NUCLEAR SYSTEMS ENGINEERING

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NuTHEL
Nuclear Thermal Hydraulic Engineering Lab.

1. INTRODUCTION TO NUCLEAR SYSTEMS

1.1 CANDU-6

❖ Heavy water

AECL

- D_2O
- Density: 1106 kg/m³ at 20 °C, 11% heavier than light water
- Melting and boiling temperature: 3.82 °C and 101.4 °C
- Price:

DEUTERIUM OXIDE (D, 99.9%)

DLM-4-1000	$1,213.00	1000 G

DLM-4 / Deuterium Oxide 99.9% / 듀테늄옥사이드

옵션:1000g 1 1,712,810원

총 상품 금액 : 1,712,810원

- CANDU-6 heavy water inventory:
  - ACR1000 (advanced CANDU reactor): 250 tons with light water coolant
  - Make up for annual losses:
- Moderation ratio

Moderator	Moderating Ratio
Light water	62
Graphite (carbon)	165
Heavy water	5,000

$$\left(\begin{array}{c}\text{neutron KE lost}\\ \text{per collisions}\end{array}\right) \times \frac{\text{elastic scattering } \sigma}{\text{neutron capture } \sigma}$$

- **Heavy water**
  - Moderation ratio

Moderator	A	α	ξ	$\rho[g/cm^3]$	from 2 MeV to 1 eV	$\xi\Sigma_s[cm^{-1}]$	$\xi\Sigma_s/\Sigma_a$
H	1	0	1	gas	14	—	—
D	2	.111	.725	gas	20	—	—
H_2O	—	—	.920	1.0	16	1.35	71
D_2O	—	—	.509	1.1	29	0.176	5670
He	4	.360	.425	gas	43	1.6×10^{-5}	83
Be	9	.640	.209	1.85	69	0.158	143
C	12	.716	.158	1.60	91	0.060	192
^{238}U	238	.983	.008	19.1	1730	0.003	.0092

Statistics

- 49 PHWRs/443 NPPs

Reactor Type	Reactor Type Descriptive Name	Number of Reactors
BWR	Boiling Light-Water Cooled and Moderated Reactor	63
FBR	Fast Breeder Reactor	3
GCR	Gas Cooled, Graphite Moderated Reactor	14
LWGR	Light-Water Cooled, Graphite Moderated Reactor	12
PHWR	Pressurized Heavy-Water Moderated and Cooled Reactor	49
PWR	Pressurized Light-Water Moderated and Cooled Reactor	302
Total		**443**

In 2020,

Country	No. of PHWRs
	19
	19/22
China	2
Argentina	1+2
Romania	2
Pakistan	1
Total	49

https://www.iaea.org/PRIS/

Statistics

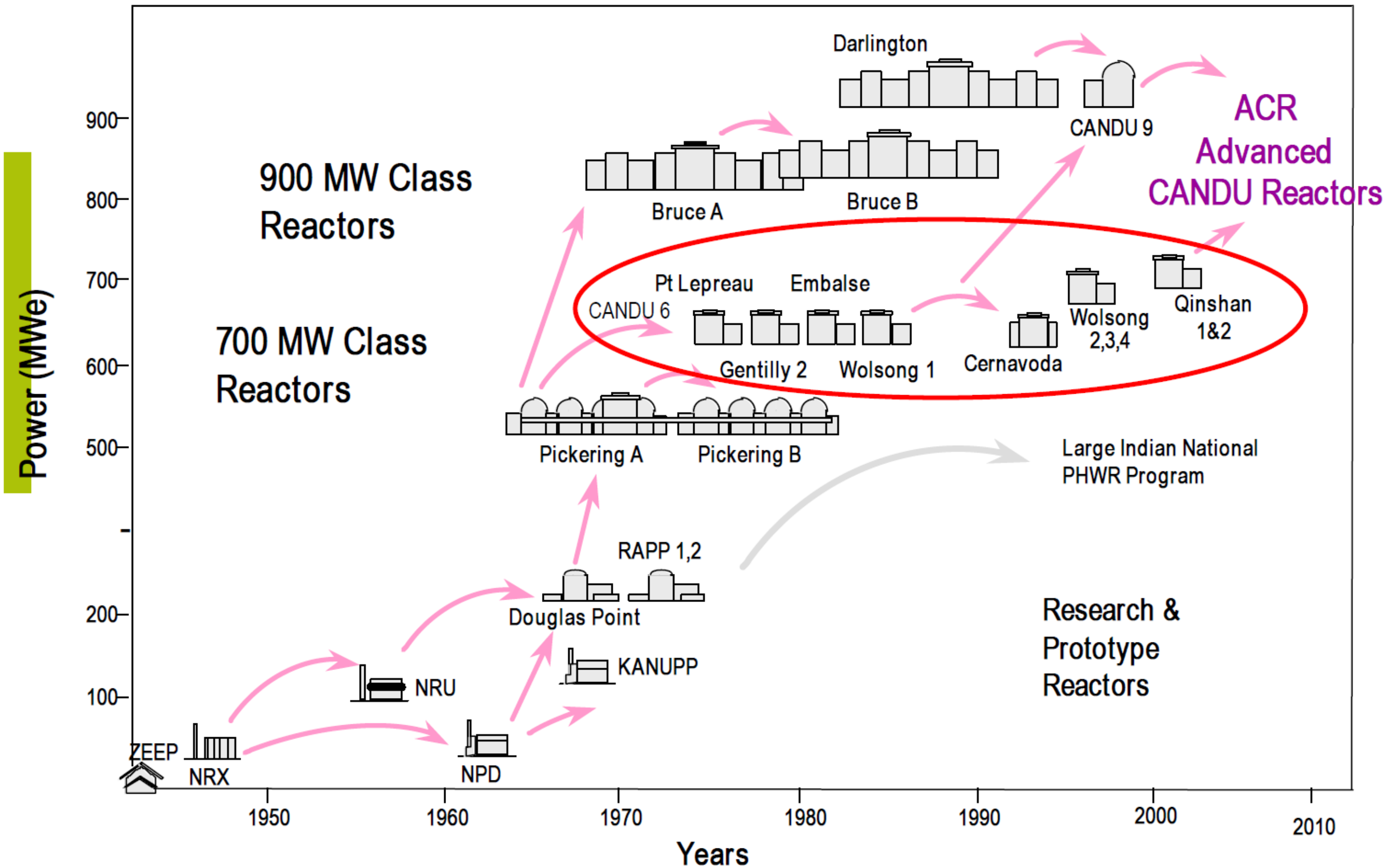
- 3 PHWRs/50 NPPs

Reactor Type	Reactor Type Descriptive Name	Number of Reactors ▼
PWR	Pressurized Light-Water Moderated and Cooled Reactor	43
PHWR	Pressurized Heavy-Water Moderated and Cooled Reactor	3
BWR	Boiling Light-Water Cooled and Moderated Reactor	2
FBR	Fast Breeder Reactor	1
HTGR	High Temperature Gas Cooled Reactor	1
Total		**50**

KAKRAPAR-4	PHWR	Under Construction	SURAT	630
KUDANKULAM-3	PWR	Under Construction	Tirunellveli-Kattabomman	917
KUDANKULAM-4	PWR	Under Construction	Tirunellveli-Kattabomman	917
PFBR	FBR	Under Construction	MADRAS	470
RAJASTHAN-7	PHWR	Under Construction	KOTA	630
RAJASTHAN-8	PHWR	Under Construction	KOTA	630

CANDU ( )

- All NPPs in CANADA are of the CANDU type.
- Marketed abroad
  - India, Pakistan, Argentina, South Korea, Romania and China



Gross generation = net generation
+ usage within the plant (also known as in-house loads)

CANDU (CANada Deuterium Uranium)

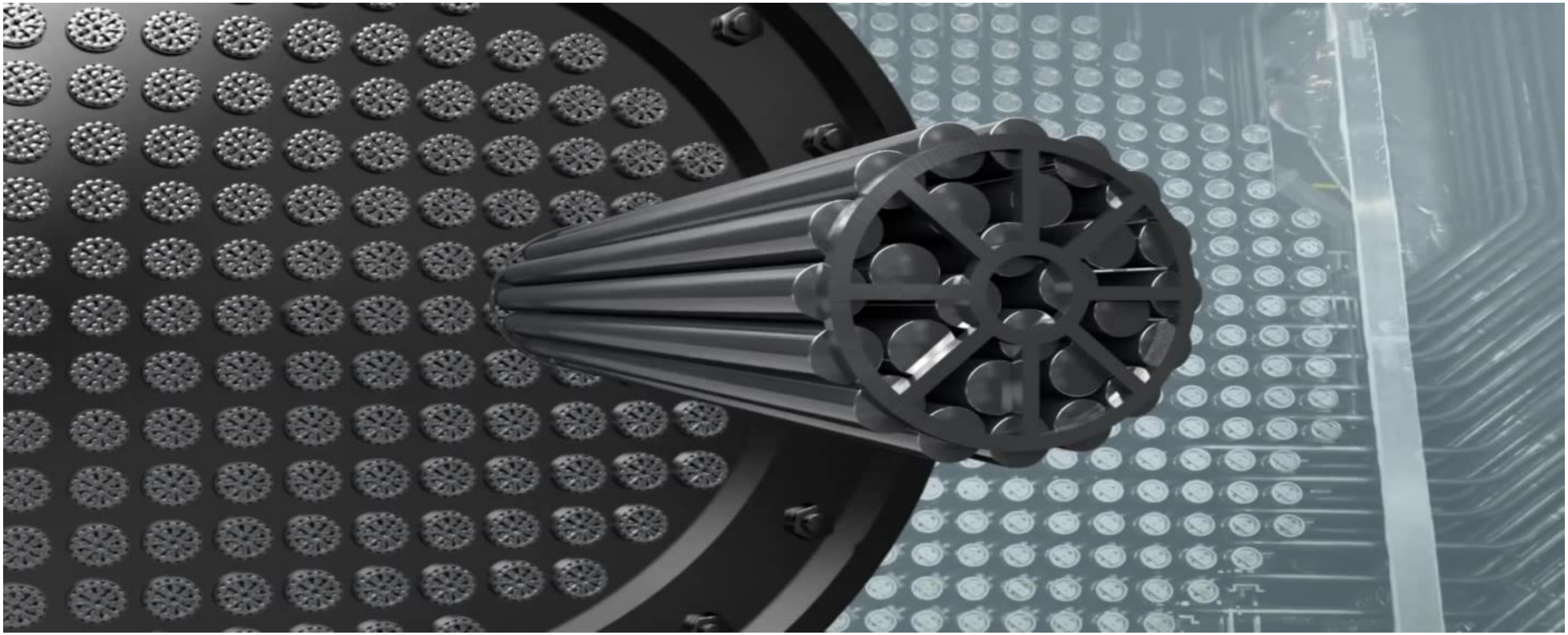
- CANDU-6
  - Power: ~ 700 MWe

A comparison of principal CANDU Heat Transport System Parameters			Heat Transport System Conditions					Heat Transport Pumps			Steam Generators		
CANDU 6 Operating stations or under construction	Electrical Output (MW) Gross/Net	Number of Fuel Channels	Elements in Fuel Bundle	Number of Loops	Outlet header Pressure (MPa)	Maximum Channel Flow (kg/s)	Outlet Header Quality (%)	Total	Operating	Motor Rating (kW)	Area (m^2) per Steam Generator	Integral Preheater	Steam Pressure (MPa)
Point Lepreau,	680/633	380	37	2	10.0	24	4	4	4	6700	3200	Yes	4.7
Gentilly 2	675/638	380	37	2	10.0	24	4	4	4	6700	3200	Yes	4.7
Wolsong 1	678/638	380	37	2	10.0	24	4	4	4	6700	3200	Yes	4.7
Embalse	648/600	380	37	2	10.0	24	4	4	4	6700	2800	Yes	4.7
Cernavoda 1, 2	710/665	380	37	2	10.0	24	4	4	4	6700	3200	Yes	4.7
Wolsong 2, 3, 4	715/668	380	37	2	10.0	24	4	4	4	6700	3200	Yes	4.7
Qinshan 1, 2	728/668	380	37	2	10.0	24	4	4	4	6700	3200	Yes	4.7
Other CANDU operating stations													
Pickering A 4 Units	542/515	390	28	2	8.7	23	-	16	12	1420	1850	Yes	4.1
Bruce A 4 Units	904/840	480	37	1	9.1	24	<1	4	4	8200	2400	No	4.4
Pickering B 4 Units	540/516	380	28	2	8.7	23	-	16	12	1420	1850	Yes	4.1
Bruce B 4 Units	915/860	480	37	1	9.1	24	<1	4	4	8200	2400	No	4.7
Darlington 4 Units	936/881	480	37	2	10.0	25	2	4	4	9600	4900	Yes	5.1

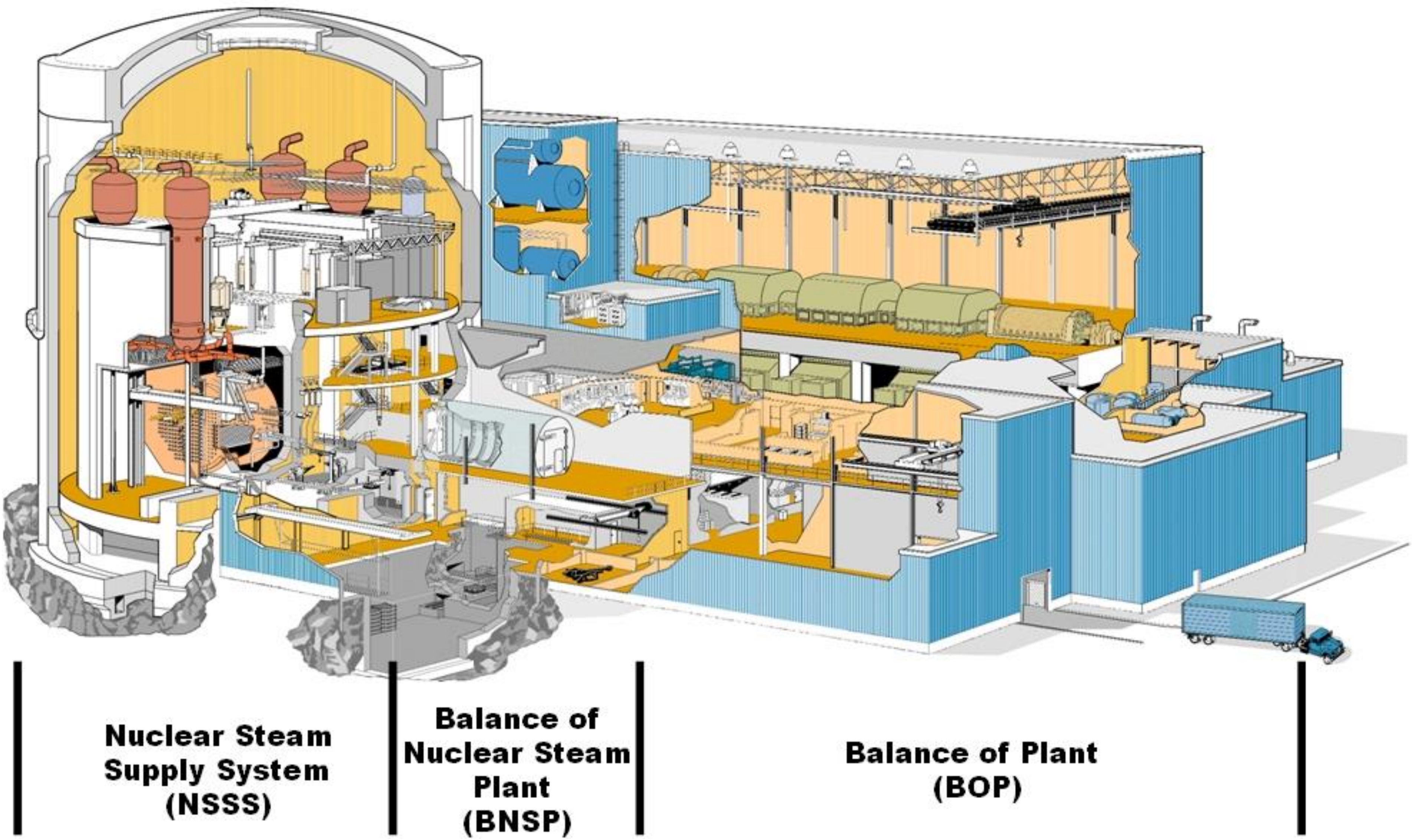
Overall

- Design features and unique characteristics
  - A reactor core comprising several hundred small diameter fuel channels rather than one huge pressure vessel.
  - Heavy water (D_2O) for moderator and coolant
  - Separate low pressure moderator and high pressure fuel cooling systems
  - On-power refuelling
  - Reactivity devices that are located in the cool low pressure moderator, and not subjected to high temperatures or pressures
  - Natural uranium fuel or other low fissile content fuel
  - Two fully capable safety shutdown systems, independent from each other and the reactor regulating system.

Overall

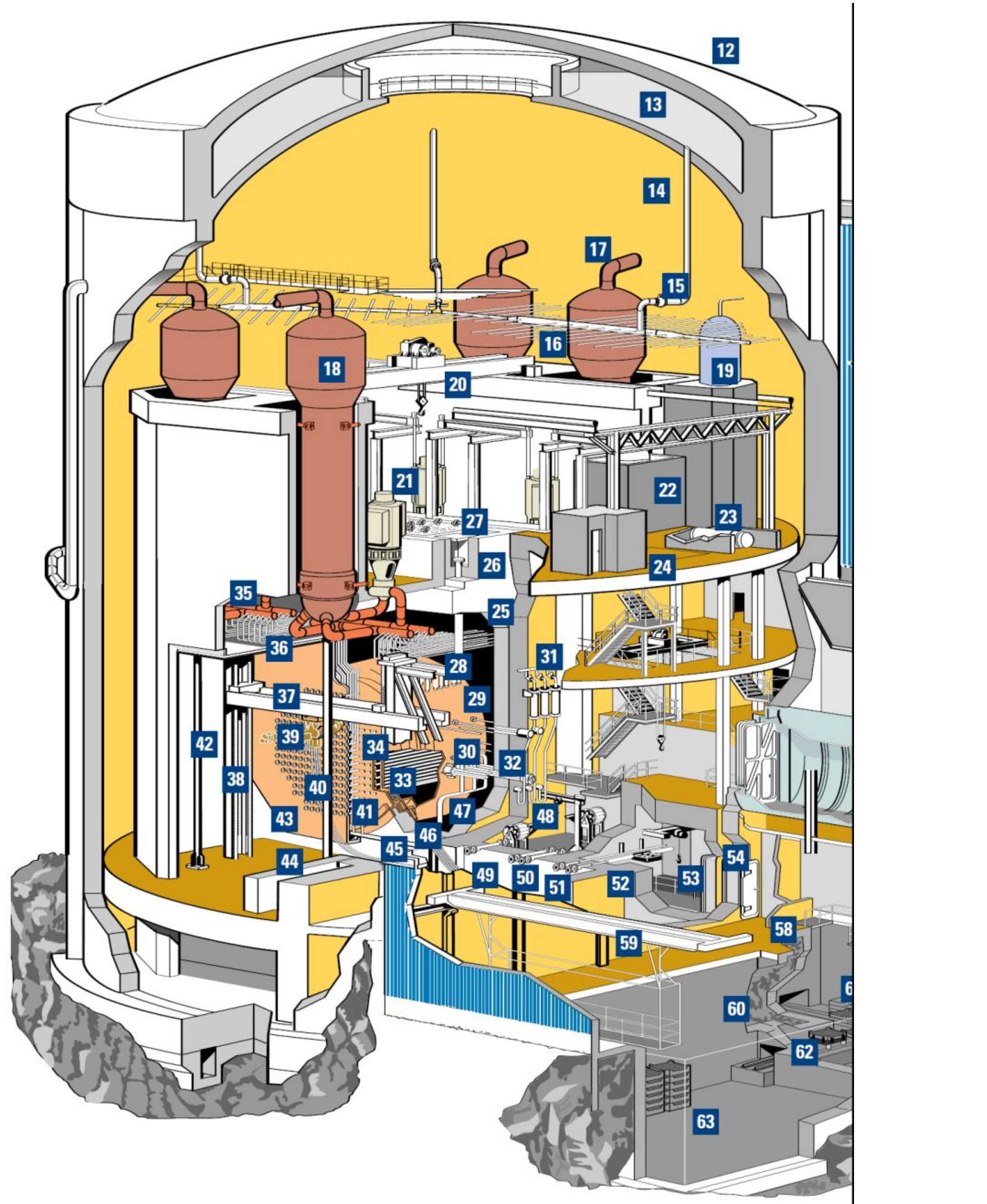


❖ **Overall**

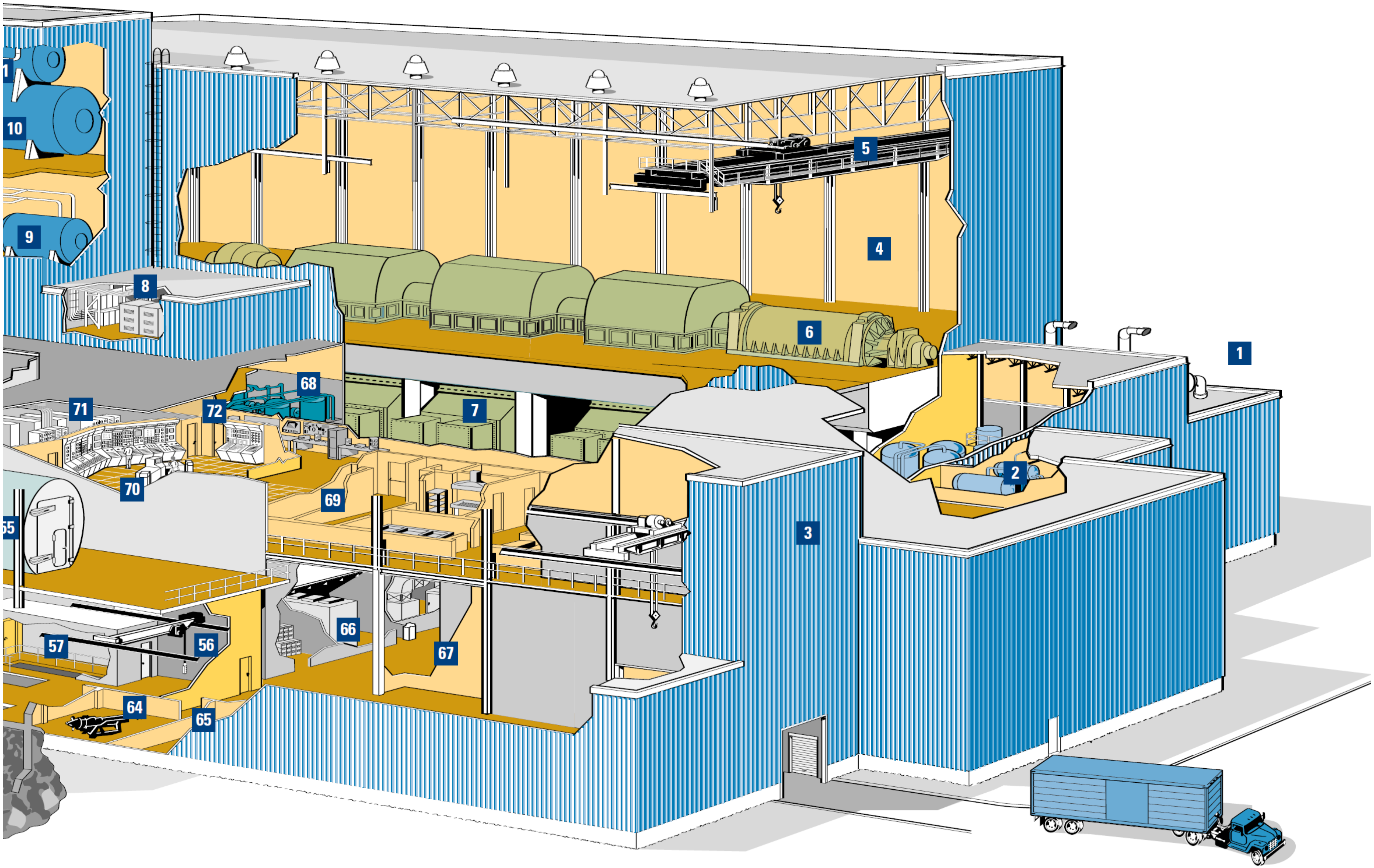

Nuclear Steam Supply System (NSSS)
Balance of Nuclear Steam Plant (BNSP)
Balance of Plant (BOP)

Overall

1. Diesel room
2. Water treatment plant *
3. Crane hall
4. Turbine building
5. Turbine building crane
6. Generator
7. Condenser
8. Battery room
9. Boiler feed water tanks
10. Deaerator storage tank
11. Deaerator
12. Reactor building
13. Dousing tank
14. Dousing water supply pipes
15. Dousing water valves
16. Dousing water spray nozzles
17. Steam pipes
18. Steam generators
19. Pressurizer
20. Crane
21. Heat transport pumps
22. Bleed condenser
23. Bleed cooler
24. Hatch
25. Reactor vault
26. Pressure relief pipes
27. Reactivity mechanism deck
28. Reactivity mechanism guide tubes
29. Calandria
30. Poison injection nozzles
31. Poison tanks
32. Ion chambers
33. Fuel channel assemblies
34. End shield
35. Headers
36. Feeder pipes
37. Fuelling machine bridge
38. Bridge support column
39. Fuelling machine
40. Catenary
41. Fuel channel end fittings
42. Steam generator support column
43. Feeder pipe insulation cabinet
44. Fuelling machine vault door
45. End shield cooling
46. Fuelling machine track
47. Moderator inlet pipe
48. New fuel handling machine
49. New fuel port
50. Fuelling machine service ports
51. Rehearsal facility
52. Spent fuel port
53. Spent fuel elevator
54. Entrance to spent fuel area
55. Airlock
56. Crane
57. Spent fuel shipping area
58. Spent fuel handling area
59. Spent fuel bay gantry
60. Spent fuel bay
61. Spent fuel transfer baskets
62. Spent fuel transfer trolley
63. Spent fuel storage baskets
64. Fuelling machine maintenance area
65. Decontamination room
66. New fuel storage
67. Tool crib
68. Vapour recovery equipment
69. Office
70. Control room *
71. Control equipment room
72. Computer room



Overall

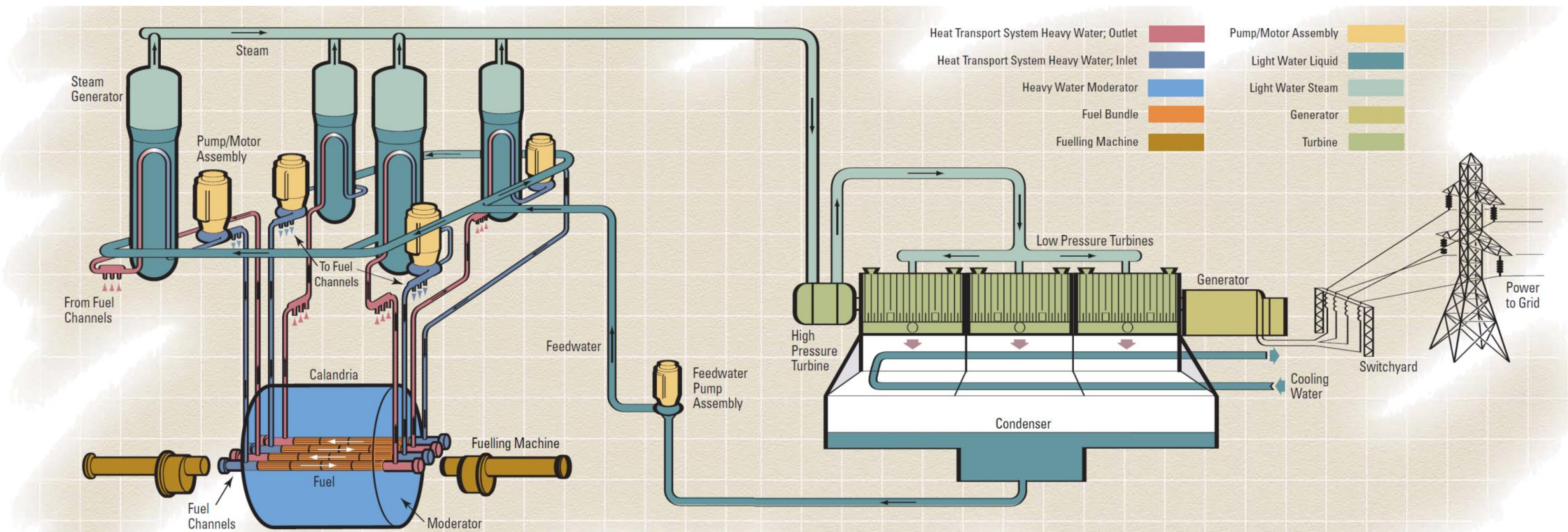


Nuclear Steam Supply System

- System configuration: 2 loops, 4 SGs
- Reactor
  - Thermal output: 2064 MW(th)
  - Coolant flow rate: 7700 kg/s
  - Operating inlet and outlet pressure: 11.75/10.0 MPa
  - Inlet and outlet temperature: 266°C/310°C

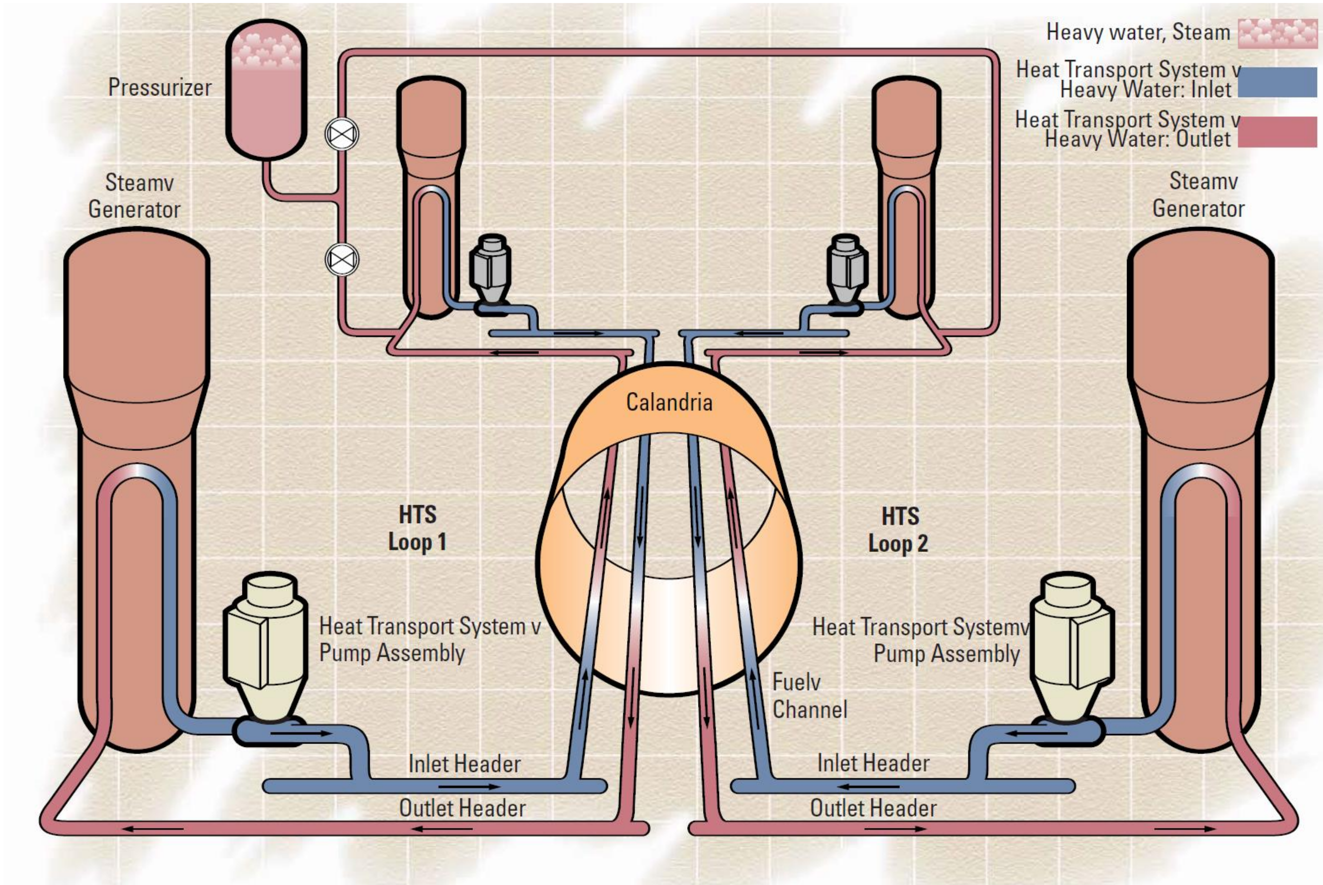
DP in CANDU > 6.2 bar in PWR

$T_{coolant}$ in CANDU < 325°C in PWR

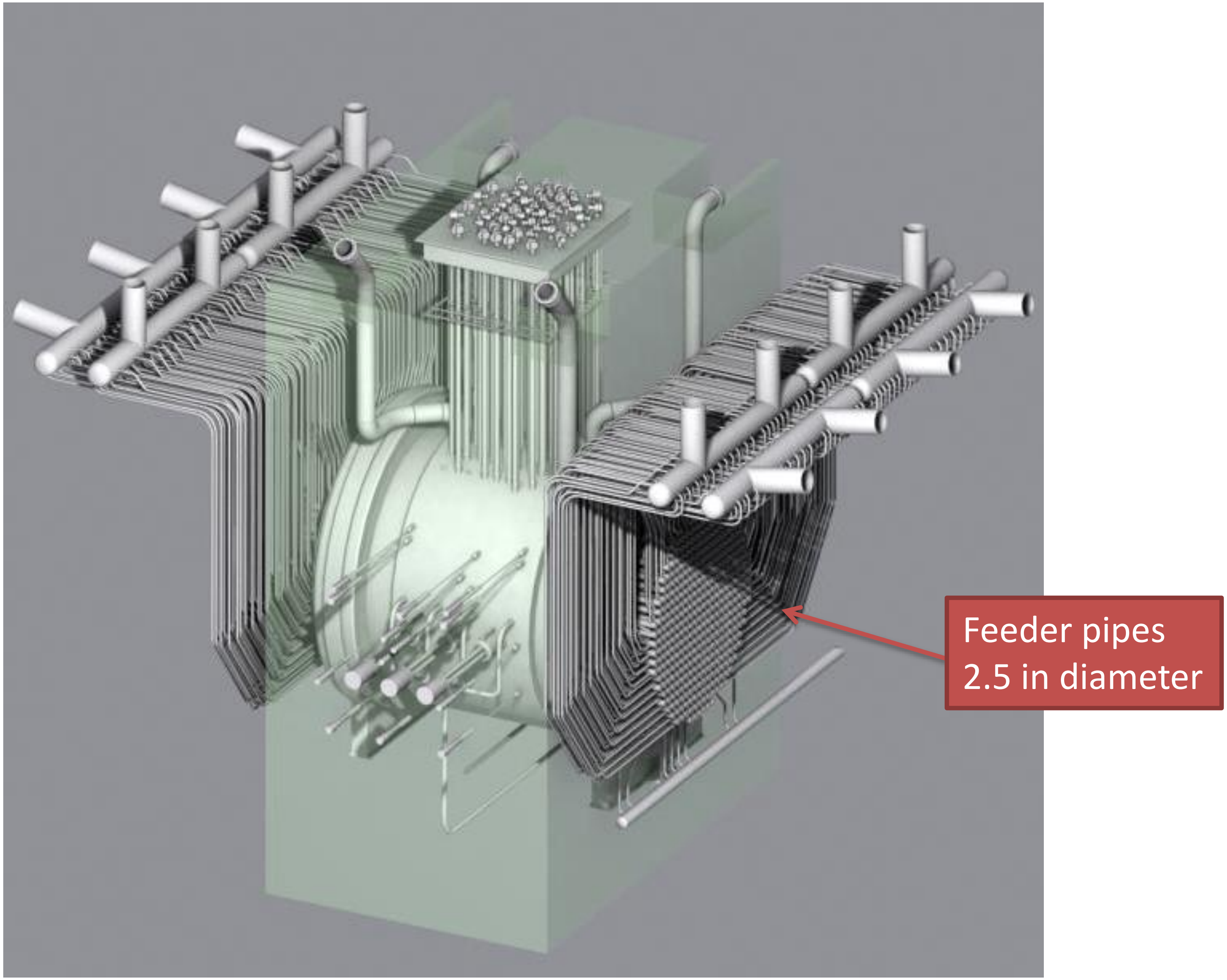


❖ Heat Transport System (Primary system)

- Two parallel coolant loops
  - Inlet headers ⇒ feeder pipes ⇒ fuel channels ⇒feeder pipes ⇒ outlet headers
- Pressurizer
- SG: U-tube shape



- **Reactor**

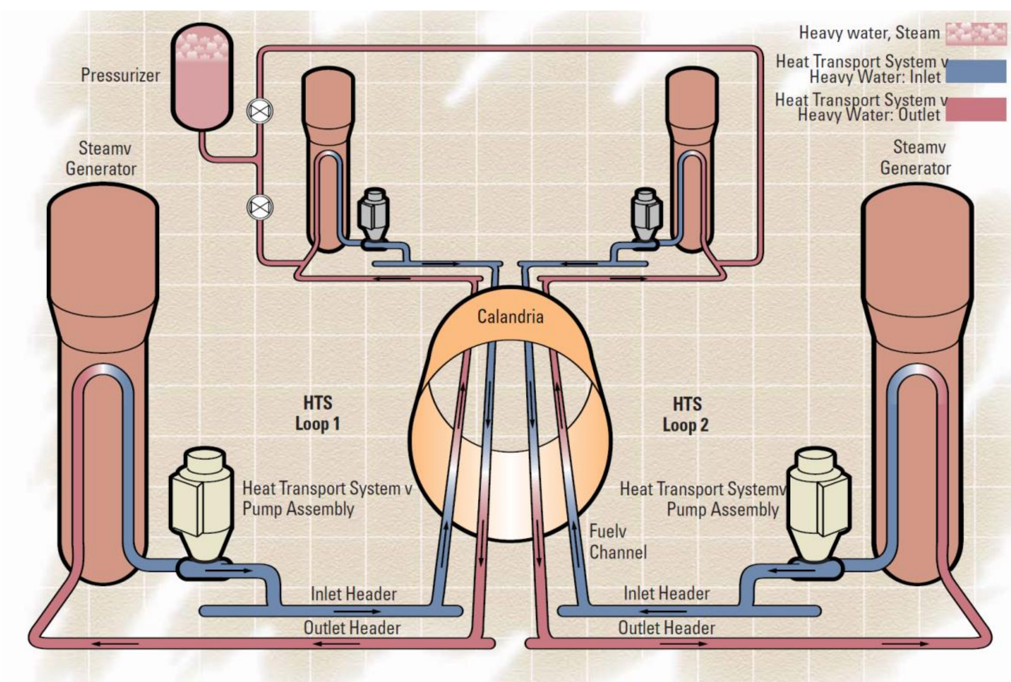


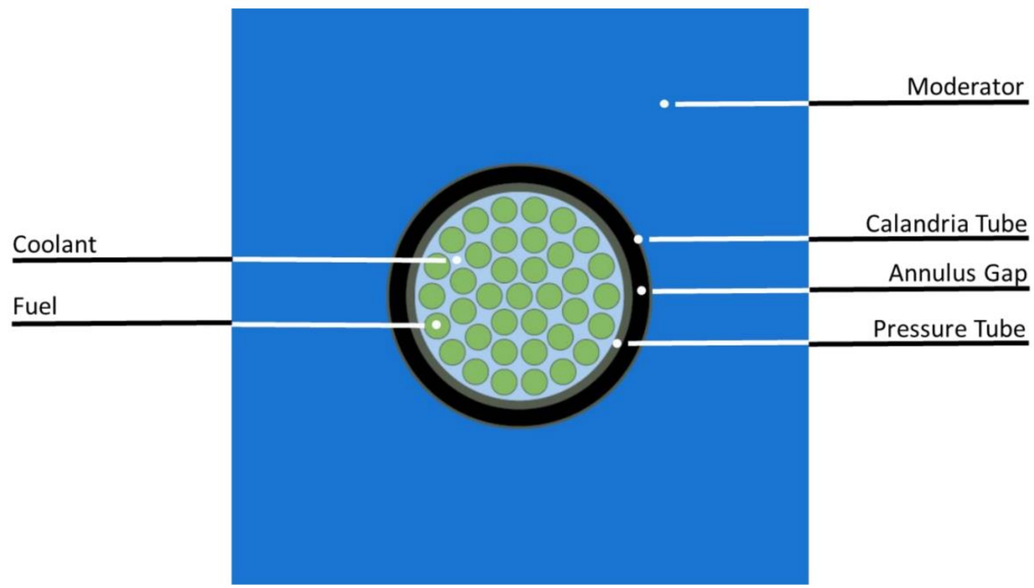
- **Heat transport system**
  - Inventory: 192 ton for whole HTS, in reactor ?
  - 2 loop

OPR1000: 206 ton
APR1400: 330 ton

- **Moderator system**
  - Inventory: 265 ton

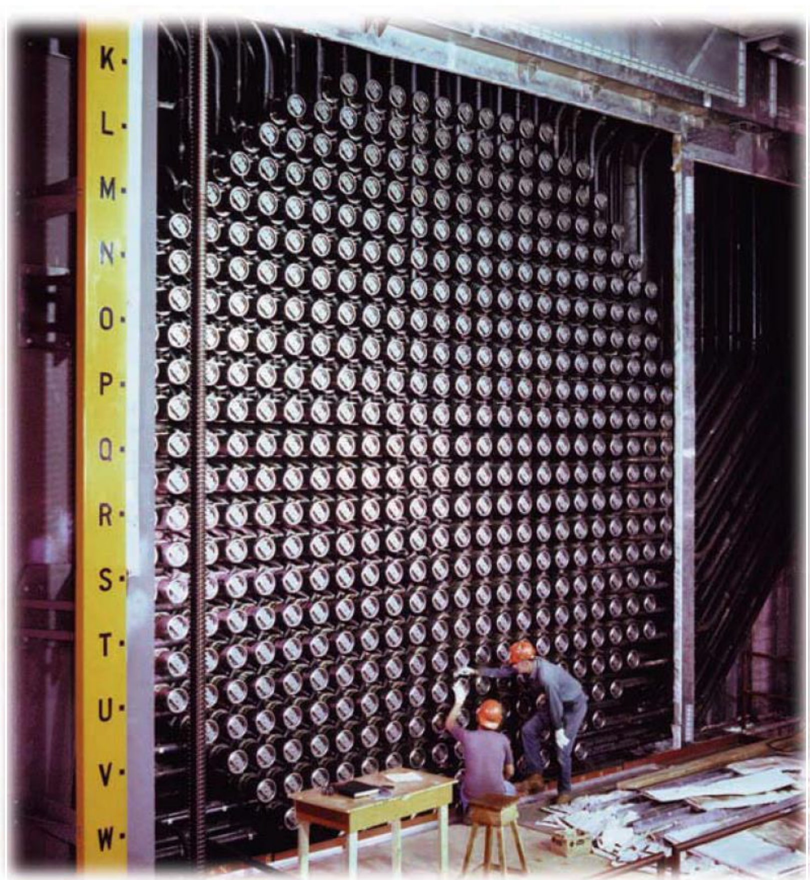
Large moderator-to-coolant ratio
In lattice cell, 4 % coolant + 83 % moderator in volume

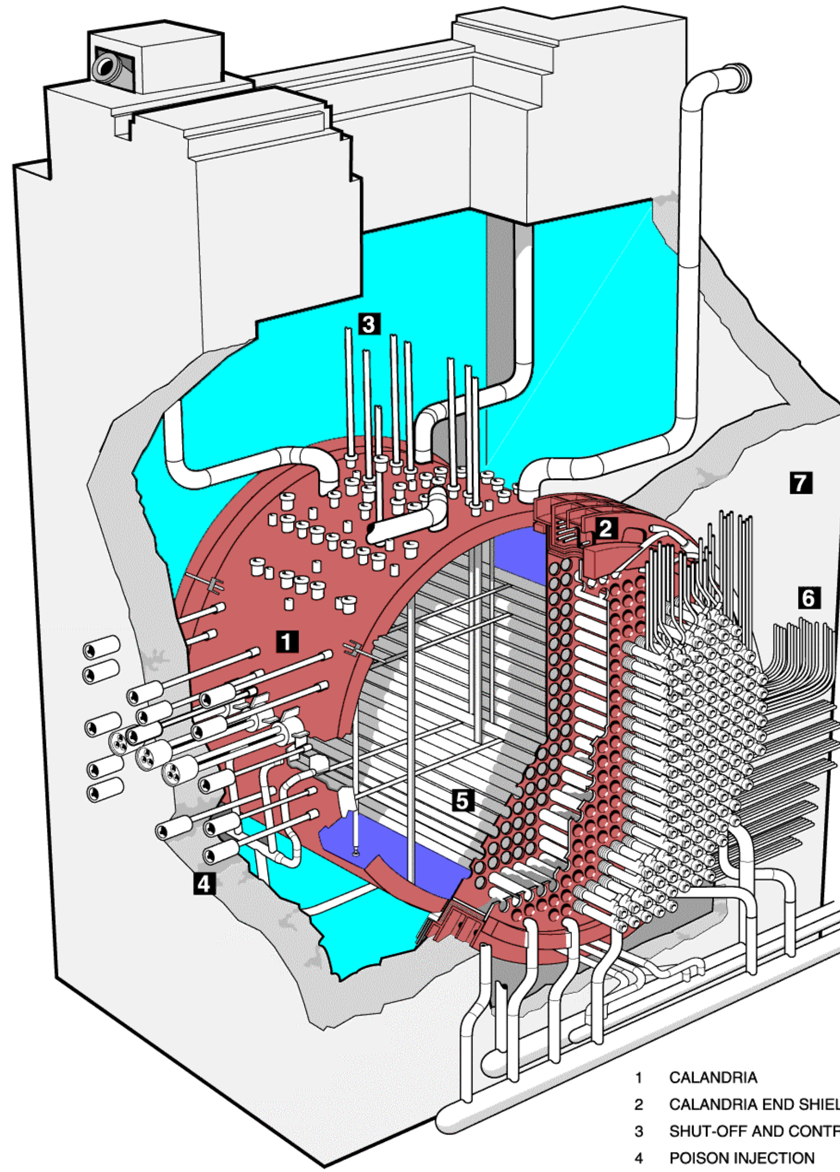




CANDU

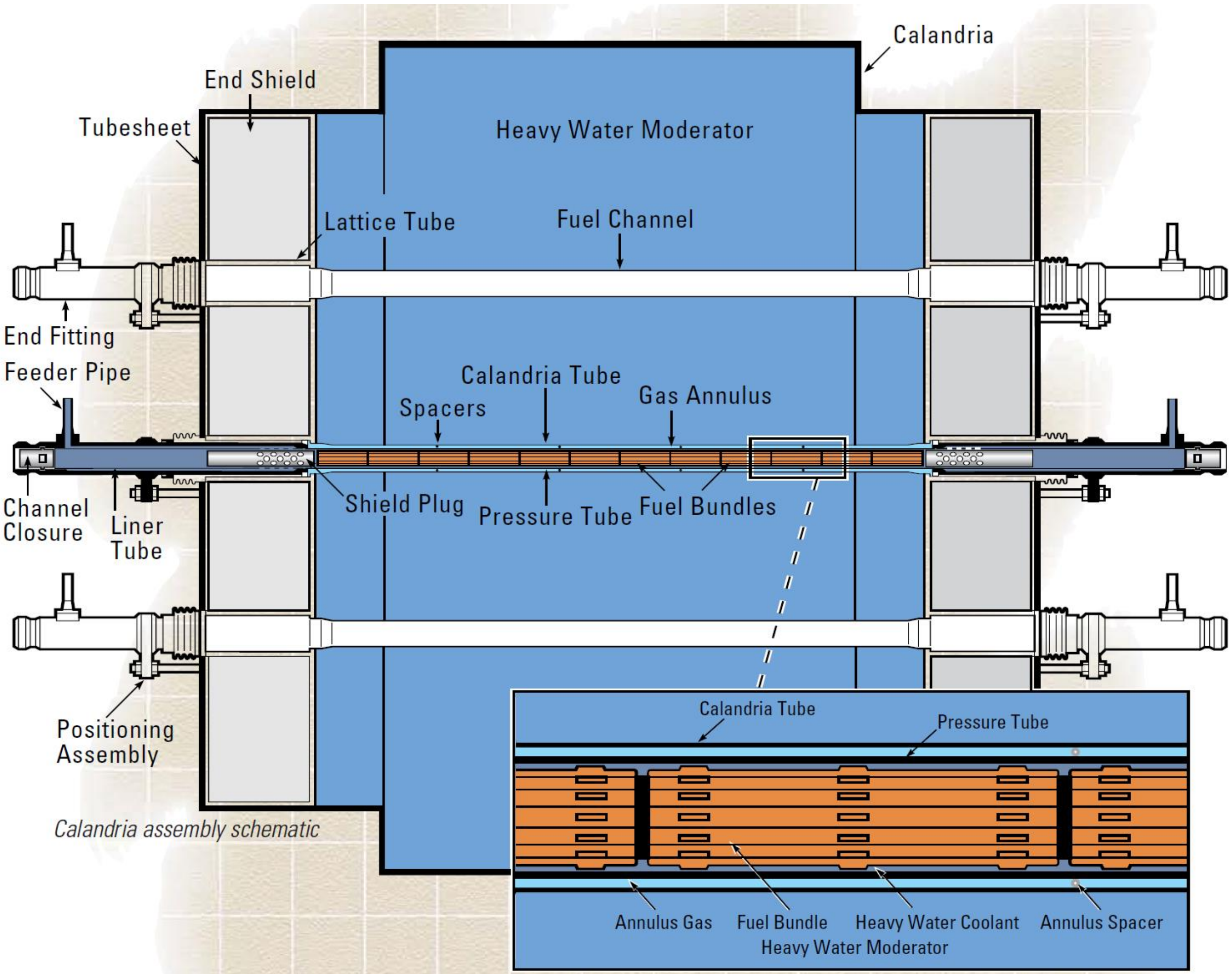
- **Reactor**
  - Calandria (calandria tank)
  - Horizontal fuel channel
  - Calandria vault (reactor vault)





Reactor

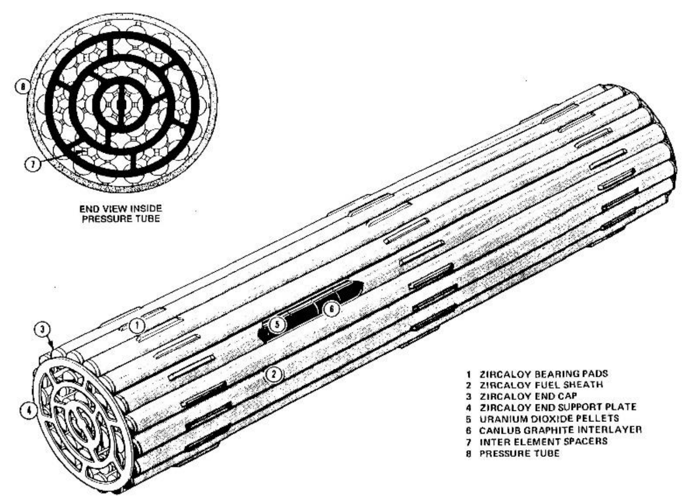
- 380 fuel channels ⇒ 12 fuel bundles ⇒ 37 elements
- Calandria length: 5.94 m, shell diameter: 7.6 m, subshell diameter: 6.7 m



Calandria assembly schematic

Fuel channel arrangement


37 Element Fuel Bundle



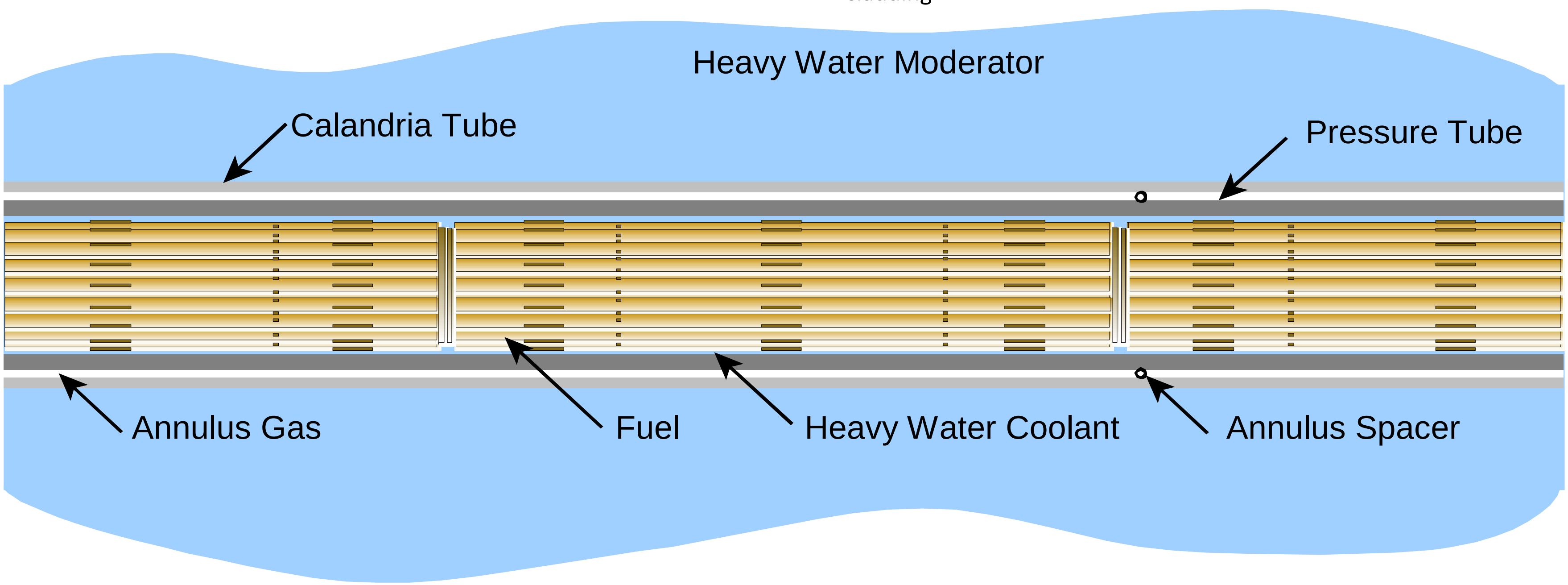
Fuel channel

- Calandria tube ⇒ pressure tube ⇒ fuel bundle
- Annulus gas: CO_2 for thermal insulation between calandria tube and pressure tube
  - Outside of calandria tube: ~ 70 °C, atmospheric pressure
  - Inside of pressure tube: ~300°C, 10 MPa
- Calandria tube outer diameter: 132 mm
- Pressure tube inner diameter: 103.4 mm
- Fuel bundle outer diameter: 102.4 mm
- Fuel sheath diameter: 13.1 mm

Annulus gas pressure ?

125~200 kPa in normal operation

$D_{cladding}$ in CANDU > 9.5mm in PWR



❖ Replacement of pressure tube

〈표 2-29〉 압력관 전량교체 사례(2011년말 현재)

발전소	준공년도	교체 시기	비고
피커링 1호기	'71	'87	교체완료
피커링 2호기	'71	'88	교체완료
피커링 3호기	'72	'91	교체완료
피커링 4호기	'73	'93	교체완료
월성1호기	'83	'09	교체완료
부루스 2호기	'77	'06~	교체 중
부루스 1호기	'77	'07~	교체 중
포인트레프르	'83	'08~	교체 중

3. 압력관에서의 재료열화 및 손상
3.1. 수소 및 수소화물(hydride)에 의한 열화, 손상
3.2. 중성자 조사에 의한 열화, 손상
3.3. 부식에 의한 열화, 손상

홍준화, "원자력재료", 한스하우스, 2012.

중수로 압력관의 안전성 평가를 위한
금속학적 영향 평가, KINS/HR-1043, 2008.

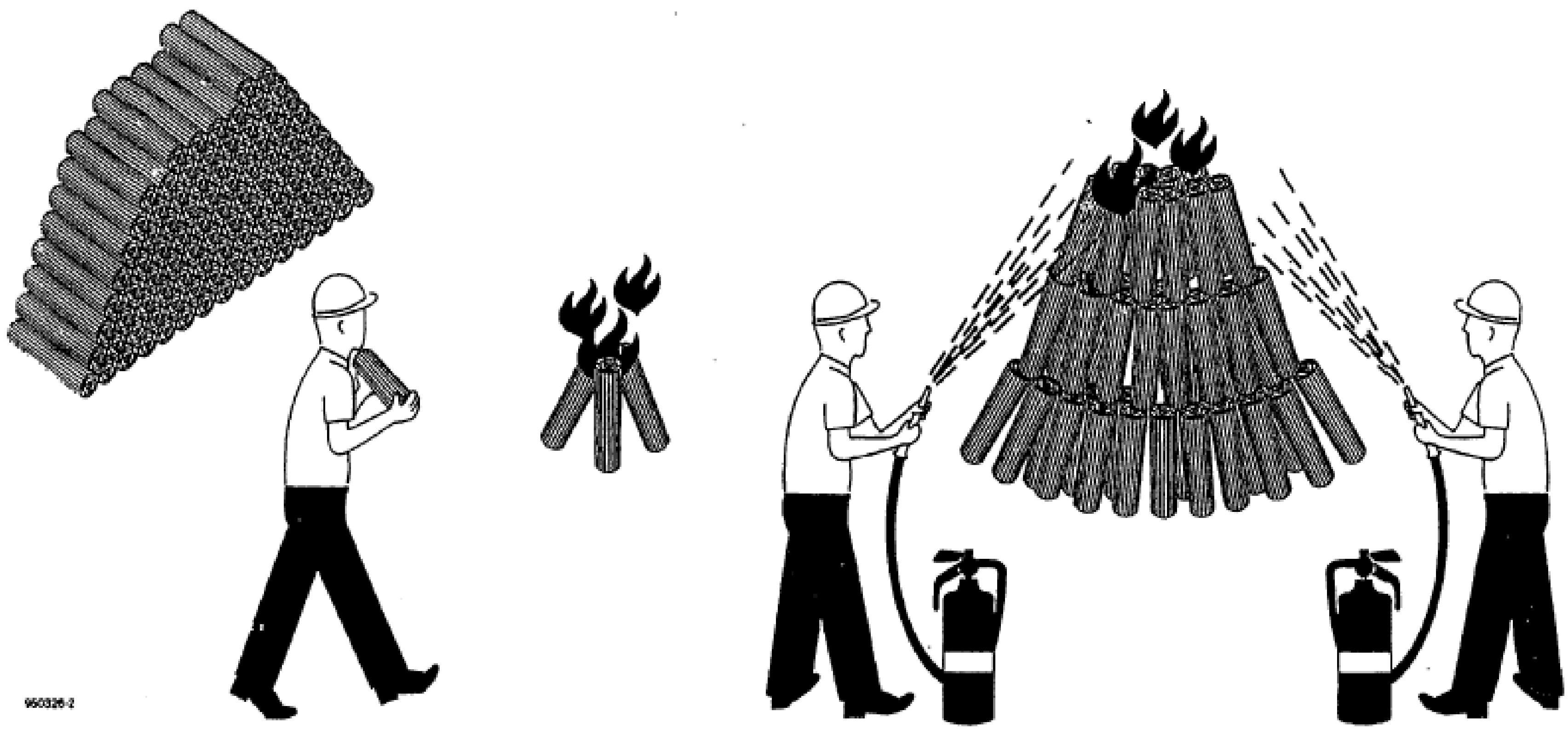
가장 많은 사고 유형

- 압력관 설치 시 부적절한 압연접합(rolled joint) 방법에 의해 형성된 과도한 인장잔류응력이 지연수소균열(DHC)을 유발하여 균열이 발생되고 이것이 전파된 경우로, Pickering-3에서 2회, Pickering-4, Bruce-2호기에서의 사고가 이에 해당

기타 사고 유형

- 압력관의 크리프변형에 의한 처짐(sagging)으로 칼란드리아관과 접촉하고 수소화물과 블리스터(blister) 형성에 따른 균열로, Pickering-2 등에서의 사고가 이에 해당
- 압력관 제조 시 압출과정 등에서 생긴 결함과 연계된 DHC 기구에 의한 균열의 생성 및 전파의 경우로, 1986년 Bruce-2에서의 사고가 이에 해당

- **Fuel handling**
  - Reactivity-Control Philosophies

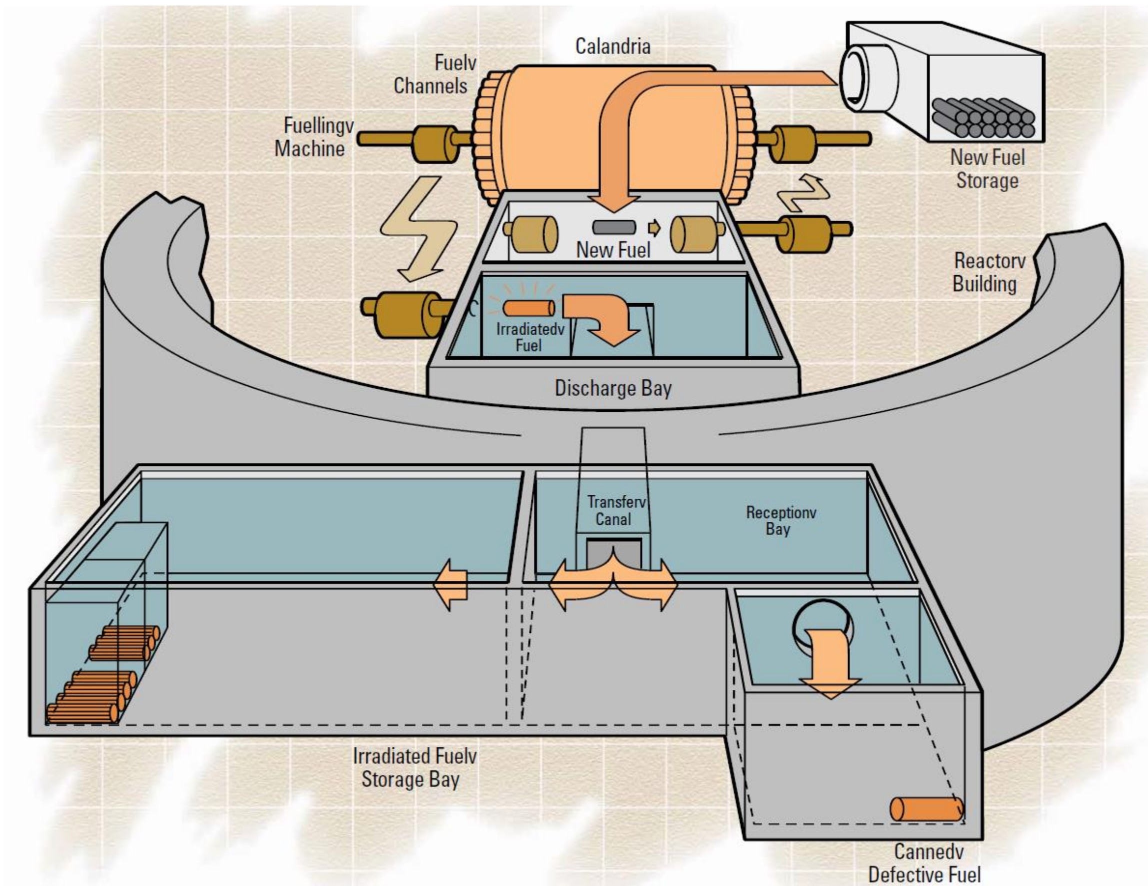


CANDU
Fuel added as needed

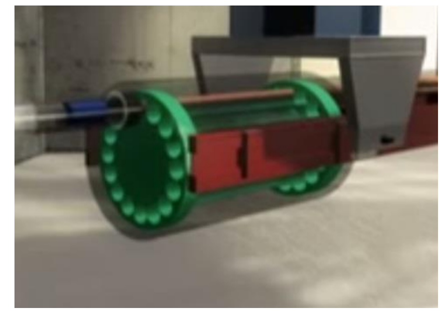
LWR
18 months fuel added at once.
Burning rate suppressed.

Fuel handling

- Refuels the reactor remotely while it is operating
- Transfers the irradiated fuel remotely from the reactor to the storage bay

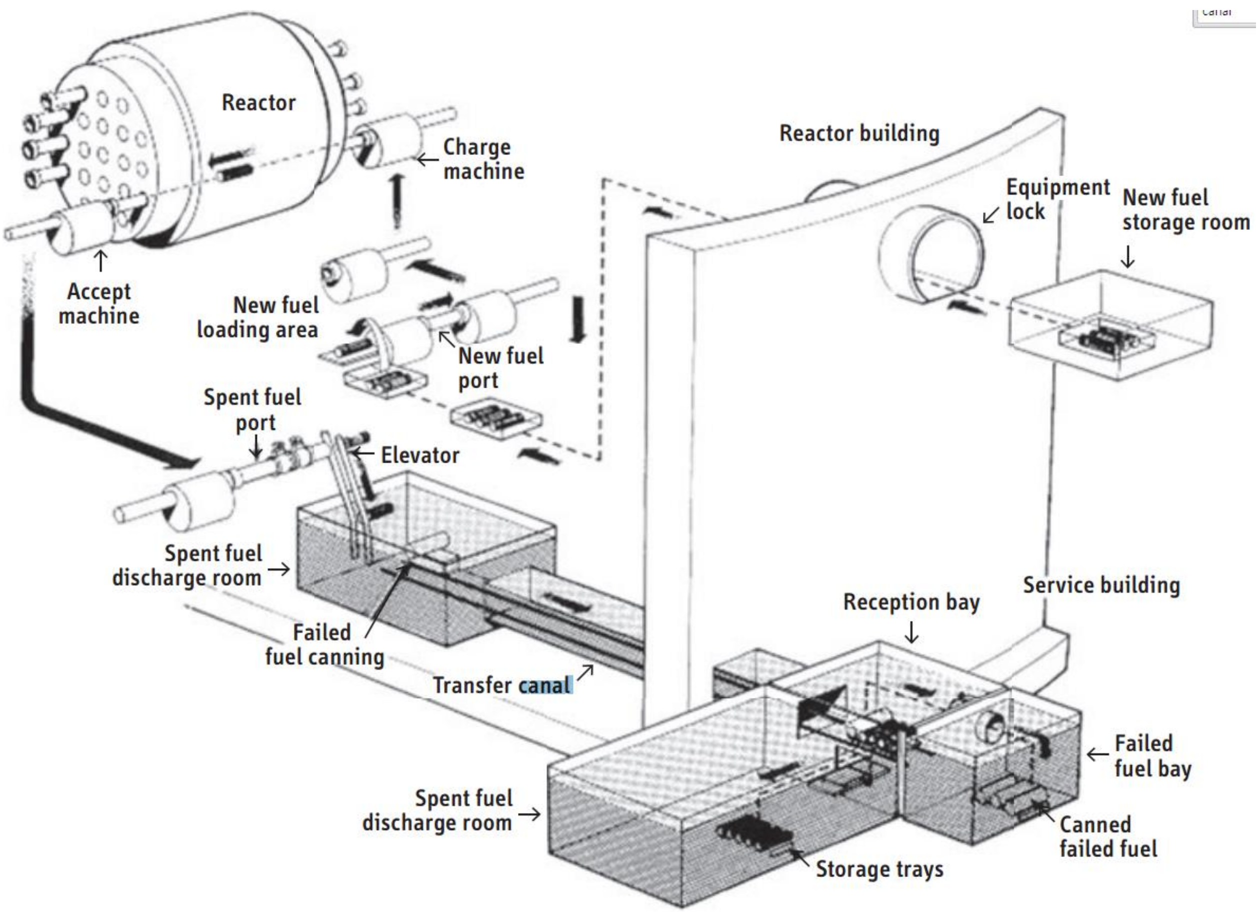


Fuelling Machine Magazine



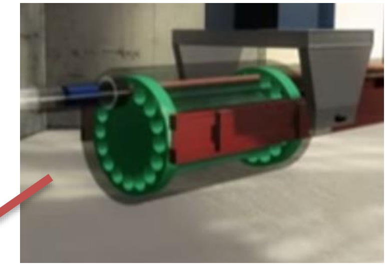
Fuel handling

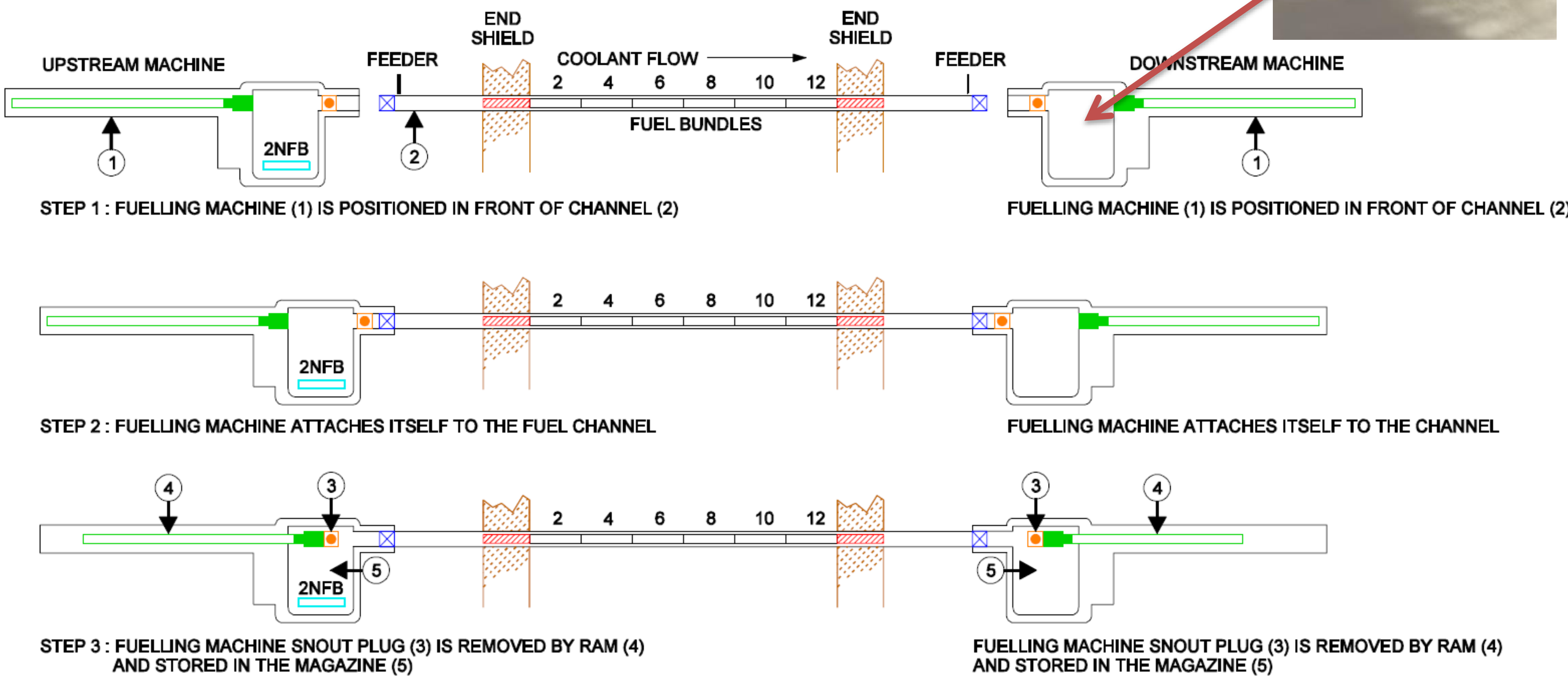
- Refuels the reactor remotely while it is operating
- Transfers the irradiated fuel remotely from the reactor to the storage bay
  - Snout plug, closure plug, shield plug/support plug, ram



Fuel handling

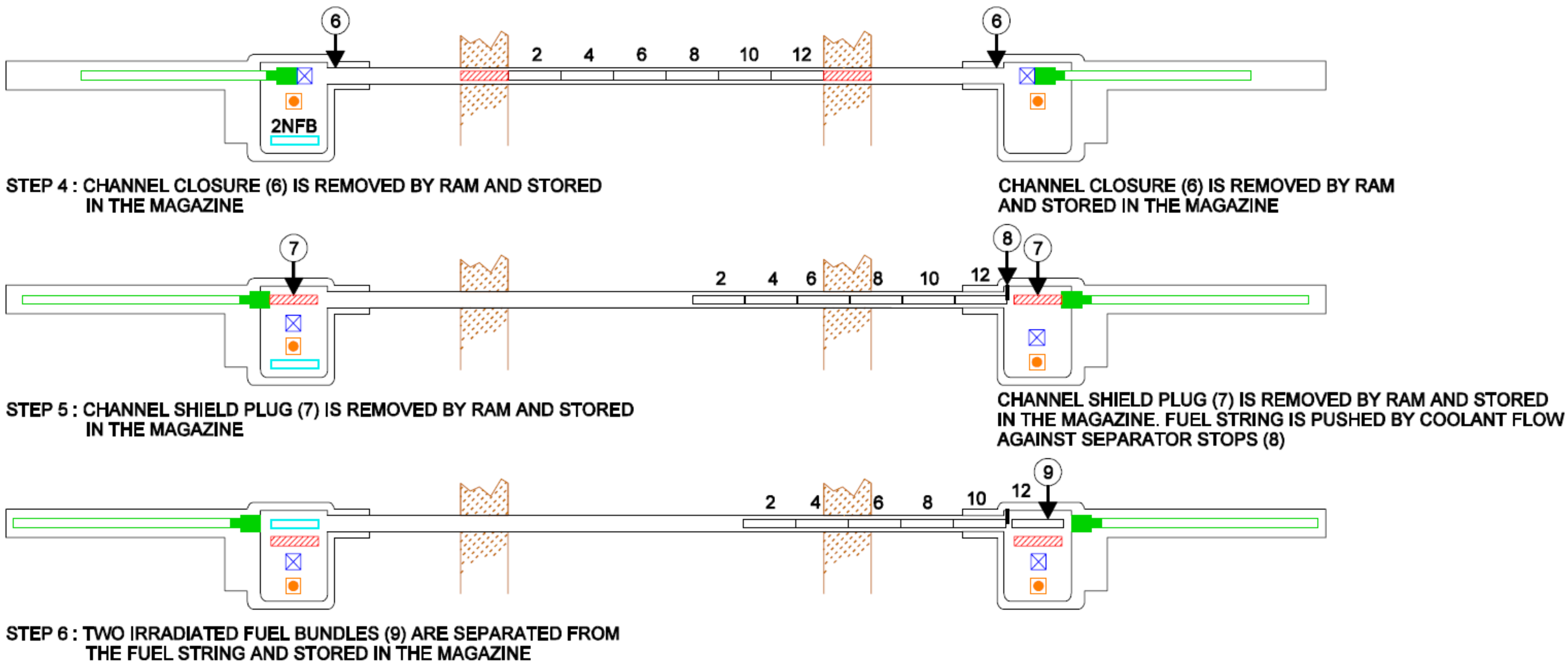
- Fuel replacement sequence





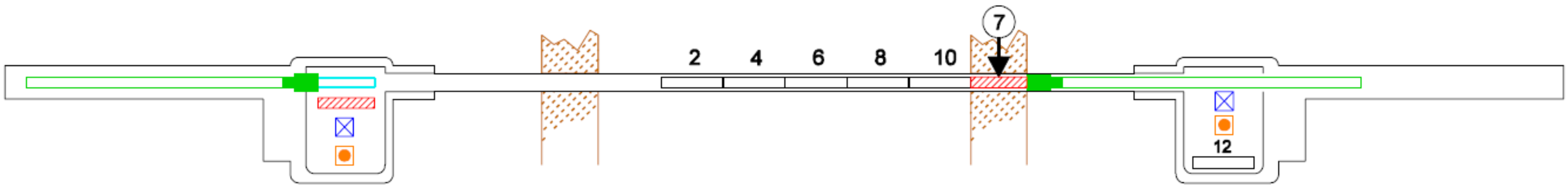
Fuel handling

- Fuel replacement sequence

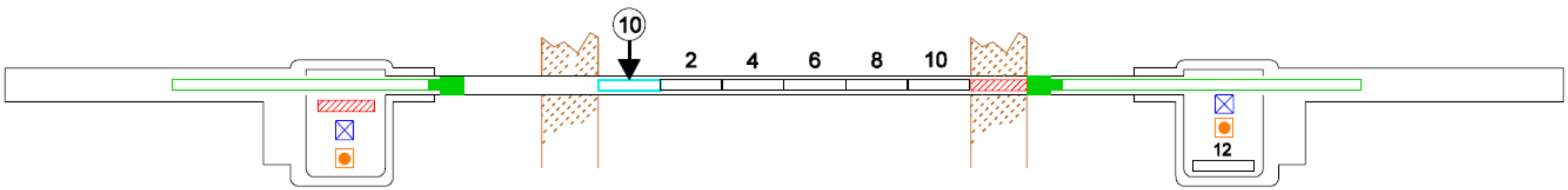


Fuel handling

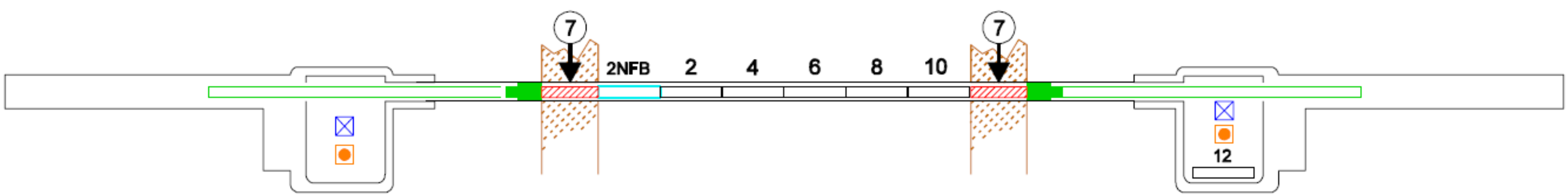
- Fuel replacement sequence



STEP 7 : RAM PUSHES THE SHIELD PLUG (7) INTO POSITION
WHICH PUSHES THE FUEL STRING BACK INTO POSITION



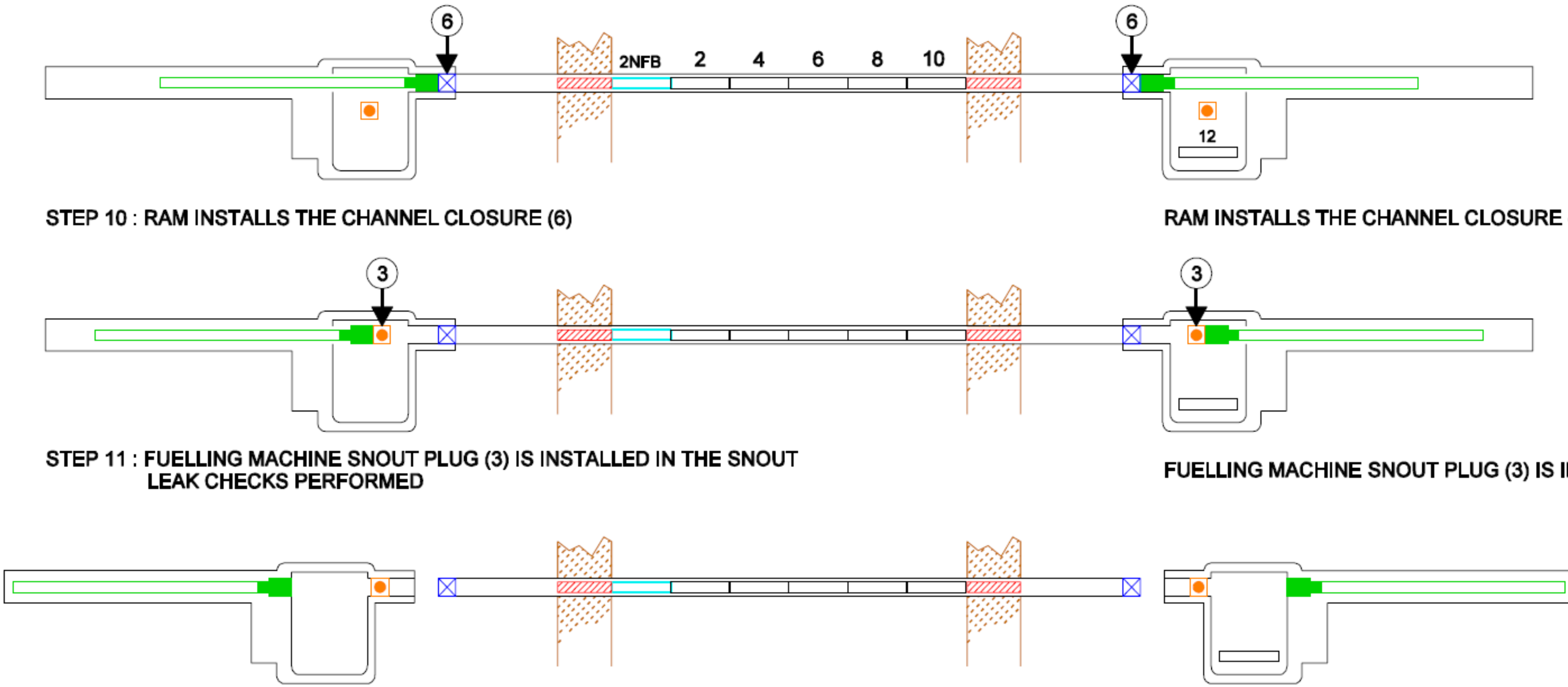
STEP 8 : RAM PUSHES TWO NEW FUEL BUNDLES (10) INTO THE FLOW



STEP 9 : EACH RAM INSTALLS THEIR RESPECTIVE SHIELD PLUGS (7)
THE PRESSURE DROP ACROSS THE CHANNEL IS MEASURED TO CHECK FLOW

Fuel handling

- Fuel replacement sequence



STEP 10 : RAM INSTALLS THE CHANNEL CLOSURE (6)

RAM INSTALLS THE CHANNEL CLOSURE (6)

STEP 11 : FUELLING MACHINE SNOUT PLUG (3) IS INSTALLED IN THE SNOUT
LEAK CHECKS PERFORMED

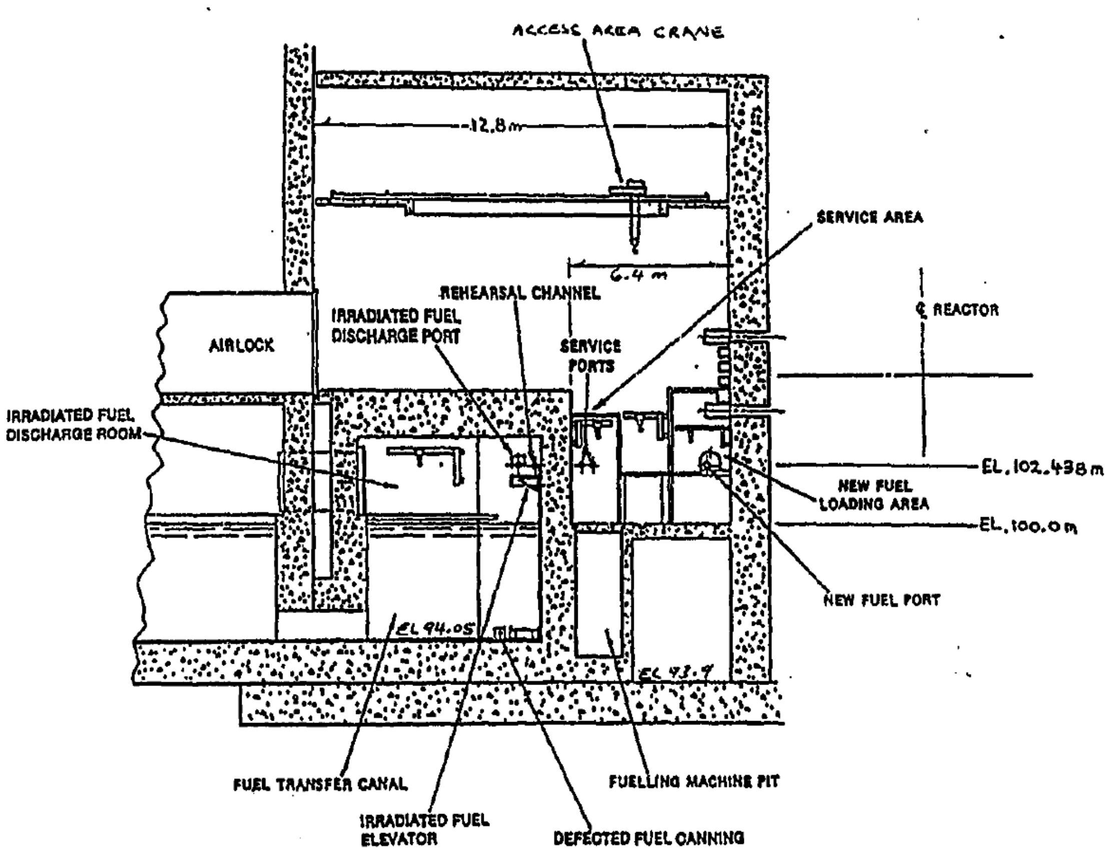
FUELLING MACHINE SNOUT PLUG (3) IS INSTALLED IN THE SNOUT

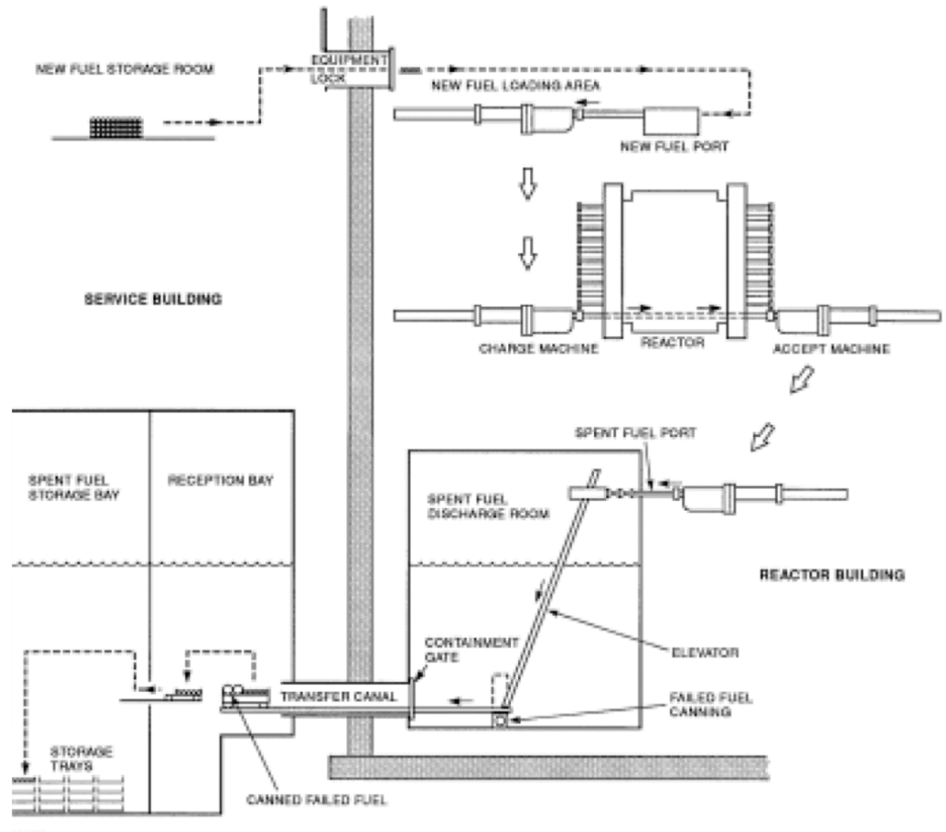
STEP 12 : FUELLING MACHINE DETACHES ITSELF FROM THE CHANNEL

FUELLING MACHINE DETACHES ITSELF FROM THE CHANNEL

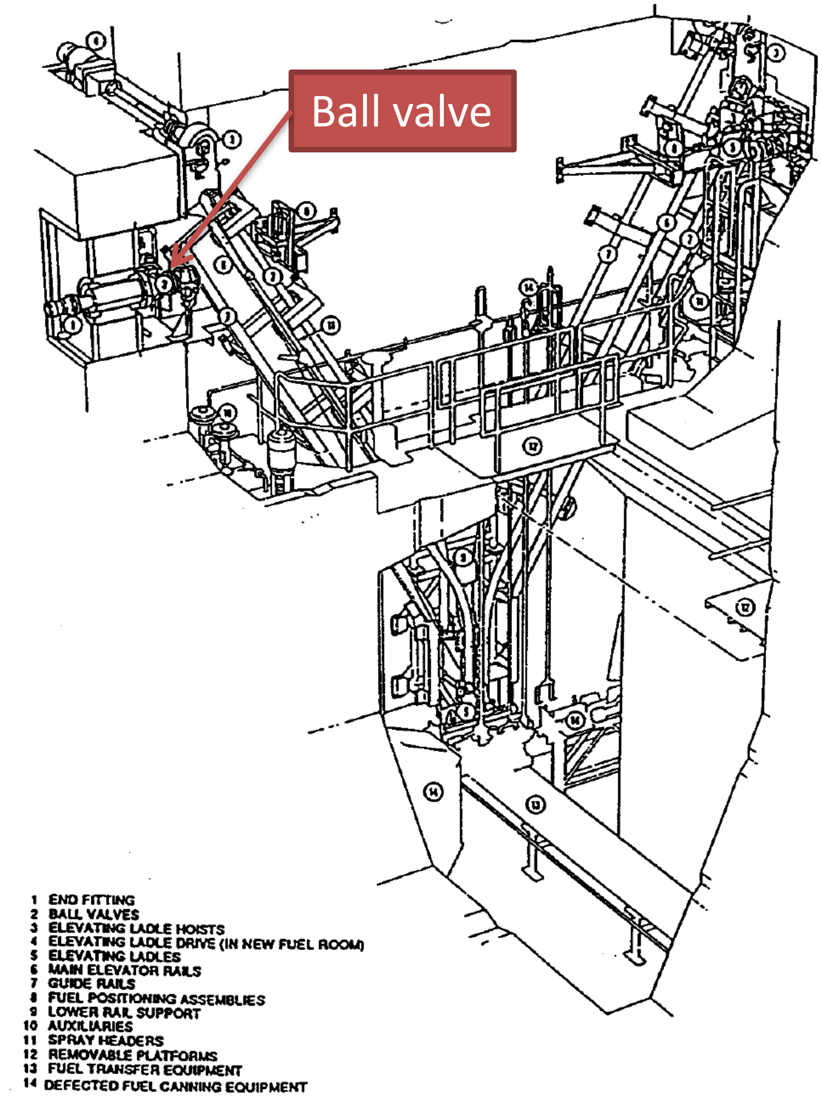
Fuel handling

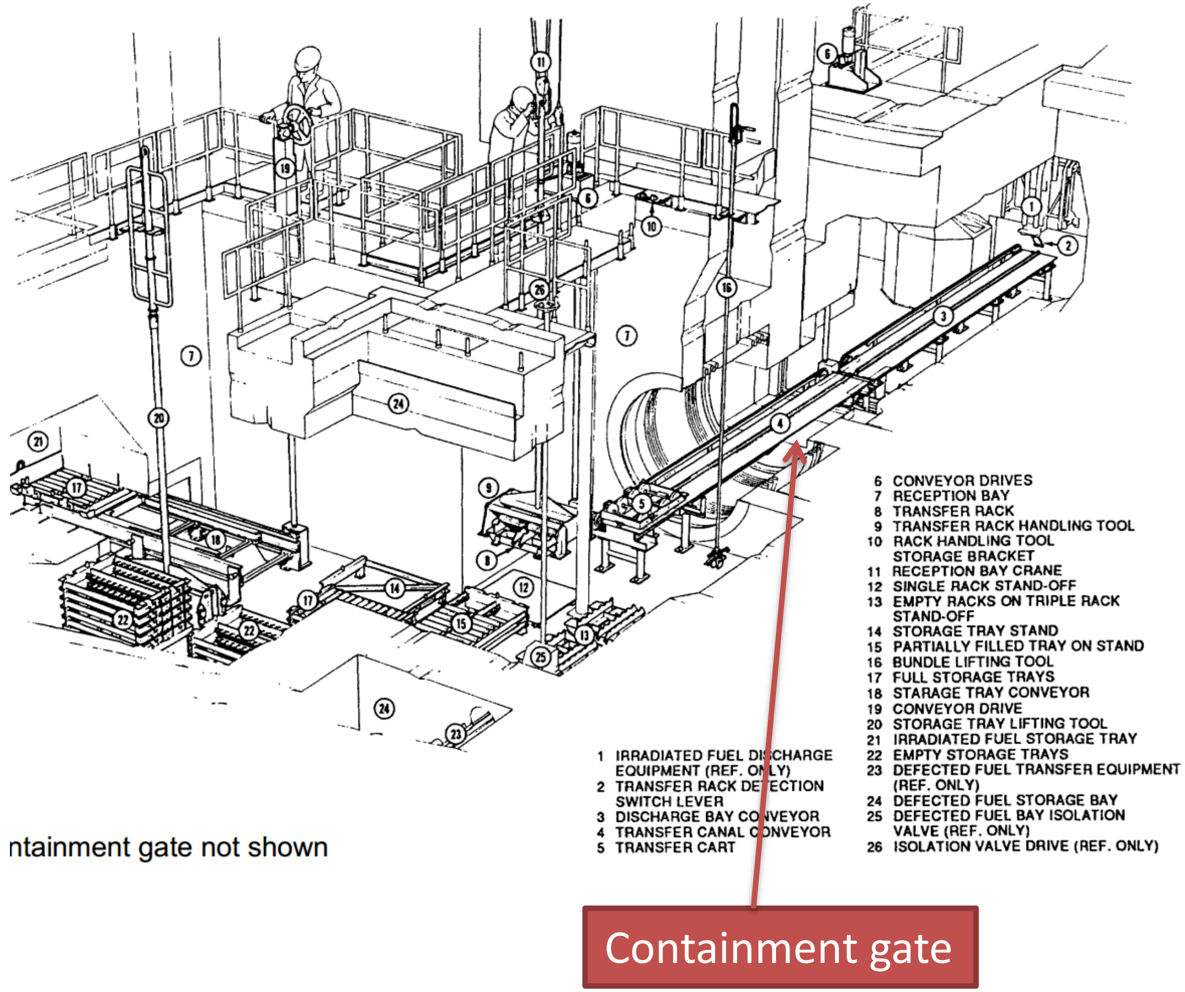
- During irradiated fuel transfer from the fuelling machine head, the containment boundary is at a containment gate at the end of the discharge bay canal.
- Earlier CANDU 6 units were not equipped with a containment gate and rely on the fuelling machine auxiliaries and the head of water in the discharge canal for the containment boundary. At all other times, the port valves are closed.





Fuel handling





CANDU

Fuel handling

- Issue of R-7
  - Requirements for Containment Systems for CANDU Nuclear Power Plants

- Pressure build-up in containment
  - LOCA
  - MSLB

Appendix 2.2 Systems Connected to Containment Atmosphere

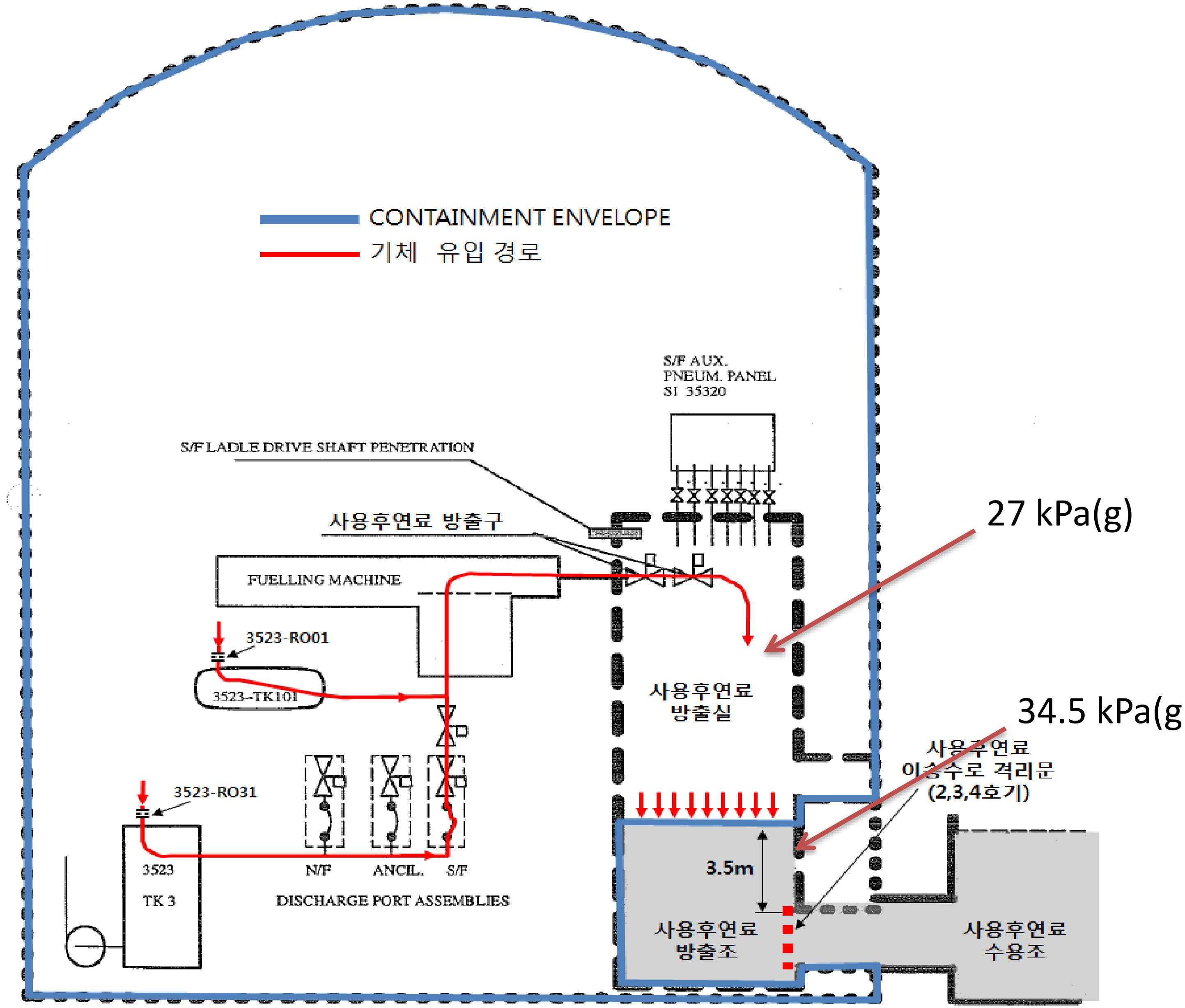
Each line that connects directly to the containment atmosphere, that penetrates the containment structure, and that is not part of a closed system, shall be provided with two isolation barriers as follows:

(a) two automatic isolation valves in series for those lines which may be open to the containment atmosphere;

(b) two closed isolation valves in series for those lines that are normally closed to the containment atmosphere;

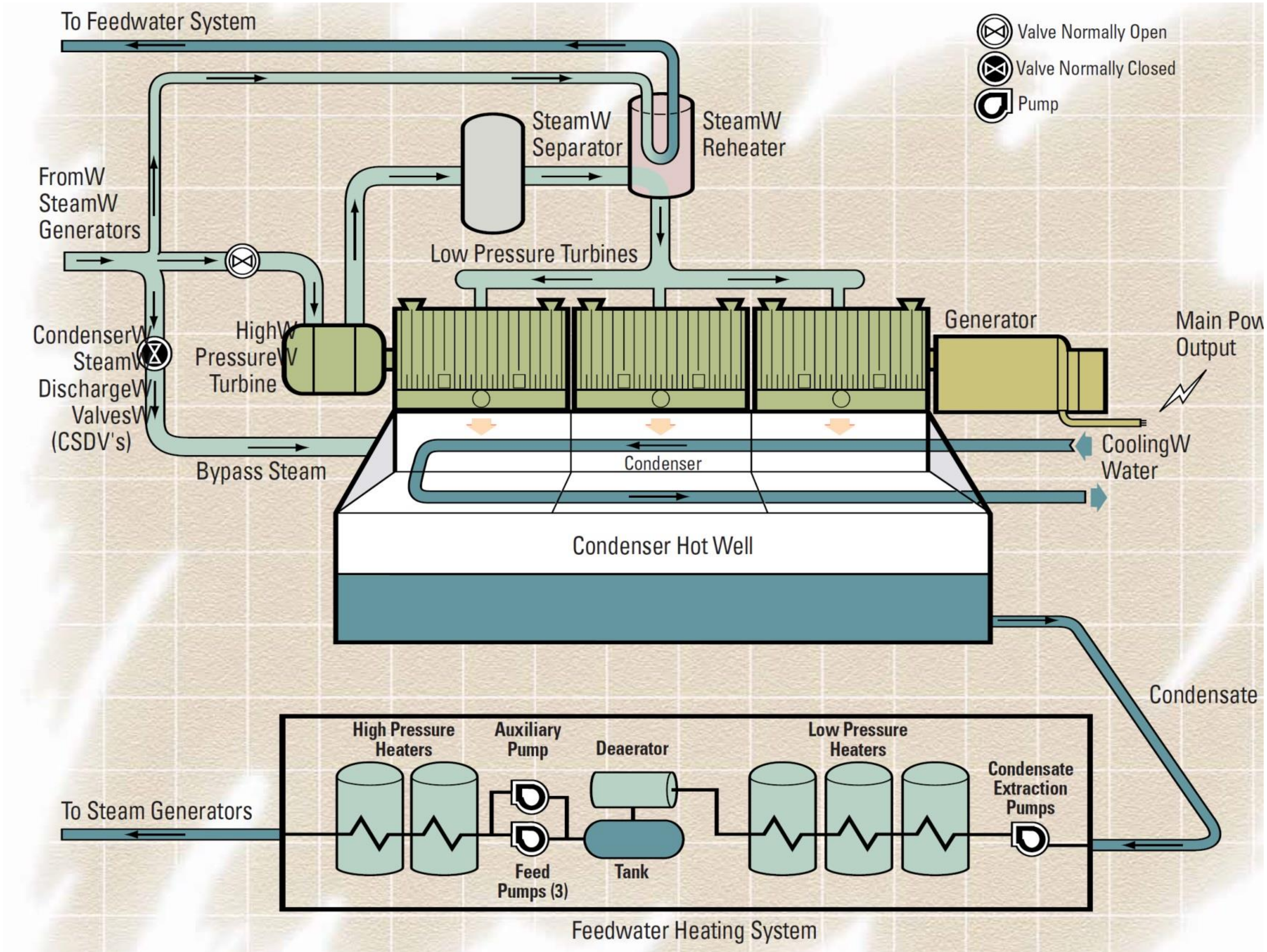
(c) one closed isolation valve for lines of 50 mm in nominal diameter or less, which are normally closed to the containment atmosphere and connected to an easily defined closed system outside containment.

CANDU



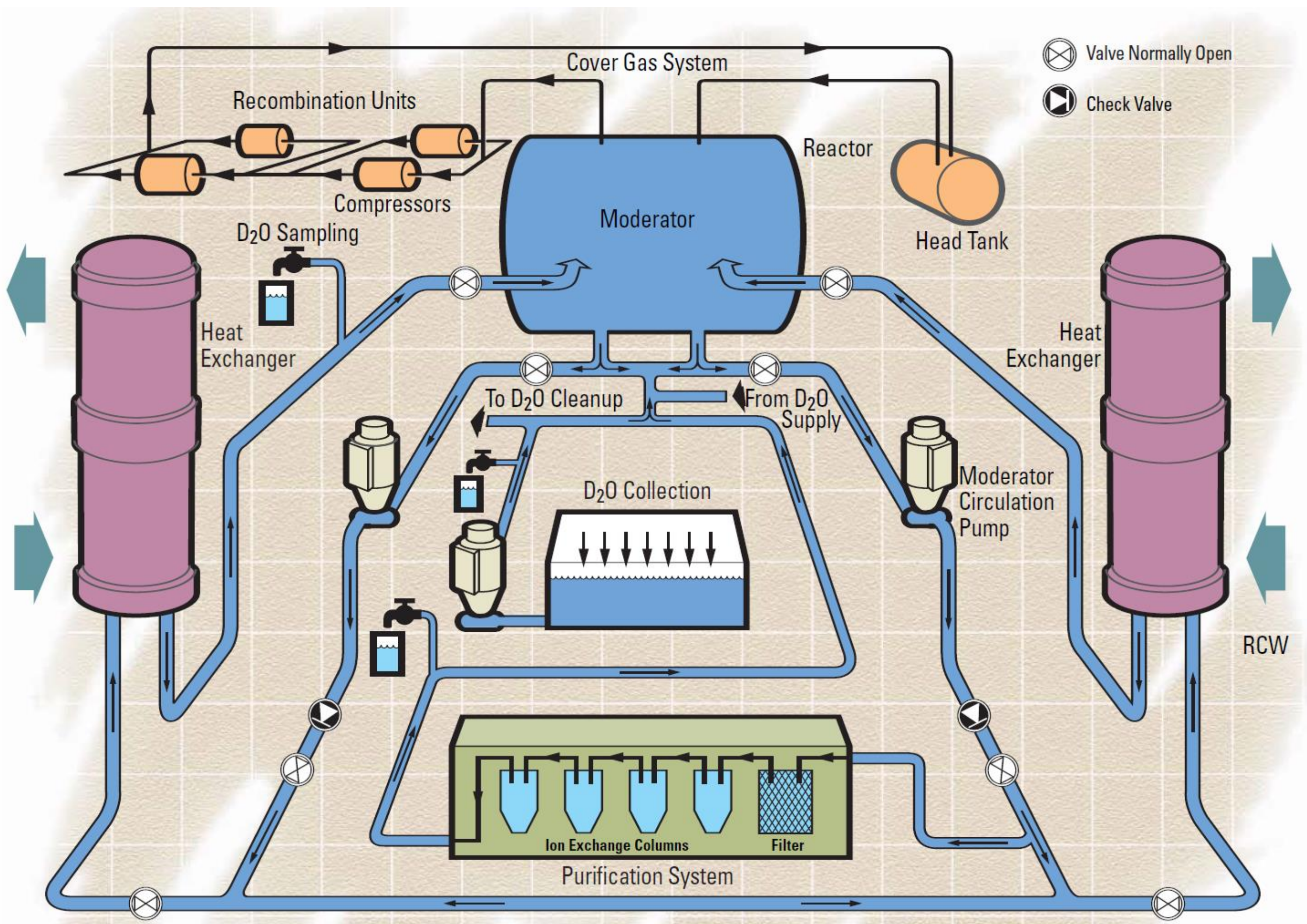
CONTAINMENT ENVELOPE
기체 유입 경로
S/F AUX.
PNEUM. PANEL
SI 35320
S/F LADLE DRIVE SHAFT PENETRATION
사용후연료 방출구
FUELLING MACHINE
3523-RO01
3523-TK101
3523-RO31
3523
TK 3
N/F
ANCIL.
S/F
DISCHARGE PORT ASSEMBLIES
27 kPa(g)
사용후연료
방출실
34.5 kPa(g)
사용후연료
이송수로 격리문
(2,3,4호기)
3.5m
사용후연료
방출조
사용후연료
수용조

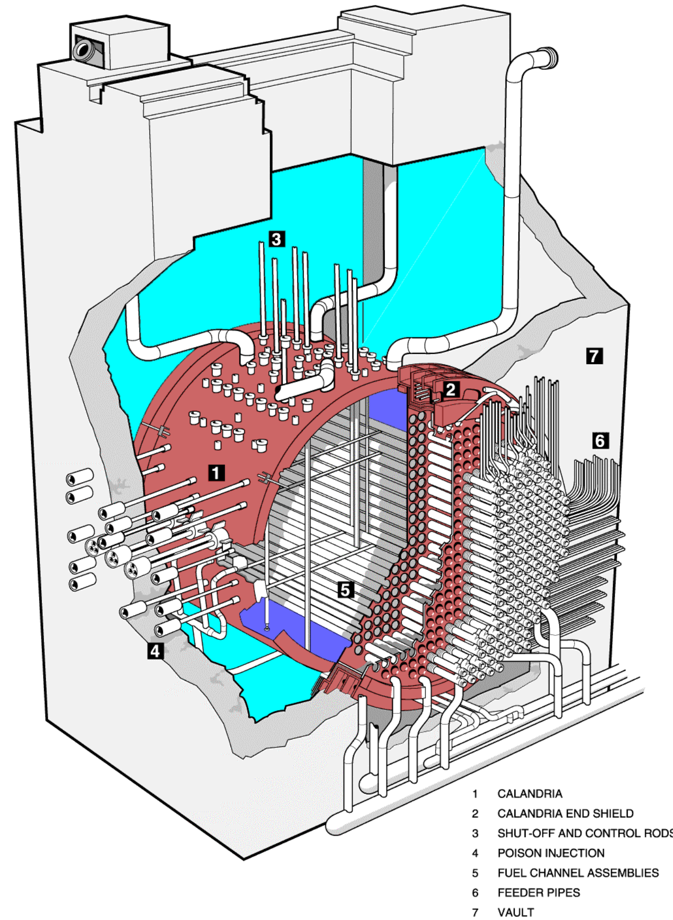
❖ Steam generation system



Moderator system

- Independent of reactor coolant system
- Normal heat removal is ~4 % of full power.
  - Gamma radiation > moderation > heat transfer from the hot pressure tubes



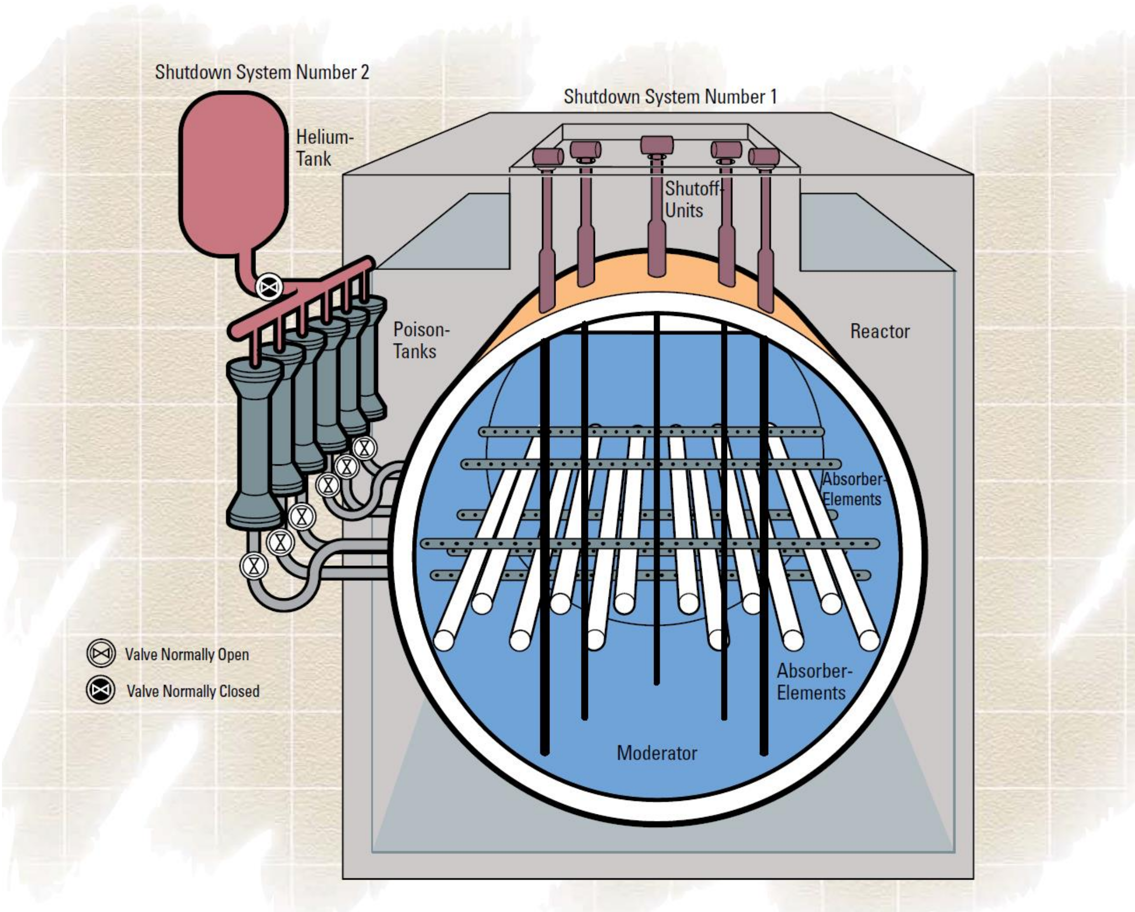


CANDU

- **Safety system**
  - Shutdown systems
  - Emergency core cooling system
  - Containment system
  - Supported by
    - Emergency power supply system
    - Emergency water supply system

Shutdown system

- Solid shutoff rods ⇐ vertical direction
- Direct liquid poison injection into the moderator ⇐ horizontal direction
- Physical and functional independency



CANDU

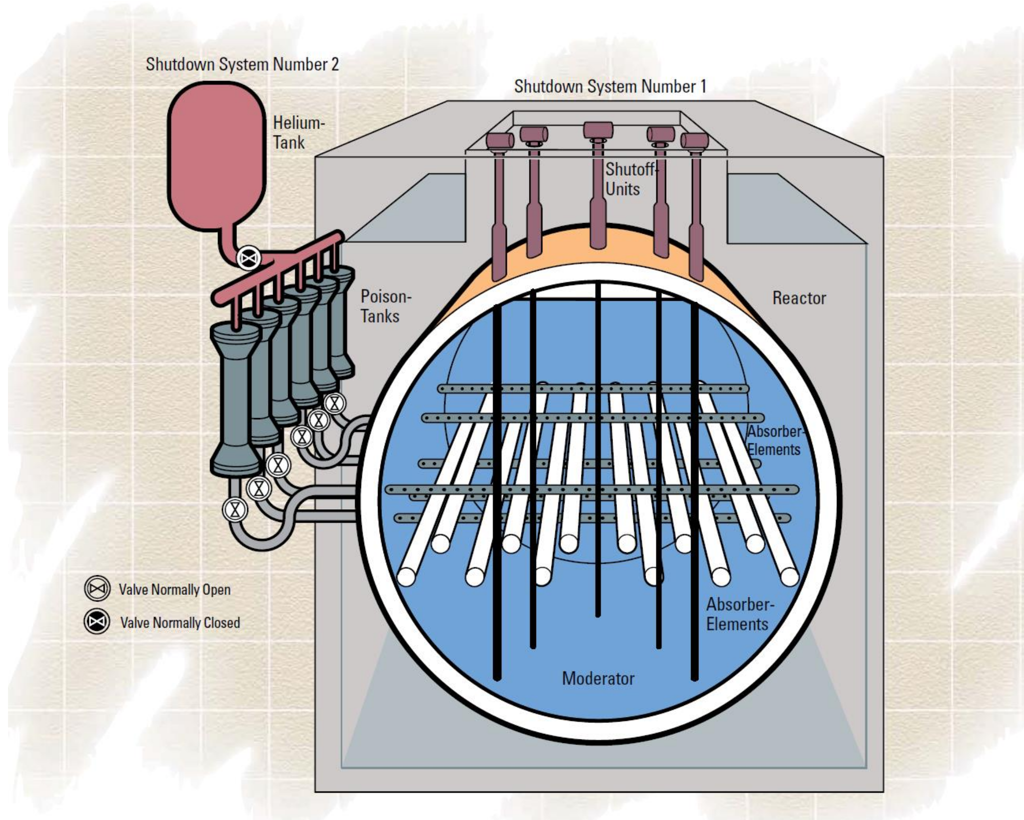
Shutdown system

Adjust rod: to control power liquid zone control absorber: light water
Shut-off rod: to shut down the reactor

Shutdown system

- Solid shutoff rods
  - High neutron power
  - Low coolant flow rate
  - High primary pressure
  - High rate log neutron power
  - High reactor building pressure
  - Low steam generator level
  - Low pressurizer level
  - High moderator temperature

- Direct liquid poison injection
  - Concentrated gadolinium nitrate solution
  - 6 poison injection nozzles
  - Valve between helium tank and the poison tanks
  - The released helium expels the poison

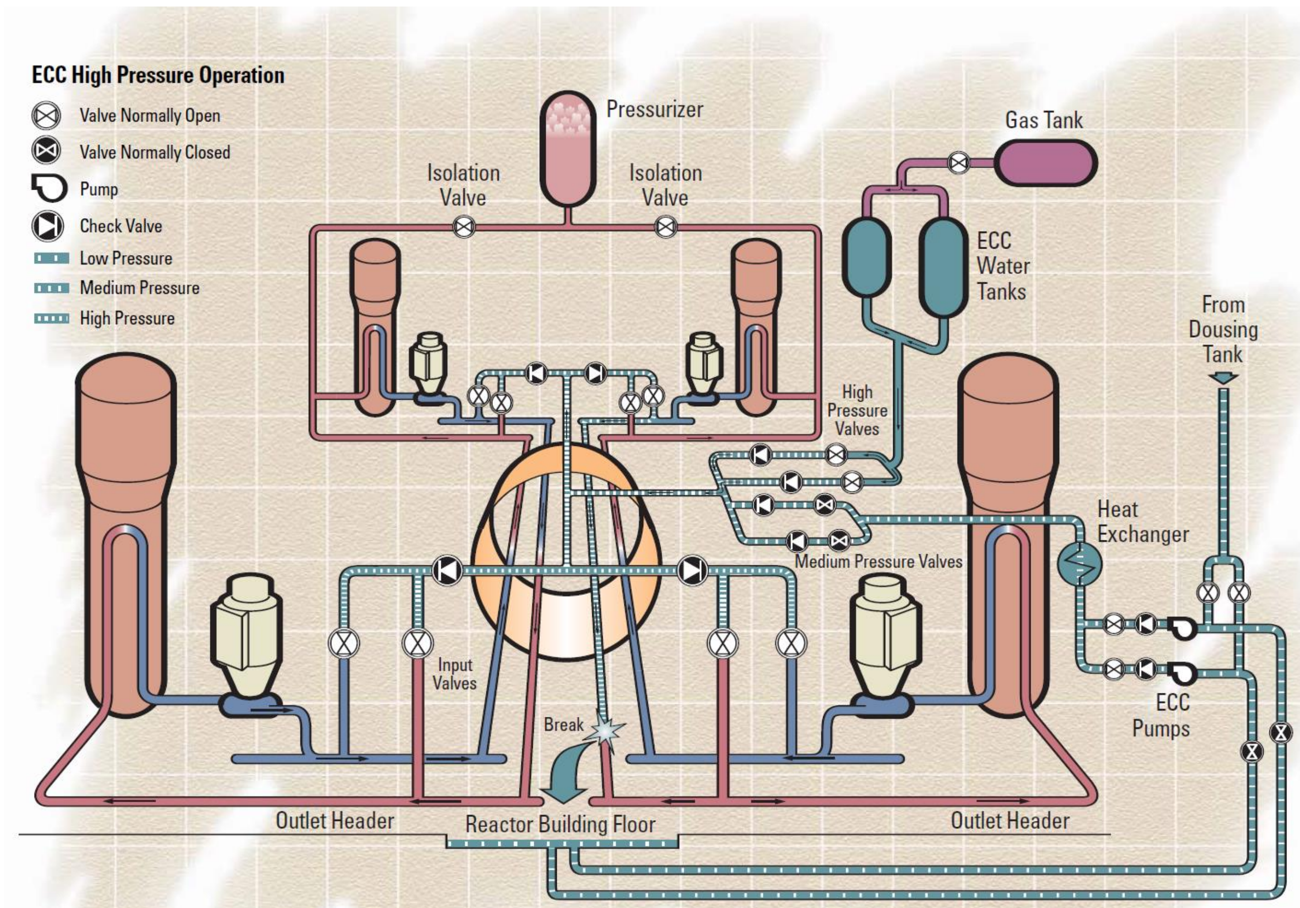


Review

- **Fuel channel configuration**
- **Working mechanism of fuel handling system**
- **Safety system**
  - Shutdown systems
  - Emergency core cooling system
  - Containment system
  - Supported by
    - Emergency power supply system
    - Emergency water supply system

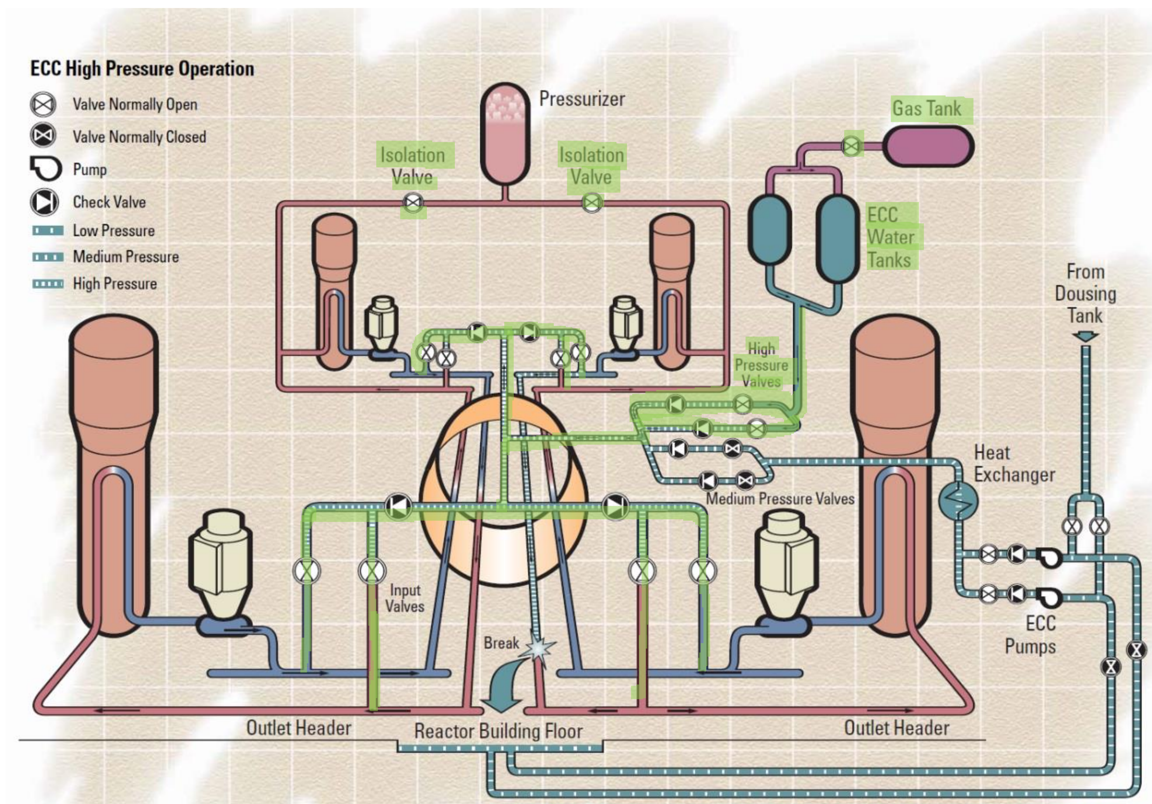
Emergency Core Cooling System (ECCS)

- Three stages of operation
  - High pressure, medium pressure, low pressure
- Operation pressure
  - Primary pressure < 5.5 Mpa



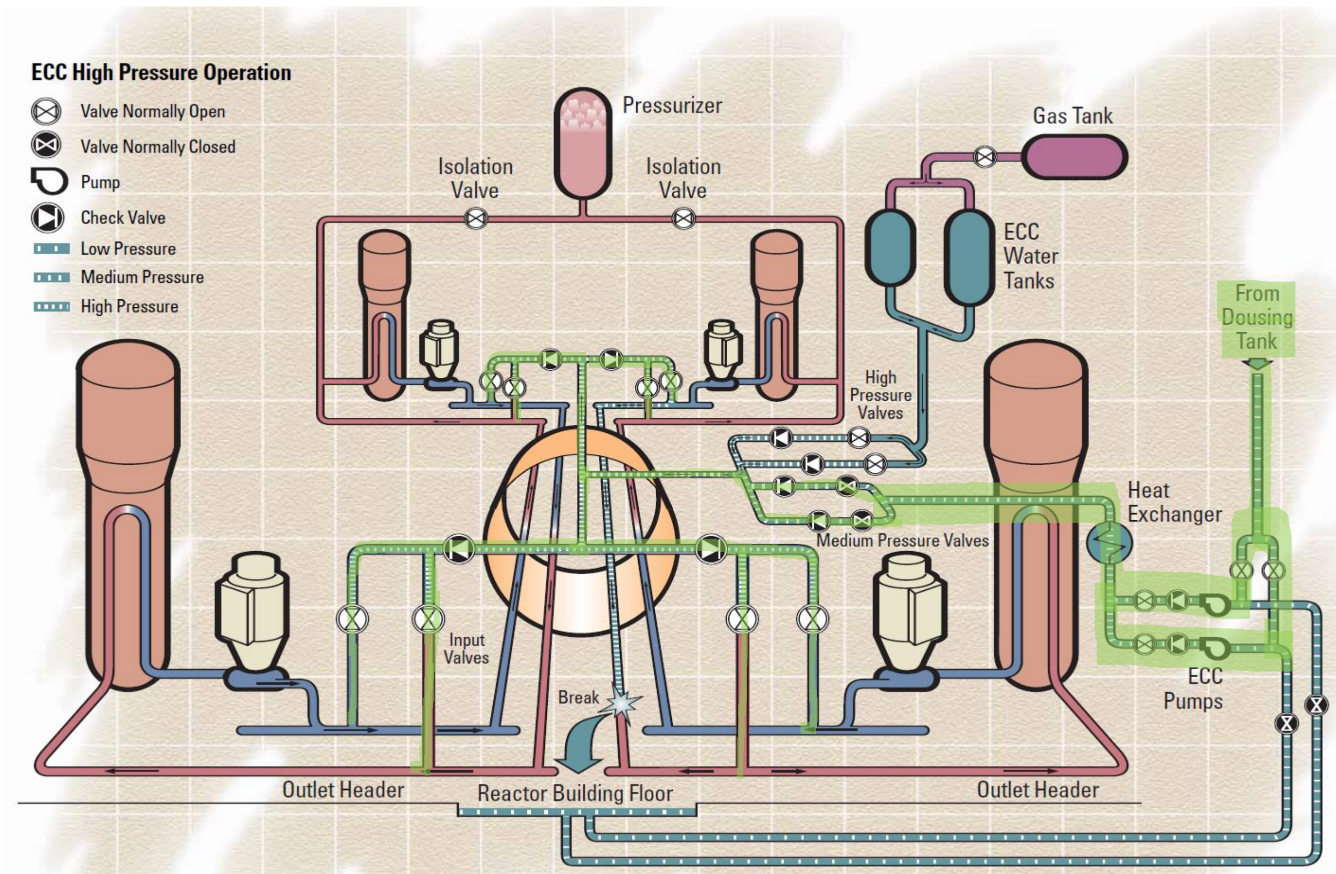
Emergency Core Cooling System (ECCS)

- High pressure operation
  - Isolation of the two HTS loops/ Opening the gas inlet valve
  - Emergency coolant (light water) is forced from the ECCS water tank into the ruptured loop.
  - Pressure: 4.014 MPa, 10 sec. after a maximum pipe size break
  - HP injection valves close automatically when the ECCS water tanks reached a predetermined low level.



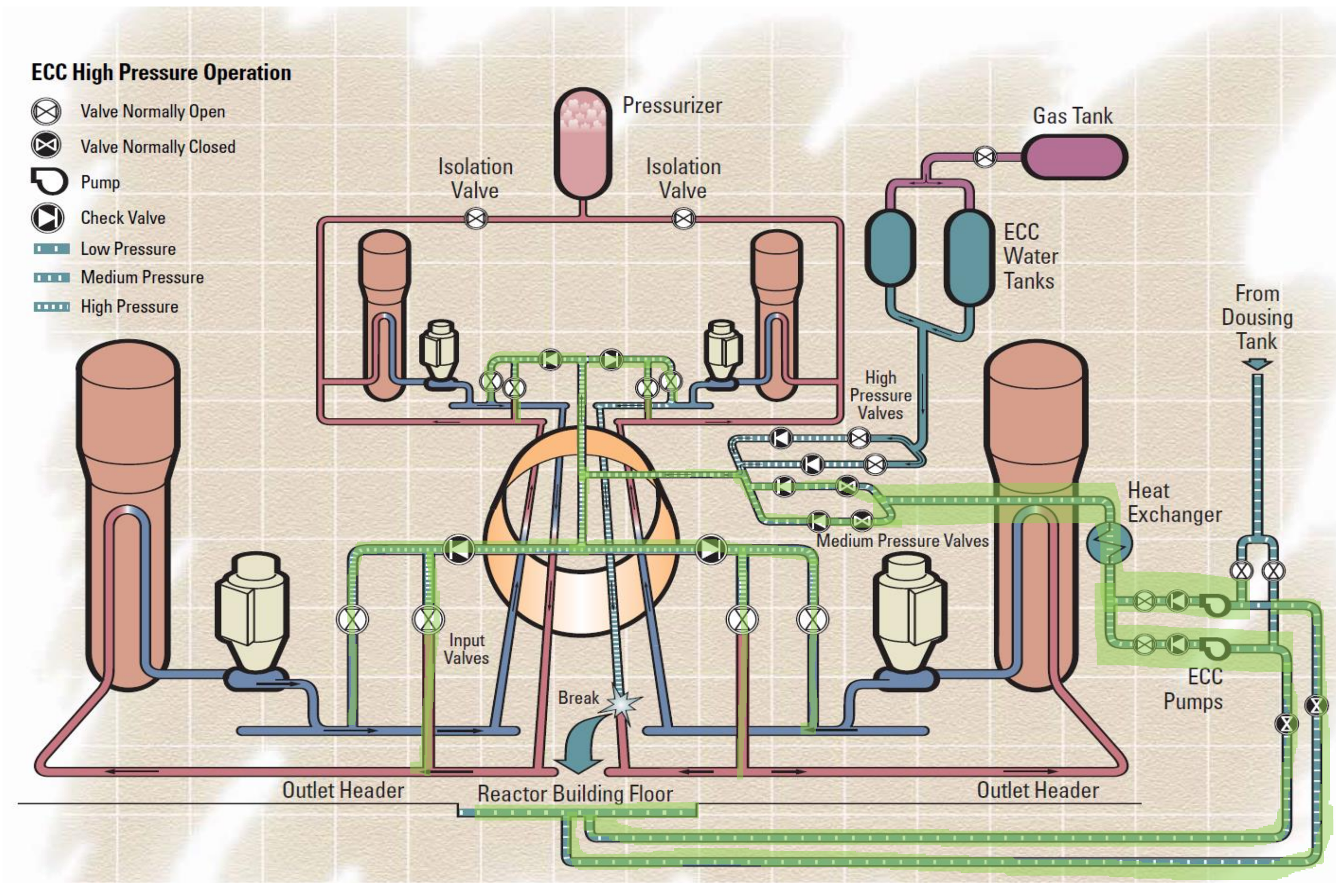
Emergency Core Cooling System (ECCS)

- Medium pressure operation
  - Water supplied from the dousing tank by ECC pumps
    - Dousing tank: 13 minutes operation with the maximum design basis break
    - 2 ECC pumps (100 % × 2)



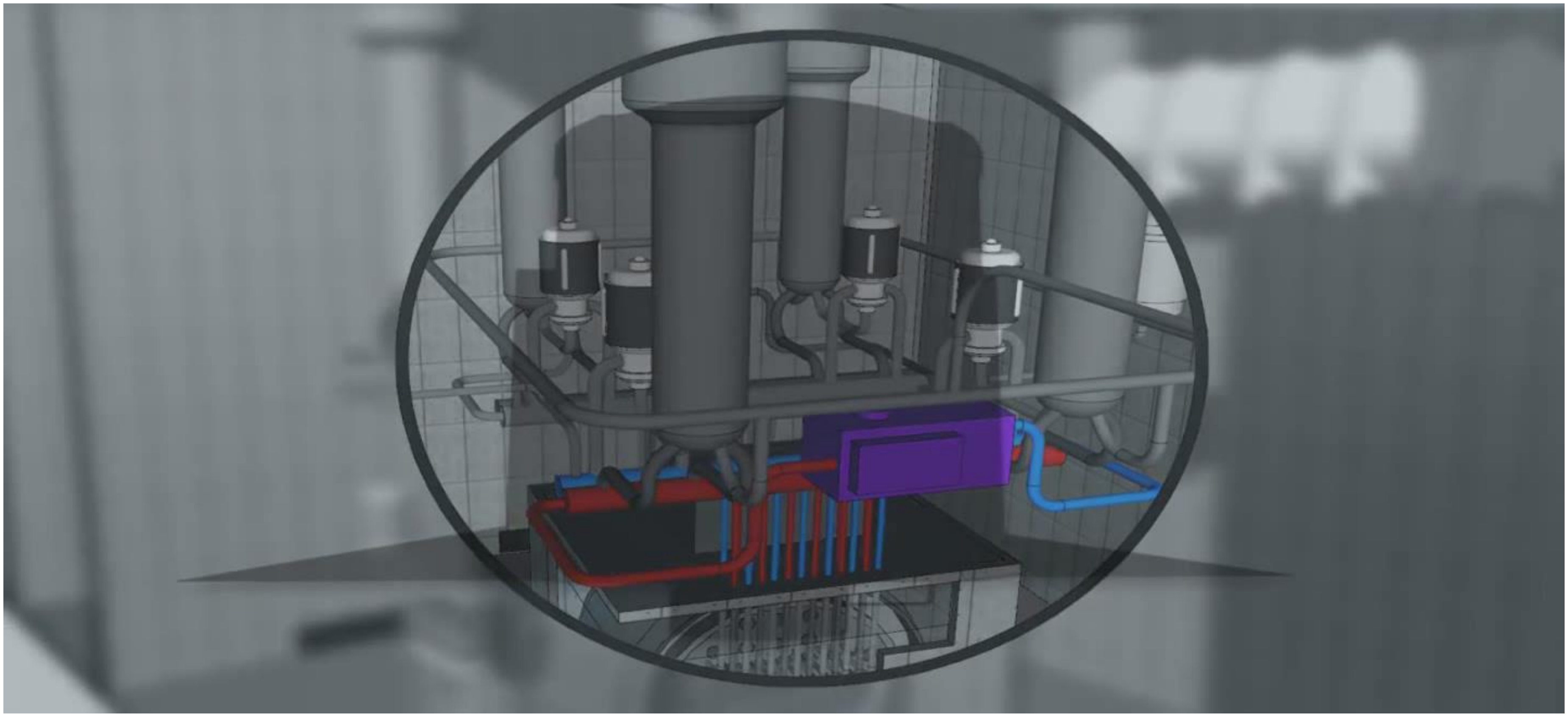
Emergency Core Cooling System (ECCS)

- Low pressure operation
  - Water collected in the reactor basement is returned to the heat transport system via heat exchangers, to provide long term fuel cooling.
  - Heat exchanger maintains the water temp. at ~ 49°C (at ECC pump, 66°C)



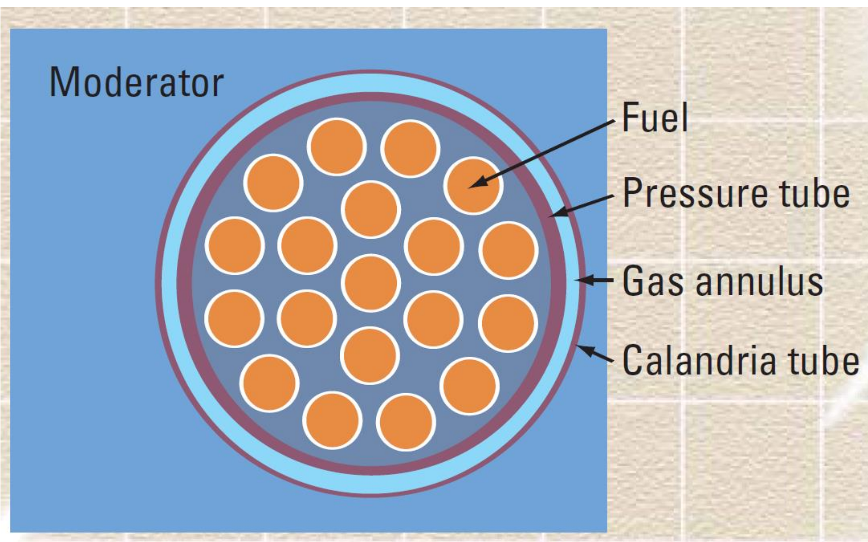
❖ Emergency Core Cooling System (ECCS)

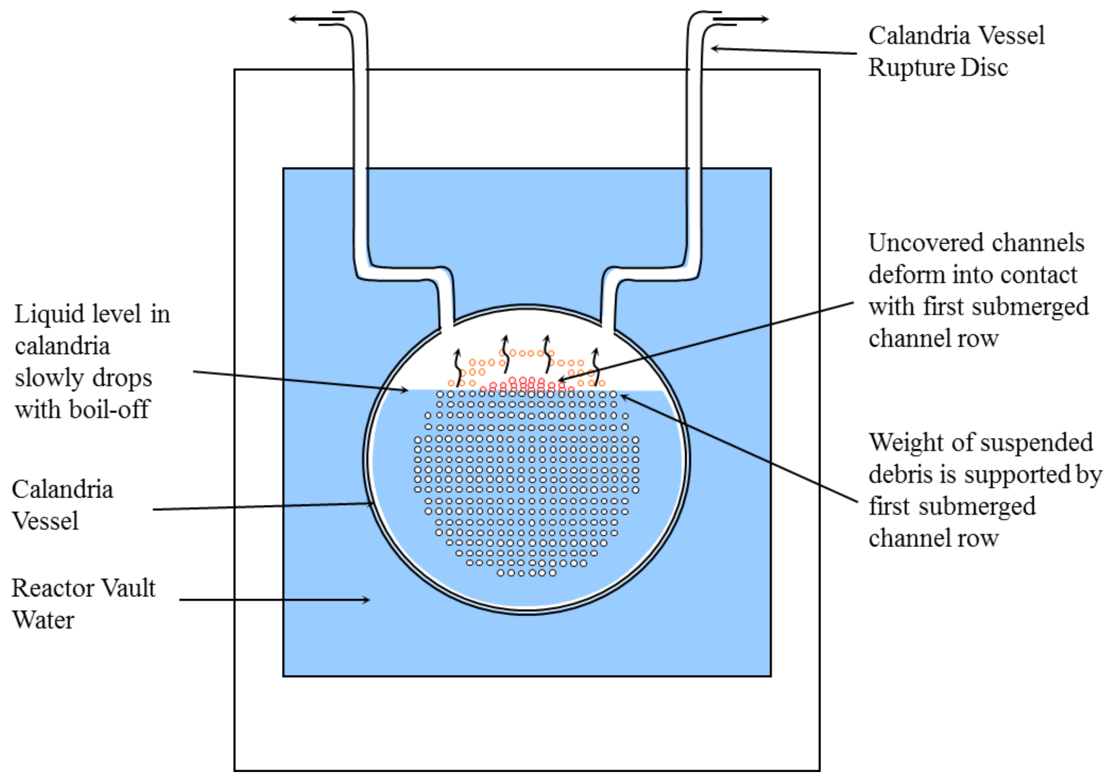
- Loss of Coolant Accident



Emergency Core Cooling System (ECCS)

- What if the ECCS fails?
  - Decay heat $\Rightarrow$ moderator by radiation and conduction





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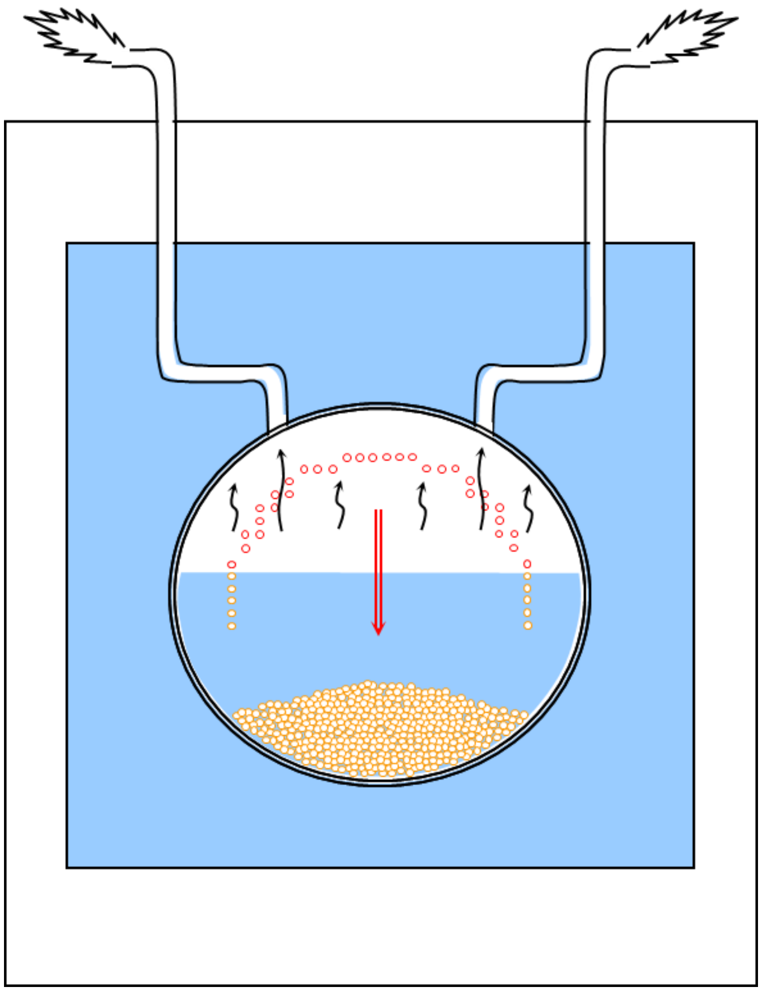

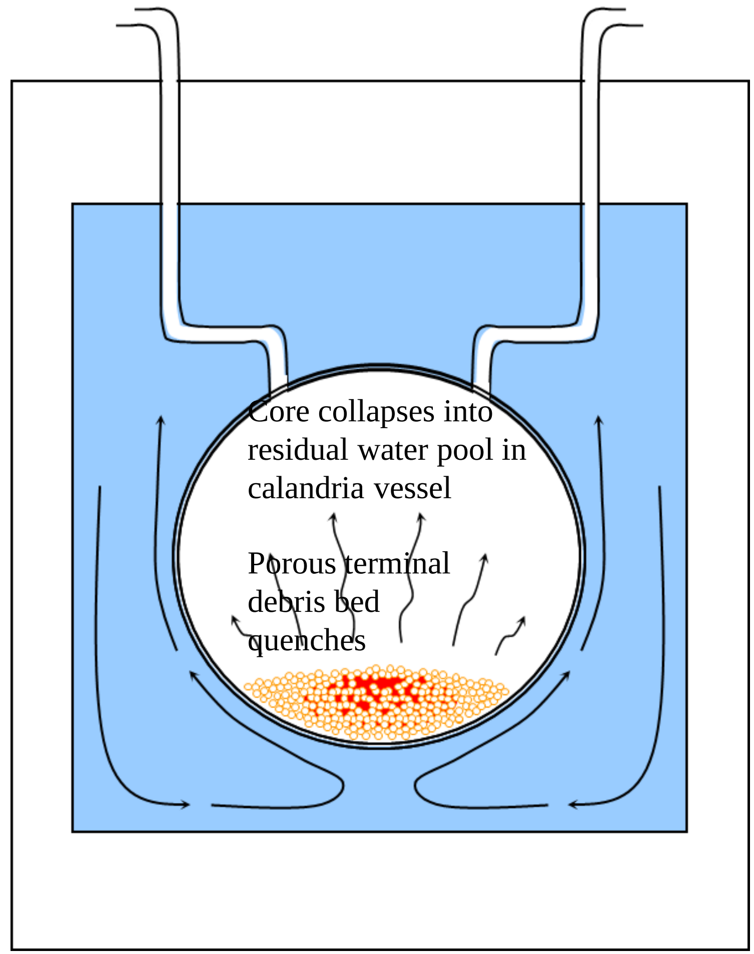

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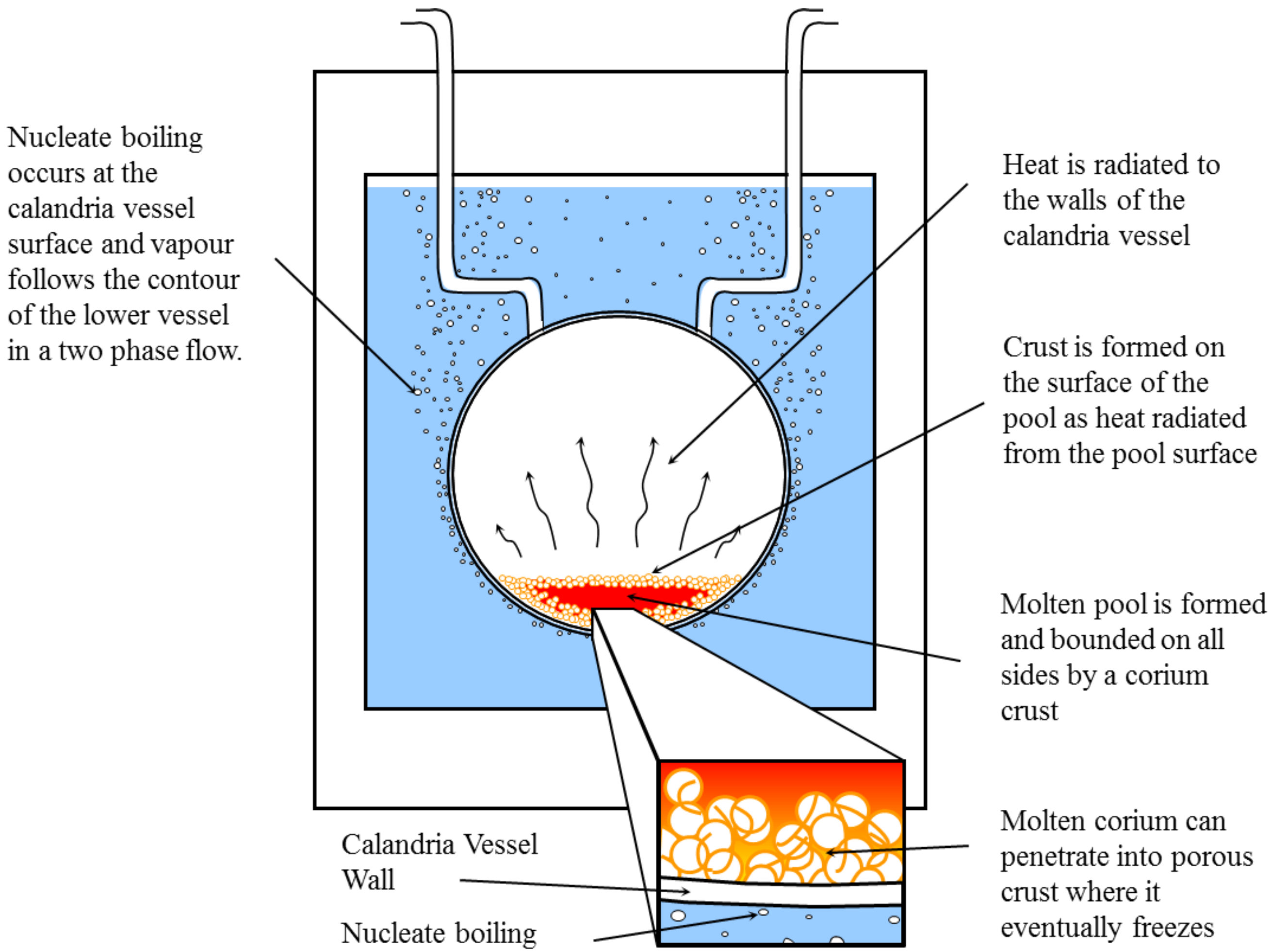
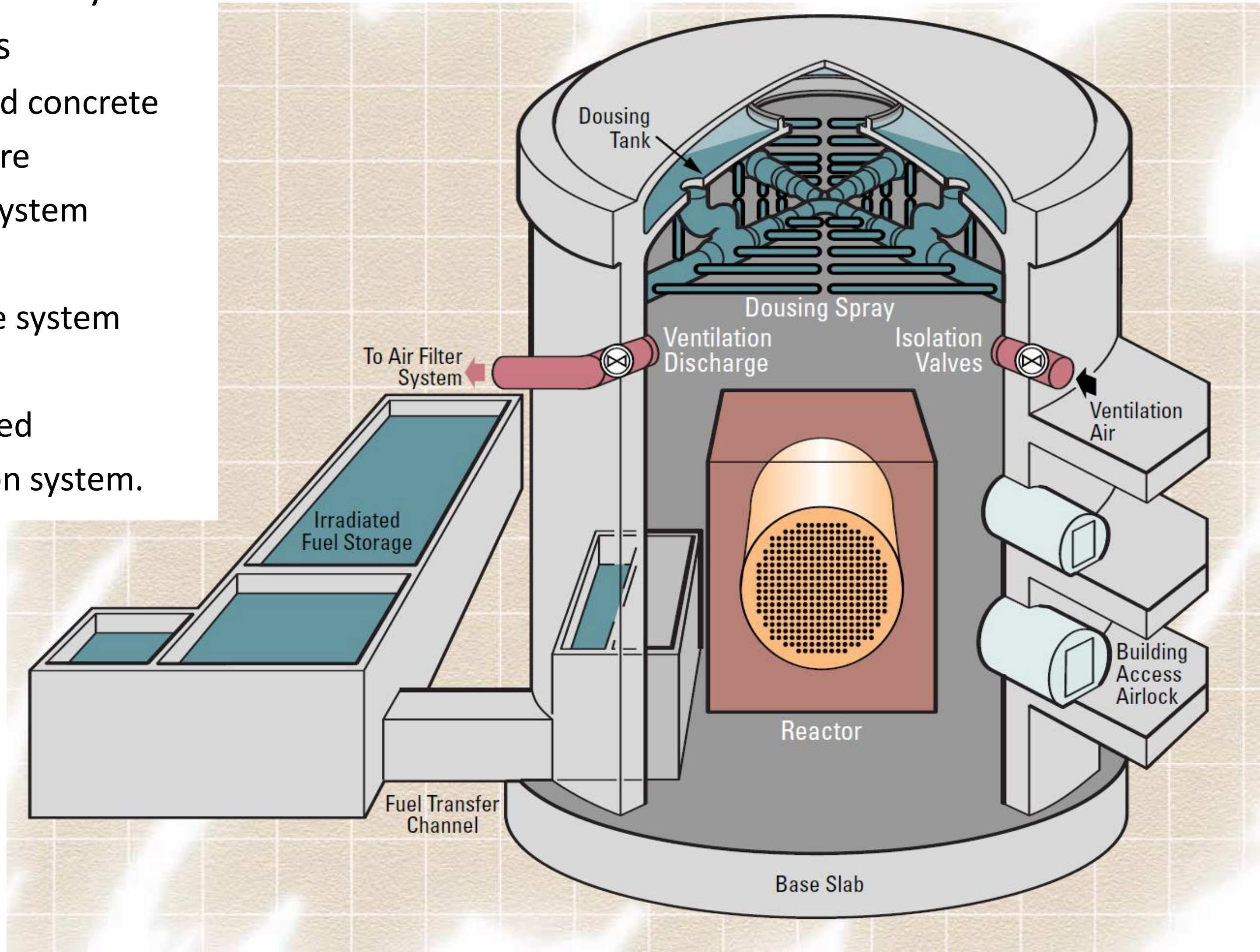


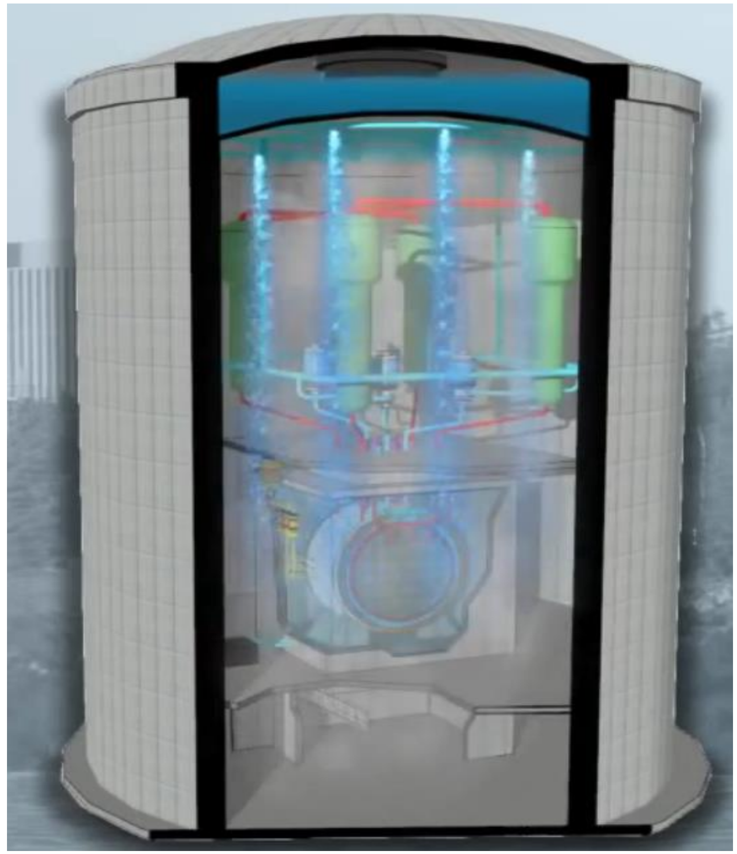
Containment system

- To provide a sealed envelope around the reactor systems if an accidental radioactivity release occurs from these systems
- Structures and systems
  - Lined, post-tensioned concrete containment structure
  - Automatic dousing system
  - Air coolers
  - Filtered air discharge system
  - Access airlocks
  - Automatically initiated containment isolation system.



Containment system

- System operation: if a large break in the heat transport system occurred
  - The building pressure would rise.
  - At an overpressure of 3.5 kPa (0.5 psig), containment closure is initiated.
  - Reactor trip and ECCS operation starts.
  - The dousing system
    - Start pressure: an overpressure of 14 kPa
    - Stop pressure: an overpressure of 7 kPa
  - Air cooler (LAC: Local Air Cooler)
    - Fan + cooling coil



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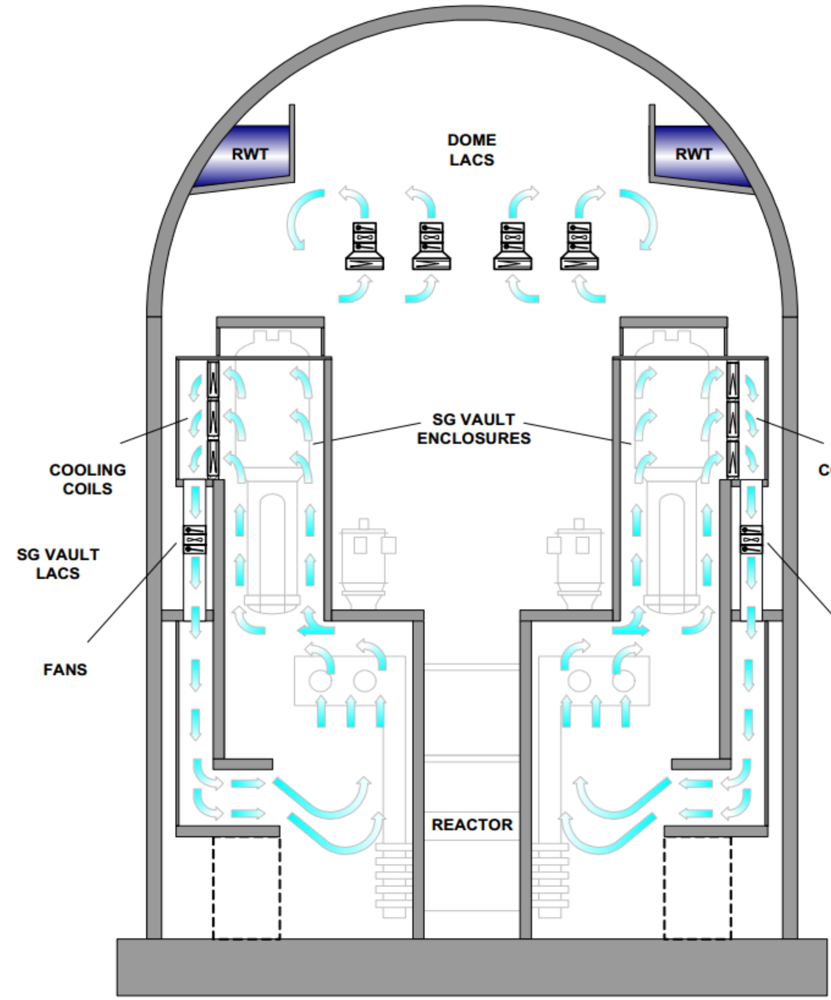


- **Containment system**

- **Containment system**

Safety Concern regarding Positive Void Coefficient

- Void coefficient of reactivity
  - positive
- Power coefficient of reactivity
  - close to zero
- Temperature coefficient of reactivity
  - close to zero

- CNSC has not set requirements for reactivity feedback
  - Positive or negative is acceptable
  - Magnitude is important

- CNSC sets performance objectives related to
  - Capacity of systems to control reactivity
  - Capacity of systems to shutdown reactor in a timely manner
  - Reliability of the control and shutdown functions

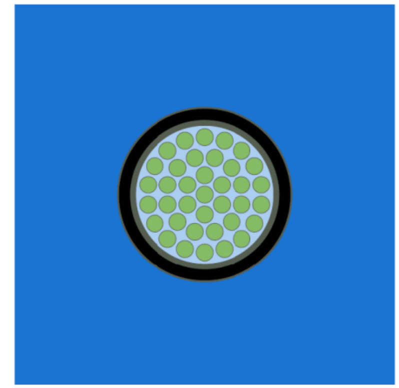
Why do CANDU reactors have a "positive void coefficient"?

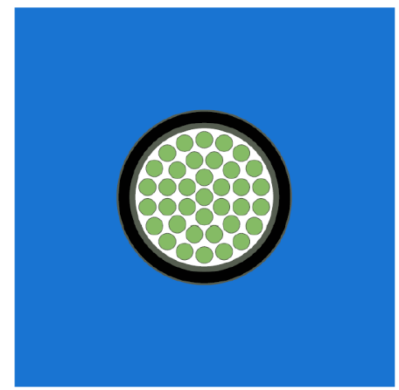
(see http://www.nuclearfaq.ca)

- In PWRs and BWRs,
  - Void coefficient: negative
  - A decrease in coolant density leads directly to a decrease in overall neutron moderation

- In CANDU
  - The coolant and moderator are separate heavy water circuits.
  - The coolant makes only a minor contribution to overall neutron moderation.
  - A loss of coolant (or a decrease in coolant density) would not be expected to affect neutron moderation significantly.
  - Furthermore, since the coolant does not significantly absorb neutrons either, one might suspect that its disappearance would have no reactivity effect at all.

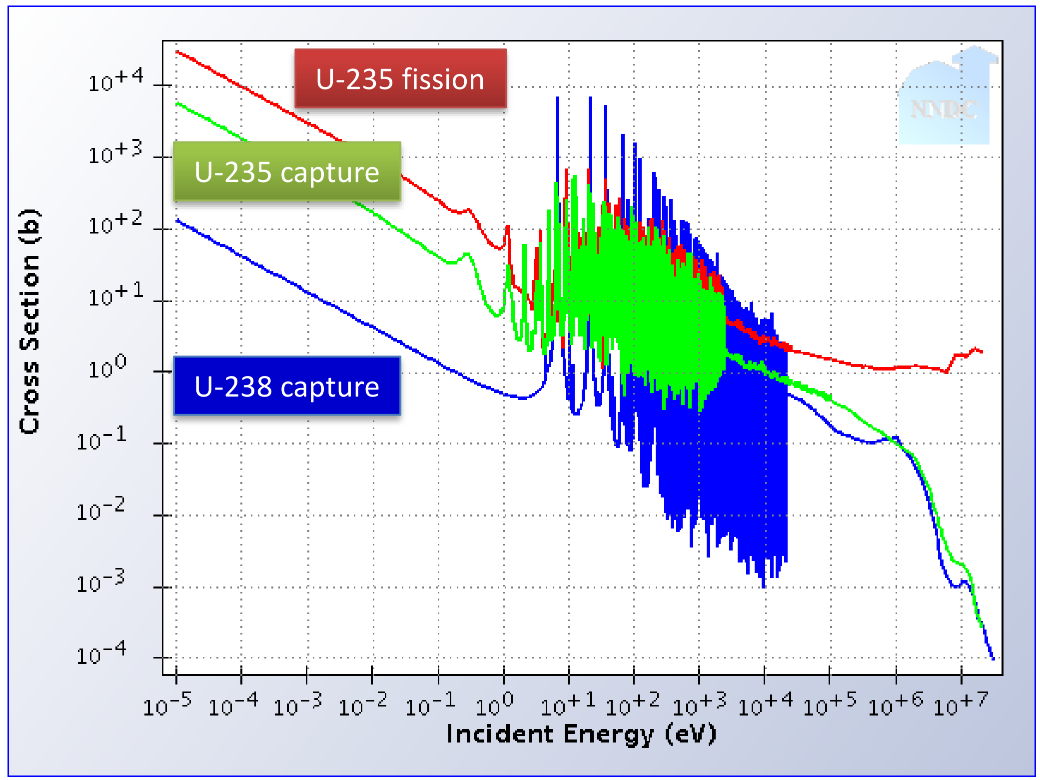
  - In fact, the coolant does account for a small amount of moderation in CANDU reactors – just enough to "knock" fast neutrons down into the resonance energy range.





Why do CANDU reactors have a "positive void coefficient"?

- In CANDU
  - This small amount of moderation takes place in a location close to the fuel.
  - Some neutrons are decelerated by "just" the right amount, and in "just" the right location.
  - LOCA
    - Resonance-energy neutrons ⇓
    - Fast neutrons ⇑ and fast fission in U-238 ⇑
  - Up-scattering
    - The coolant increases energy of thermal neutrons
    - $T_{coolant}$ (300 °C) >> $T_{moderator}$ (75 °C)
    - Less fission in U-235
  - LOCA
    - No up-scattering
    - Fission in U-235 ⇑



How do CANDU reactors meet high safety standards, despite having a "positive void coefficient"?

- Combined effect
  - Small feedback under power ramping
- Shut-down system
  - Shut-off rods
  - High pressure liquid poison injection
- Core's dynamic response
  - Neutron moderation is sufficiently sluggish to enable control.

- "*Any* feedback coefficient is a challenge for safe control, if it is large enough or highly variable - regardless of its sign."
  - A BWR, for instance, has a large negative void coefficient which must be handled by the safety system in scenarios involving massive void collapse (inducing positive reactivity).

- "It is far more important to keep feedback coefficients small and independent of operating conditions."

❖ Safety Concern regarding Positive Void Coefficient