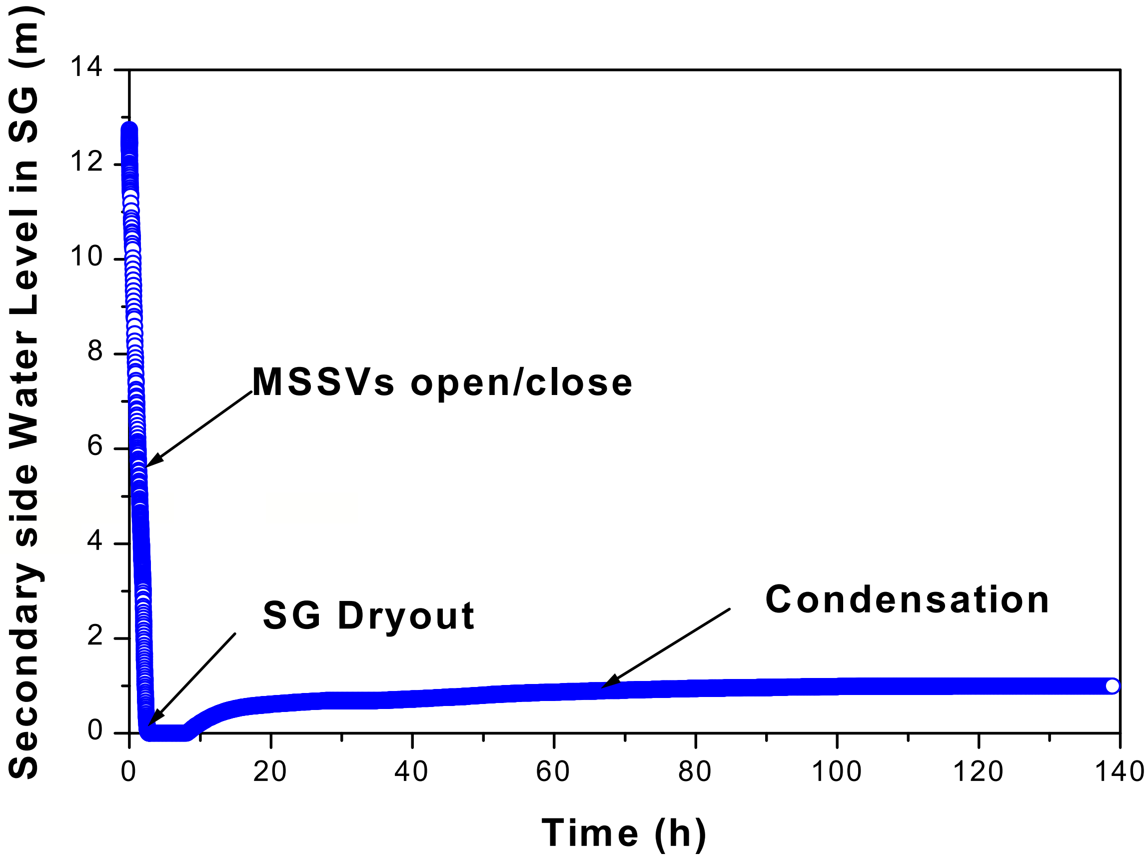
- AC power and all onsite standby/emergency power unavailable
- Reactor shutdown after accident initiation
- Moderator-, Shield-, Shutdown cooling unavailable
- Main and Auxiliary Feed water unavailable
- ECCS (high, medium and low pressure) unavailable
- Dousing and Crash cool-down not credited
- LACS not available
- All Operator Interventions not credited

Generic CANDU 6 SBO Analysis Sample Results

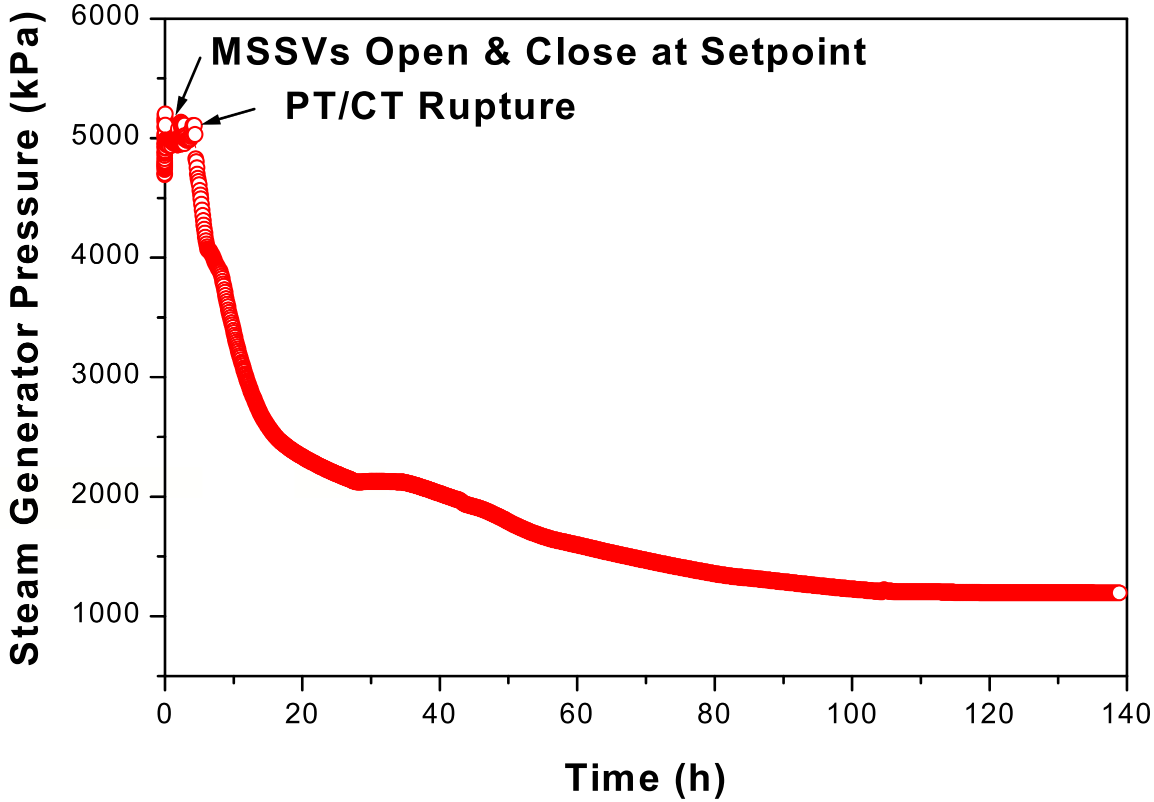
Pressure in RCS (PHTS) Loops and Pressurizer (Generic CANDU 6 SBO)



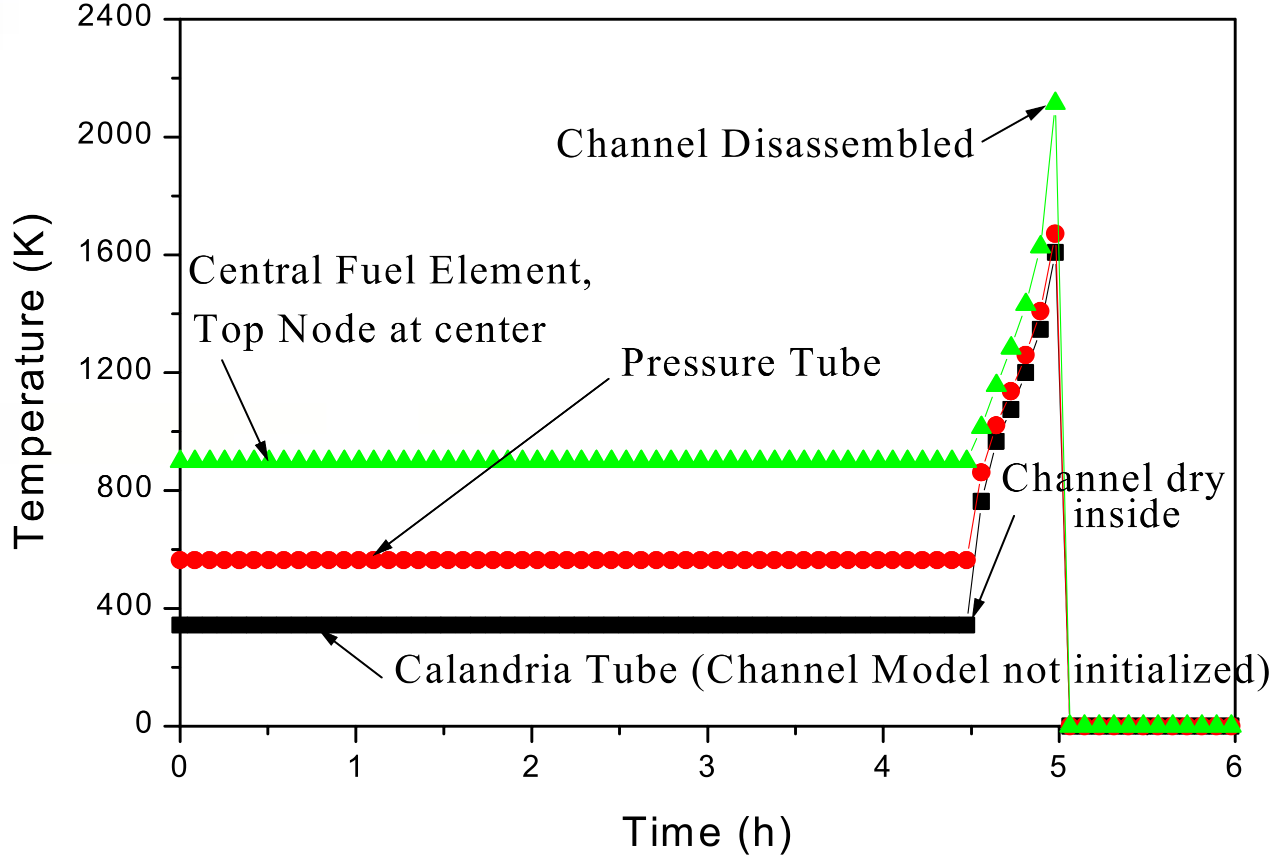
Steam Generator Water Level (Generic CANDU 6 SBO)



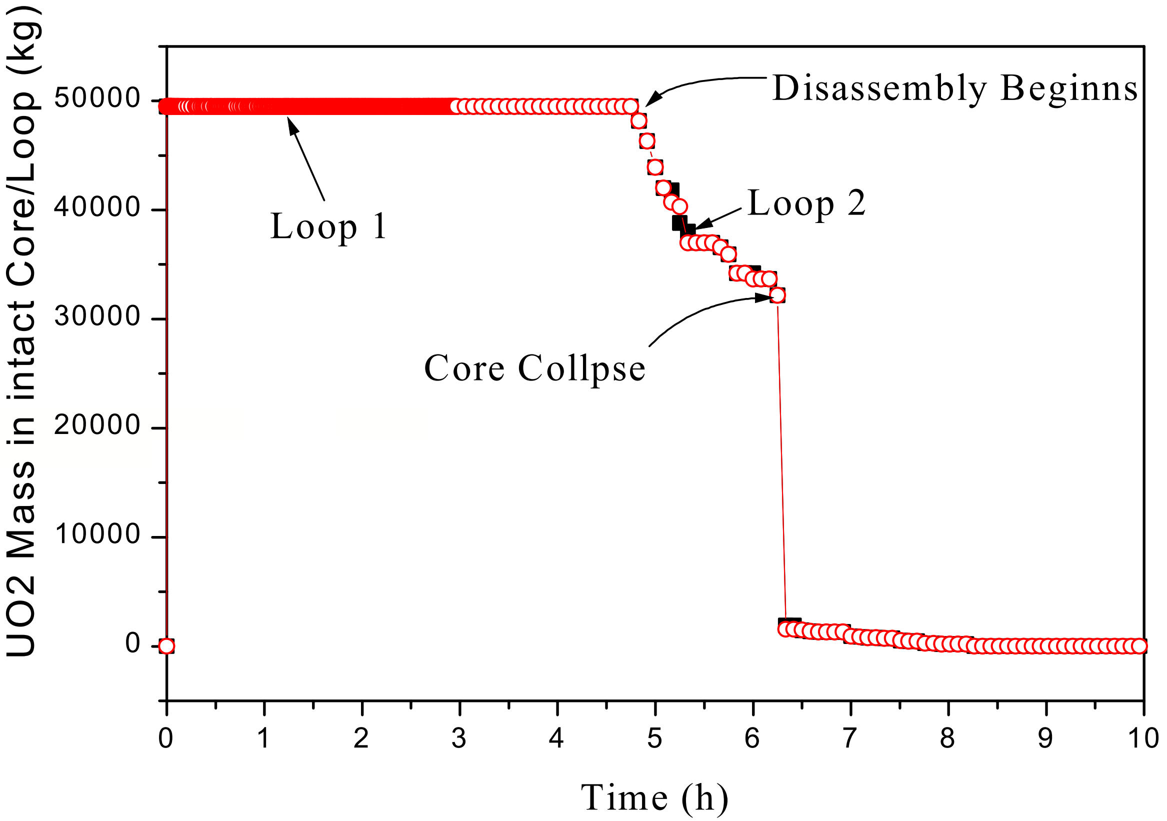
Steam Generator Pressure (Generic CANDU 6 SBO)



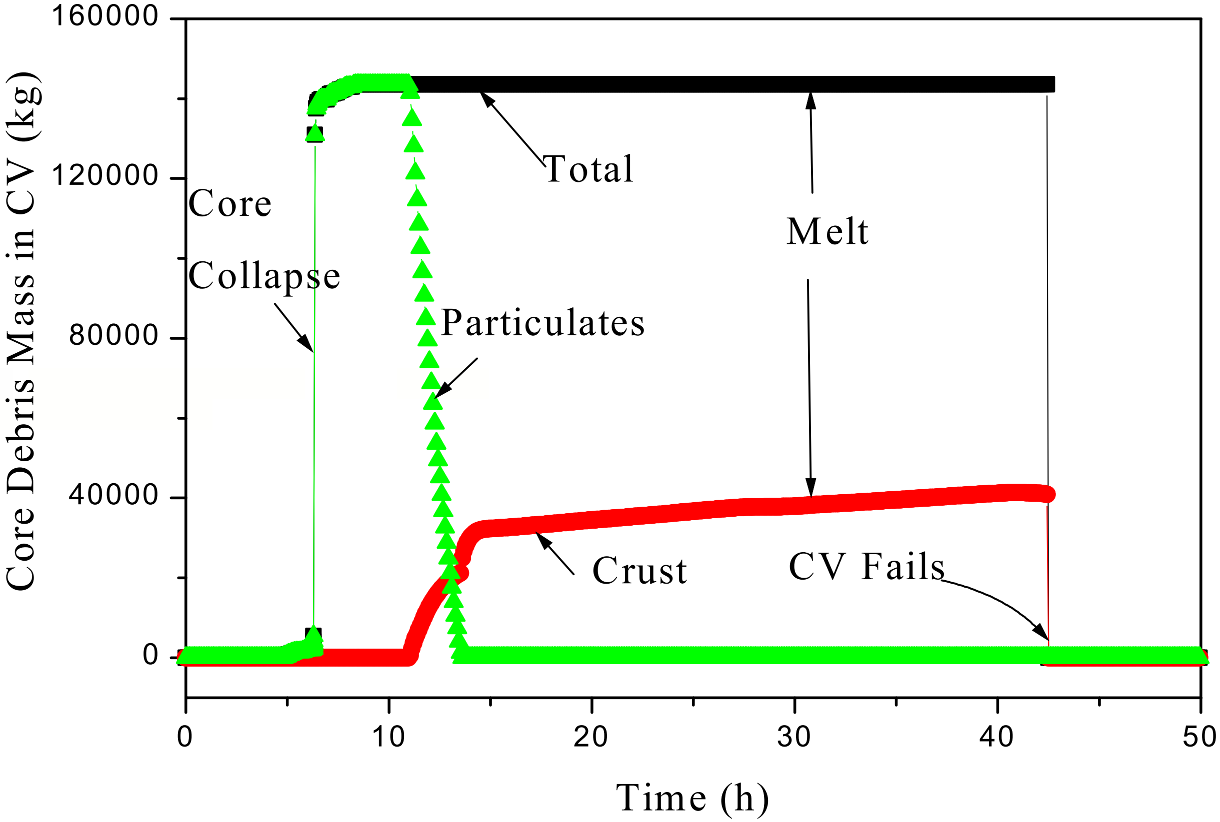
PT, CT and Fuel Temperature of Top Node, Mid-Channel (Generic CANDU 6 SBO)



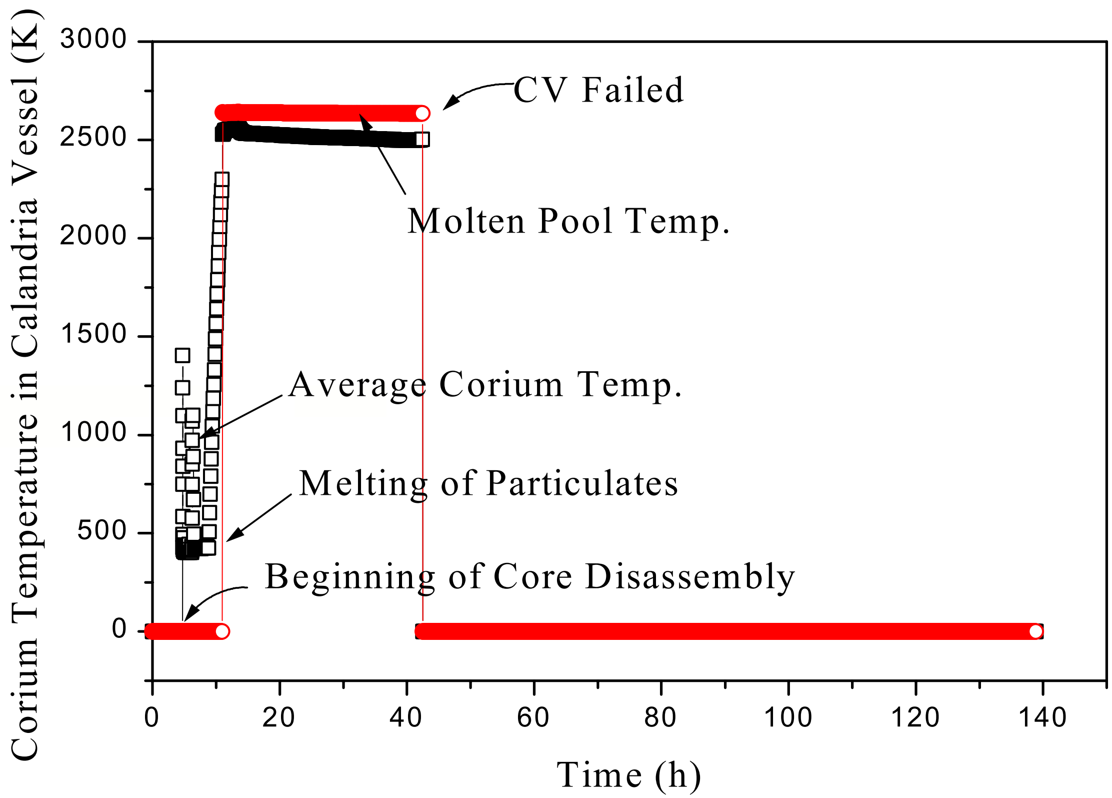
UO2 Mass/Loop
(Generic CANDU 6 SBO)



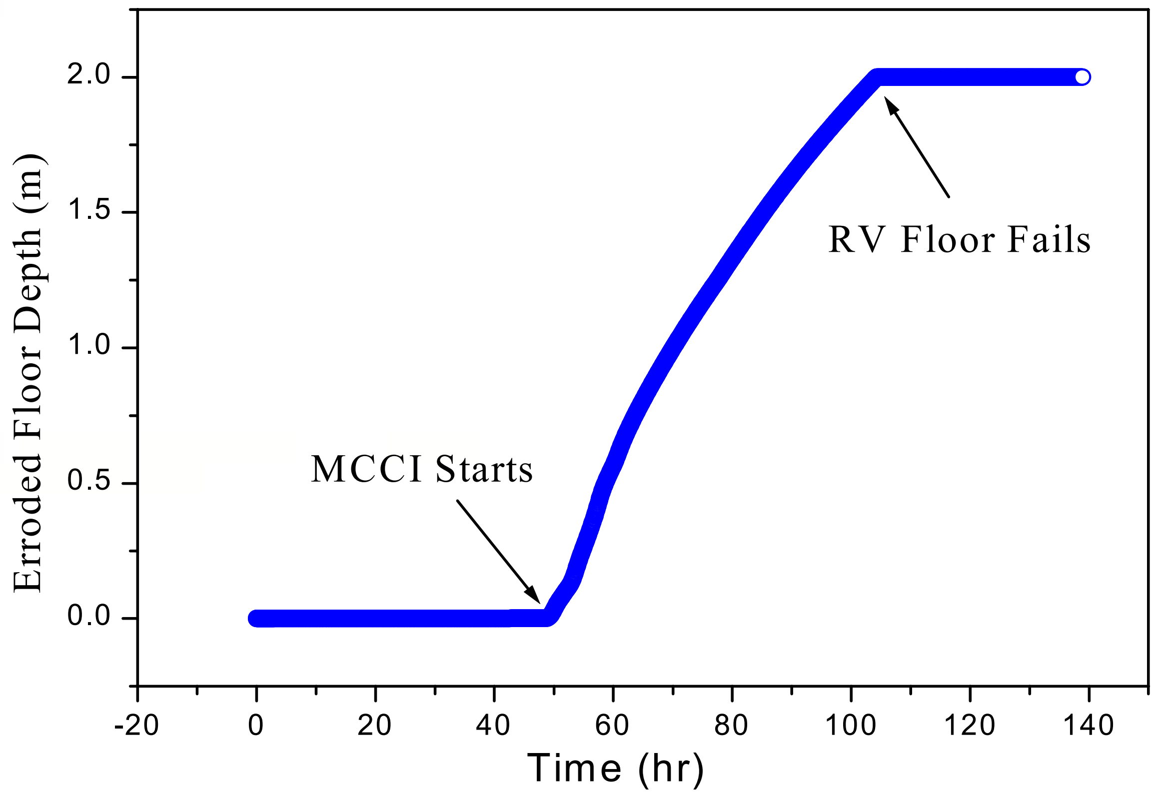
Corium Mass in Calandria Vessel (Generic CANDU 6 SBO)



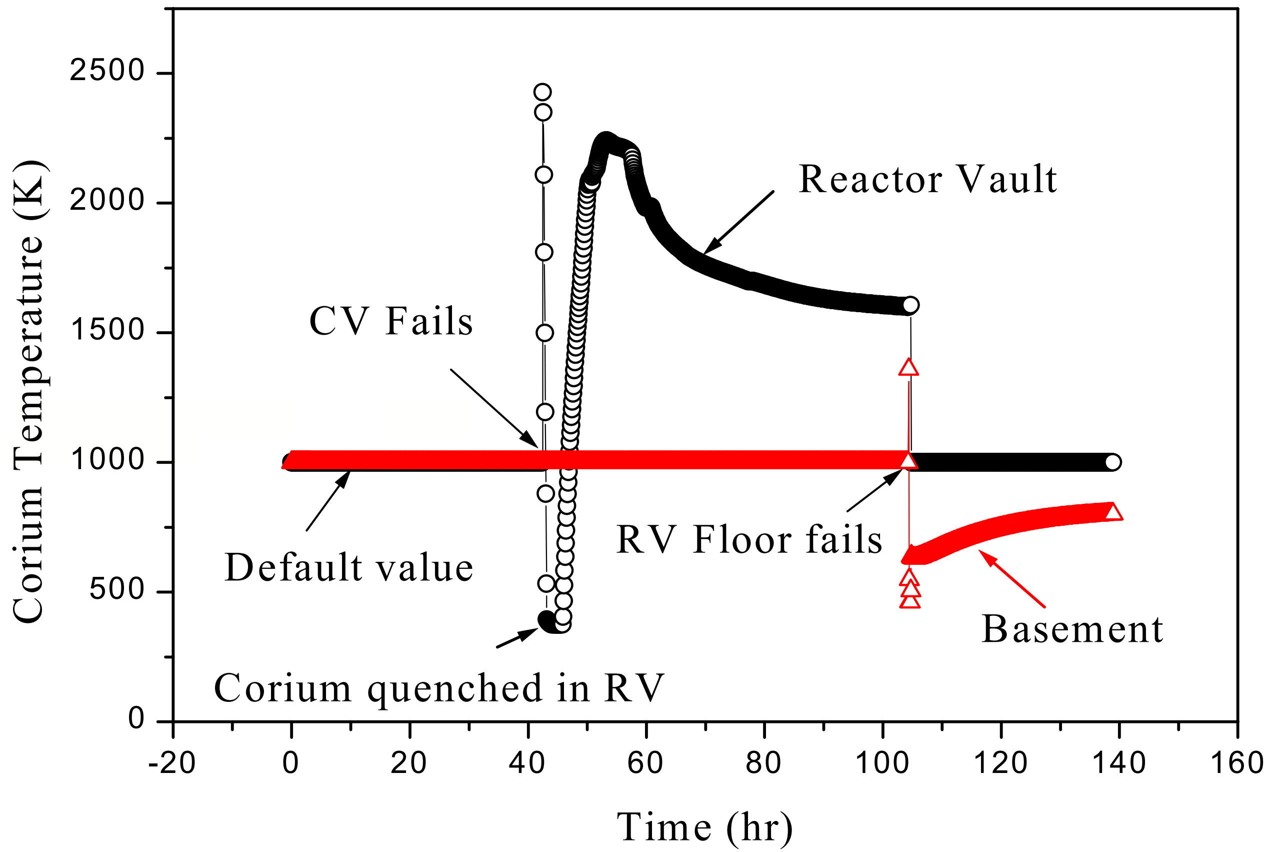
Corium Temperature in Calandria Vessel (Generic CANDU 6 SBO)



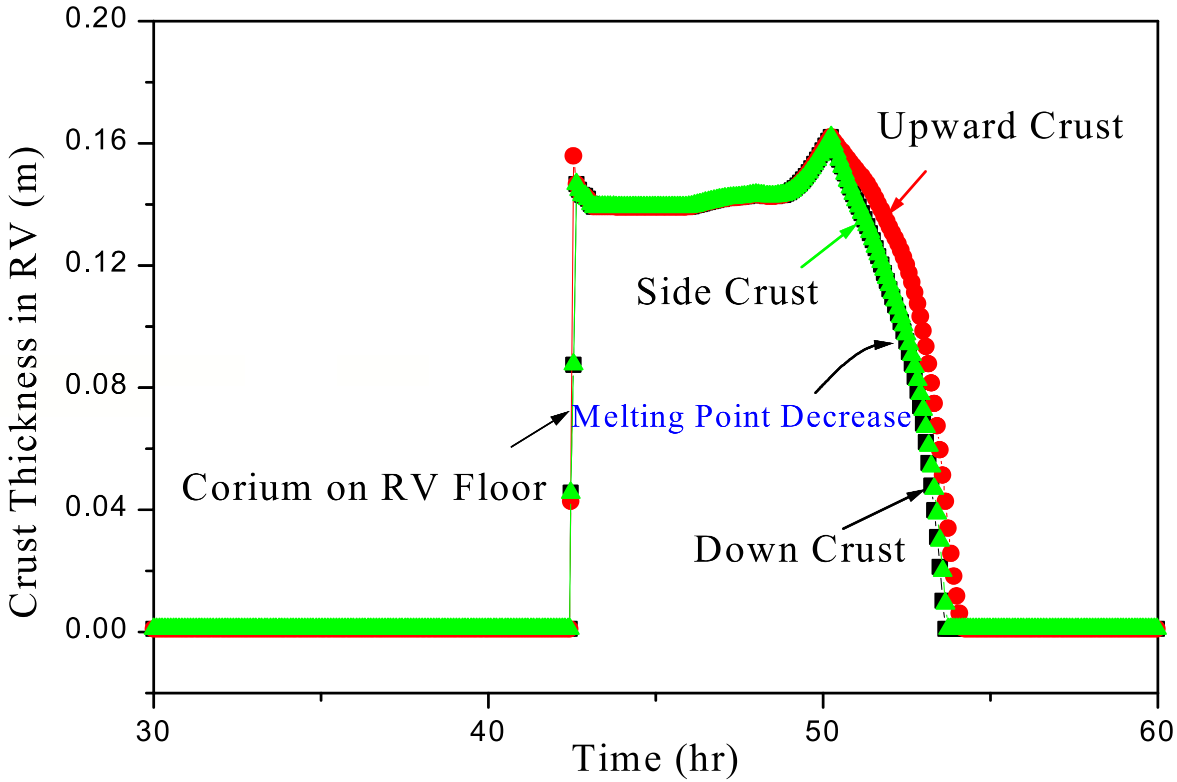
Eroded Concrete Depth in Reactor Vault (Generic CANDU 6 SBO)



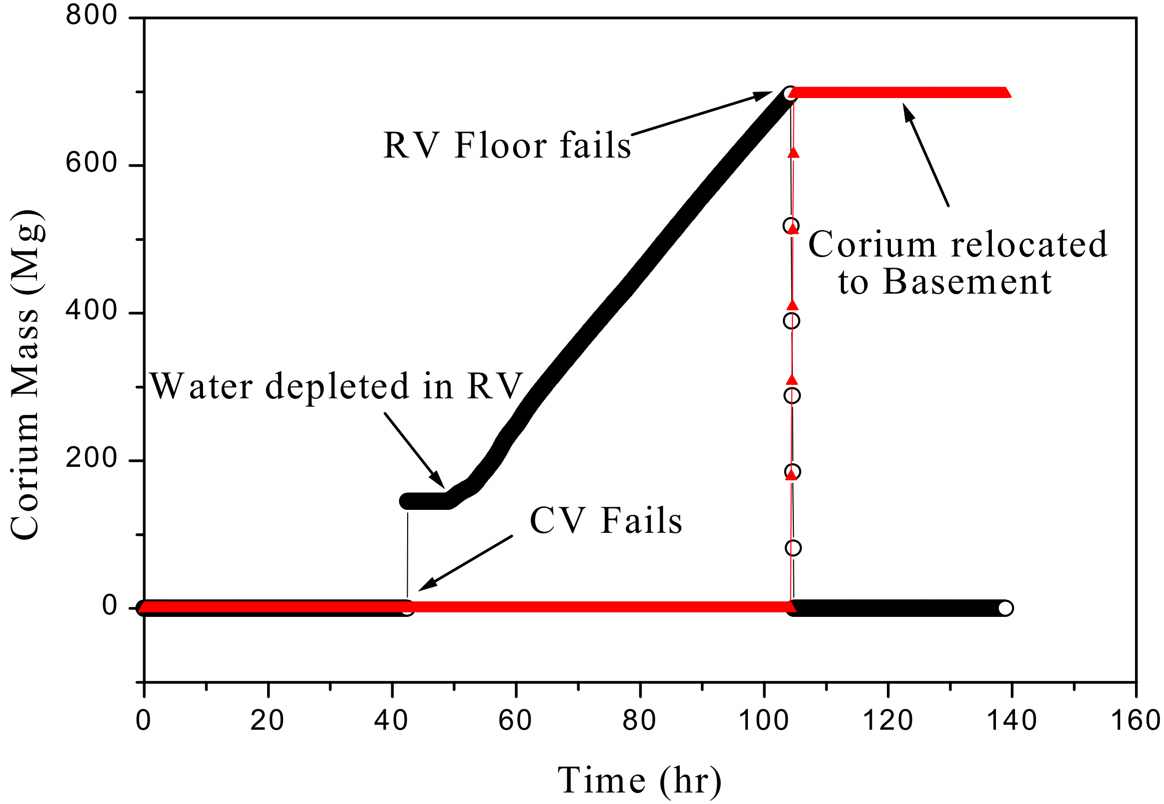
Corium Temperature in Reactor Vault and Basement (Generic CANDU 6 SBO)



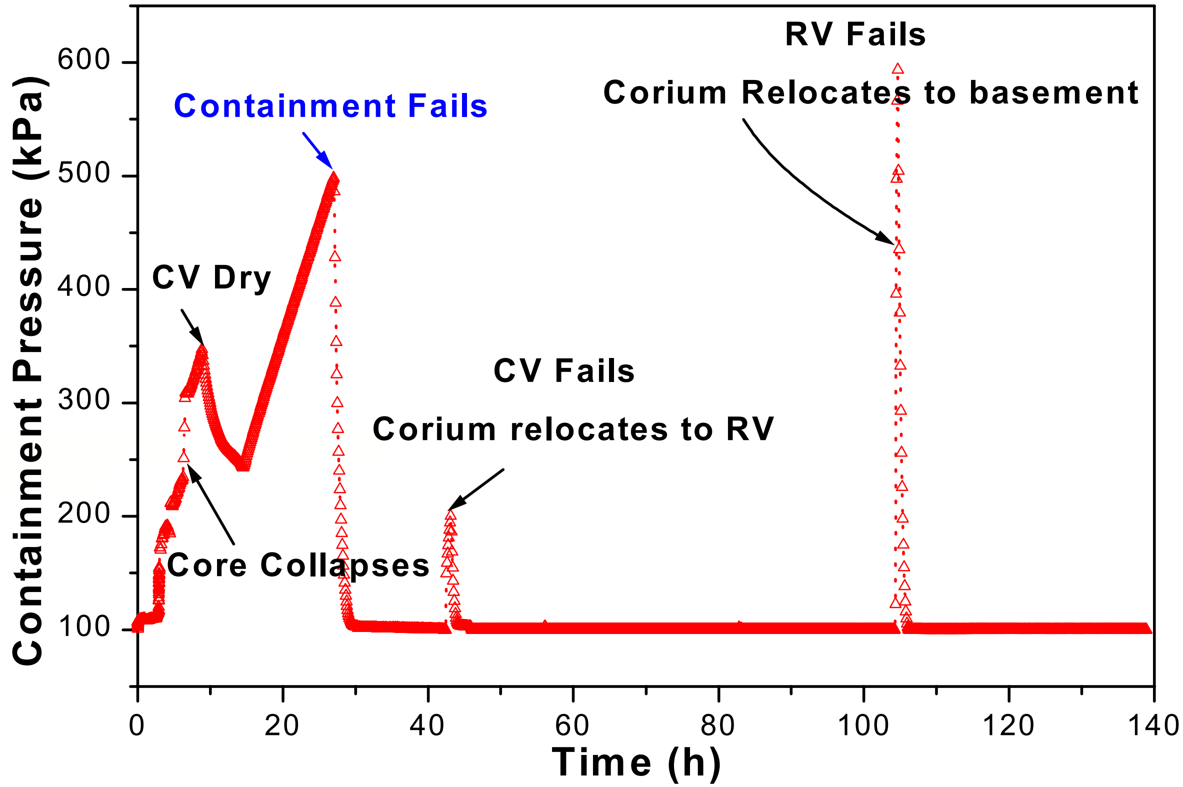
Corium Crust Thickness
in Reactor Vault (Generic CANDU 6 SBO)



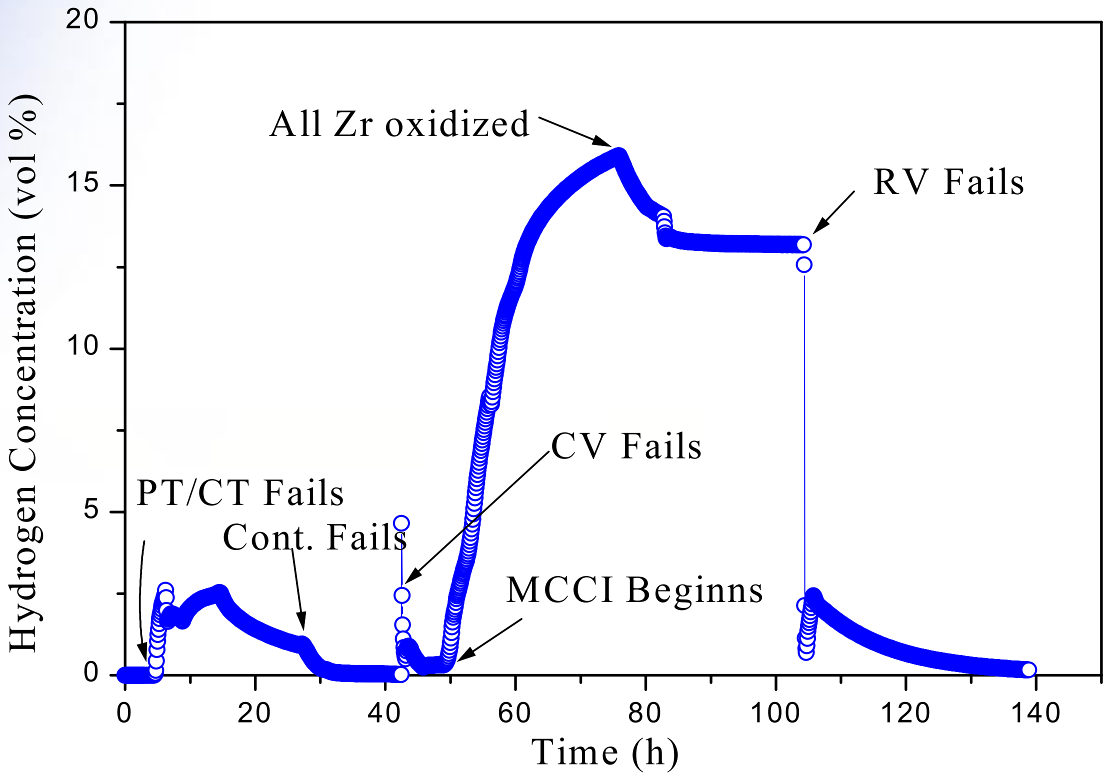
Corium Mass in Reactor Vault and Basement (Generic CANDU 6 SBO)



Containment Pressure (Generic CANDU 6 SBO)

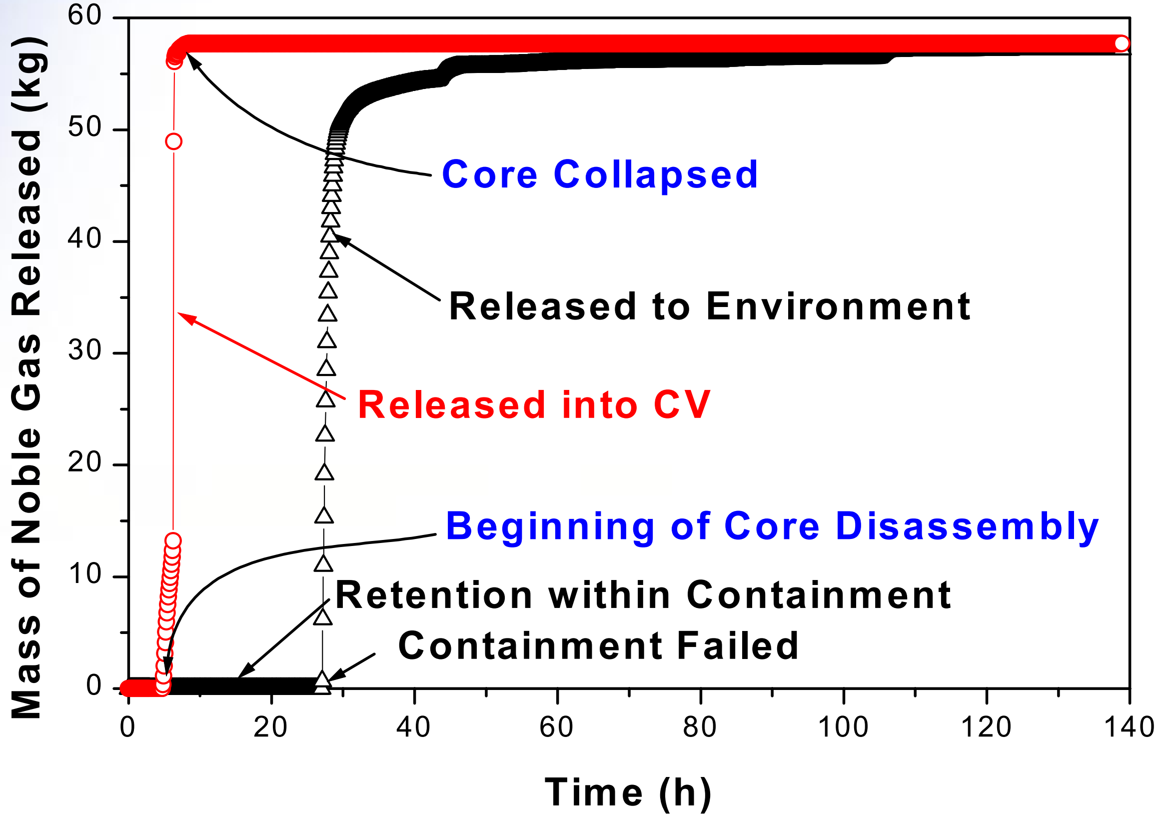


Hydrogen in Containment (Generic CANDU 6 SBO)



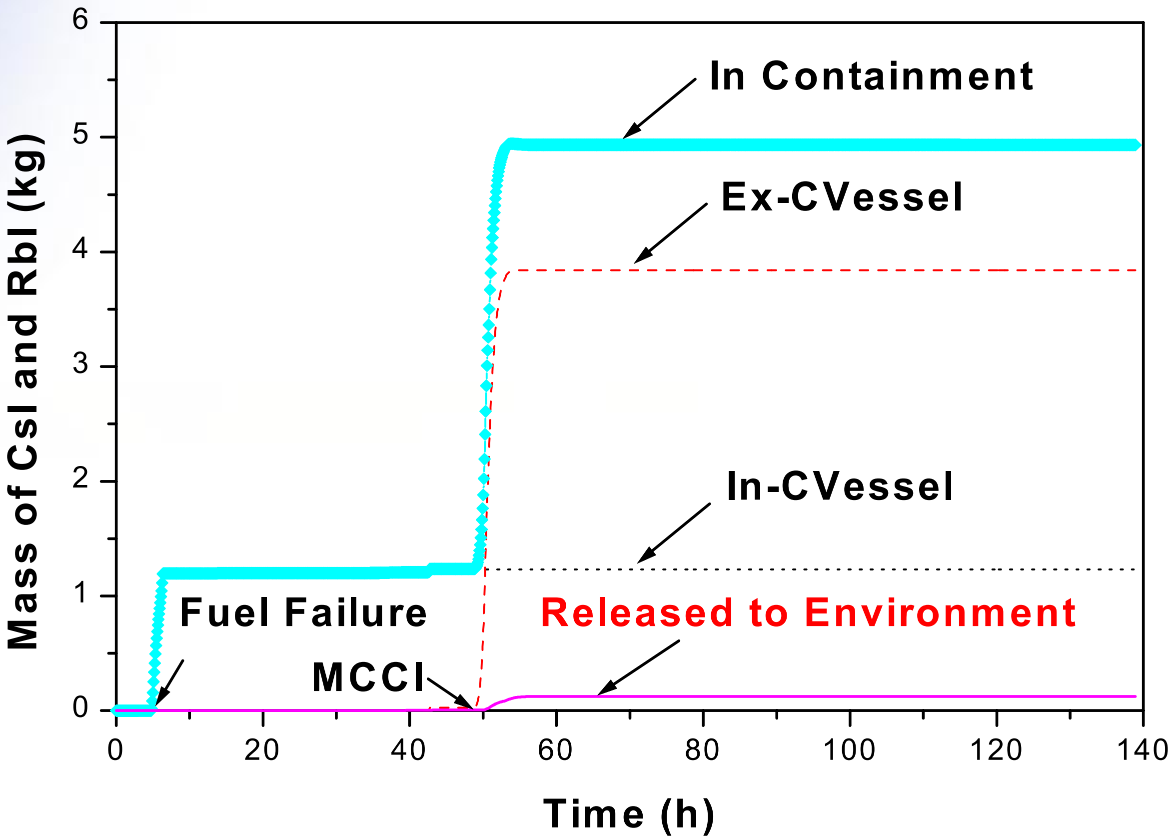
Until MCCI begins (~50 h), hydrogen concentration within DDT limits

Mass of Noble Gas Released (Generic CANDU 6 SBO)



Almost all Noble gases released, when containment fails

Mass of CsI and RbI released and in various locations (Generic CANDU 6 SBO)



Total Cs + I released in form of CSI+RbI+CsOH <0.01%, when containment fails. Significant retention in wet CANDU containment.

When concrete floor fails, total Cs + I released in form of CSI+RbI+CsOH is ~1.8%.

Summary of Sequence of Events (Generic CANDU 6 SBO with Loss of all Engineered Mitigating Systems)

Sequence of Events

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Loss of AC and all backup power

SG Secondary side dry

Fuel Bundles uncovered within Fuel channels

One fuel channel is dry inside

PT and CT ruptured

Moderator reaches saturation Temperature in CV

Beginning of Core Disassembly

Core Collapse onto CV Bottom

CV Water depleted

RV begins to boil off

Containment failed after a day

CV failed due to creep (Corium can be contained in CV as long as it is cooled by water in shield tank)

MAAP4 Validation

- **MAAP4 is validated by FAI using: Separate Effects Experiments, Integral Experiments, Industry Experience and detailed analysis for a large number (21) of physical processes such as:**
    - Core Heatup
    - Clad Oxidation
    - Fission Product Release
    - Aerosol Transport and Deposition
    - Hydrogen Combustion
    - In-Vessel Cooling
    - RPV External Cooling
    - Molten Debris Heat Transfer
    - Debris Fragmentation
    - Debris Dispersal and so on…

Sample MAAP4 Validation Matrix

		Type of Comparison						Documentation			
Phenomenon Name		**Separate Effects Experiments**		**Integral Experiments**		**Industry Experience (TMI-2)**	**Detailed Analysis**				
Physical Process	*Experiment/Code*	*Small Scale*	*Large Scale*	*Out-of-Reactor*	*In-reactor*			*Open Literature*	*IDCOR Reports*	*MAAP User's Manual*	*EPRI Reports*
Core Heatup (SA1)	PBF-SFD Tests			✓					✓		
	LOFT FP-2			✓							✓
	TMI-2					✓			✓		✓
	BWR Heatup code						✓		✓	✓	
	PWR Heatup code						✓	✓	✓	✓	
	CORA		✓	✓				✓		✓	
Clad Oxidation (SA2)	Numerous Expts	✓							✓		
	LOFT FP-2				✓						✓
	TMI-2					✓			✓		✓
	BWR Heatup code						✓	✓	✓	✓	
	PWR Heatup code						✓	✓	✓	✓	
	CORA		✓	✓				✓		✓	
Fission Product Release (SA3)	ORNL Experiments	✓						✓	✓	✓	
	SASCHA Expts	✓						✓	✓	✓	
	PBF-SFD Tests				✓				✓		
	LOFT FP-2				✓						✓
	TMI-2					✓			✓	✓	

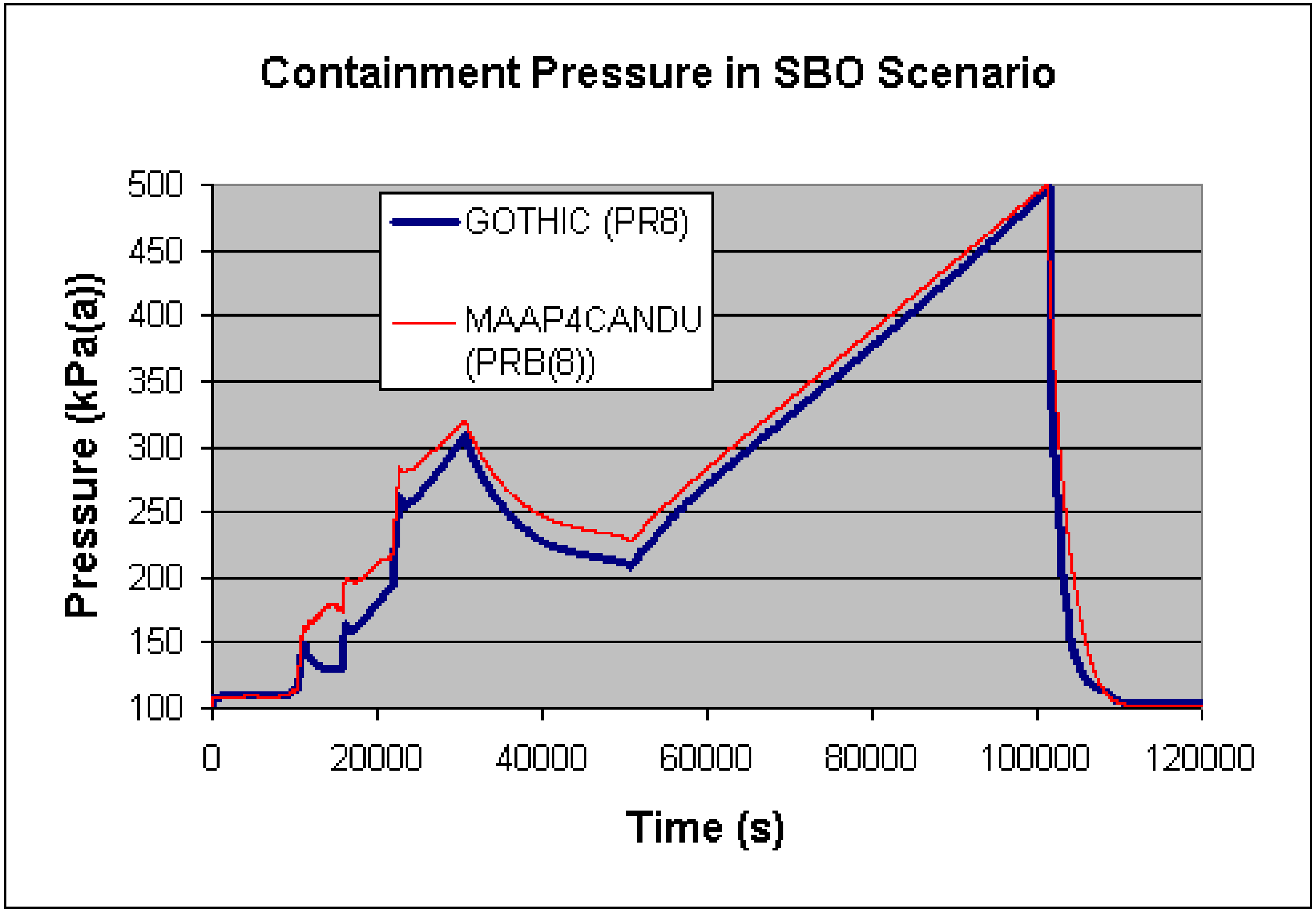
MAAP4 CANDU Validation

- Validation Method
  - There are no known “exact results” available to validate MAAP4 CANDU for the response of a CANDU to a Severe Core Damage Accident
  - Several Systems in MAAP4 CANDU are hard-coded and cannot be validated individually. Therefore, system response is compared with simplified analytical solutions for selected accident sequences. (Sample Results to follow here)
  - Validate separate modules, where possible, against other validated codes (GOTHIC for Containment Response)
  - Use CANDU-specific experimental results as they become available

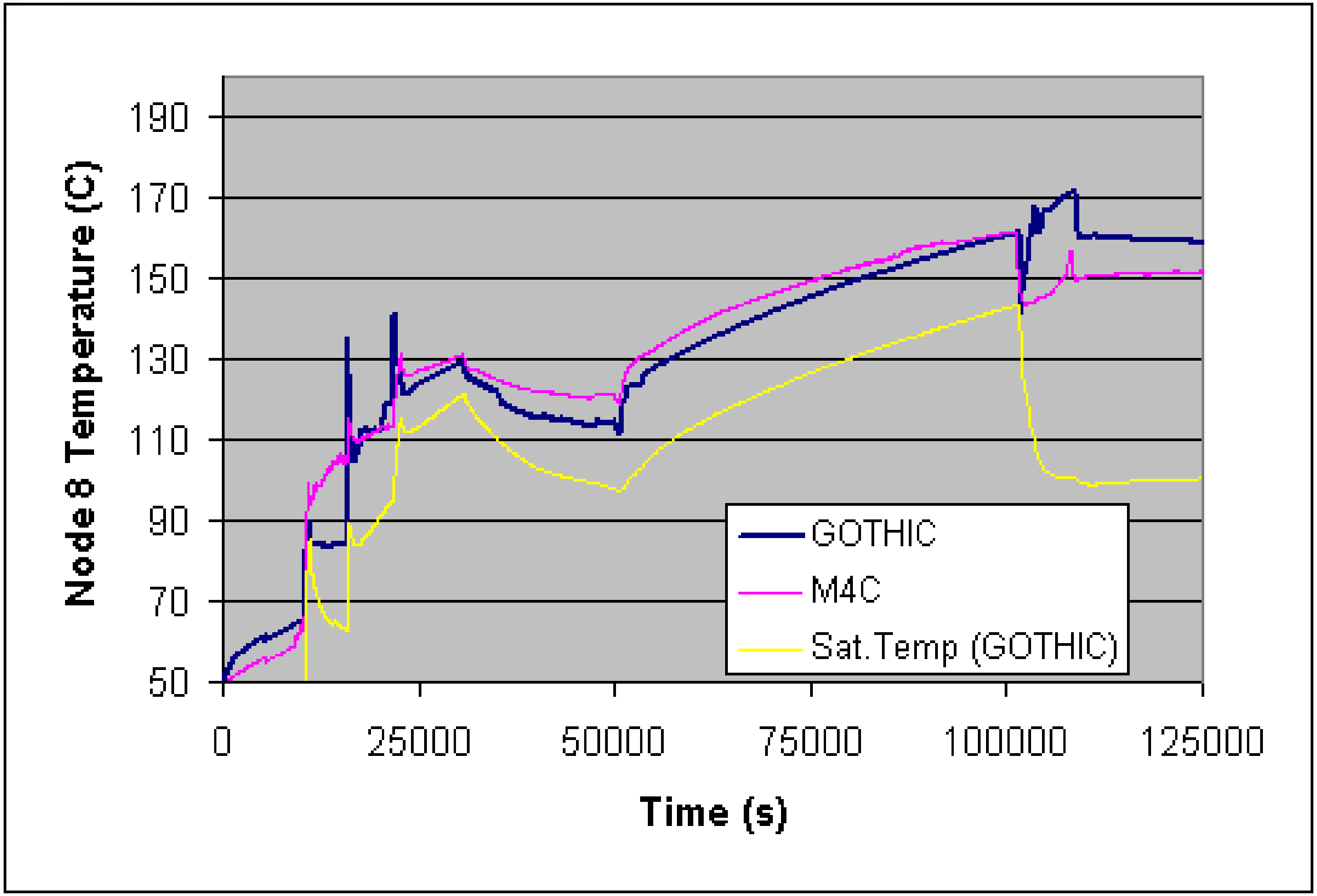
MAAP4 CANDU Containment Response Validation: Comparison with GOTHIC

- **CANDU 6 SBO scenario from reactor trip until containment failure**
  - Containment model Geometry of MAAP CANDU was replicated in GOTHIC (same number of volumes and connections between rooms)
  - Used MAAP CANDU results of mass and energy input to containment from RCS (PHTS), calandria vessel, and hot structures for GOTHIC
  - Transient containment pressure and temperature compared

MAAP4 CANDU Validation: Containment Response with GOTHIC



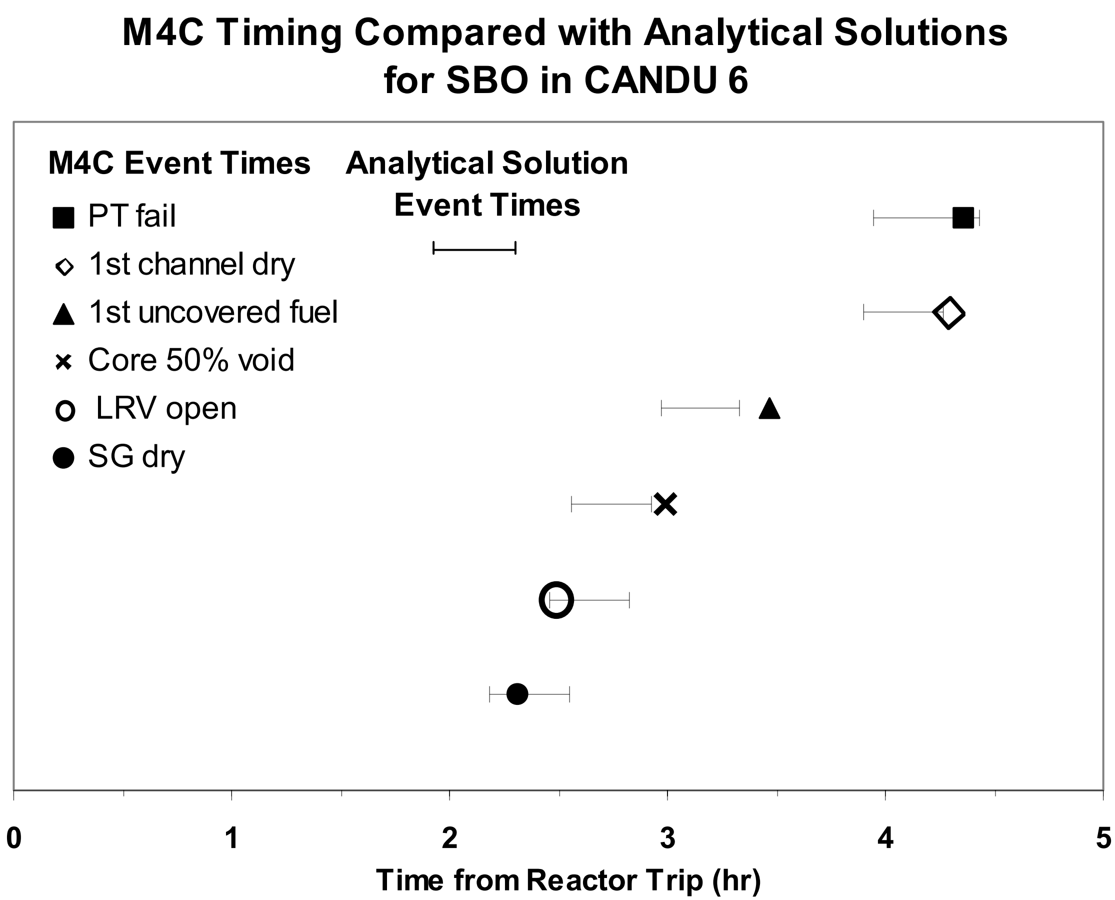
MAAP4 CANDU Validation: Containment Response with GOTHIC



MAAP4 CANDU Validation: Event Times Compared with Analytical Solutions

- CANDU 6 SBO scenario from reactor trip until high pressure lead-channel rupture
  – Analytical models for decay heat transfer from the core to the RCS (PHTS) including steam generator and moderator
  – In analytical model the RCS (PHTS) swell, loss through Liquid Relief valves also modeled
  – Timing of significant events compared

MAAP4 CANDU Validation: Event Times Compared with Analytical Solutions



Summary

- MAAP4 CANDU will be a useful tool to calculate severe accident progression in ACR after necessary code modifications are made
- MAAP4 CANDU results to-date show the slow nature of CANDU severe core damage accident progression
- Cooling provided by the shield tank prevents failure of the calandria, thereby containing the core
- Significant time will be available to arrest accident progression.
- MAAP4 CANDU validation activities are in progress. ACR-specific validation plan in progress

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