Overview of MAAP4-CANDU Code

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Presentation Outline

- Introduction
- Key Safety Features of ACR-700 Design
- MAAP4-CANDU Code Features and ACR-700 plant nodalization
- Failure criteria
- Sample Results for a CANDU 6
- MAAP4-CANDU Qualification/Validation
- Summary

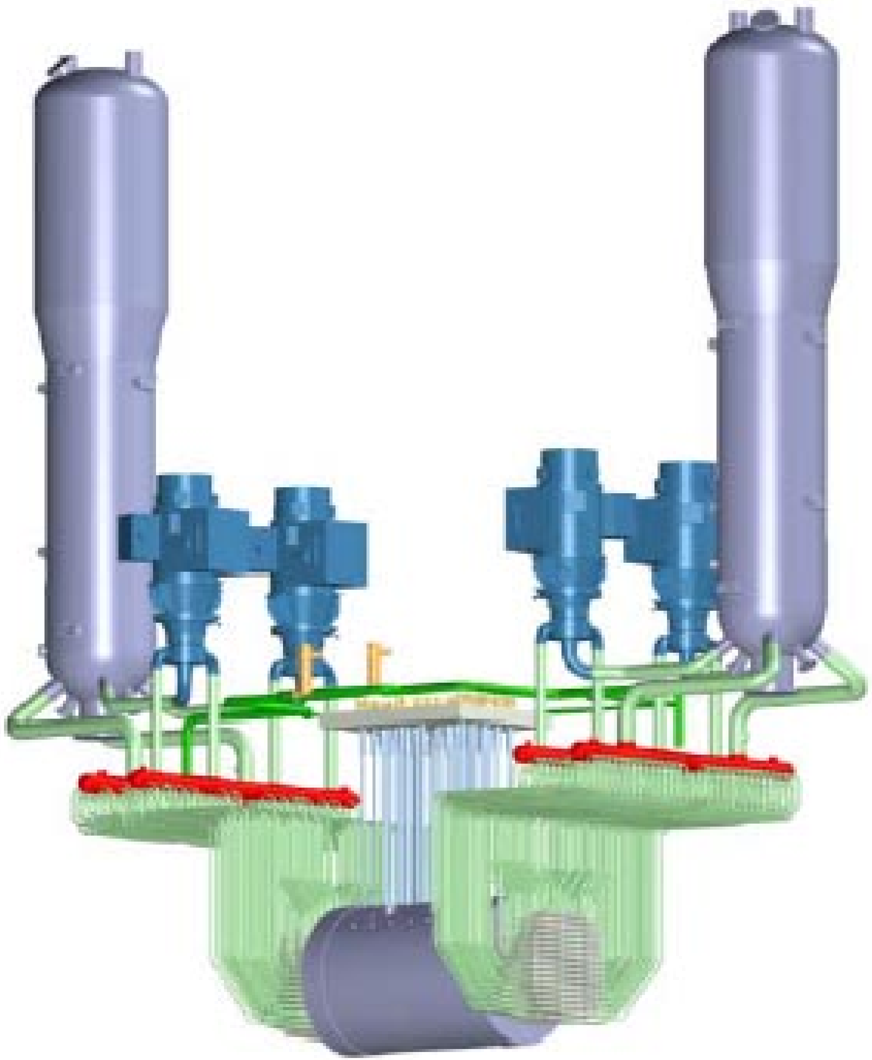
Introduction

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

Introduction (cont'd)

- The main distinguishing features of MAAP4-CANDU are models of the horizontal CANDU-type fuel channels and CANDU-specific systems such as Calandria Vessel, RCS, containment systems: dousing, local air coolers, hydrogen igniters and recombiners, reserve water system etc.
- MAAP4-CANDU has been used for Severe Core Damage Accident Consequence Analysis:
  - OPG Nuclear Generating Stations (Darlington, etc.);
  - Generic CANDU 6 Station
- Version 4.0.5 of MAAP4-CANDU developed for ACR-700 Plant.
- This presentation focuses on:
  - CANDU and ACR Enhancements
  - Qualification Activities

ACR-700 RCS System Layout



	CANDU 6	ACR-700
RCS Loops	2	1
RCS Pumps	4	4
Steam Generators	4	2
React. Inlet Headers	4	2
React. Outlet Headers	4	2
Fuel Channels	380	292

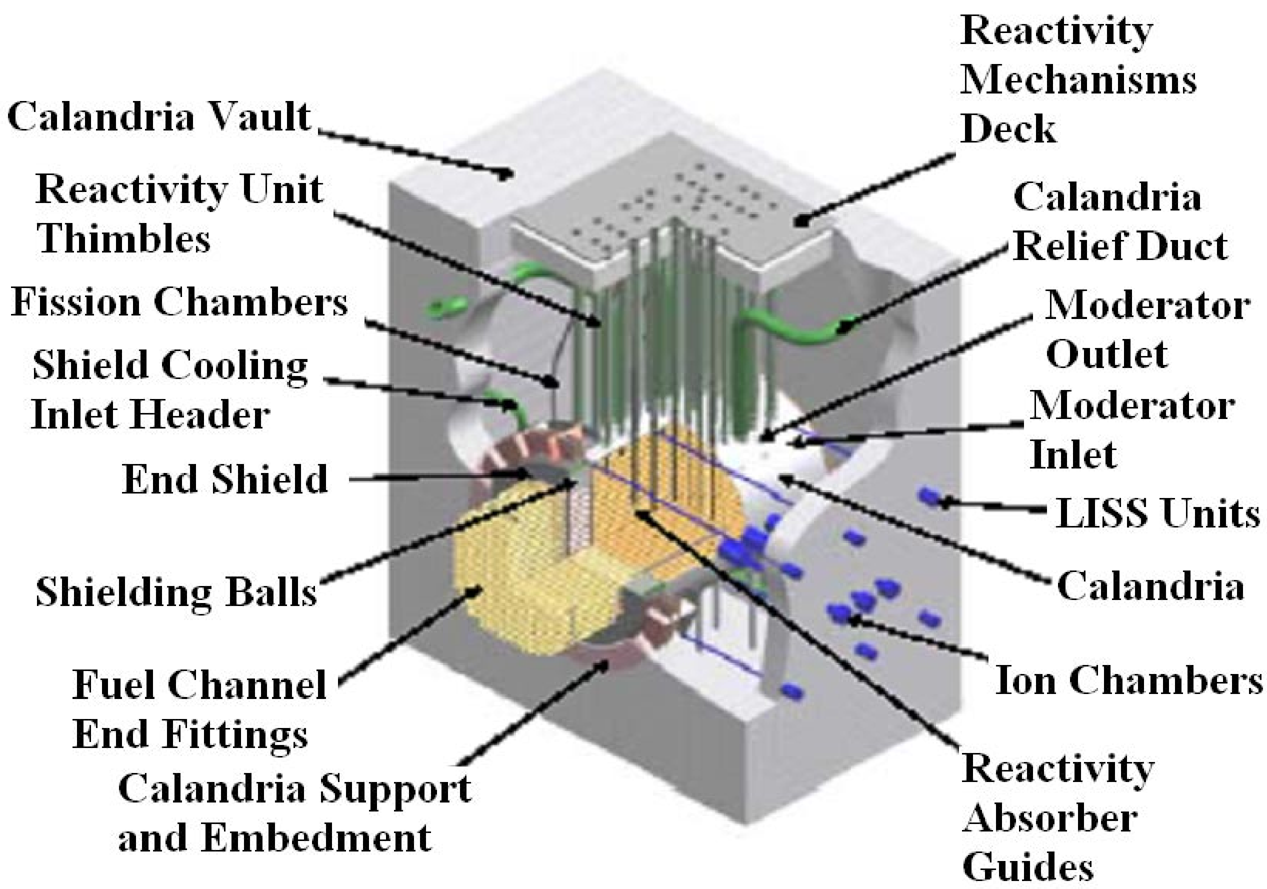
- RCS in CANDU 6 version of MAAP4-CANDU is “hard-wired”.
- MAAP4-CANDU ACR-700 version is an evolution of the CANDU 6 version

Key Safety Design Features of ACR-700

- The moderator system and the reactor (calandria) vault water inventory have significant capacity to slow down severe core damage progression so that the operator will have sufficient time to implement severe accident management measures

- The addition of the Reserve Water System in ACR-700 provides Steam Generators, the Calandria Vessel and the Reactor Vault with water to extend the duration of plant heat removal capability for Severe Core Damage Accidents

ACR-700 Reactor Assembly



292 fuel channels

Inventory:

H_2O in HTS:	~118 Mg
H_2O in RV:	630 Mg
D_2O in CV:	102 Mg
RWT:	2500 Mg

- Severe Core Damage progression in CANDU is slow

ACR-700 Safety Systems

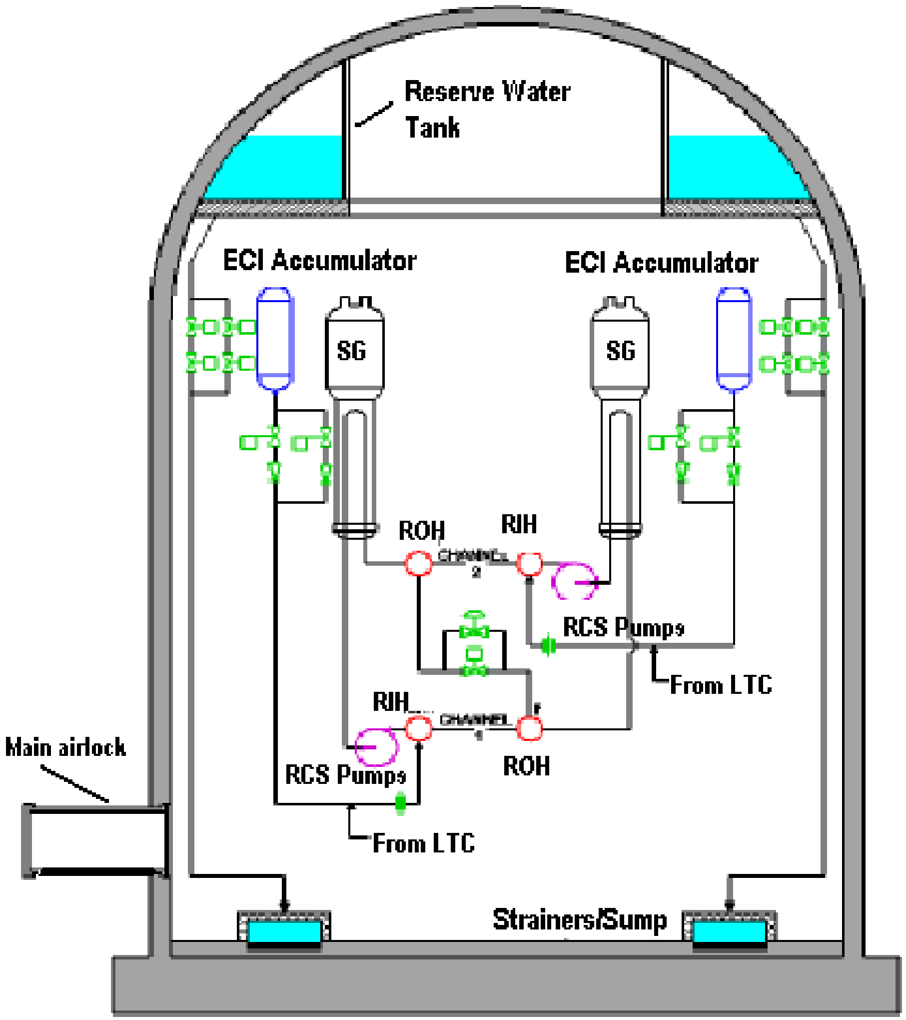
- **Safety Systems:**
  - Shutdown Systems (1&2);
  - Emergency Core Cooling System (Emergency Core Injection System and Long Term Cooling System);
  - Containment Systems (Containment Cooling System, Hydrogen Control System);

- **Safety-related Systems**:
  - Reserve Water System;
  - Recirculated Cooling Water System;
  - Compressed Air System, etc.

CANDU Safety Approach

- For reactor shutdown, in addition to the control system, two independent shutdown systems can shut the plant down for any design basis or severe-accident.
- For accidents with the reactor coolant pressure boundary intact, the decay heat is removed by the steam generator heat sink or a dedicated decay heat removal system. Even in the unlikely failure of the decay heat removal systems, the moderator will prevent gross damage to the fuel and maintain fuel channel integrity in the absence of sustained film boiling.
- In the unlikely event that the moderator cooling system also fails ("Severe Core Damage Accidents"), the fuel channels would sag and collapse as the moderator boils off; but the core debris would still be contained within the calandria vessel as long as it remains cooled by the calandria vault water outside.
- If the calandria vault water boils off, the calandria vessel can fail, which will take a long time, typically more than 24 hours in a CANDU 6. Additionally, in ACR-700 plant, a Reserve Water Tank with a capacity of 2500 m3 is available, which will provide light water make-up to the various systems, which will extend the failure time of the calandria vessel. The Reserve Water Tank can also be replenished from an external water source, if required.

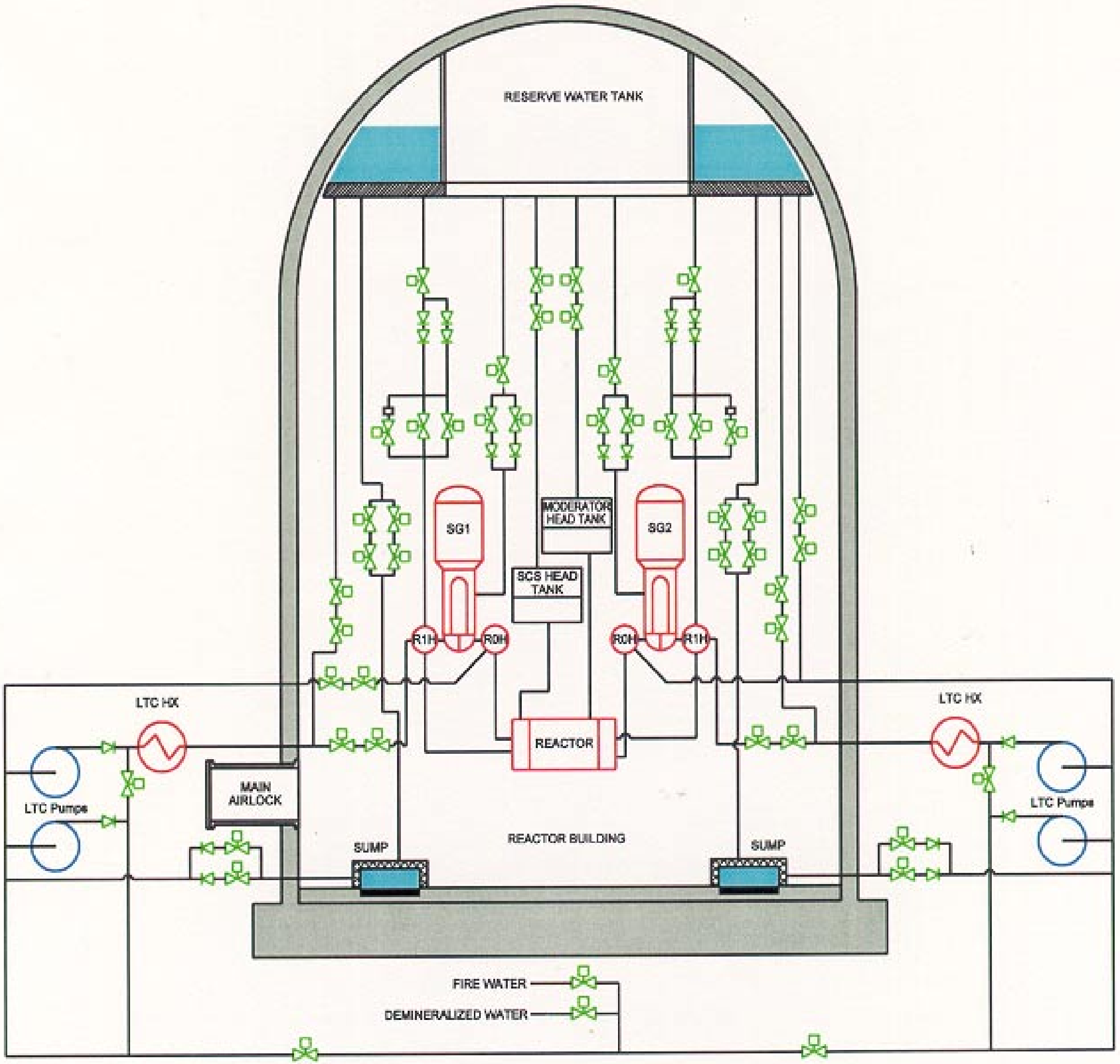
ACR-700 Emergency Coolant Injection Flow Diagram



ECIS:

- 2 ECI High Pressure water Tanks : Injection into Reactor inlet headers at RCS high pressure
- Water Dump from the Reserve Water Tank on the containment floor on LOCA signal (this water will be used by LTC system at low RCS pressure).

ACR-700 Reserve Water System (RWS) Flow Diagram



6 main RWS functions

1.) Low pressure ECCS
2.) Make-up to the shield cooling system (calandria vault/end-shields)
3.) Make-up to the calandria vessel
4.) RCS make-up
5.) SG make-up
6.) RWT make-up from an external source and RWS recovery.

MAAP4-CANDU CAPABILITIES

- Typical severe core damage accident scenarios, which can be analyzed with MAAP4-CANDU:
  - Large Break Loss-of-Coolant Accident (LOCA)
  - Small Break LOCA
  - In-core LOCA
  - Transients such as Loss of AC and DC power
  - Steam Generator Tube Rupture

- MAAP4-CANDU computations are more than one hundred times faster than real time (for Pentium-4 PC 1.7 MHz)

Physical Processes modeled in MAAP4-CANDU

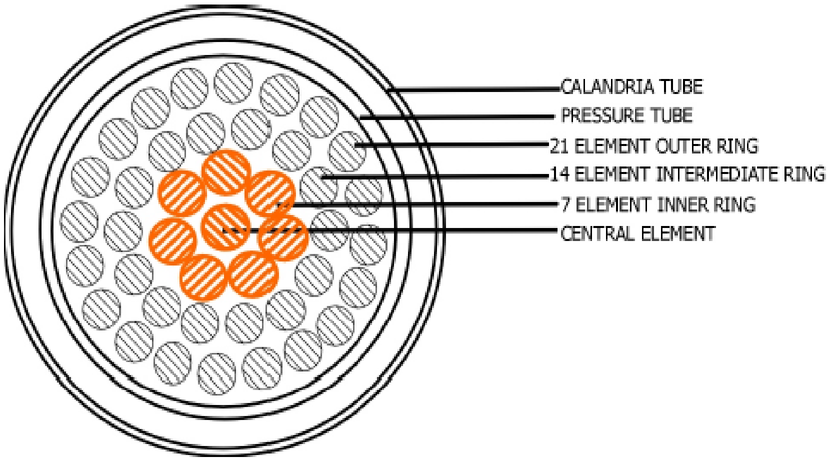
- Simple Thermal hydraulics in RCS, calandria vessel, calandria vault, end-shields, containment compartments
- Core heat-up (initiated, when the fuel channel is dry)
- Core melting and disassembly
- Zr oxidation by steam and hydrogen generation
- Material creep and possible rupture of pressure and calandria tubes, calandria vessel wall
- Ignition of combustible gases
- Energetic corium-coolant interactions
- Molten corium-concrete interaction
- Fission product release, transport and deposition

Nodalization of ACR-700 Plant

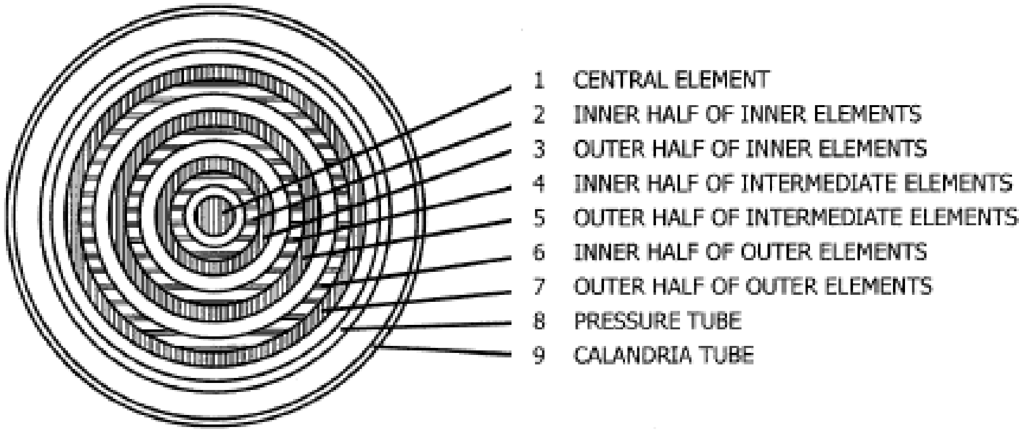
- Generalized Containment Model
  - Compartments represented by 13 nodes connected by 34 flow junctions
  - Containment walls/structures modeled as 90 “heat sinks”
- RCS
  - RCS loop following “figure of 8”; 14 major nodes: ROH, RIH, SG inlet piping, etc.
- Core
  - 292 fuel channels arranged in 18 rows and 18 columns, represented by 6 vertical nodes, 3 power groups (L,M,H) of channels in each vertical core node: 18 characteristic channels
  - 12 fuel bundles represented by 12 axial nodes
  - 43 fuel elements, pressure and calandria tube modeled as 9 concentric rings
- Calandria Vessel cylindrical Wall
  - 15 vertical nodes, 1 axial node
- Steam Generator
  - Primary side modeled as 2 nodes (“hot “ and “cold”); secondary side- as 1 node

ACR-700 Fuel Channel Cross-Section Diagram and Nodalization Scheme

ACR-700 Fuel Channel Cross-section



7 fuel rings + pressure tube & calandria tube



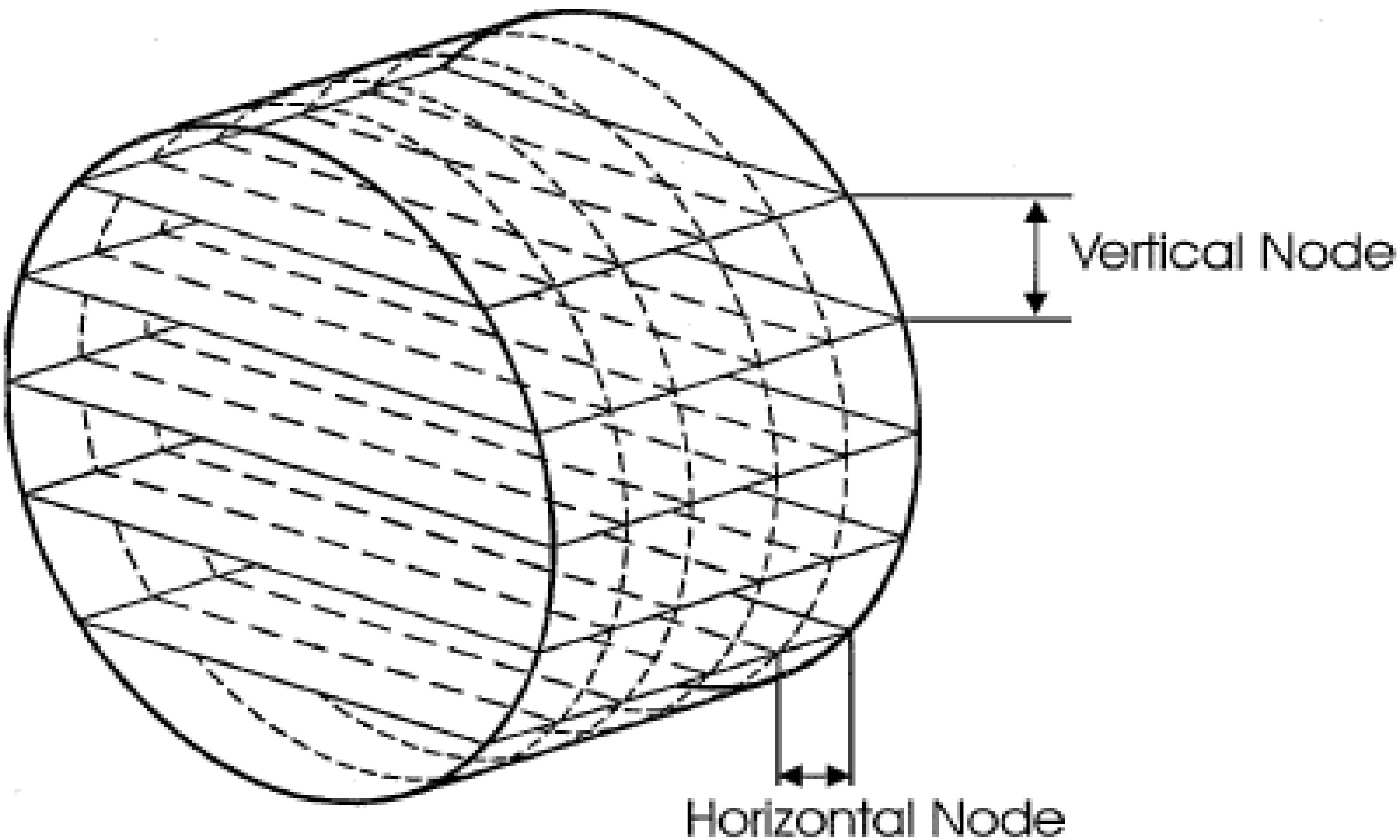
Time-Averaged Channel Power Map for ACR-700 Core

(preliminary)

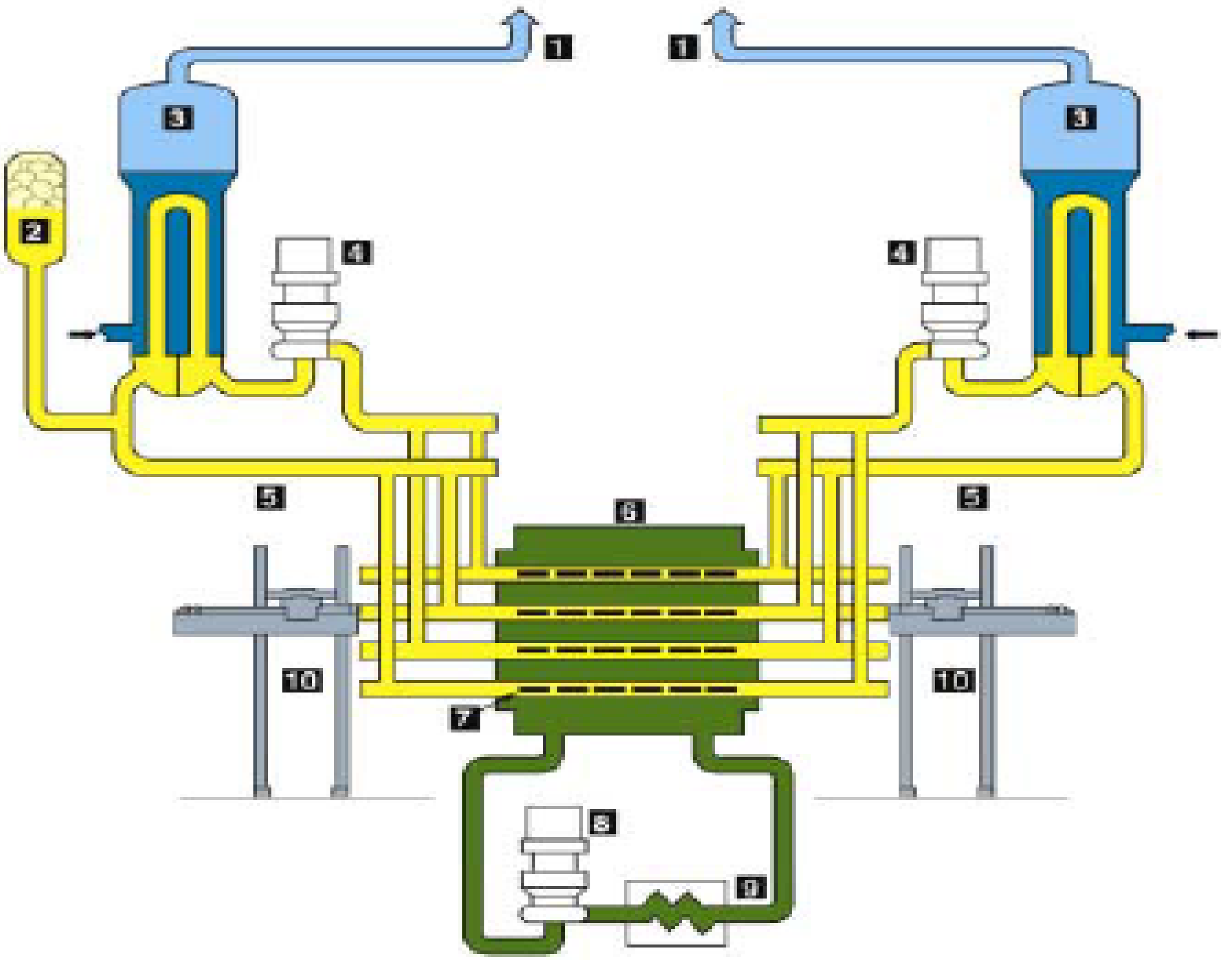
CHANNEL POWER DISTRIBUTION

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	
A					5665	6371	7001	7286	7203	7203	7285	7000	6370	5664					
B				6123	6224	6853	7290	7474	7372	7372	7474	7289	6852	6223	6122	1			
C			6331	6245	6616	6641	6902	7003	6896	6896	7002	6901	6641	6615	6244	6330			
D		6250	6443	6744	6577	6844	7012	7081	7023	7023	7081	7012	6843	6576	6742	6441	6248		
E	5800	6288	6831	6682	6855	7017	7134	7194	7196	7196	7193	7133	7016	6853	6680	6830	6287	5798	2
F	6249	6693	6691	6893	7036	7161	7250	7294	7306	7306	7293	7249	7160	7035	6892	6689	6691	6248	
G	6813	7068	6920	7060	7173	7273	7338	7379	7391	7391	7379	7337	7273	7172	7059	6918	7066	6811	
H	7196	7334	7095	7186	7271	7345	7403	7439	7449	7448	7439	7403	7344	7270	7185	7093	7332	7195	3
J	7388	7471	7188	7254	7322	7385	7438	7470	7478	7478	7470	7437	7385	7321	7253	7187	7469	7387	
K	7388	7470	7188	7254	7321	7385	7437	7470	7478	7478	7469	7437	7385	7321	7253	7187	7469	7387	
L	7196	7333	7094	7186	7270	7345	7403	7439	7448	7448	7439	7403	7344	7269	7185	7093	7332	7195	4
M	6812	7067	6919	7060	7173	7273	7337	7379	7390	7390	7379	7337	7272	7172	7059	6918	7066	6811	
N	6249	6692	6690	6892	7035	7160	7249	7293	7305	7305	7293	7249	7159	7035	6891	6689	6691	6248	
O	5799	6287	6830	6680	6853	7015	7133	7193	7195	7195	7193	7133	7015	6853	6680	6829	6286	5798	5
P		6249	6441	6742	6575	6842	7011	7080	7022	7022	7080	7011	6842	6575	6741	6441	6248		
Q			6329	6244	6614	6640	6900	7002	6895	6895	7001	6900	6640	6614	6243	6329			
R				6121	6222	6851	7269	7473	7371	7371	7473	7288	6851	6222	6121				
S					5663	6370	7000	7284	7201	7201	7284	6999	6369	5663			6		

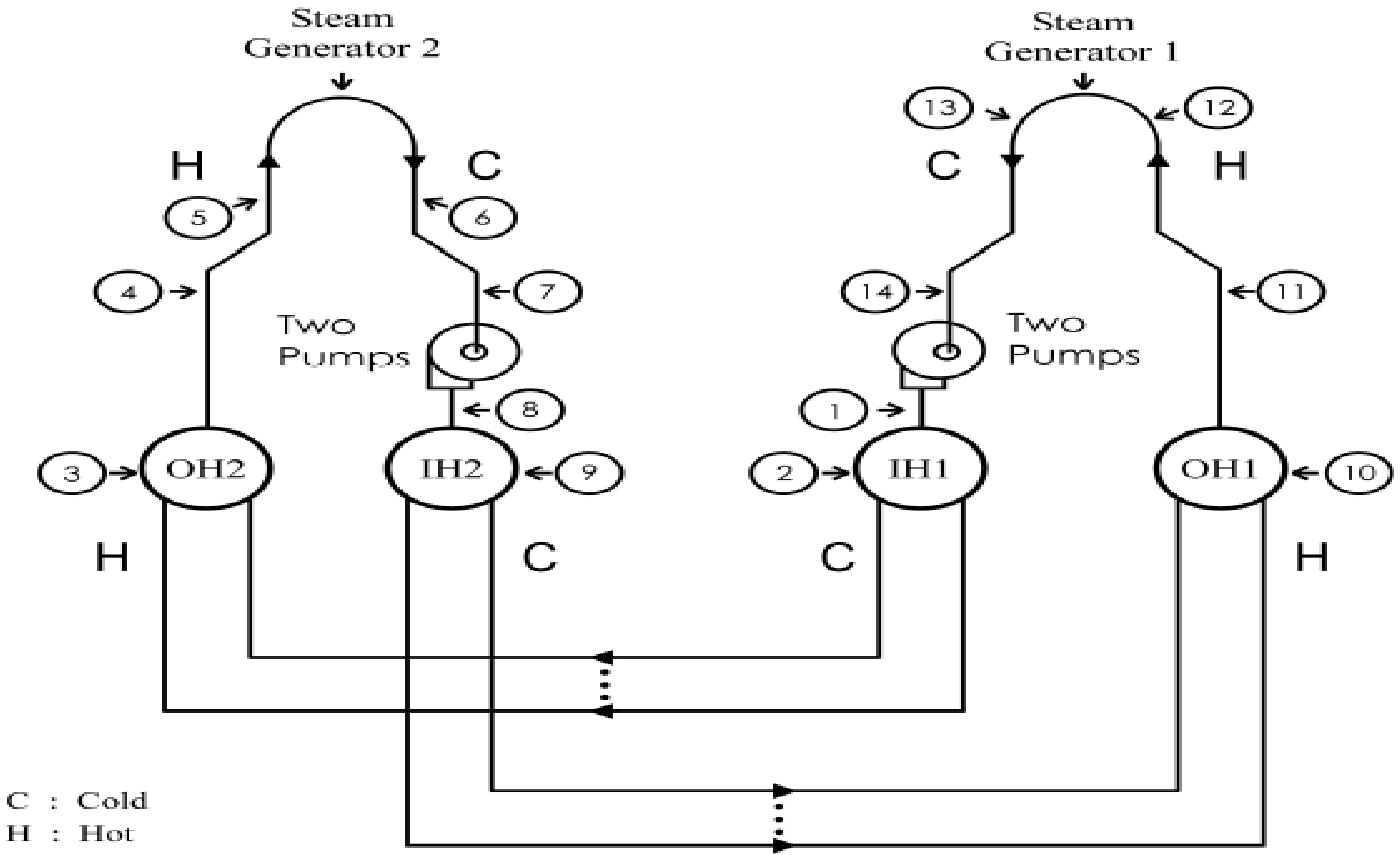
Nodalization Scheme for ACR-700 Core



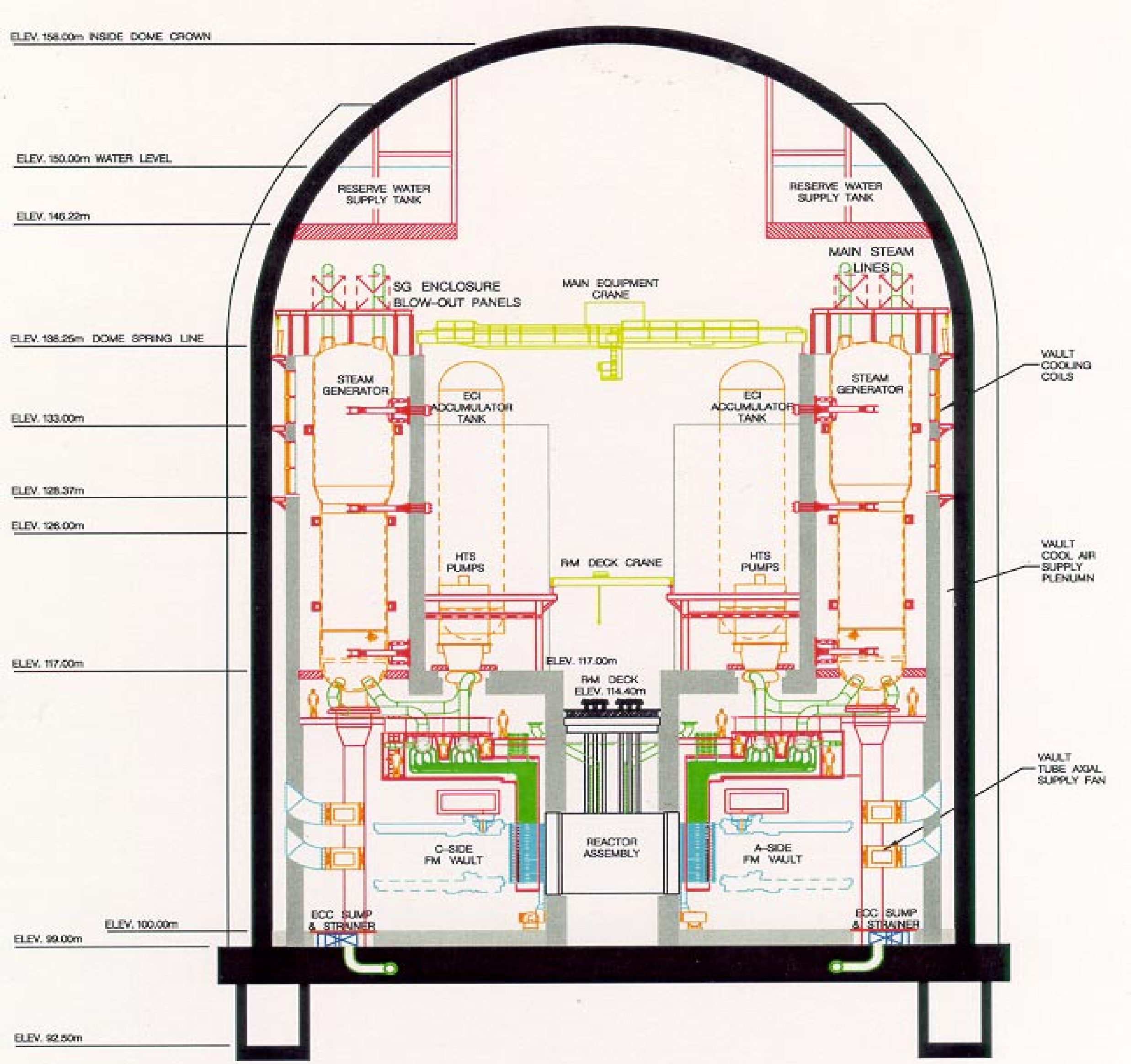
ACR-700 Reactor Coolant System



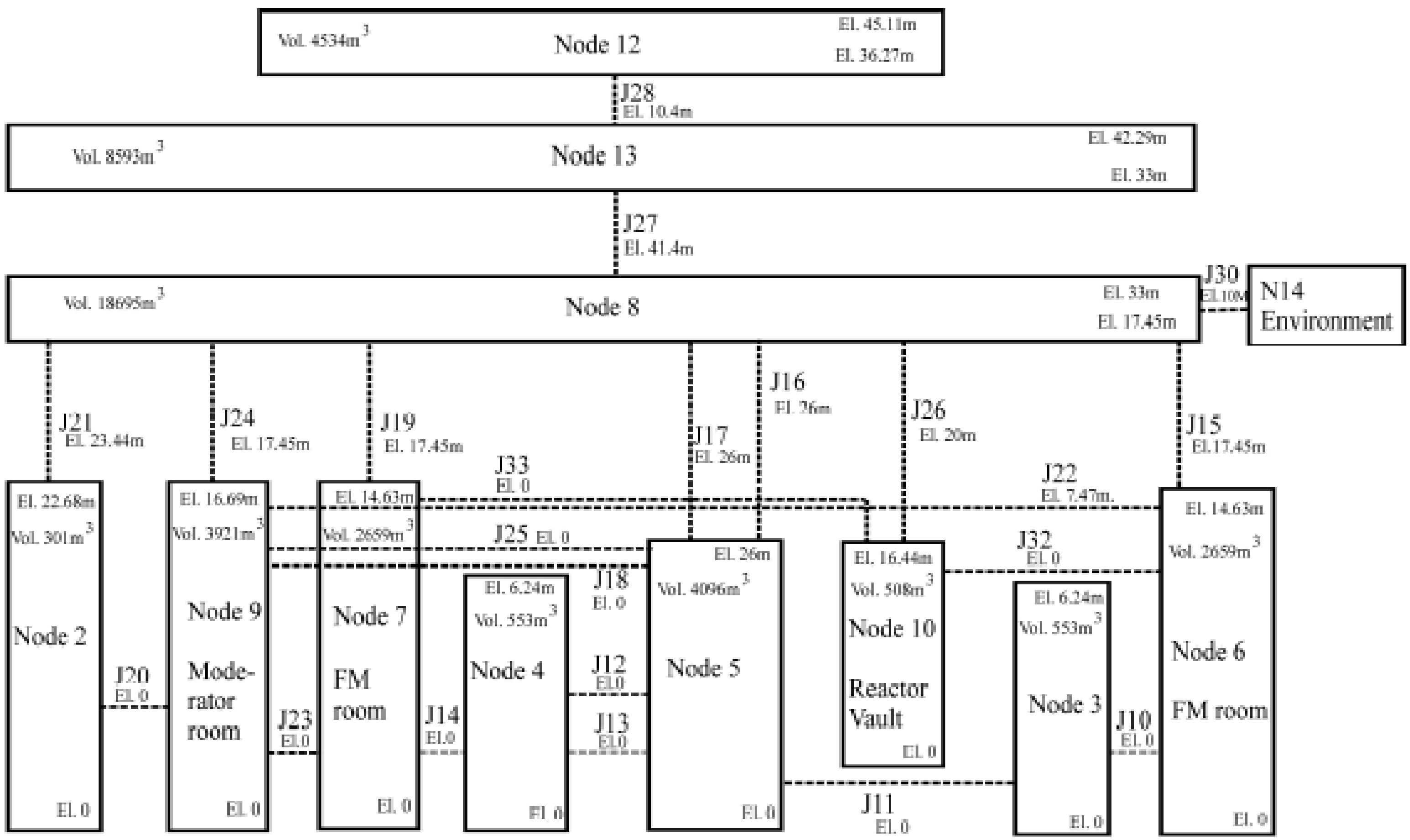
Nodalization Scheme for ACR-700 RCS: 14 major nodes



ACR-700 Reactor Building



Nodalization for ACR-700 Containment (preliminary)
13 nodes, 34 flow junctions



Note: Elevations are given w.r.t. containment floor, which is set at zero meter.

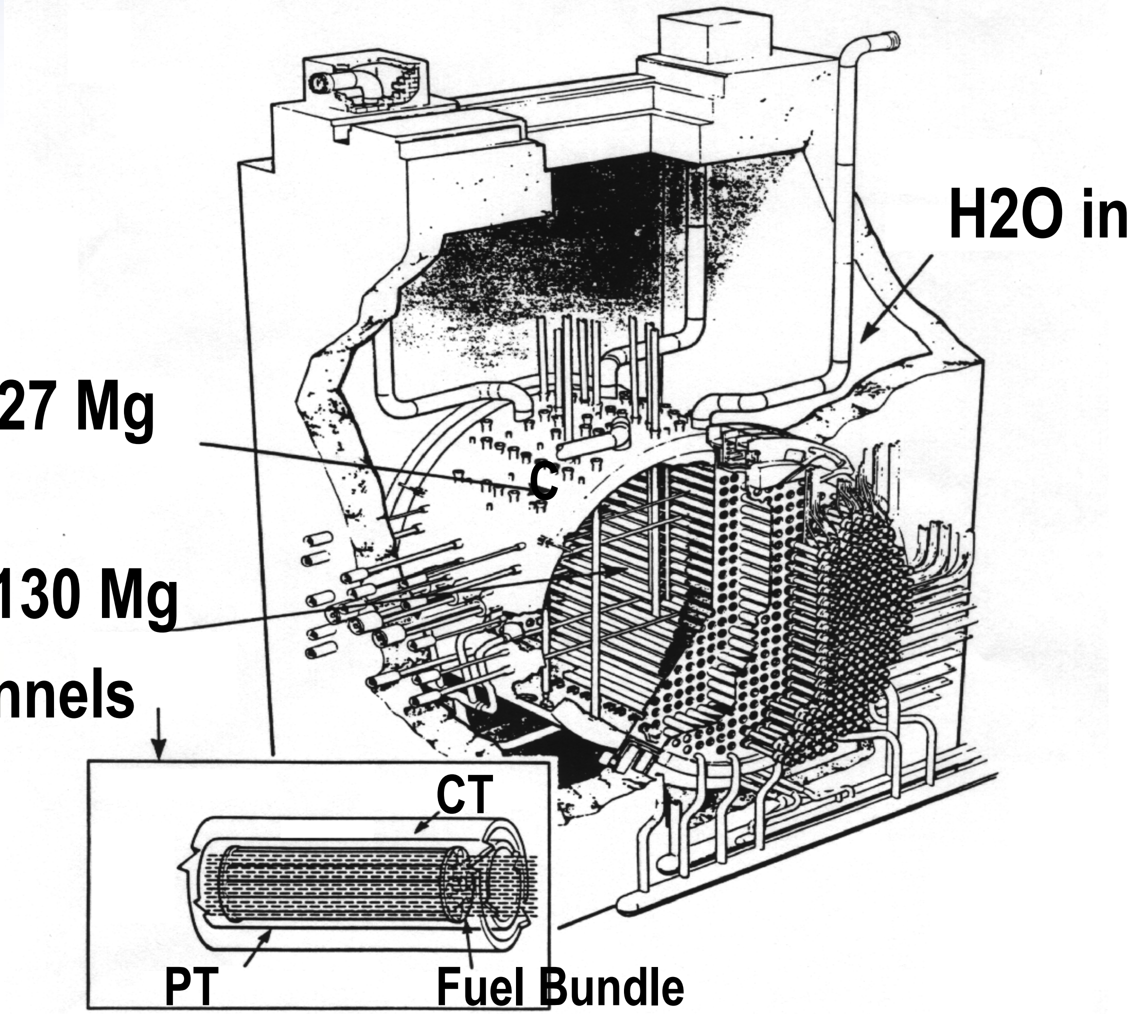
FAILURE CRITERIA

- **Containment:**
  - Simplified containment failure model based on gas pressure in containment:
    - when gas pressure in containment is higher than a user-input value, containment fails
- **Calandria Vessel (CV):**
  - Several options of failure models available in the code, such as creep rupture of the CV wall using Larson-Miller parameter parameter etc. from generic MAAP and CANDU-specific option for CV failure, when water level in reactor vault reaches corium level inside calandria (low pressure).
- **Calandria Vault:**
  - When the eroded concrete thickness reaches a certain “critical” depth (user input), the calandria vault is considered failed.

FAILURE CRITERIA (Cont'd)

- **Fuel channel failure:** (Perforation of both PT&CT, followed by mass transfer between the channel inside and CV)
  - *High RCS Pressure:*
    - *Pressure tube balloons and fails when hoop stress exceeds failure stress derived from Mochizuki et al experiments*
  - *Low RCS Pressure:*
    - *Local melt-through based on PT/CT average temperature*
    - *Failure by sagging: if PT/CT average temperature exceeds user input value*
- **Fuel channel disassembly:** (Separation of channel segments from original channel)
  - An axial fuel channel segment is considered disassembled when the average PT&CT temp. reaches melting temp. of oxygenated Zr.
- **Fission product (FP) release:**
  - If the combined fuel cladding/UO2 temperature for a core node is greater than 1000 K (user input), noble gases from the gap are released. FP from the fuel matrix are released depending on the fuel temperature.

CANDU 6 Reactor Core



CANDU 6 SBO Analysis Assumptions

- AC Power & 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 Actions not credited

Summary of Significant Events (Generic CANDU 6 SBO)

Time (h)	comments
0	Loss of AC and all backup power
2.5	SG Secondary side dry
3.5	Fuel Bundles dry within Fuel Channels
4.4	PT and CT ruptured
4.4	Moderator in CV begins to boil
4.8	Beginning of Core Disassembly
6.3	Beginning of Core Debris relocation to CV Bottom
8.3	Core Collapse onto CV Bottom
8.9	CV Water depleted
14.5	Calandria Vault begins to boil off
27.1	Containment failed
42.4	CV failed due to creep

Generic MAAP4 Validation

- **Generic MAAP4 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**

MAAP4-CANDU Validation

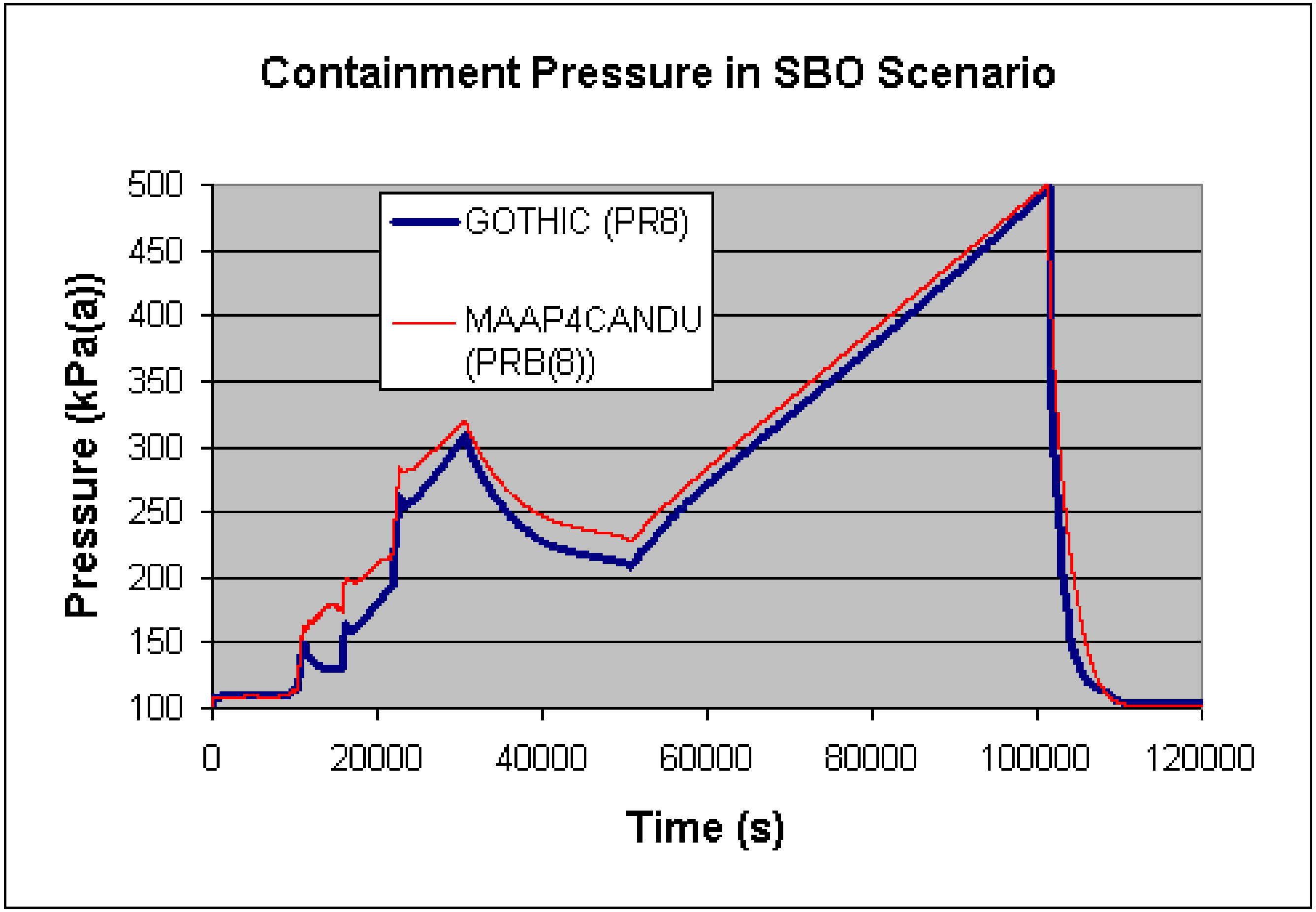
- **Validation Method**

  - **No known "exact results" available to validate CANDU response to a Severe Core Damage accident using MAAP4-CANDU**
  - **Some CANDU-specific phenomena (e.g. core disassembly, debris movement, debris suspension) have no experimental data for model validation.**
    - **Some 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.**
    - **Validate separate modules, where possible, against other validated codes (eg. GOTHIC for containment response)**

MAAP4-CANDU Validation: Containment Pressure Response Compared with GOTHIC

- **CANDU 6 Station Blackout (SBO) until containment failure**
  - **Containment model geometry of MAAP4-CANDU replicated in GOTHIC (same number of volumes & connections between rooms)**
  - **Used MAAP4-CANDU mass and energy results as input to containment from RCS, calandria vessel, and hot structures for GOTHIC**

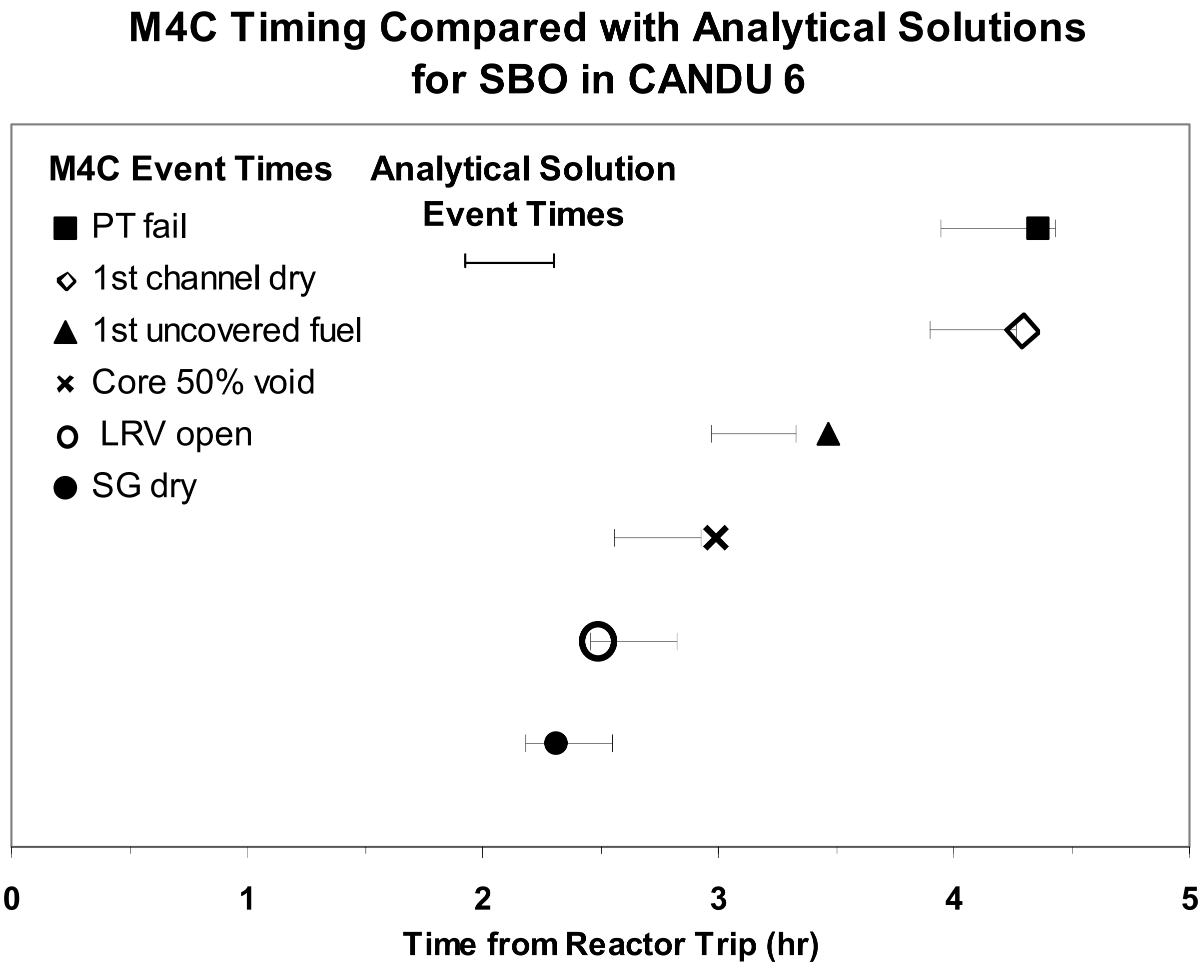
MAAP4-CANDU Validation: Containment Pressure Response Compared with GOTHIC



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 including steam generator and moderator
  - In analytical model the RCS level swell, loss through liquid relief valves also modeled
  - Timing of significant events compared
  - Good agreement obtained

MAAP4 CANDU Validation: Event Times Compared with Analytical Solutions



In Progress: MAAP4-CANDU Validation

- **MAAP4-CANDU validation activities are underway for previous CANDU 6 version**
- **No new phenomena for MAAP4-CANDU ACR-700 version validation identified.**
  - **New MAAP4-CANDU ACR-700 version requires incremental validation for ACR-specific systems such as RWS.**

Summary

- MAAP4-CANDU Version 4.0.5 contains all major models required for ACR-700 plant severe accident analysis: one-loop RCS, Core, Pressure and Inventory Control System, ECIS, LTC, Reserve Water System and Containment Systems
- ACR-700 Failure criteria are being developed and will be incorporated in the MAAP4-CANDU ACR Version
- Confirmatory validation activities are underway for validating the code

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