Bharat Small Modular Reactor (BSMR)

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Outline

1. Introduction to Indigenous SMRs
2. Design & Basic Overview of Bharat Small Modular Reactor (BSMR-200)
    - Salient Features of BSMR-200
    - Reactor Internals & Major Reactor Equipment
    - Nuclear Layout
    - BSMR-200 Plant Layout
    - APURVA Forgings Developed by BARC
    - Technological Readiness
    - Reactor Core Thermal Hydraulics
3. Safety Analysis

Indigenous SMRs: Nuclear Energy Mission of Union Budget 2025-26

S No.	Reactor	Remarks
1	BSMR-200	Role: captive power generation for the industry / deployment in brown fields ▪ Fully indigenous design and technology, including fuel ▪ To be uprated subsequently upto 300 MWe *Design upgradation to 700 MWe planned*
2	SMR-55	Role: power generation in remote / brown field sites ▪ Based on modular reactor plant design ▪ Fully indigenous design and technology, including fuel

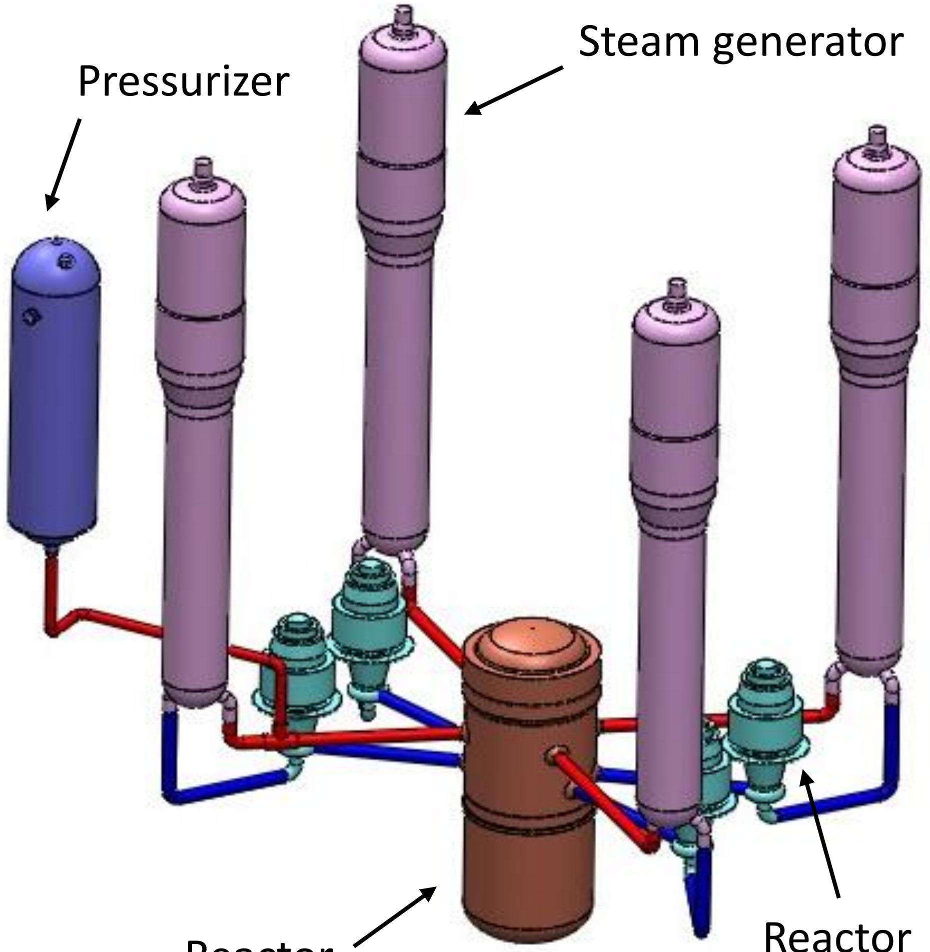
Bharat Small Modular Reactor (BSMR-200)

- BSMR-200, a medium sized pressurised water reactor based nuclear power plant
  - 200 – 300 MWe NPPs, an attractive option as a captive power plant
  - Also, as a replacement power plant in sites with retiring coal power plants
- Establishment of BSMR-200 (SMR) through a R&D project to demonstrate indigenous PWR technology prior to commercial launching
- Overall schedule of construction and operationalisation inline with the nuclear energy mission
- Lead unit will be in a DAE site: Tarapur Atomic Power Station site

Salient Features

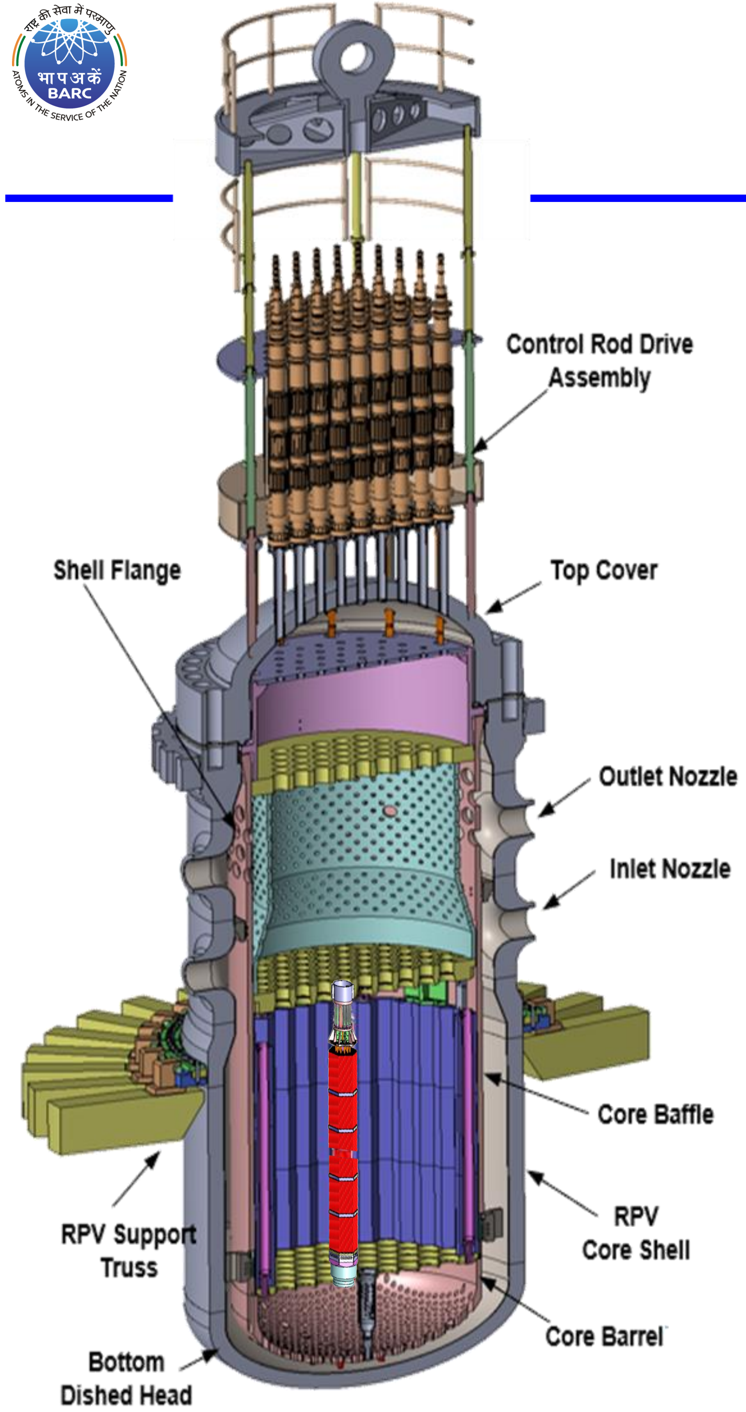
- Nuclear systems of pressurized heavy water reactors (PHWRs) and BSMR differ
  - BSMR uses enriched fuel, needs smaller fuel quantity and reduced spent fuel burden
  - PHWR requires more frequent online refueling
  - No heavy water moderator system in BSMR
  - BSMR core has burnable neutron poisons to manage excess reactivity
  - Fuel clusters housed in pressure tubes in PHWRs; BSMR uses reactor pressure vessel
  - BSMR uses soluble boron for reactivity control

Salient Design Features of BSMR-200



Parameter	Value
Power	~220 MWe / 760 MW_{th}
Primary coolant pressure	120 bar
Coolant temp. at reactor inlet	253 °C
Coolant temp. at reactor outlet	288 °C
Primary coolant flow rate	20,000 m^3/h
Fuel	UO_2 (enrichment < 5%)
Cycle length	600 days
Discharge burnup of fuel	46 GWd/tHM
Steam flow rate, kg/s	~92
Steam temp at SG exit, °C	250
Steam pressure at SG exit, kg/cm²	40

Reactor Internals



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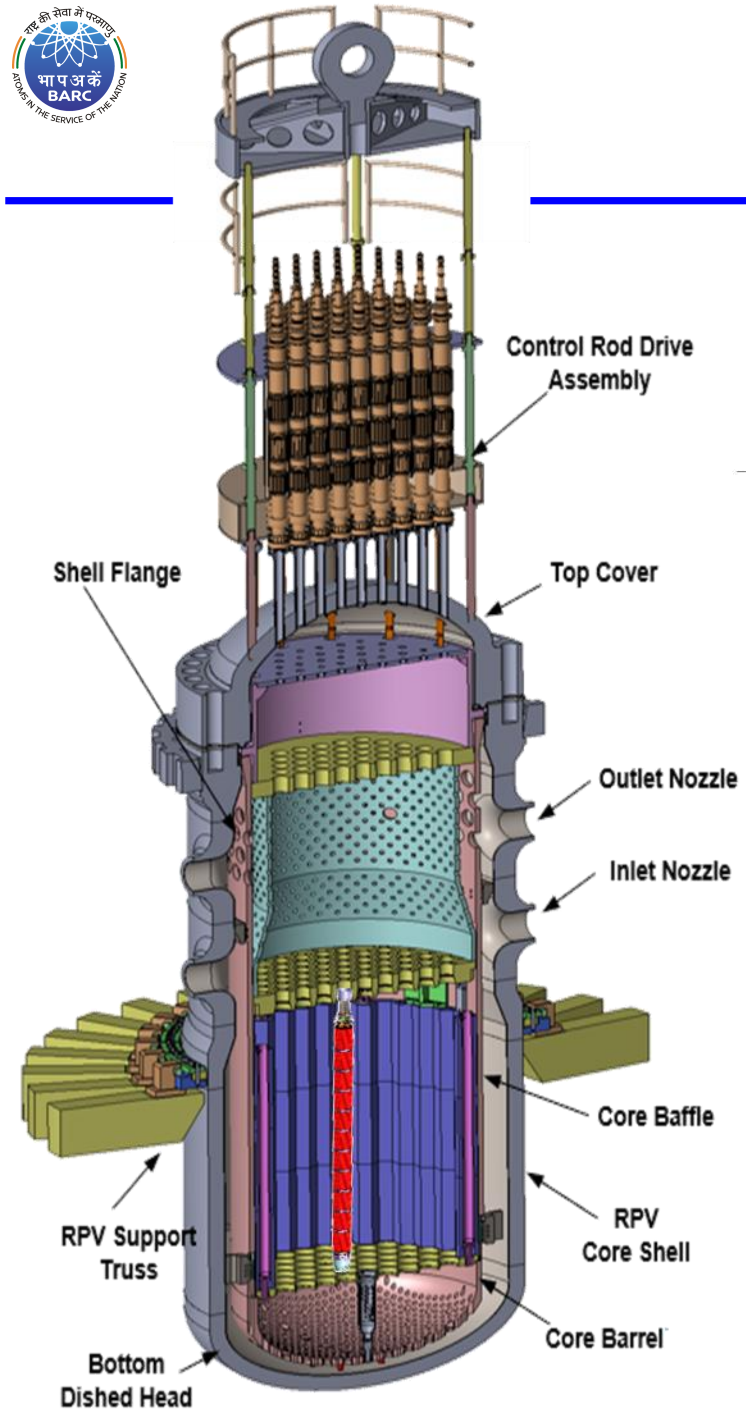
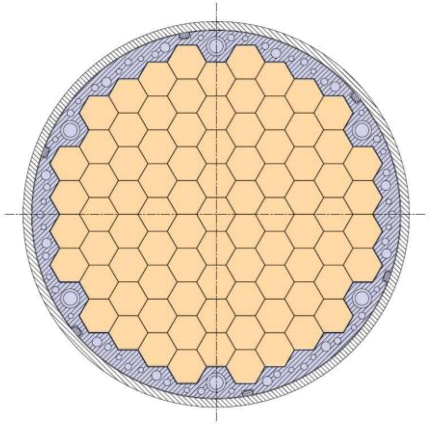
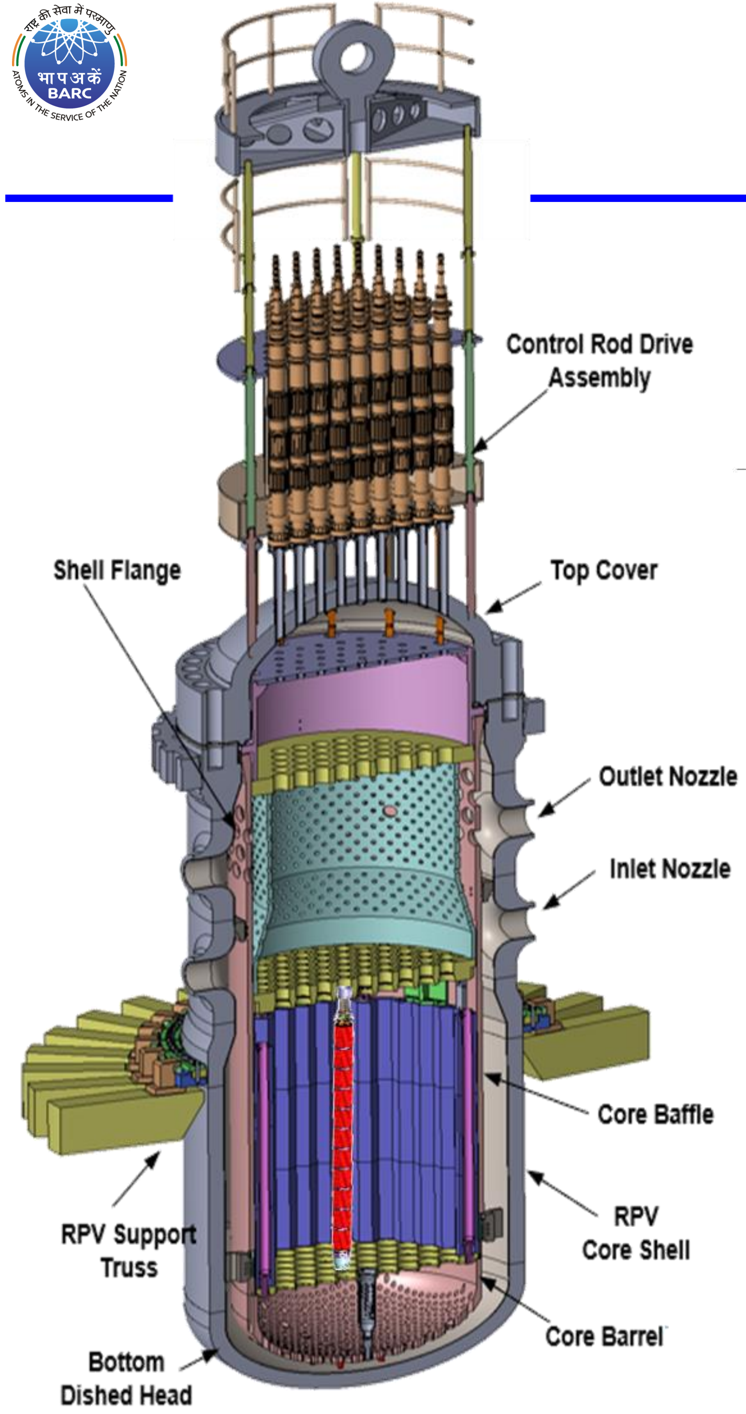


Core cross section



Fuel assembly

Reactor Internals



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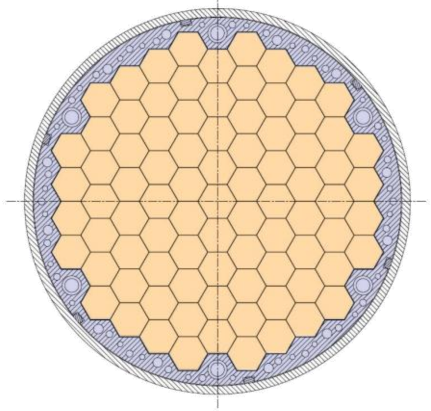

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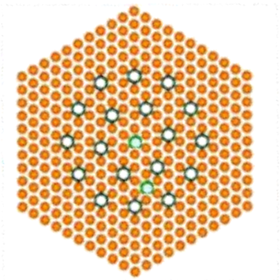
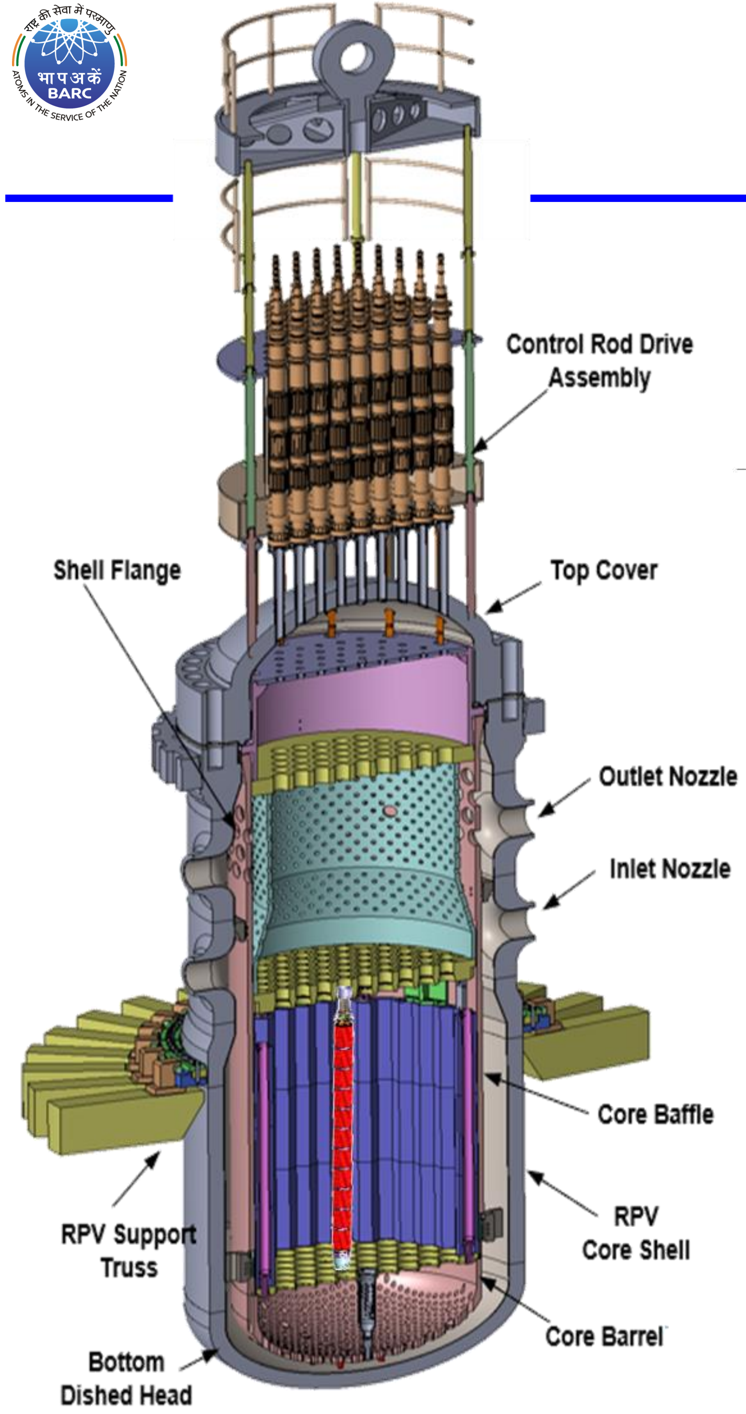


Fuel assembly cross section

Fuel rod

Reactor Internals



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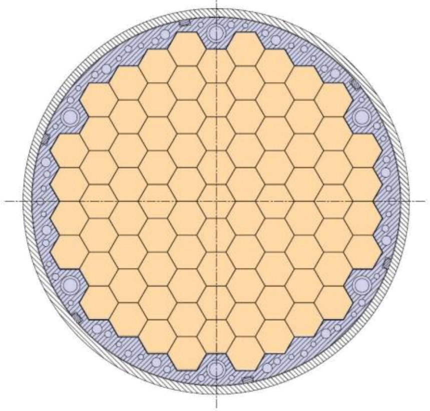

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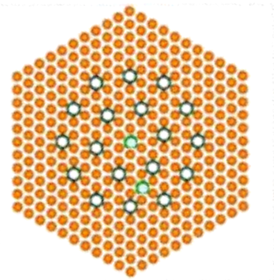
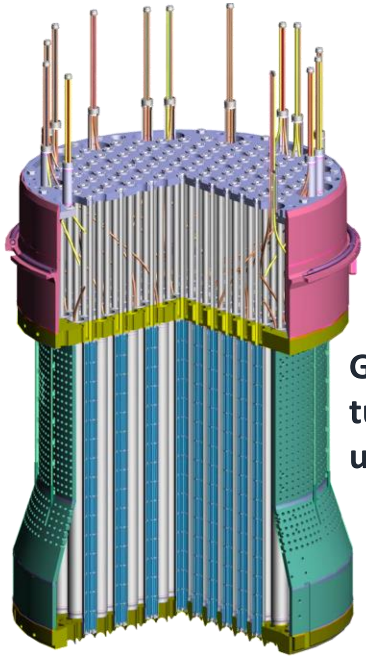
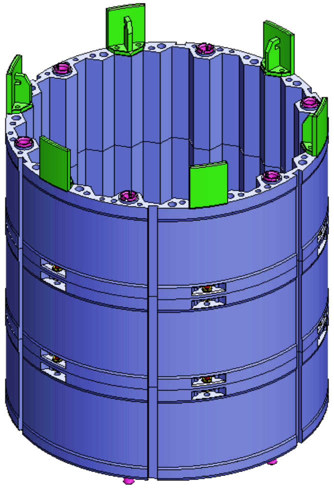


Fuel assembly cross section

Fuel rod

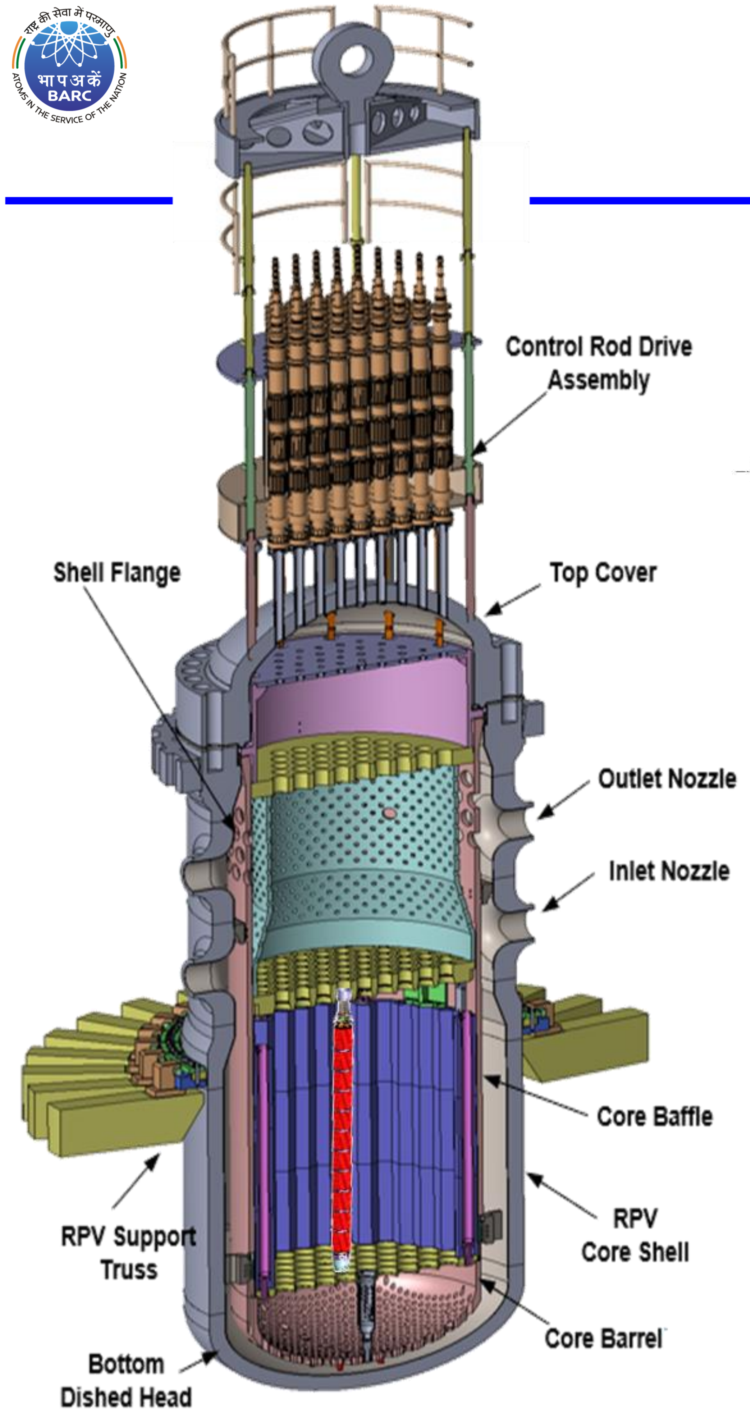


Guide tube unit



Core baffle

Reactor Internals



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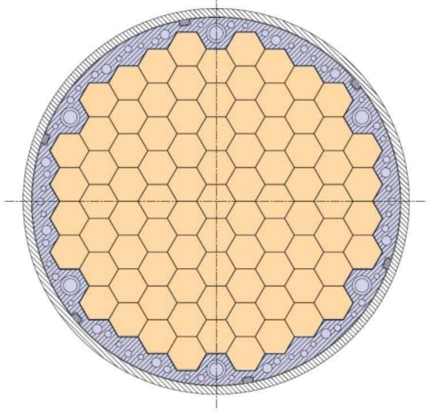
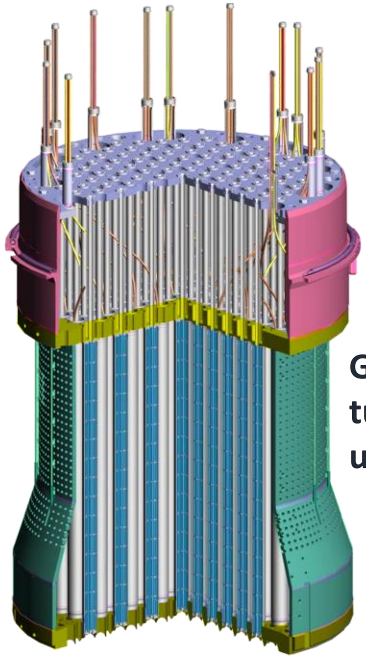


Control rod cluster



Guide tube unit

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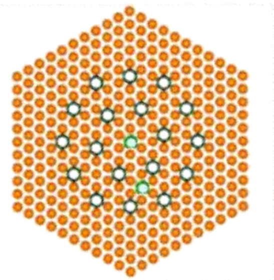
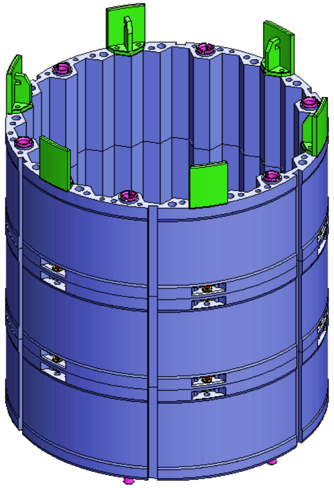


Fuel assembly cross section

Fuel rod



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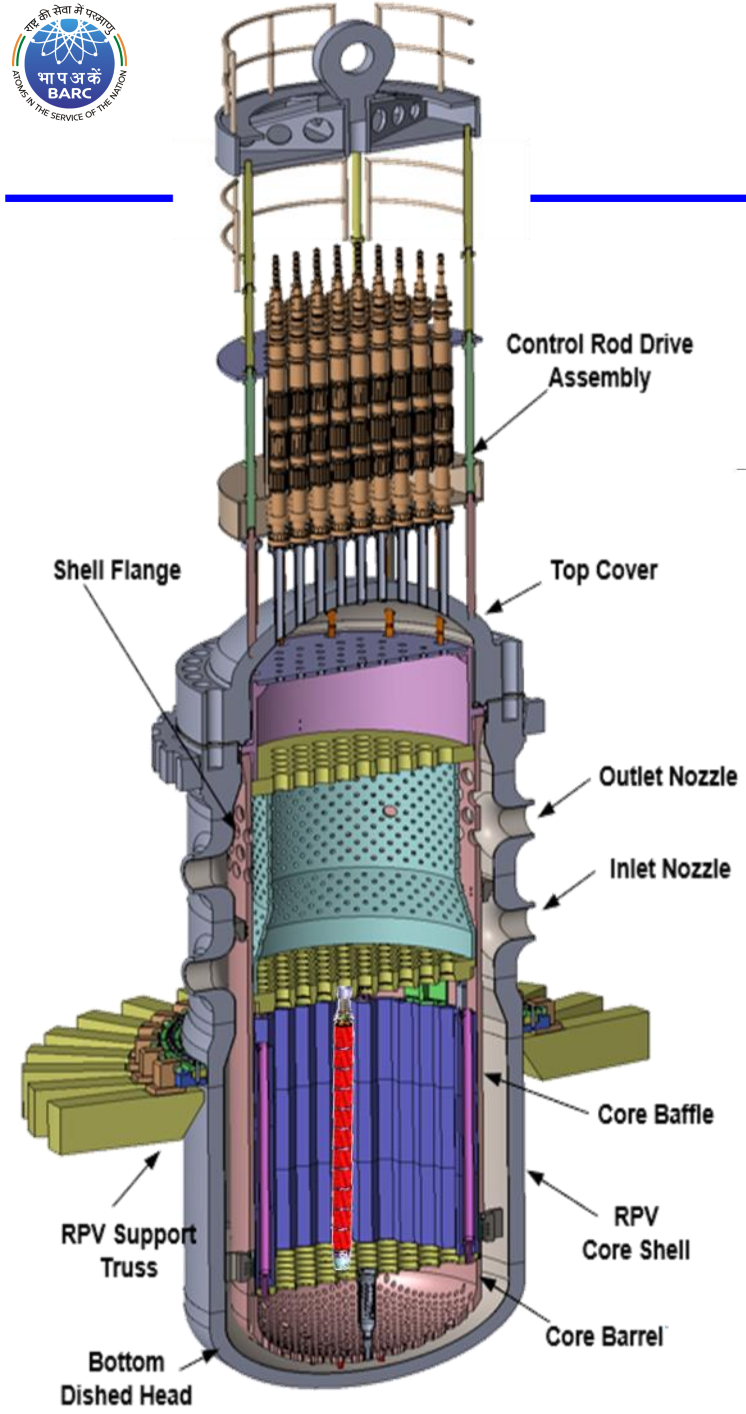

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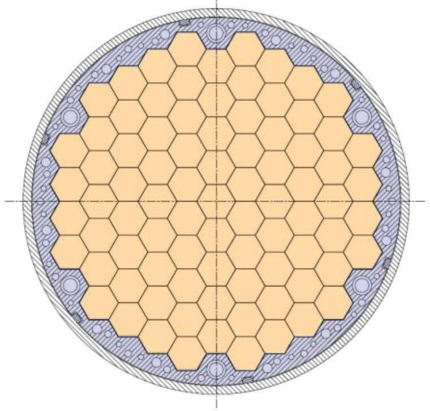
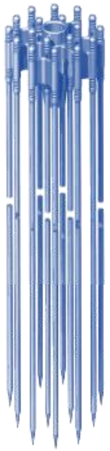
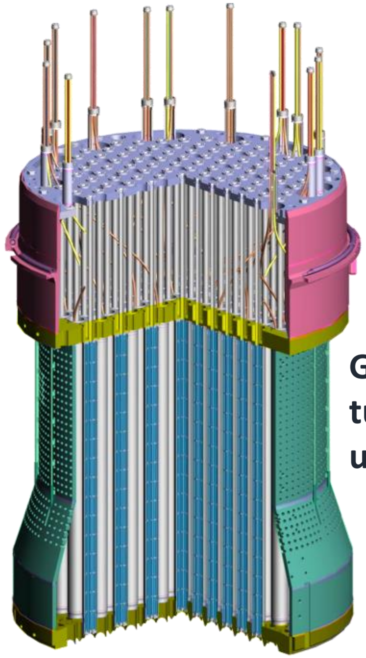




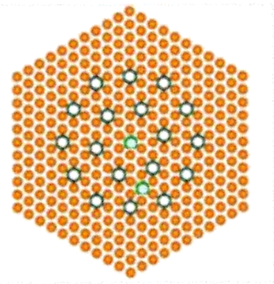
Control rod cluster



Guide tube unit

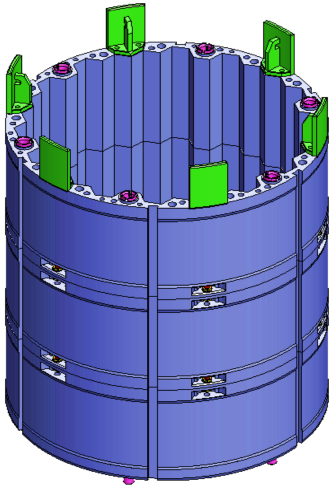
Core Barrel

Fuel assembly



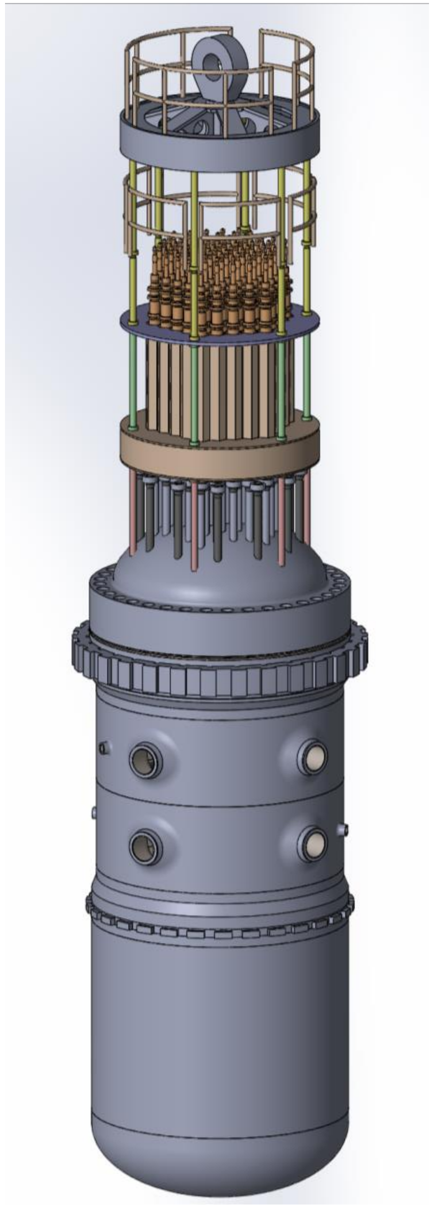
Fuel assembly cross section

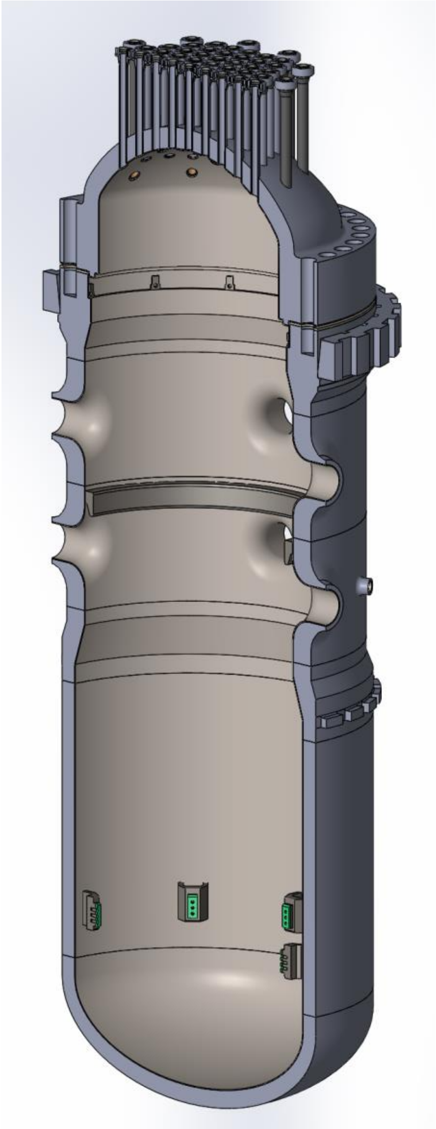
Fuel rod

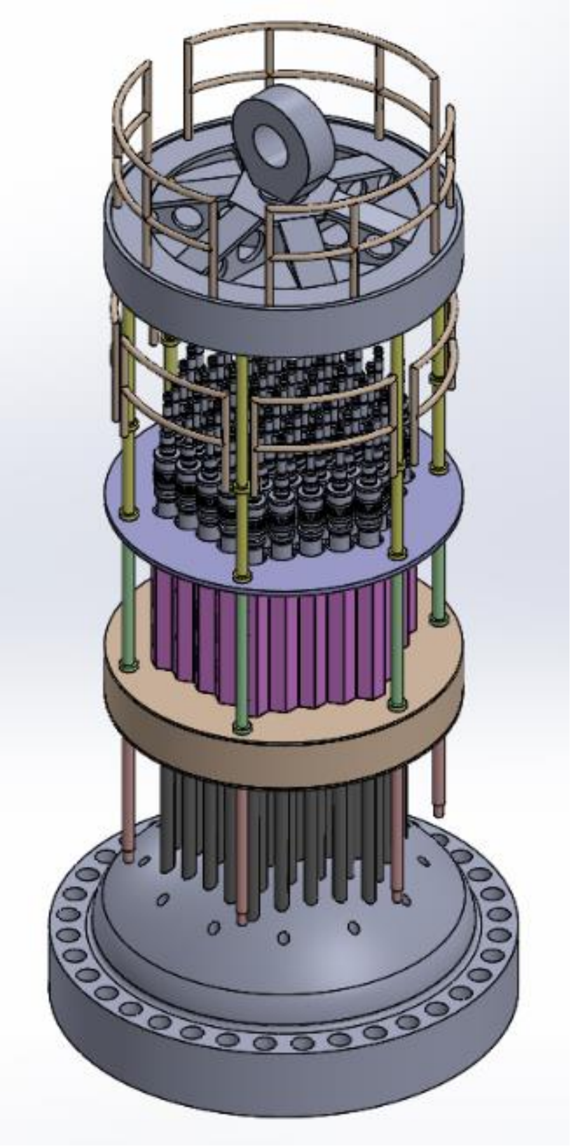


Core baffle

Reactor Pressure Vessel

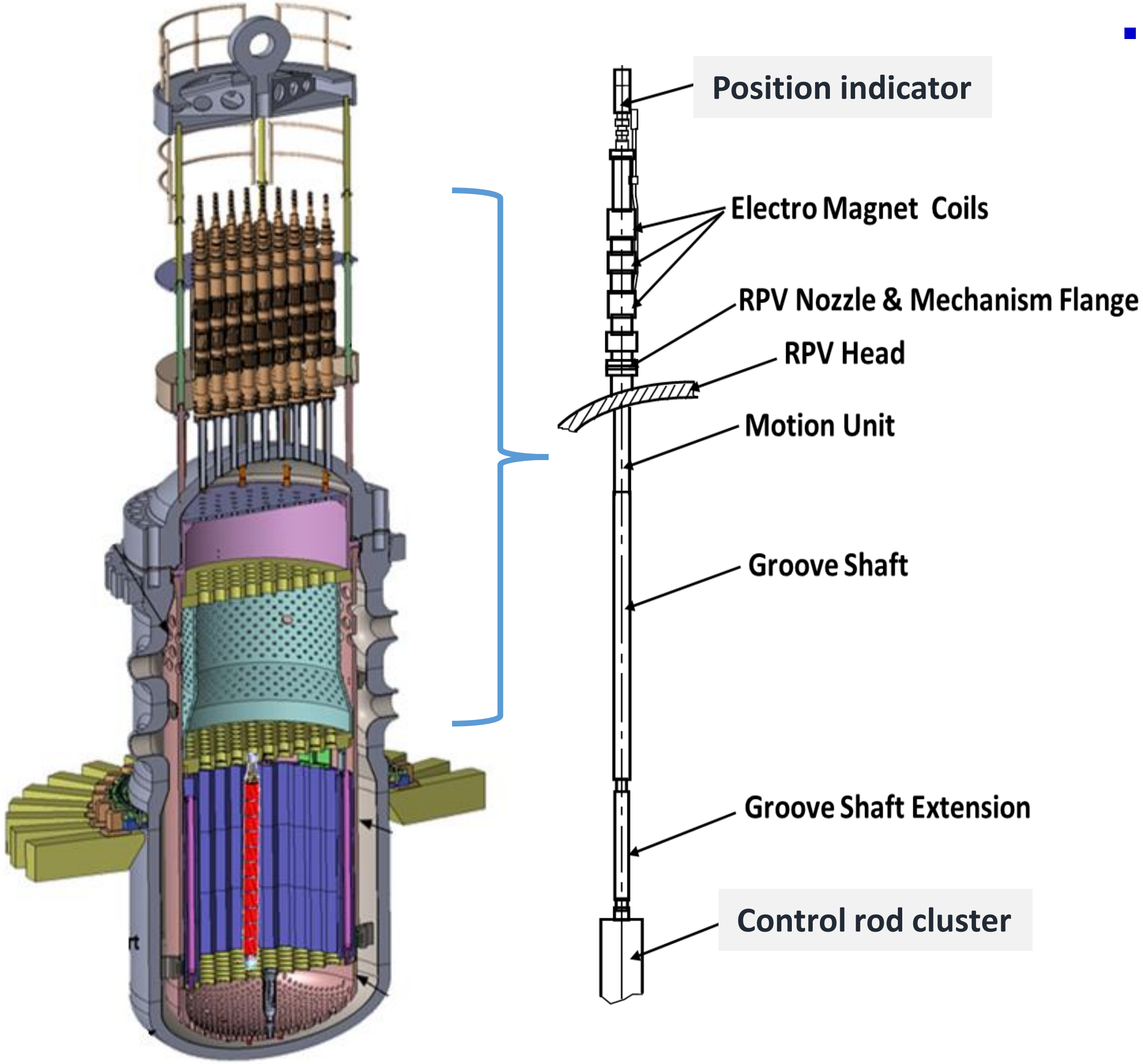




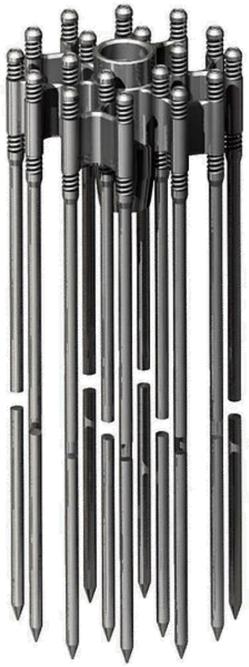


Parameter	Value / Approach
Design pressure	17.7 MPa
Design temperature	350 °C
Service Life	60 y
RPV material	APURVA

Reactivity Control Mechanism



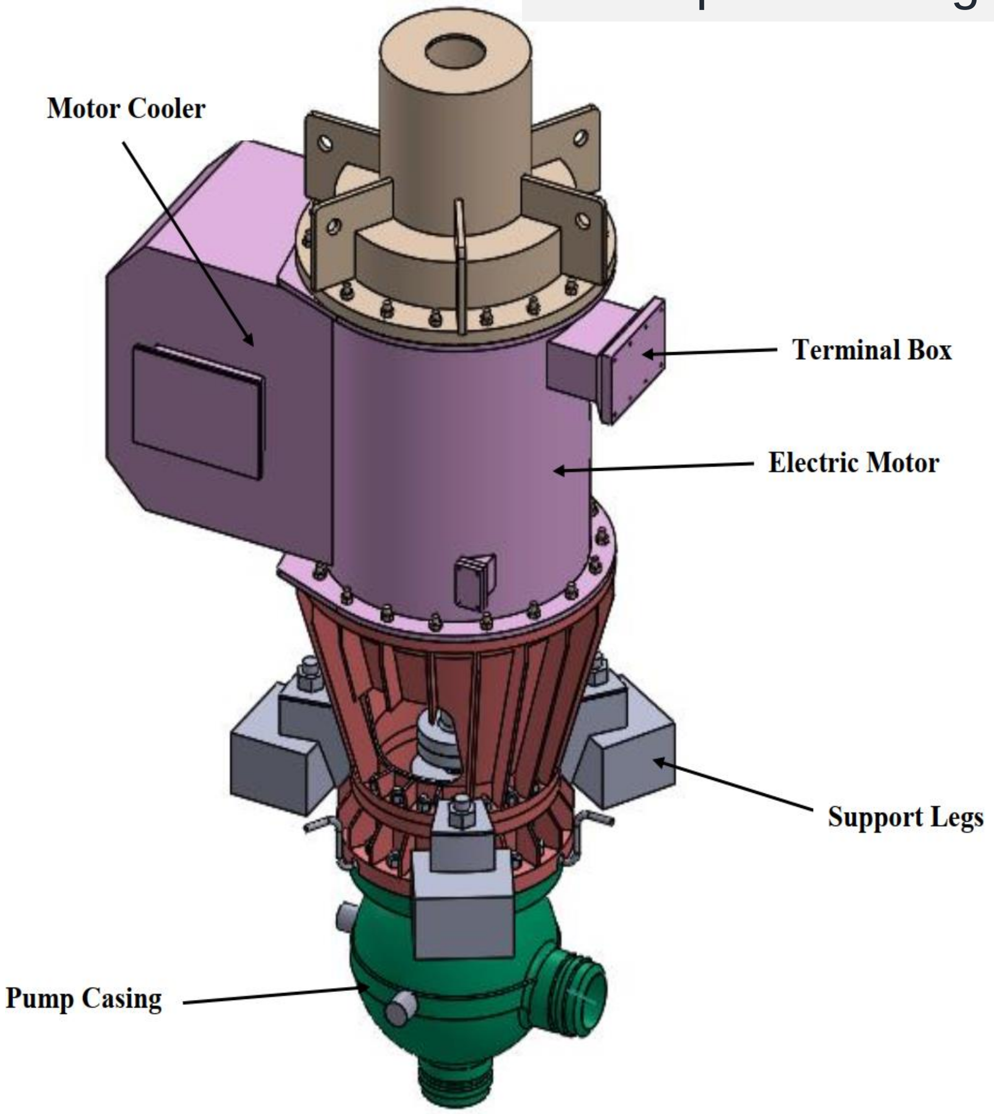
- Prototype mechanisms realized for operating conditions (325 ºC and 162 bar)

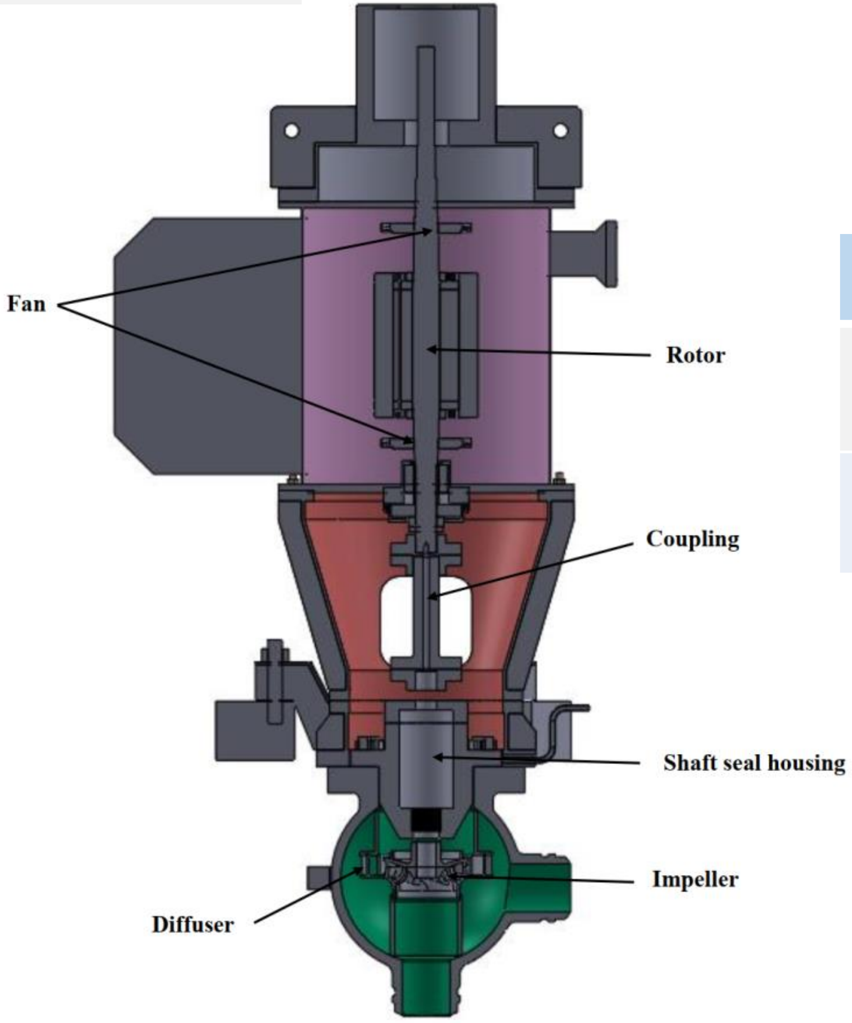

Control rod cluster


Prototype Mechanism

Reactor Coolant Pump

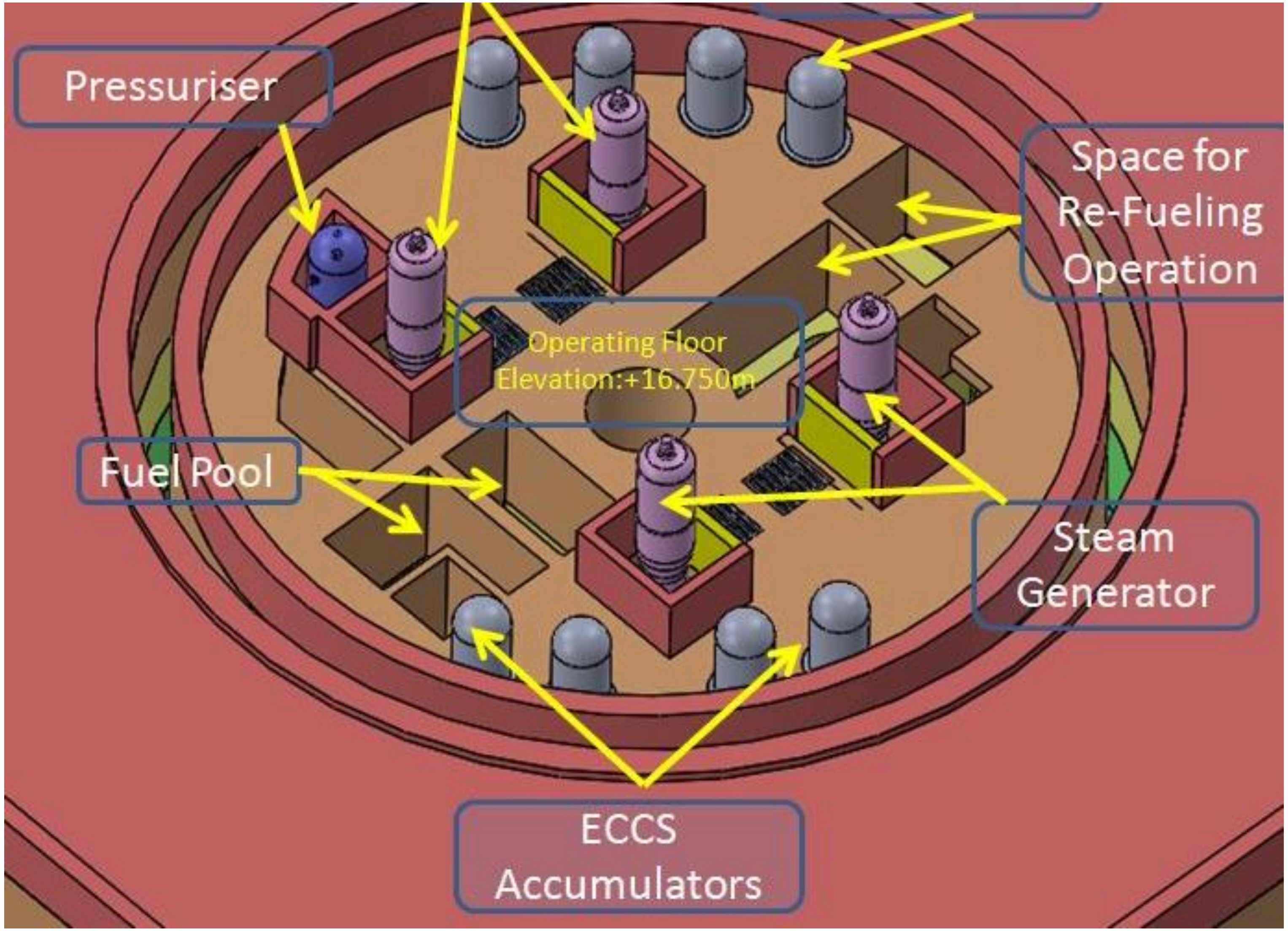
Conceptual Configuration - RCP



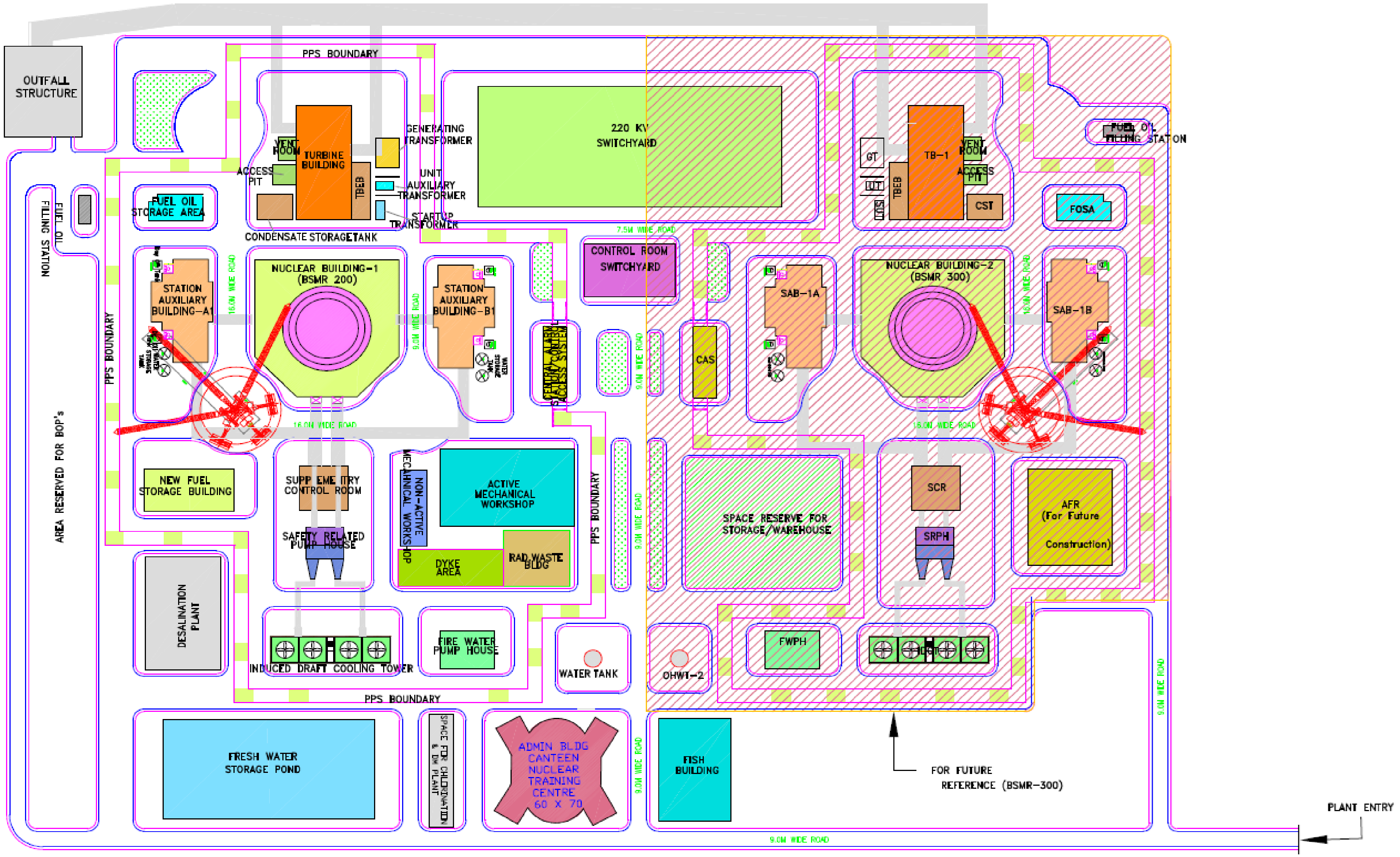


Parameter	Value
Rated Flow (m^3/h)	~5000
Pump Head (m)	~100

Nuclear Layout



BSMR-200 Plant Layout



APURVA Forgings Developed by BARC

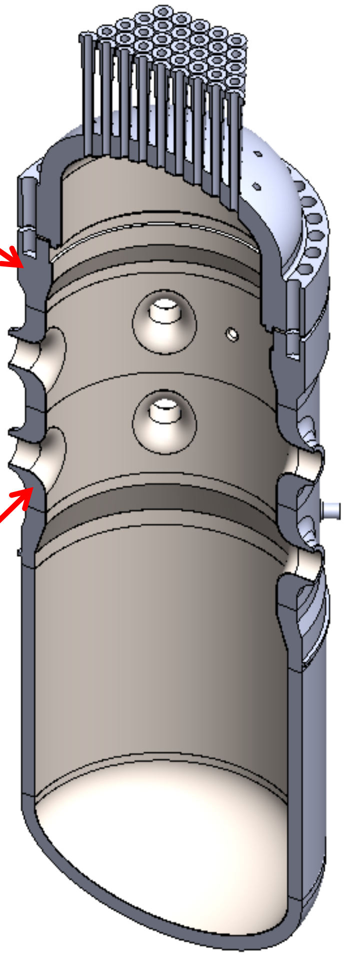
Advanced Purified Reactor Vessel Alloy

750 thick Ring Forging

(Φ3800 ID x 750 thk x 1200 L ; 100 t)

340 thick Shell Forging

(Φ4200 IDx340 thk x 3000 L ; 113 t)



Salient Features of APURVA Steel Forgings:

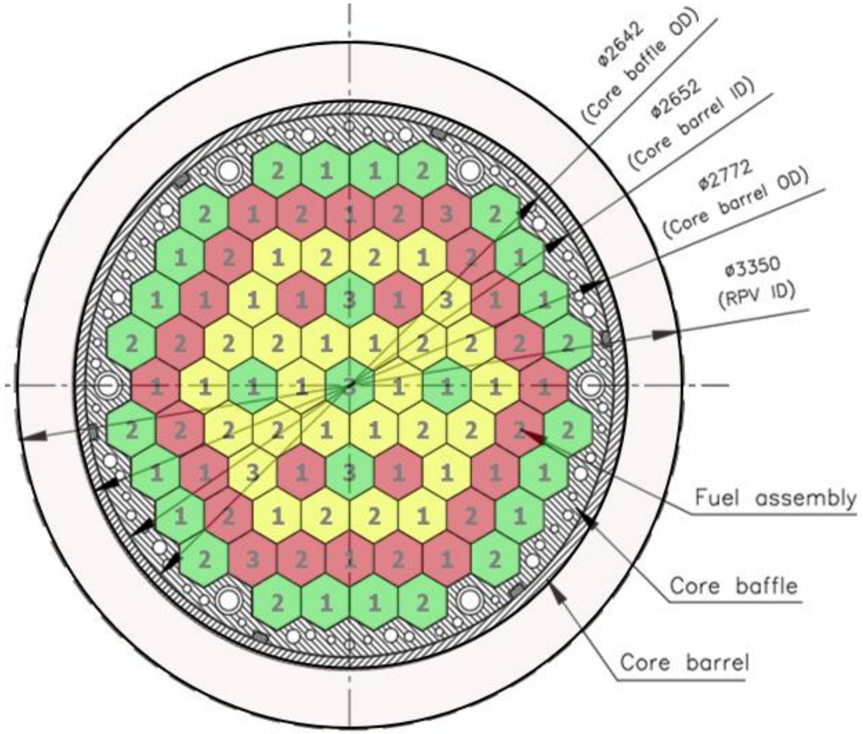
- *Si-modified* **Mn-Mo-Ni** Steel
- Equivalent to American & German RPV Steels
- Ultra-clean→ Very low P, S, Cu, As, Sb, Sn
- Hydrogen < 1.0 ppm
- High Strength & Toughness throughout section thickness
- High resistance to temper embrittlement
- High resistance to irradiation embrittlement
- Good Weldability

BSMR-200 : Technological Readiness

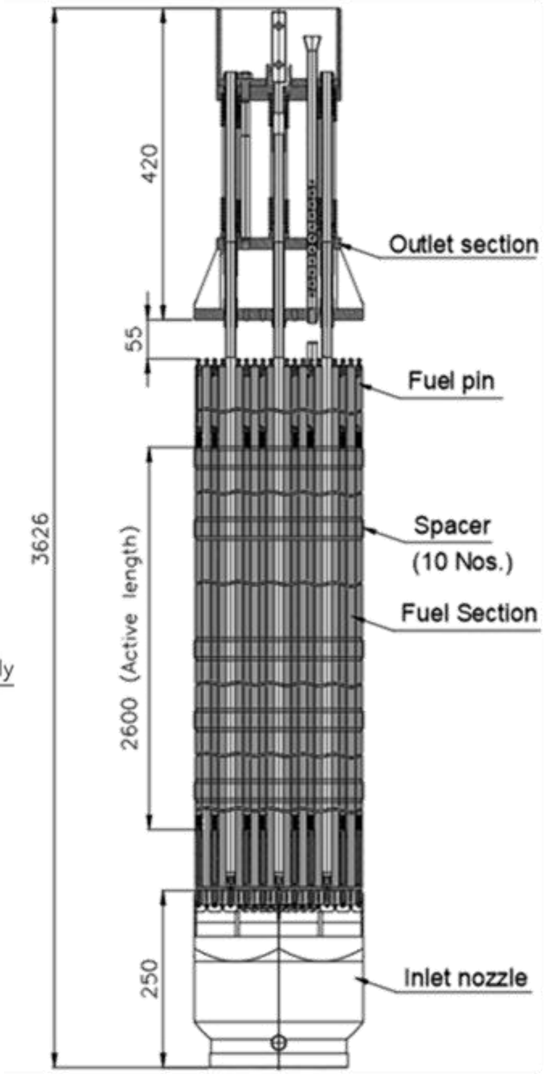
1. Reactor pressure vessel (RPV), the critical process equipment
   - APURVA grade low alloy steel forgings, developed by BARC
2. Steam generators (SGs), thermal link between reactor coolant system and secondary plant
   - PHWR-220 SGs will be suitable for BSMR-200
3. Reactivity control mechanism (RCM), for reactor control & protection
   - Prototype RCM developed by BARC
4. Reactor coolant pump (RCP)
   - Indigenous manufacturing
5. Steam turbine plant and electrical systems:
   - Existing systems of PHWR 220 suitable
6. Nuclear fuel
   - Based on conventional UO_2 pellets

Reactor Core Thermal Hydraulics of BSMR 200

Reactor Core and Thermal Hydraulics



Reactor Assembly

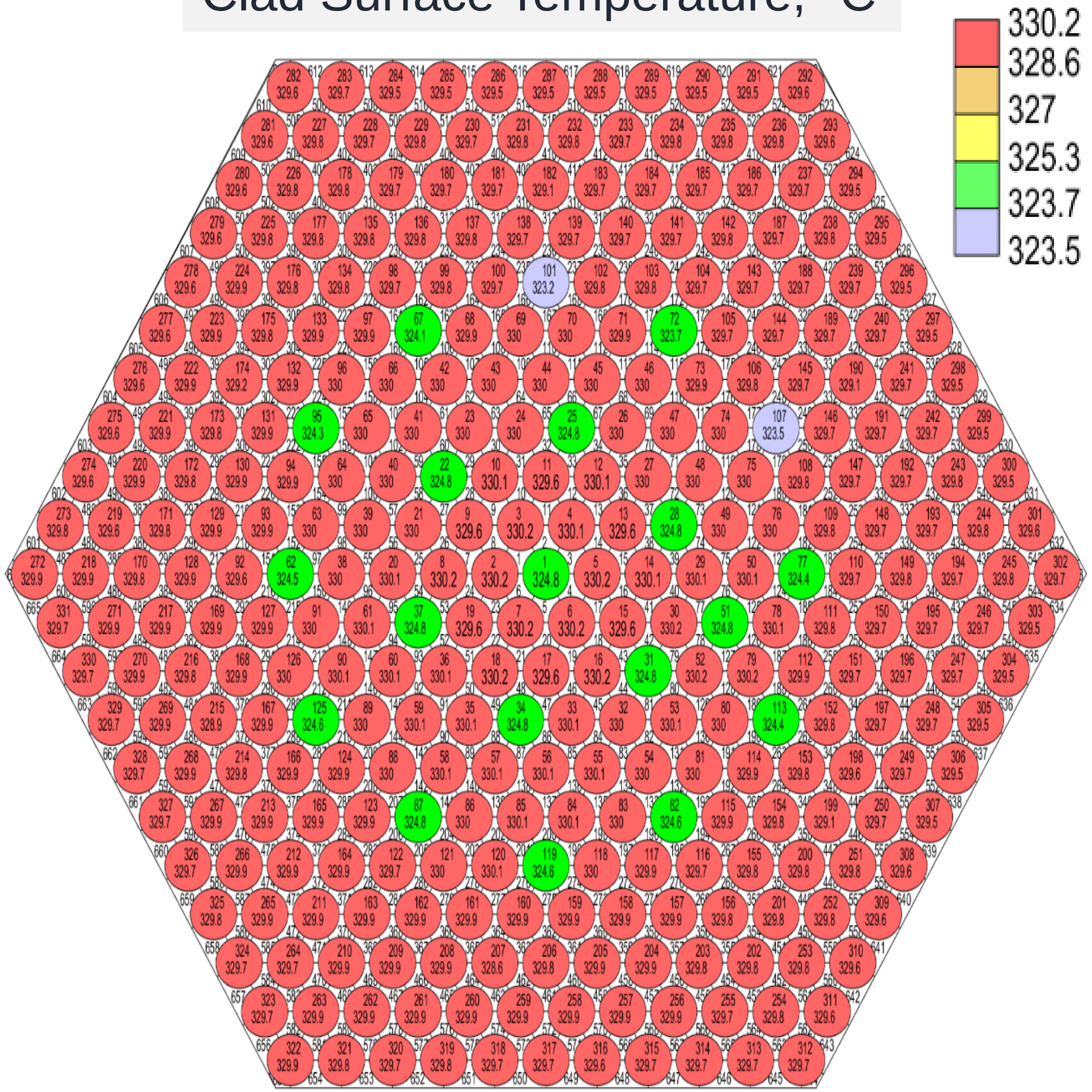


Fuel Assembly

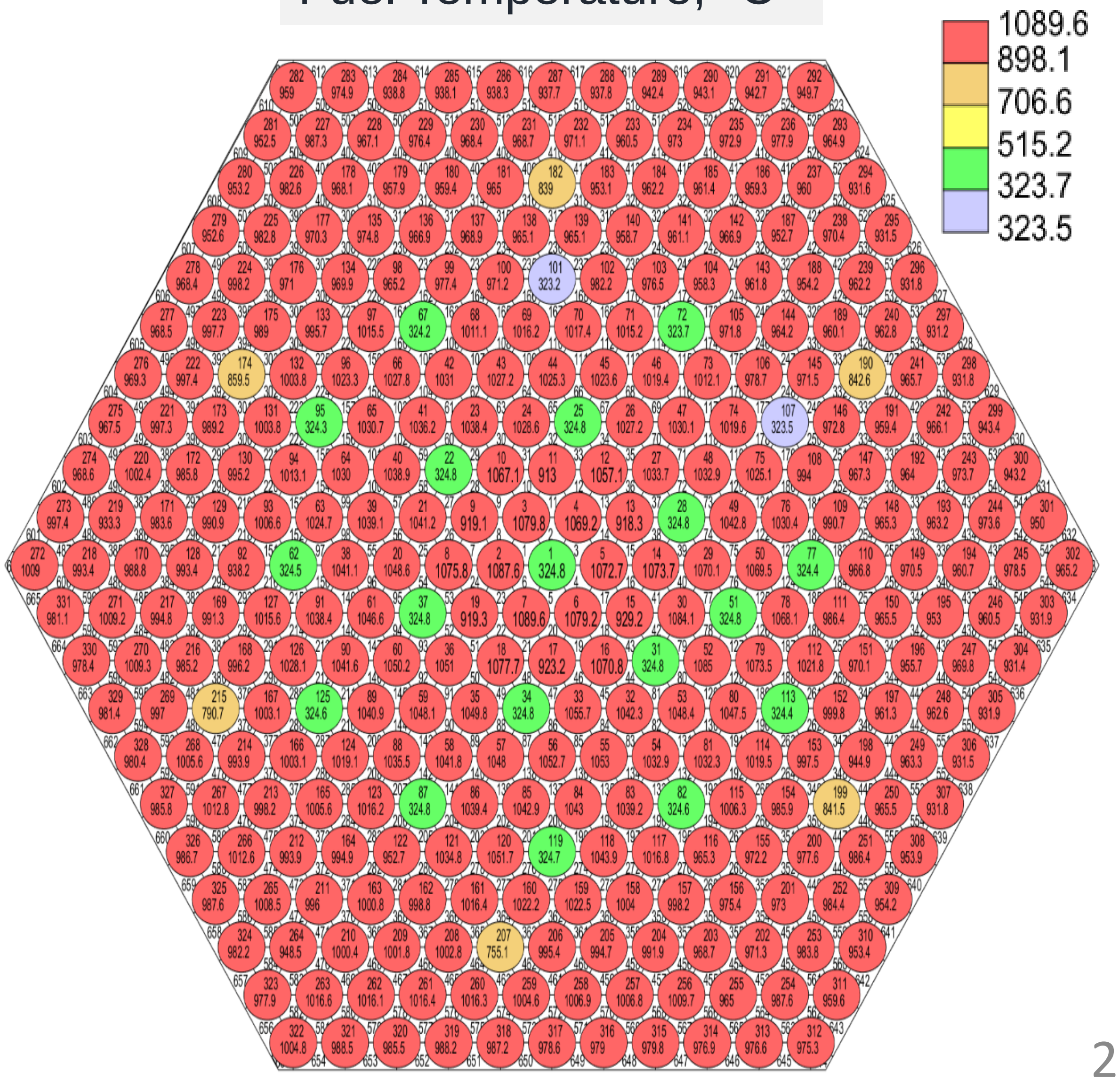
Thermal-hydraulic Parameters (hottest FA)	
Parameter	**Value**
FA power, MW	15.67
FA inlet flow, kg/s	44.8
Average/peak LHR ,W/cm	194/272
Average FA outlet temperature, ^{o}C	320.4
Maximum heat flux, MW/m^2	0.92
Maximum clad temperature, ^{o}C	330
Maximum fuel temperature, ^{o}C	1090

Core Thermal hydraulics: Results

Clad Surface Temperature, °C



Fuel Temperature, °C



Safety Assessment of BSMR 200

Outline

1. Introduction to Indigenous SMRs

2. Design & Basic Overview of Bharat Small Modular Reactor (BSMR-200)

3. Fundamental of Reactor Safety

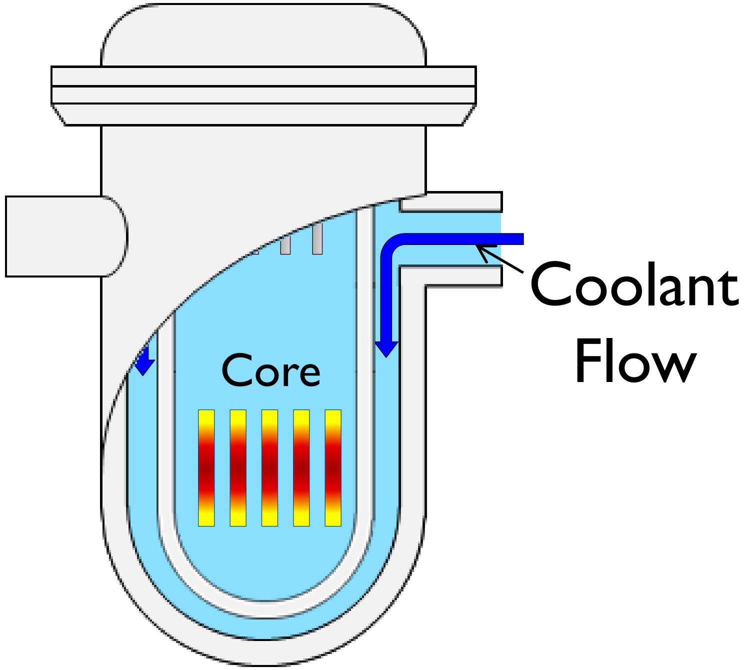
4. Design and Design Validation Assessment
   - Containment Peak Pressure Analysis
   - RPV Design Qualification under Pressurised Thermal Shock
   - Emergency Core Cooling System of BSMR 200

5. Safety Analysis
   - Design Basis Accident
   - Design Extension Condition
   - Radiological Impact Assessment

6. Summary

Fundamental of Reactor Safety

Heat Generation = Heat Removal → Normal Operation



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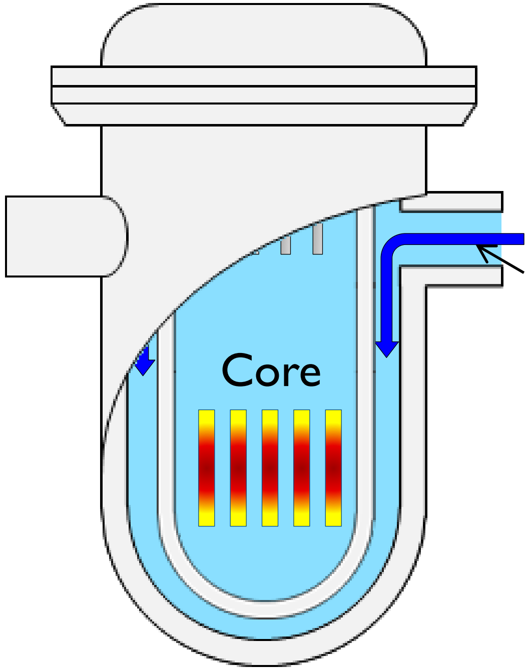
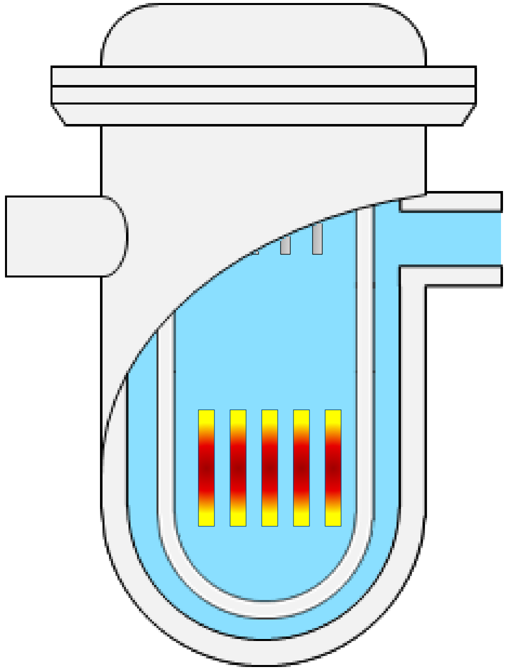


If, <u>deviation</u> from Normal Operation

Fundamental of Reactor Safety



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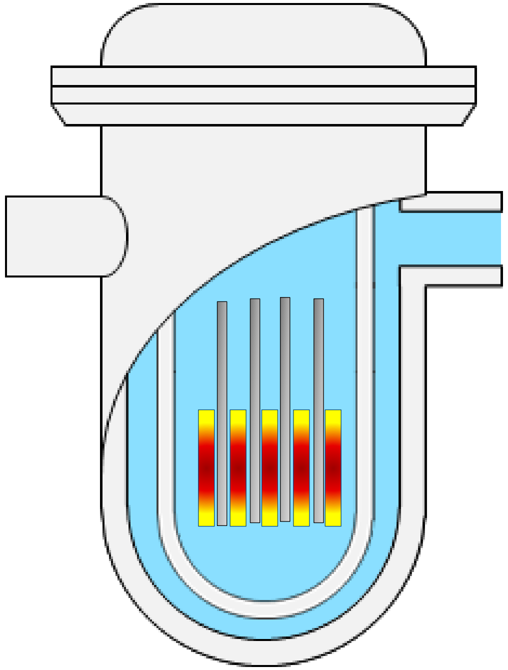
↓

<u>Accident Postulate:</u> Break in primary heat transport system

Fundamental of Reactor Safety



Heat Generation = Heat Removal → Normal Operation

If, deviation from Normal Operation

Accident Postulate: Break in primary heat transport system

Control

Control & Protection System

Fundamental of Reactor Safety

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If, deviation from Normal Operation

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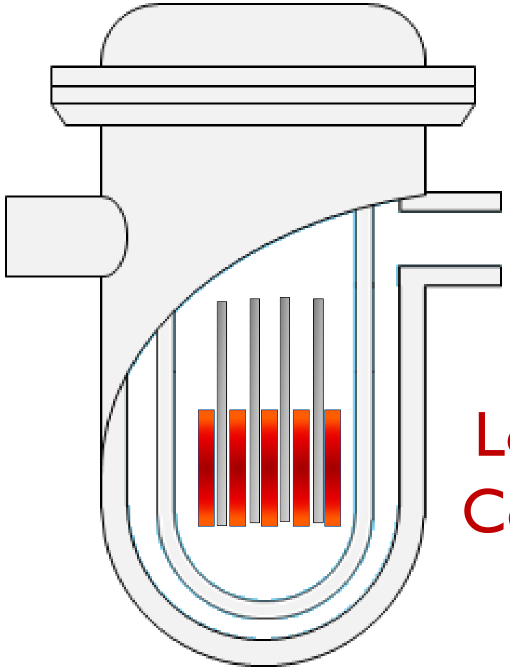

Loss of Coolant

Control

Control & Protection System

Cool

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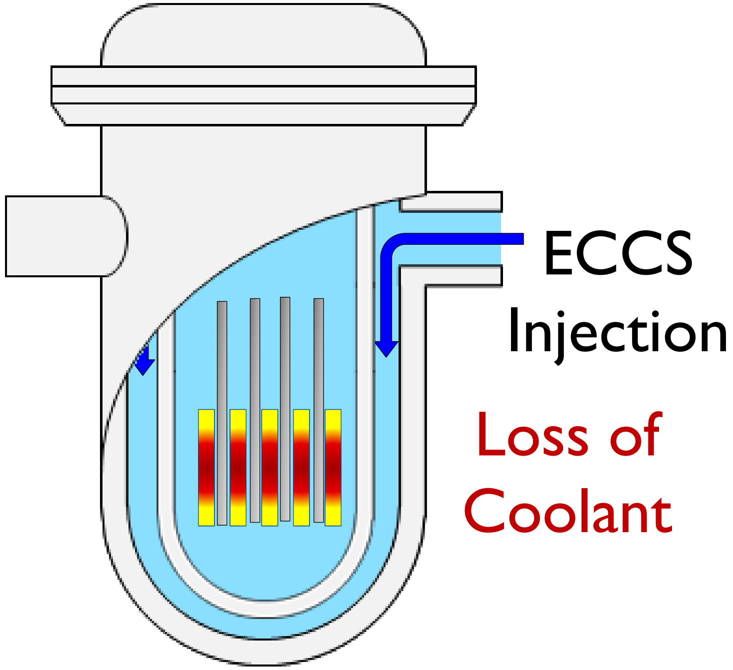

Control

Control & Protection System

Cool

PHT & Engineered Safety Features

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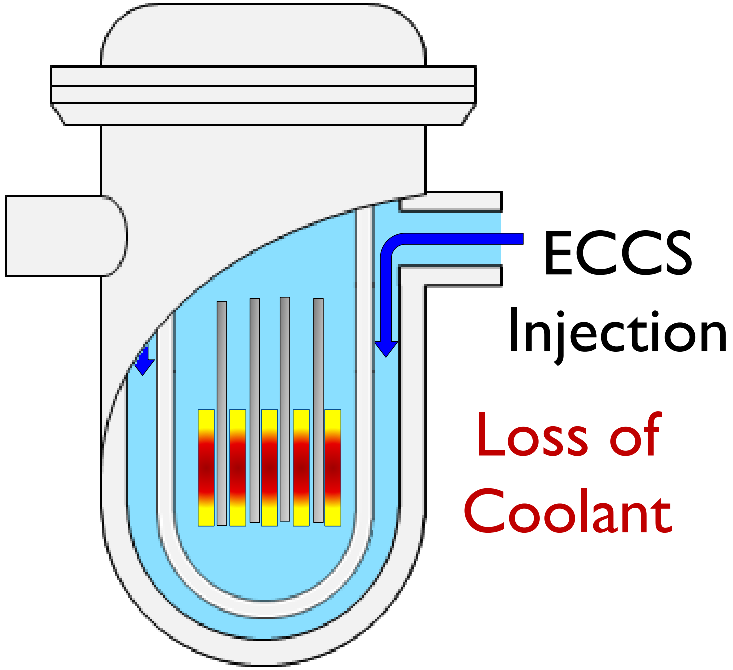

Control
- Control & Protection System

Cool
- PHT & Engineered Safety Features

Confine
- Multiple physical barriers

Fundamental of Reactor Safety

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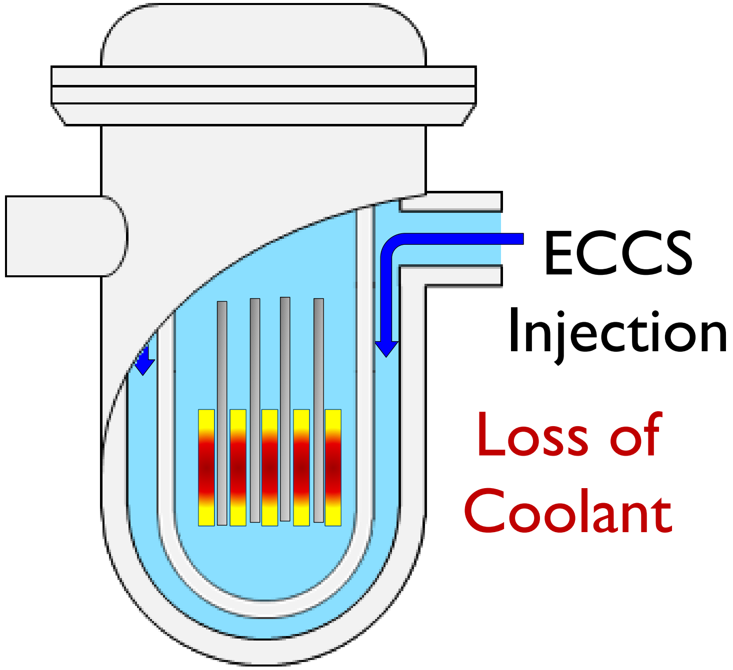

Control
Control & Protection System

Cool
PHT & Engineered Safety Features

Confine
Multiple physical barriers

3Cs achieved by deploying defence-in-depth (DiD) in design

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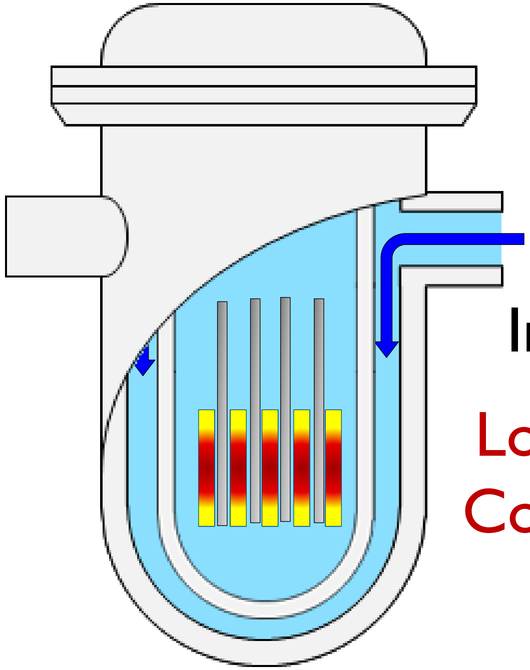

Control → Control & Protection System

Cool → PHT & Engineered Safety Features

Confine → Multiple physical barriers

3Cs achieved by deploying defence-in-depth (DiD) in design

Aim to limit radiation dose to public & hazard to environment

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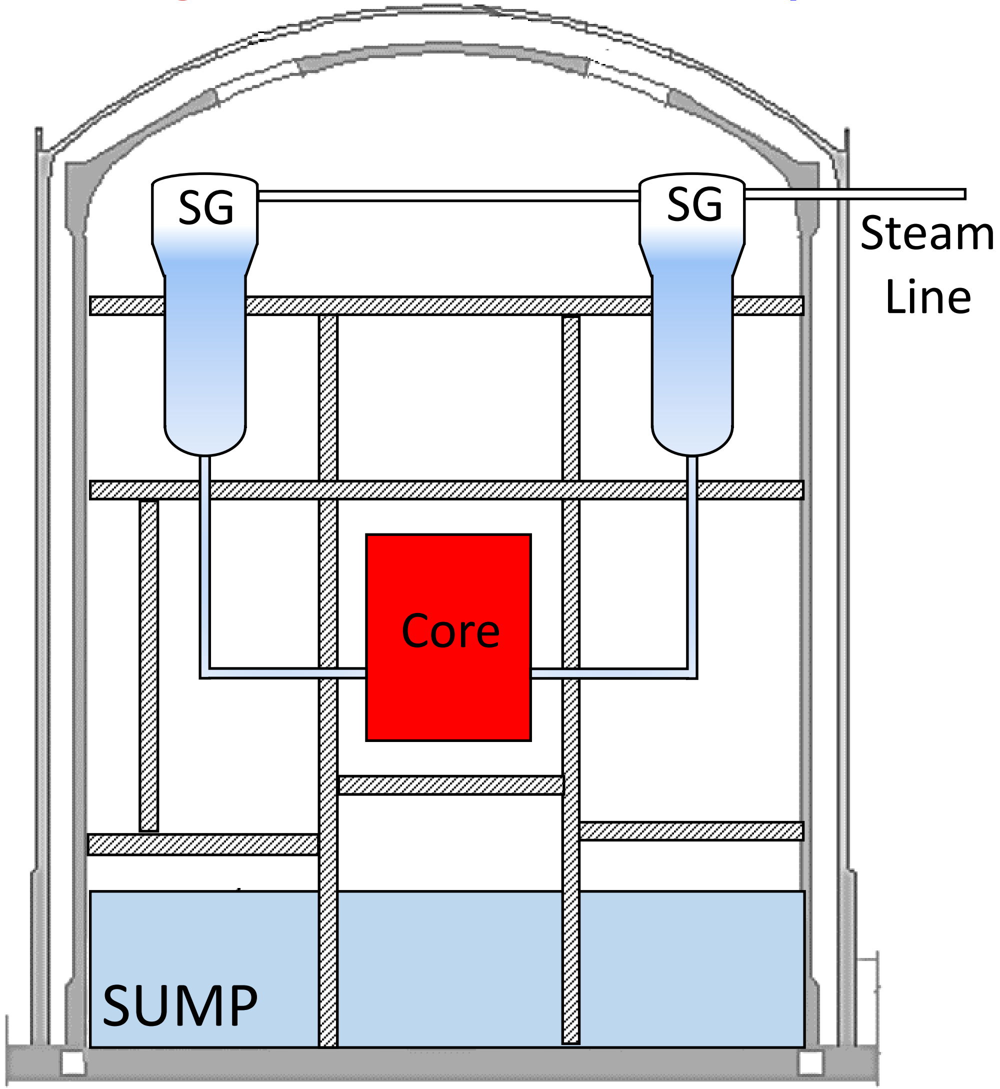
Design of the ultimate physical barrier: *Containment*

Hermetically sealed containment: Steel-lined ➡ low leakage ➡ low risk to the public



Design of the ultimate physical barrier: ***Containment***

Hermetically sealed containment: Steel-lined → low leakage → low risk to the public

Containment Thermal Hydraulic Analysis

Identify Postulated Initiating Event

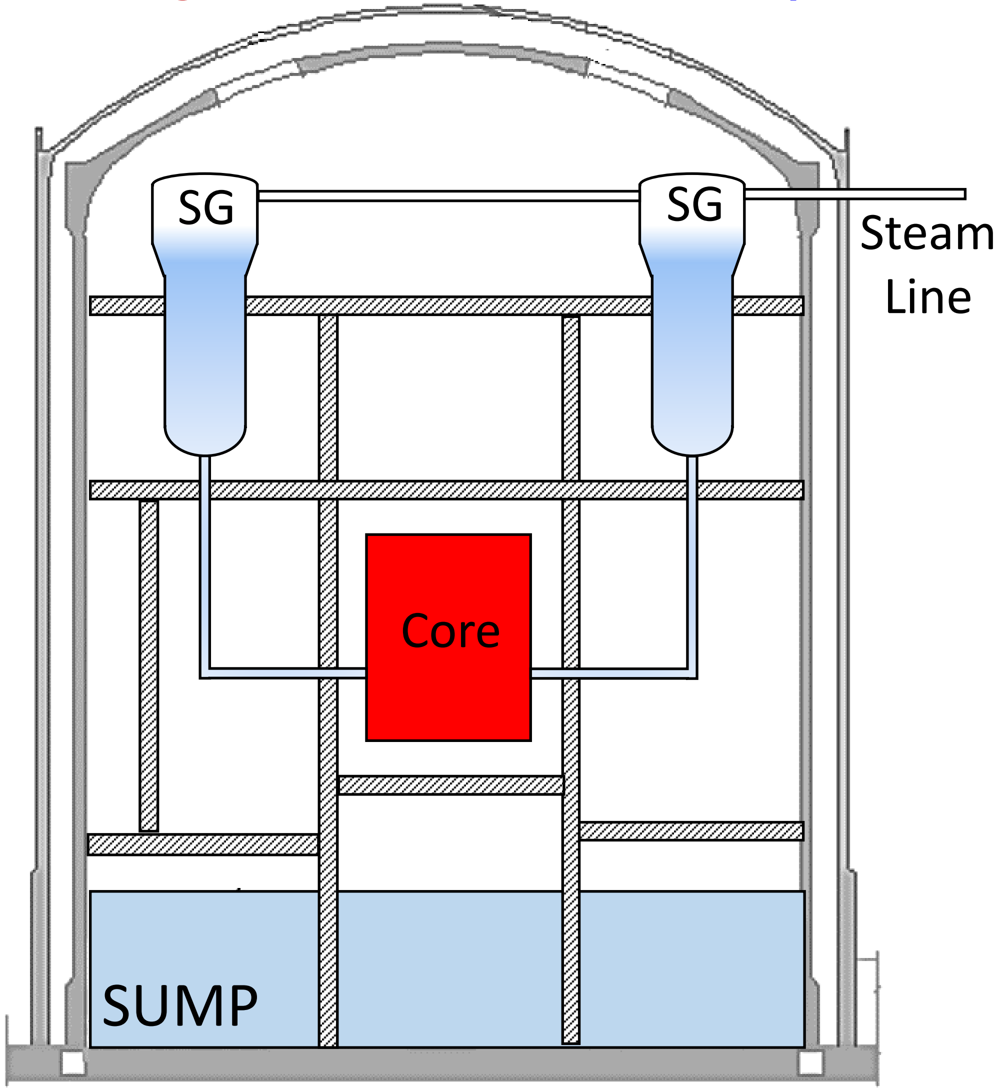
Critical ones:
1. LOCA
2. MSLB

Plant Modelling

Detailed Containment Geometry

Numerical Simulation

Solve for Mass, Momentum & Energy Conservation



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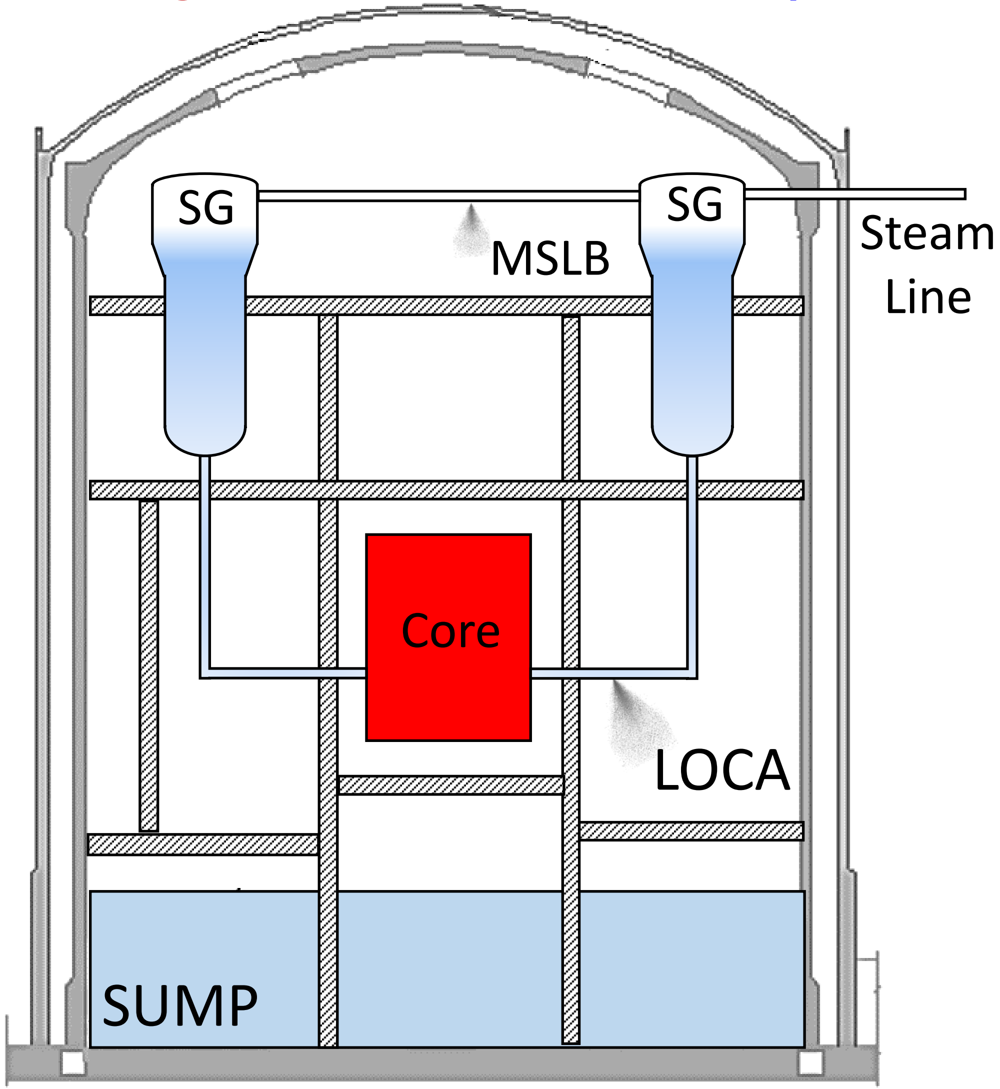

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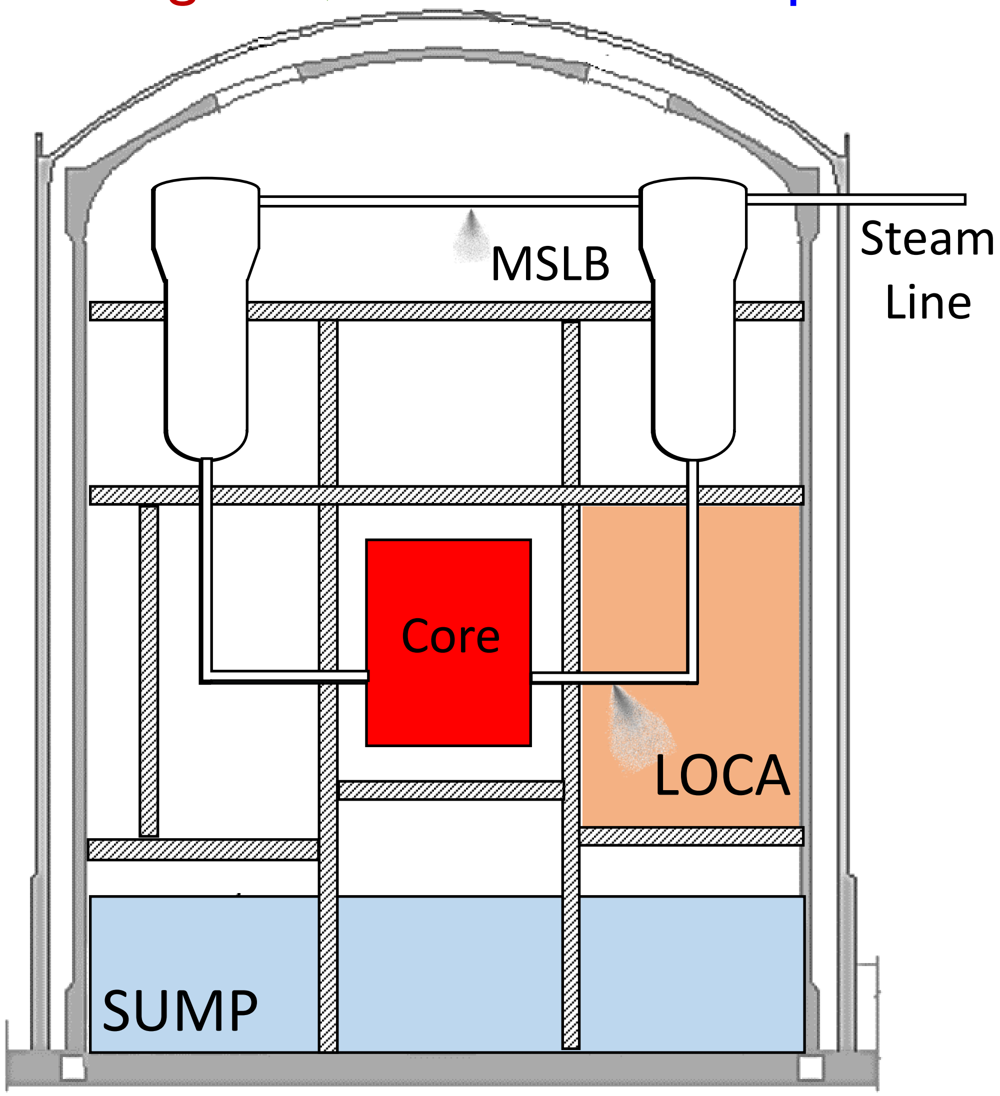

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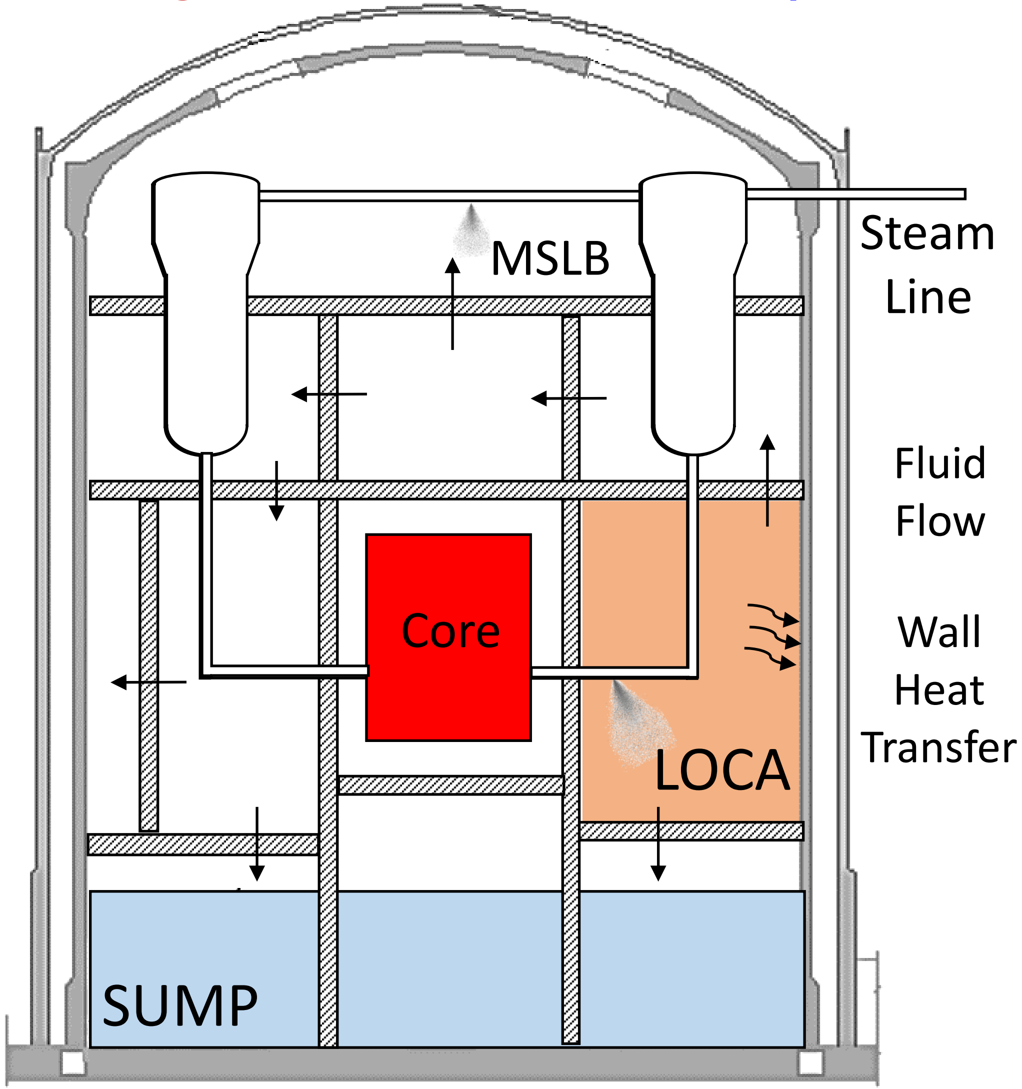
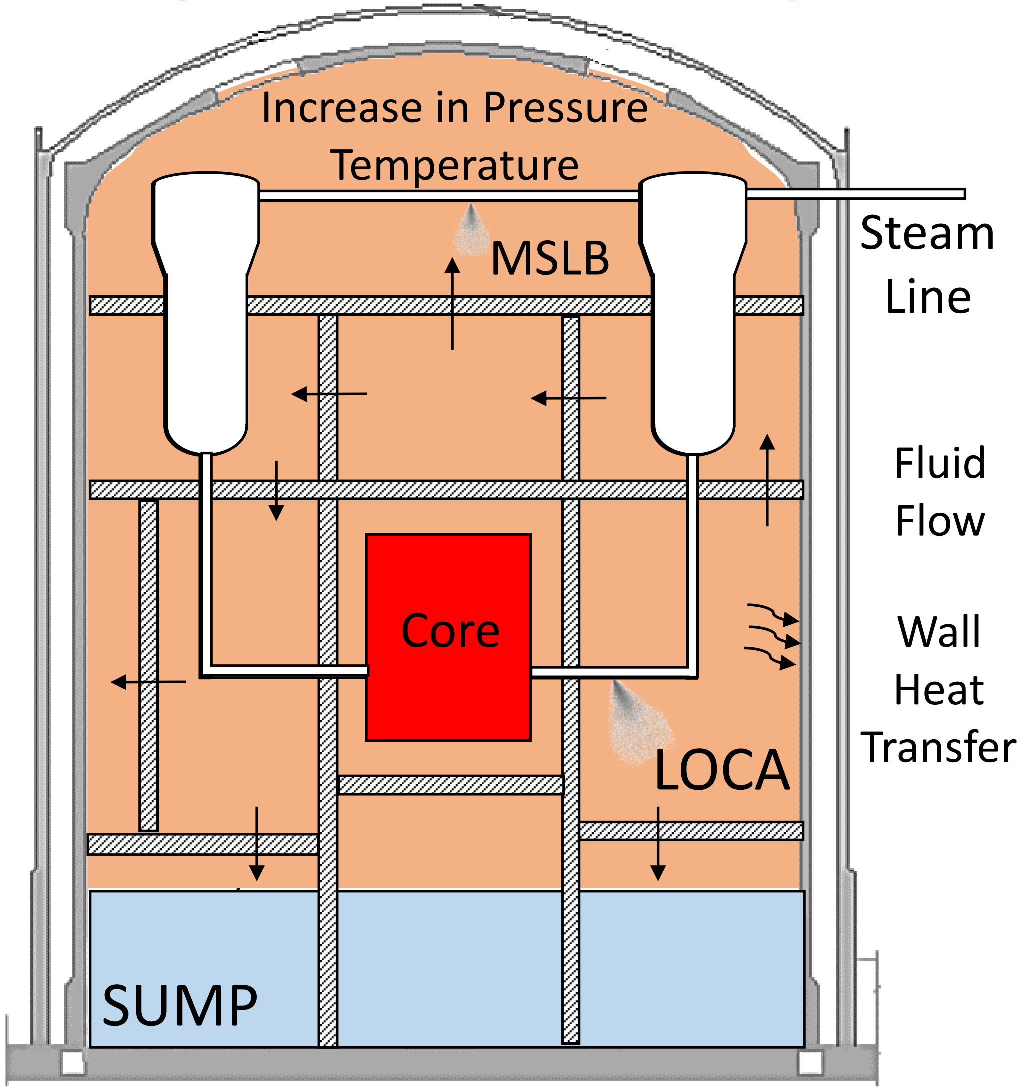


Design of the ultimate physical barrier: *Containment*

Hermetically sealed containment: Steel-lined → low leakage → low risk to the public

Containment Thermal Hydraulic Analysis

Identify Postulated Initiating Event	Plant Modelling	Numerical Simulation
Critical ones: 1. LOCA 2. MSLB	Detailed Containment Geometry	Solve for Mass, Momentum & Energy Conservation



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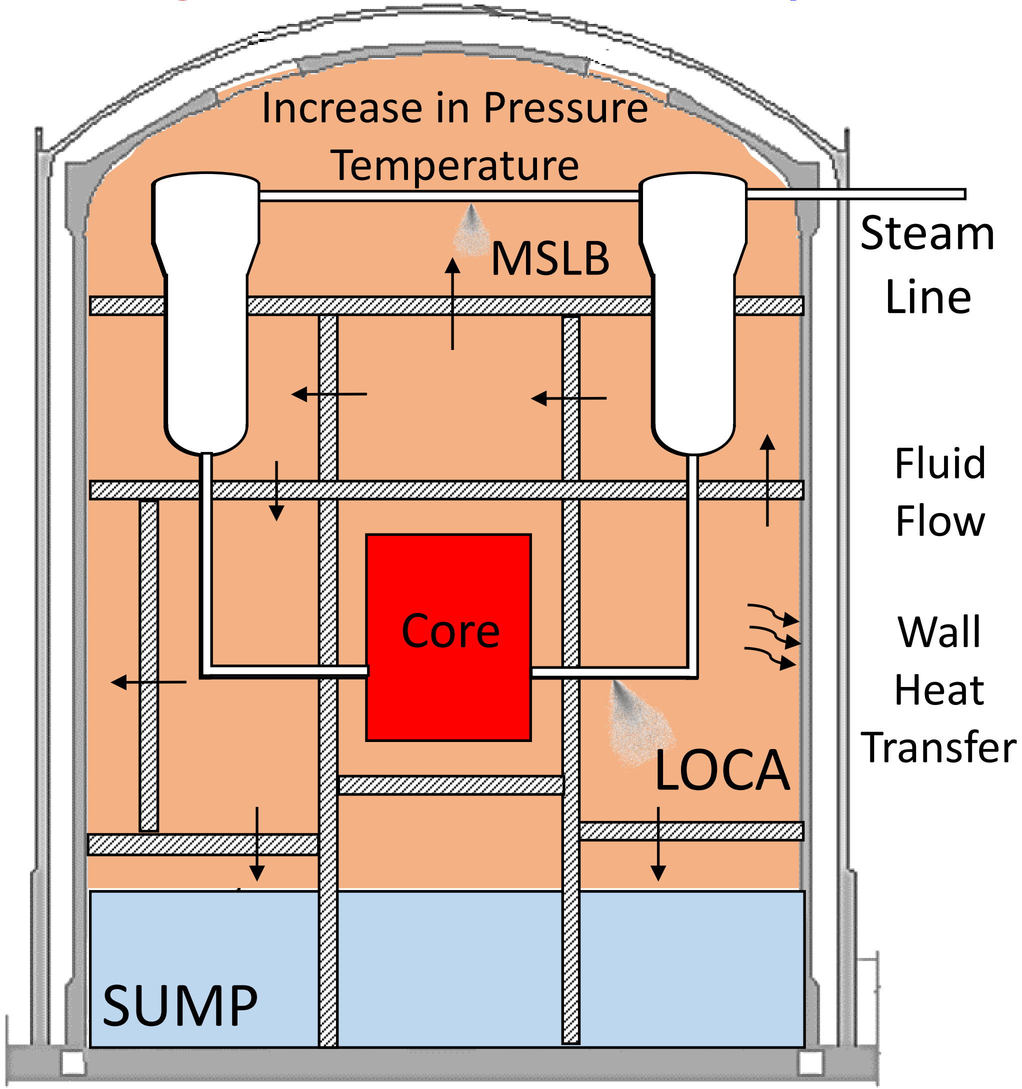
<u>*Containment Thermal Hydraulic Analysis*</u>

Identify Postulated Initiating Event
Critical ones:
1. LOCA
2. MSLB

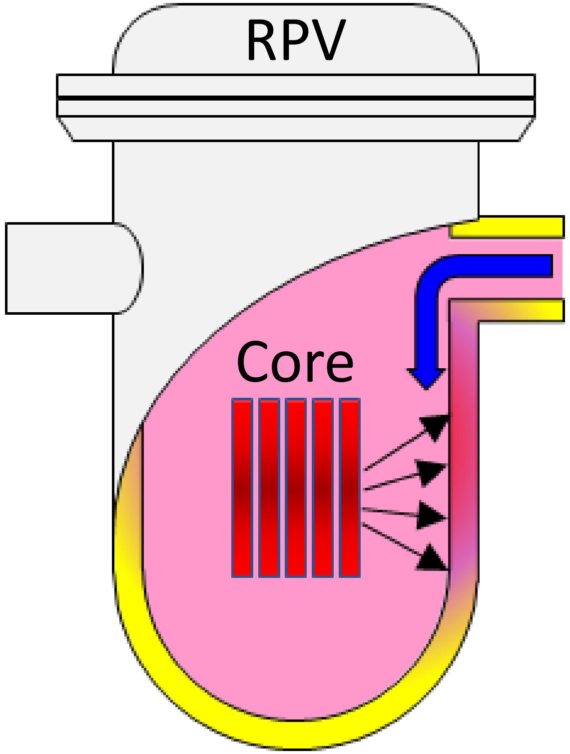
Plant Modelling
Detailed Containment Geometry

Numerical Simulation
Solve for Mass, Momentum & Energy Conservation

Scenario	Peak Pressure
Main Steam Line Break (MSLB)	1.73 kg/cm^2 (g)
Large Break Loss of Coolant Accident (LB-LOCA)	1.4 kg/cm^2 (g)



RPV Design Qualification for Pressurized Thermal Shock



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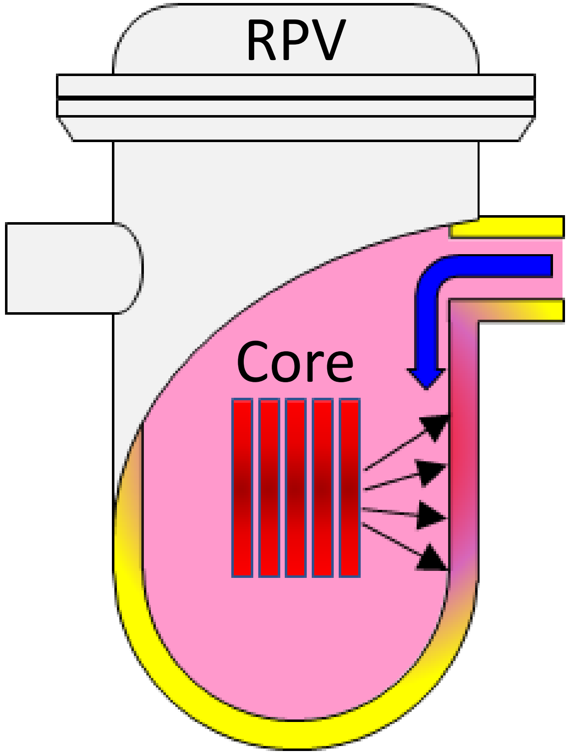
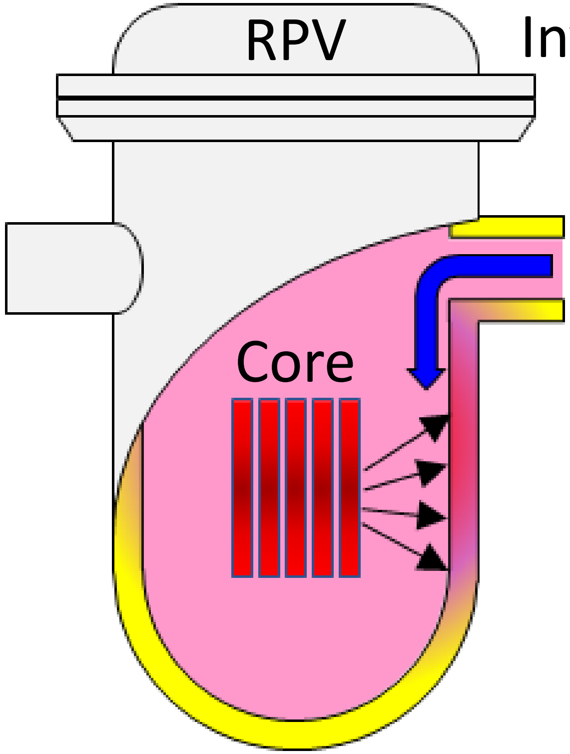


Initiating Event

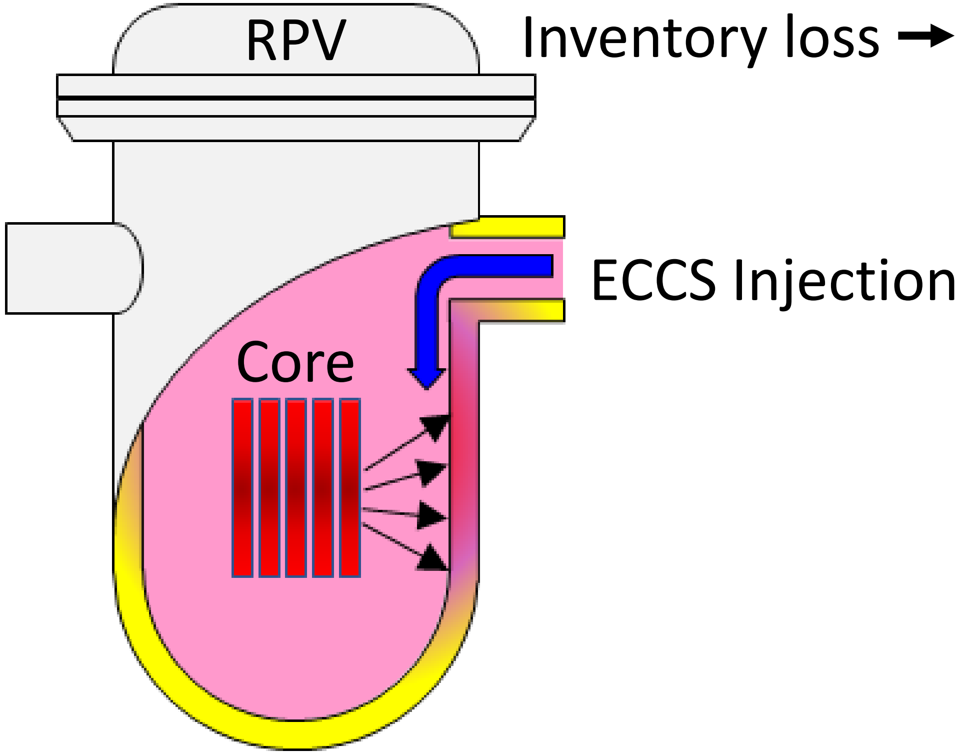
RPV Design Qualification for Pressurized Thermal Shock



Inventory loss → Steep fall in PHT pressure

Initiating Event

RPV Design Qualification for Pressurized Thermal Shock



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Initiating Event

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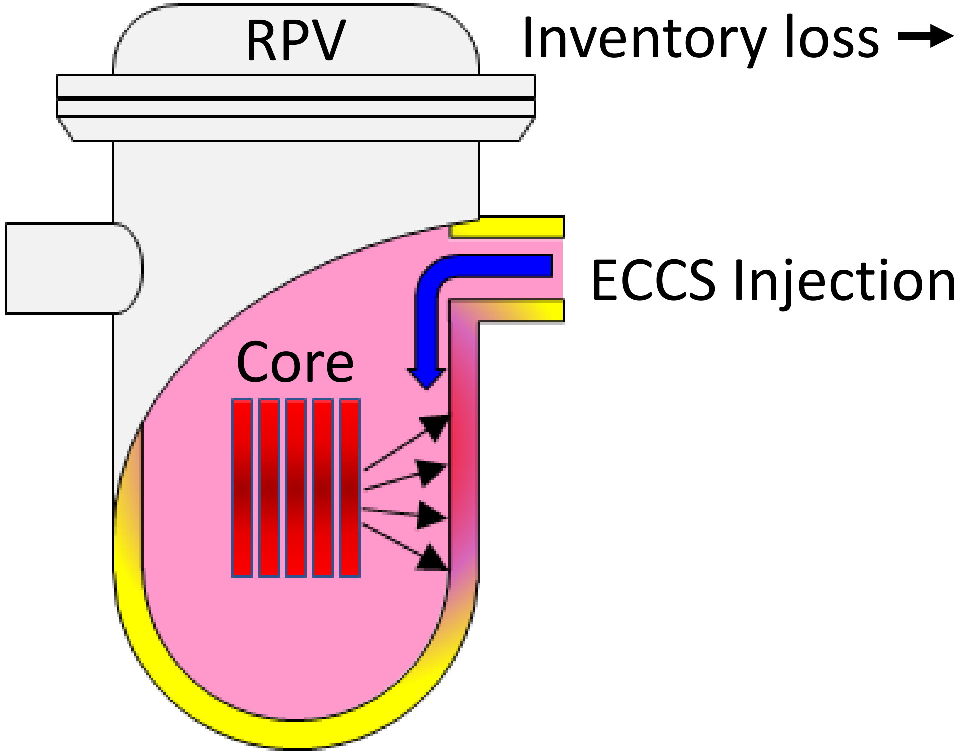


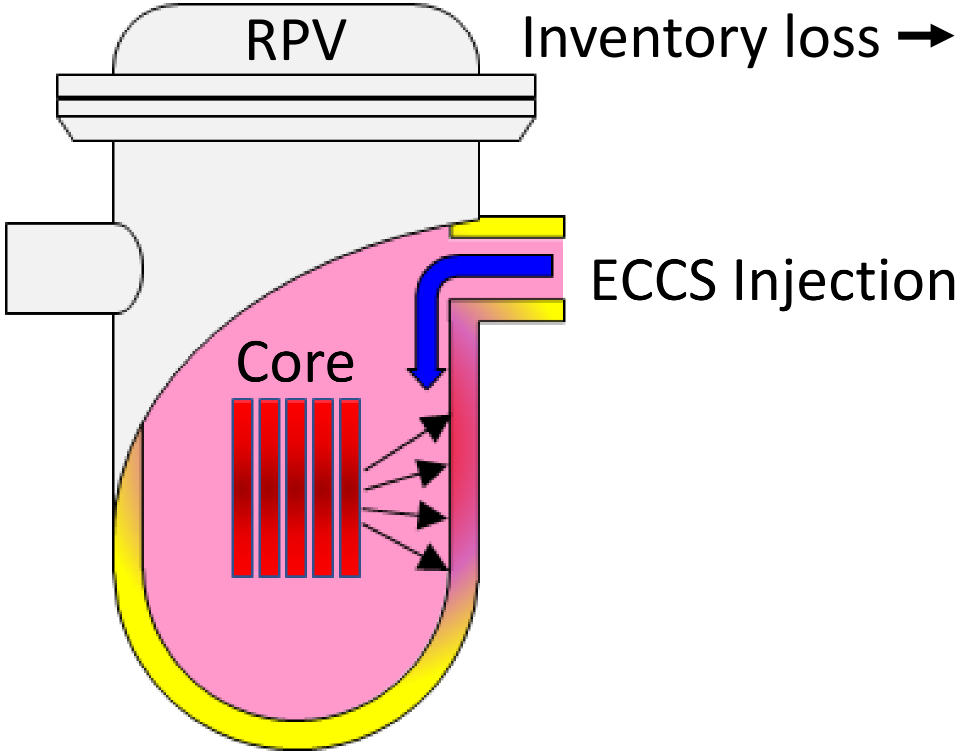
Inventory loss → Steep fall in PHT pressure

Initiating Event

Internal Pressure

Overcooling of RPV Wall

RPV Design Qualification for Pressurized Thermal Shock



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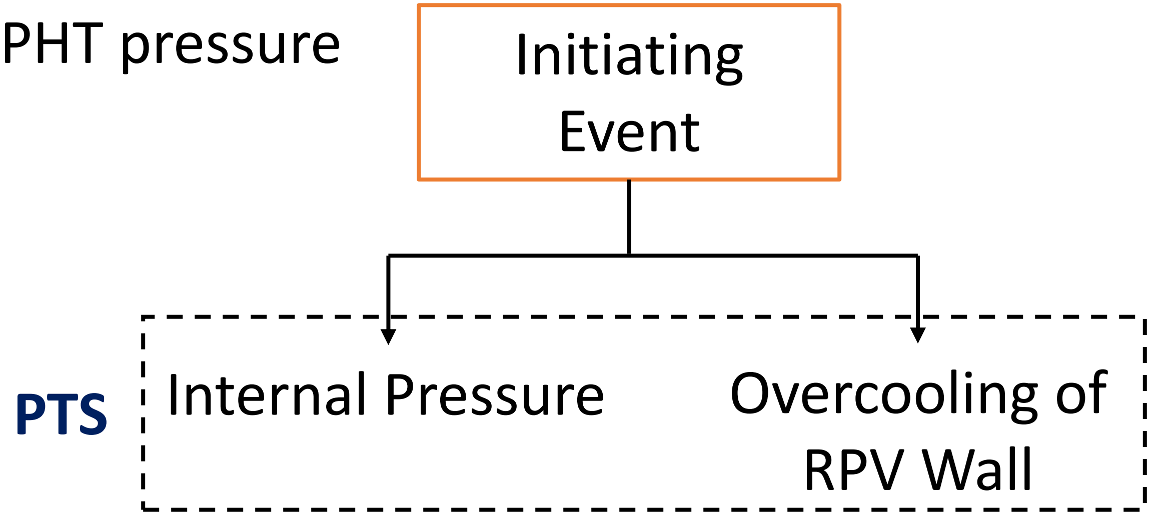

RPV Design Qualification for Pressurized Thermal Shock

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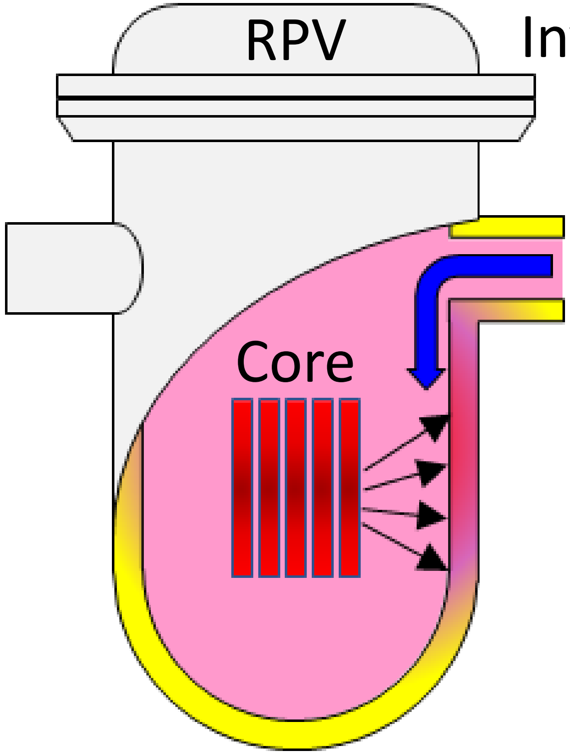


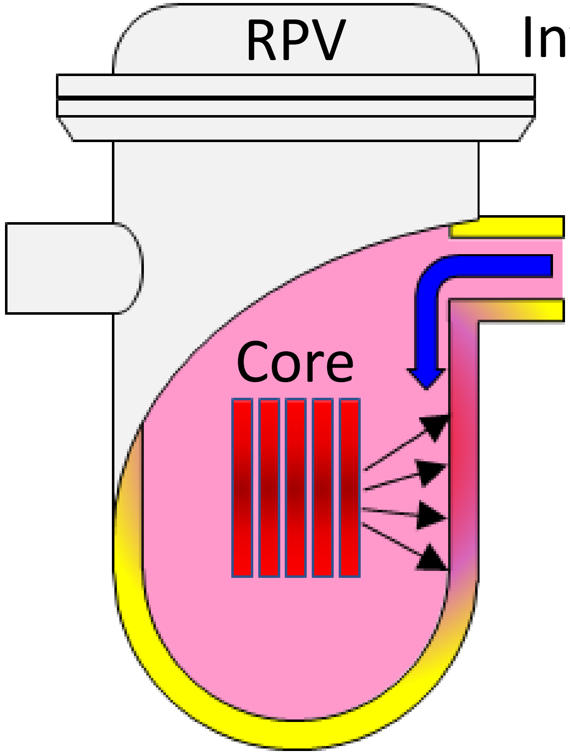
Initiating Event

PTS

- Internal Pressure → Mechanical Stress
- Overcooling of RPV Wall

RPV Design Qualification for Pressurized Thermal Shock

Inventory loss → Steep fall in PHT pressure



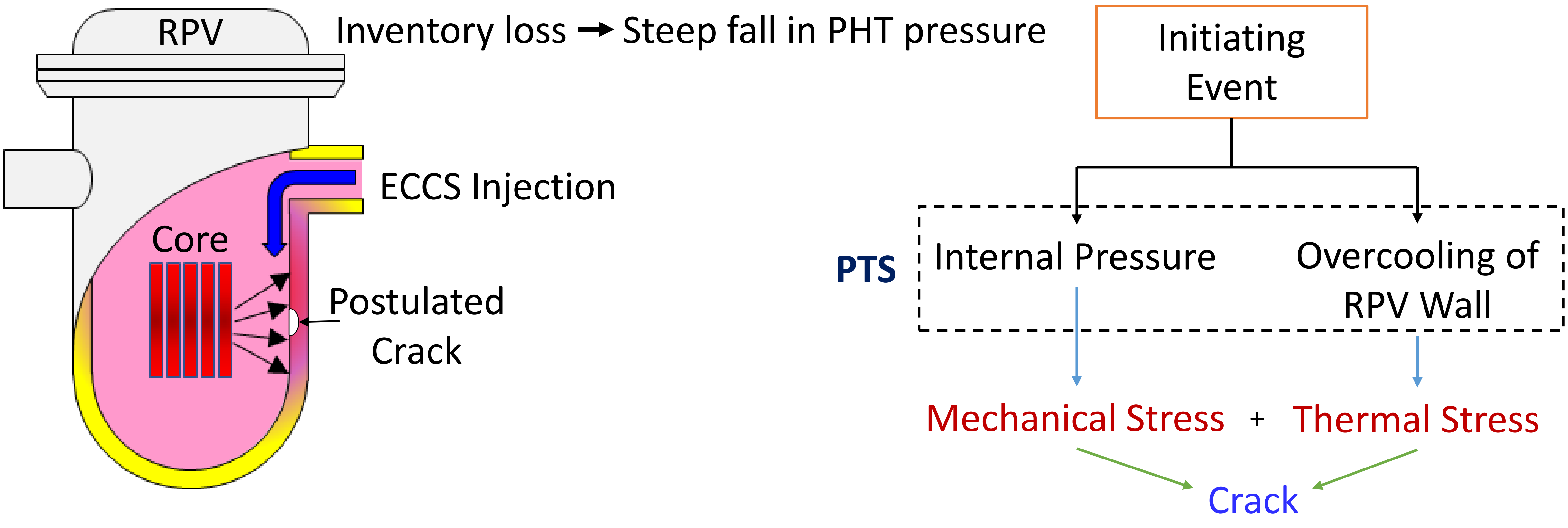
ECCS Injection

Initiating Event

PTS

- Internal Pressure → Mechanical Stress
- Overcooling of RPV Wall → Thermal Stress

RPV Design Qualification for Pressurized Thermal Shock



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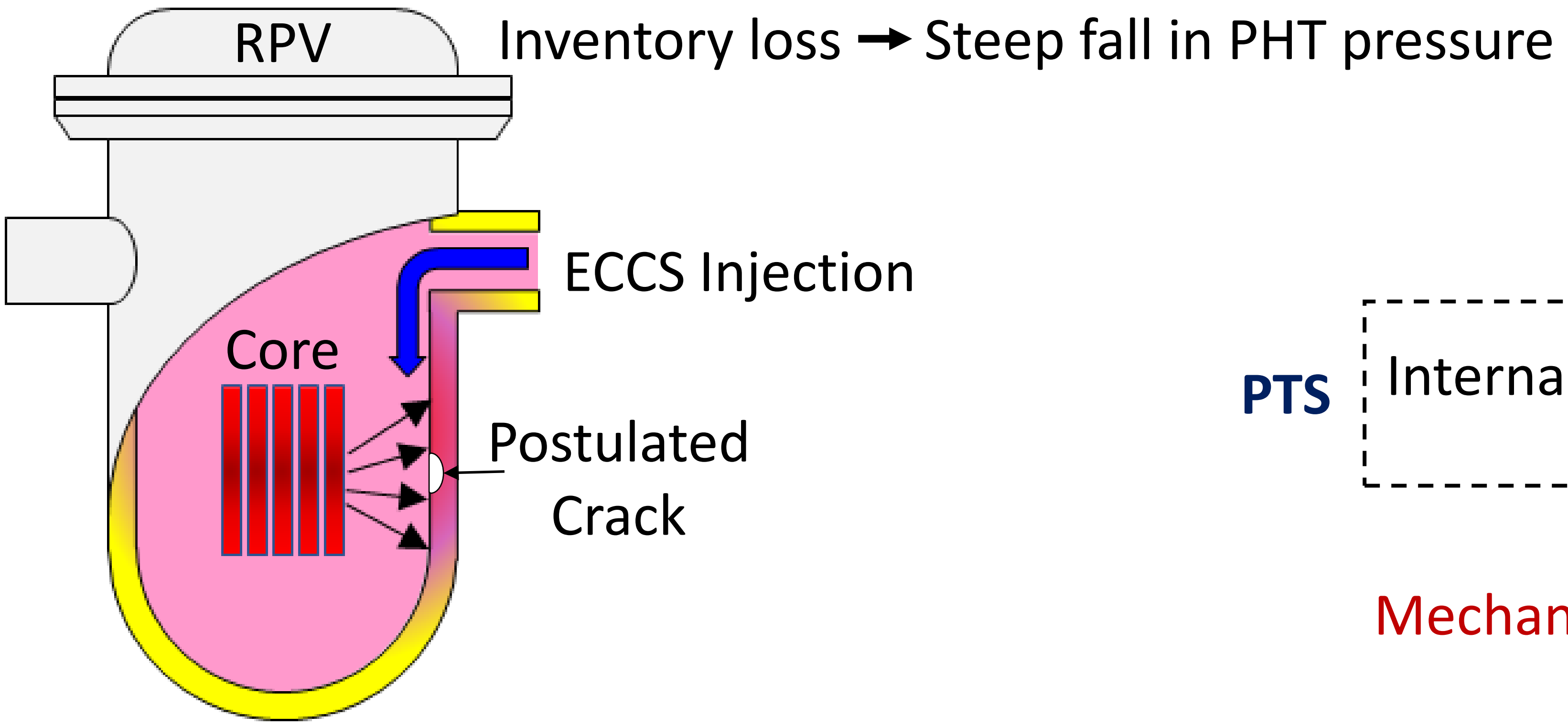
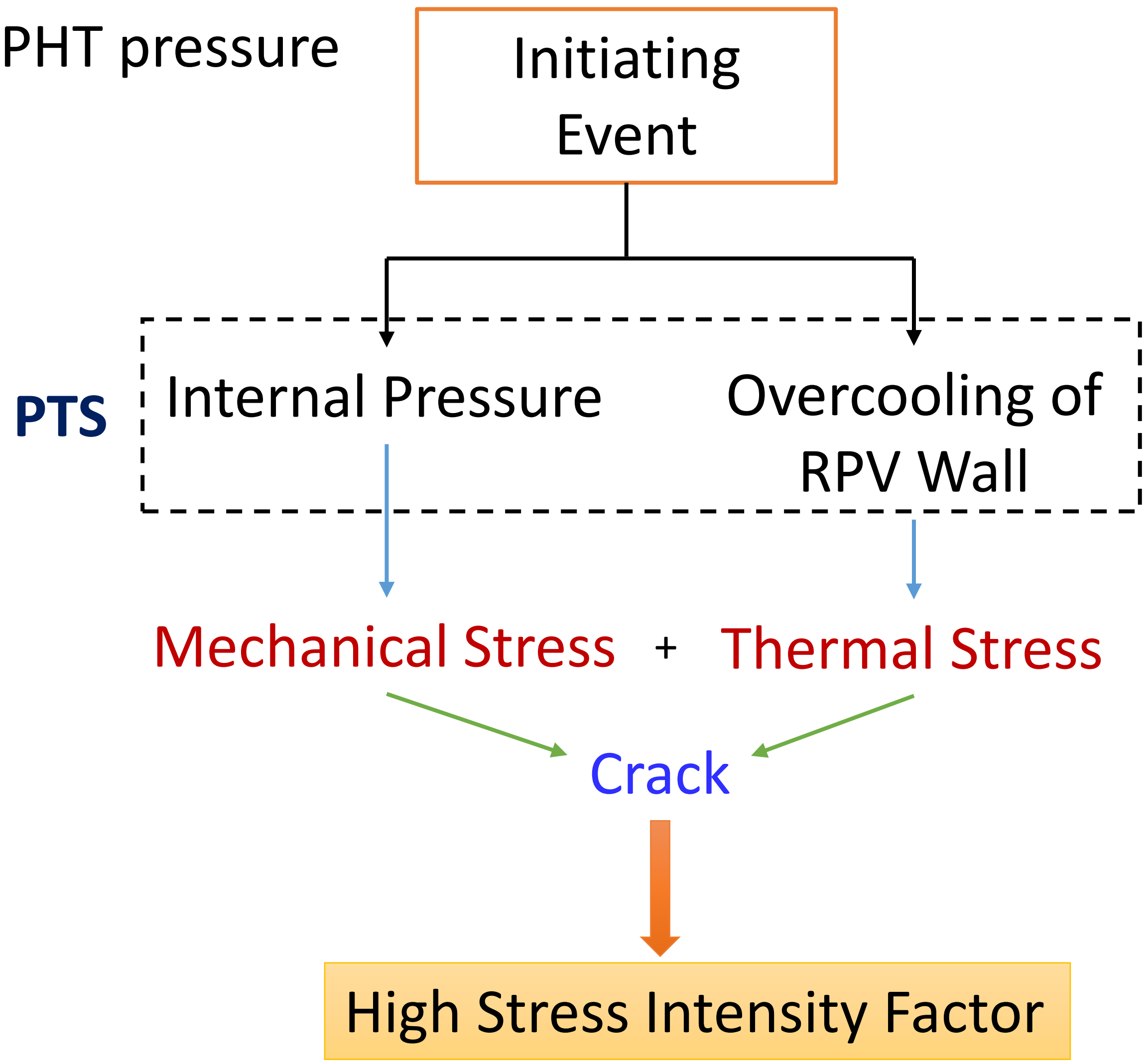
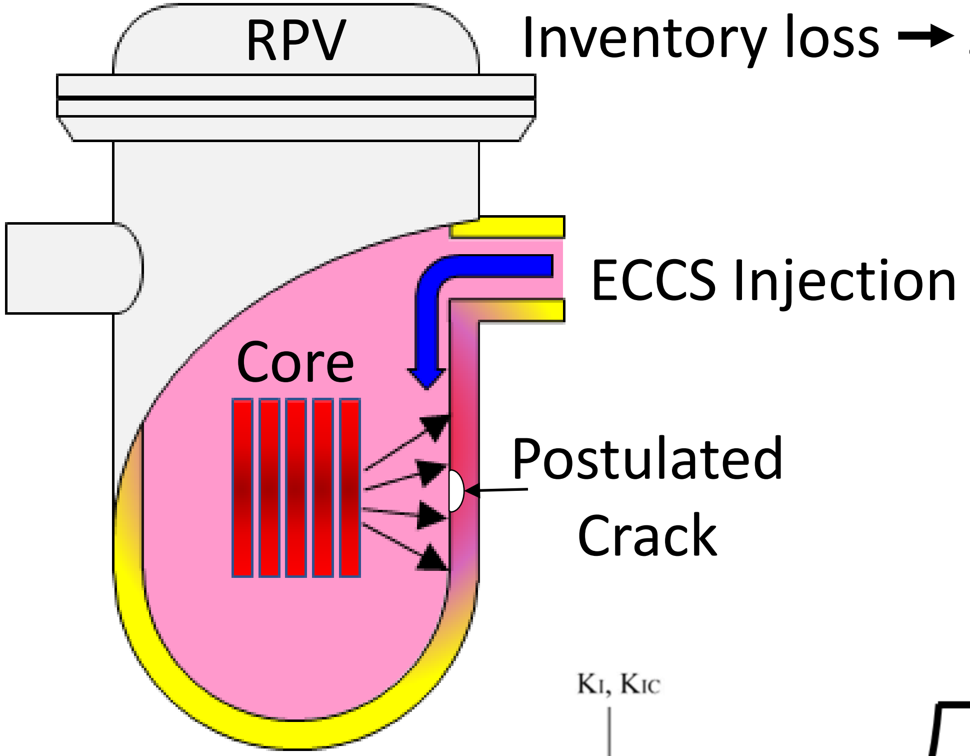




RPV Design Qualification for Pressurized Thermal Shock

Inventory loss → Steep fall in PHT pressure



Initiating Event

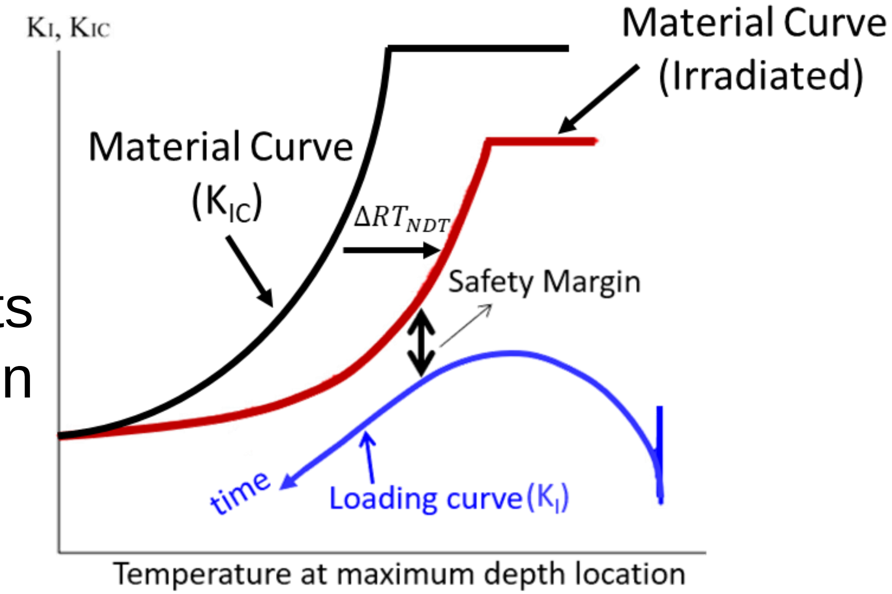
PTS

Internal Pressure | Overcooling of RPV Wall

Mechanical Stress + Thermal Stress

Crack

High Stress Intensity Factor

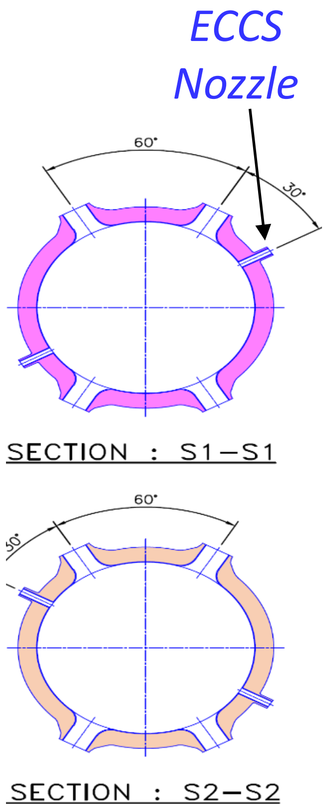


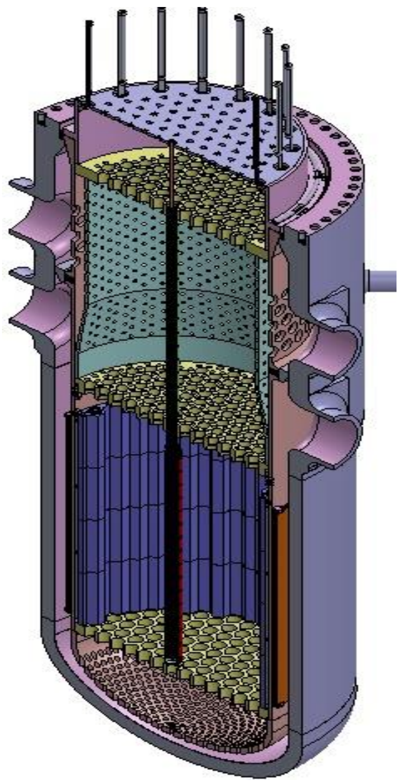
Material curve accounts for maximum irradiation induced degradation

Emergency Core Cooling System (ECCS) of BSMR 200

ECCS is a safety systems that inject water into the reactor core to prevent fuel overheating and meltdown.

- Active — High Pressure Injection System
- Passive — First Stage Hydro Accumulator
- Active — Long Term Recirculation System
- Passive — Additional Passive Core Flooding





Flow Direction during ECCS Injection

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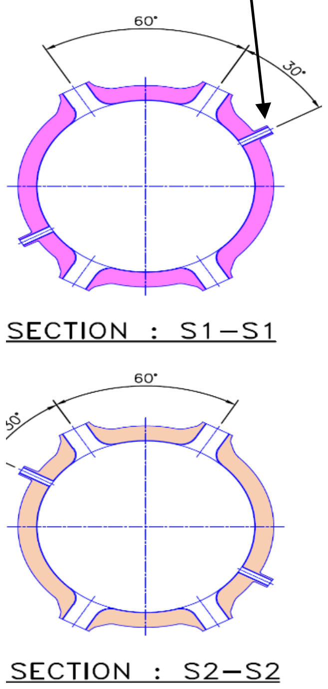
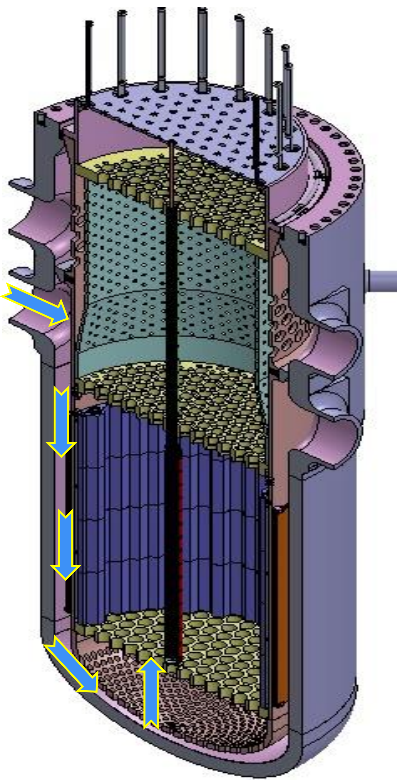
ECCS Nozzle





Bottom Flooding

Flow Direction during ECCS Injection

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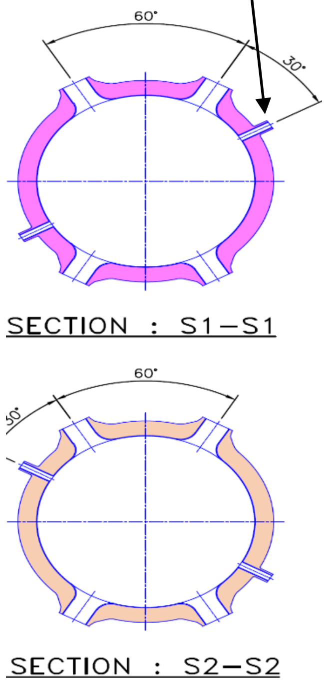

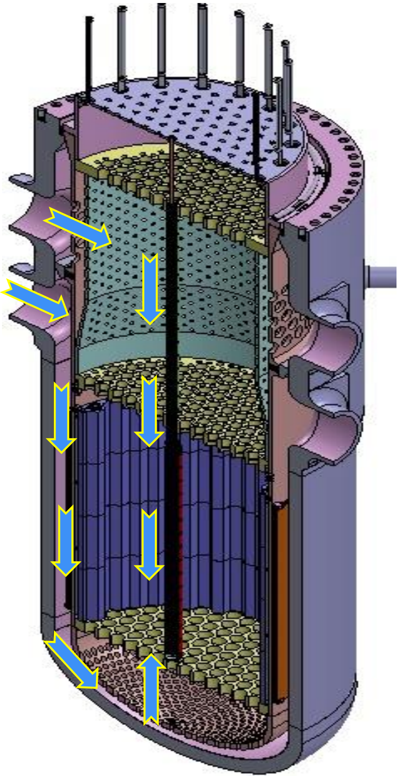

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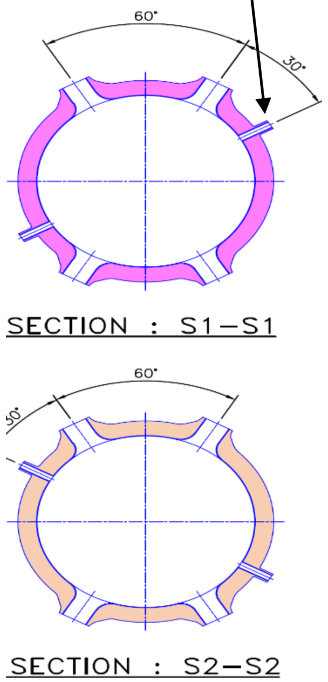

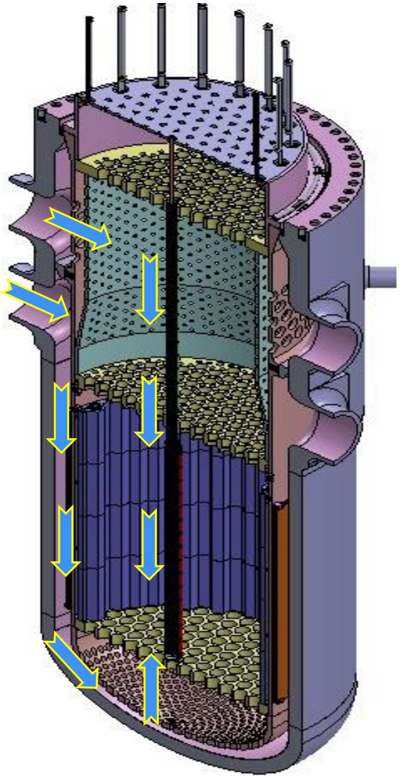

Flow Direction during ECCS Injection

- Advantage of <u>additional passive core flooding</u> **(24 h)** during **In-Vessel Melt Retention (IVMR) &** Radiological impact under **Design Extension Condition (DEC)**

Outline

1. Introduction to Indigenous SMRs
2. Design & Basic Overview of Bharat Small Modular Reactor (BSMR-200)
3. Fundamental of Reactor Safety
4. Design and Design Validation Assessment
    - Containment Peak Pressure Analysis
    - RPV Design Qualification under Pressurised Thermal Shock
    - Emergency Core Cooling System of BSMR 200
5. **Safety Analysis**
    - Design Basis Accident
    - Design Extension Condition
    - Radiological Impact Assessment
6. Summary

Safety Analysis

Safety analysis of the plant design is to assess the challenges to safety under various categories of plant states.

Deterministic Safety Assessment

Evaluation of consequences of Postulated Initiating Event

- Estimation of Fuel Temperature
- Check for Fuel Failure
- If failed, estimate radioactivity release
- Evaluate Dose to public

Probabilistic Safety Assessment

Evaluation of likelihood of occurrence of undesirable event and its consequence

- Level 1: Core Damage Frequency (CDF)
- Level 2: Large Early Release Frequency (LERF)

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Scope of Analysis

Accident Condition

Design Basis Accident (DBA)

Design Extension Condition (DEC)

DEC-A (Limited core melt)

DEC-B (Severe core melt)

Prevention of Core Damage

Mitigation of Core Damage

Design Basis Accident

Preliminary Safety Assessment (Enveloping Cases)

Pipe Ruptures

Postulated Accident	*Max. Clad Temp.*
Double-ended cold leg break	535 ℃
Double-ended hot leg break	320 ℃
Main Steam Line Break	320 ℃

Reactivity Insertion Accident

Postulated Accident	*Outcome*
Loss of regulation	Manageable reactivity excursions
Control rod ejection	
Main Steam Line Break	

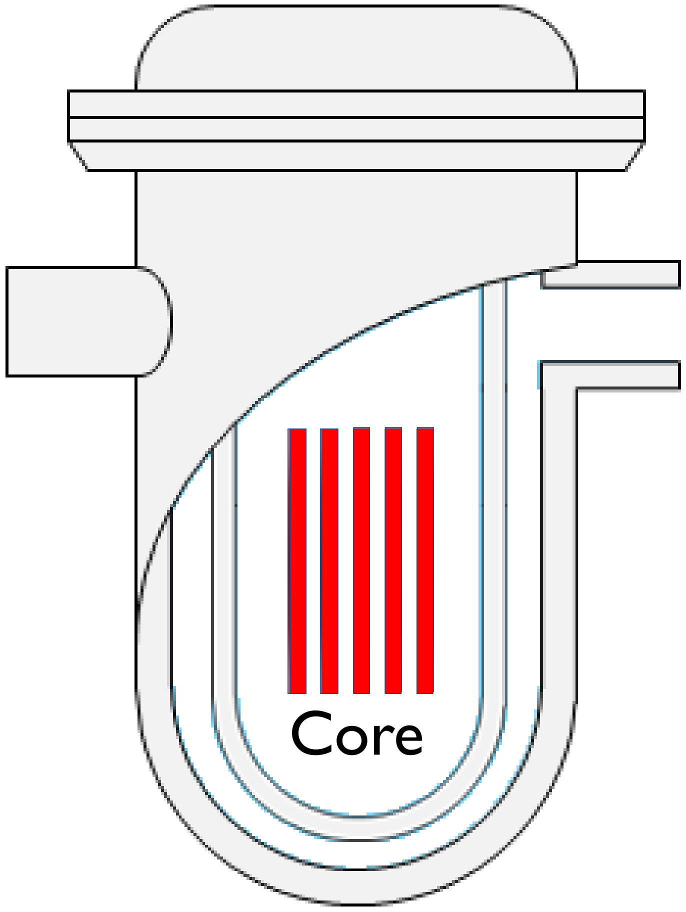
- All ECCS acceptance criteria met (Peak Clad Temperature <1200°C …)
- Fuel temperature much below fuel failure criteria

Design Extension Condition (DEC)

Strategies for Accident Management

1. In-vessel Injection (Preventive)

2. Ex-vessel Injection (Mitigation)



Design Extension Condition (DEC)

Strategies for Accident Management

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Direct injection to RPV under Design Extension Condition

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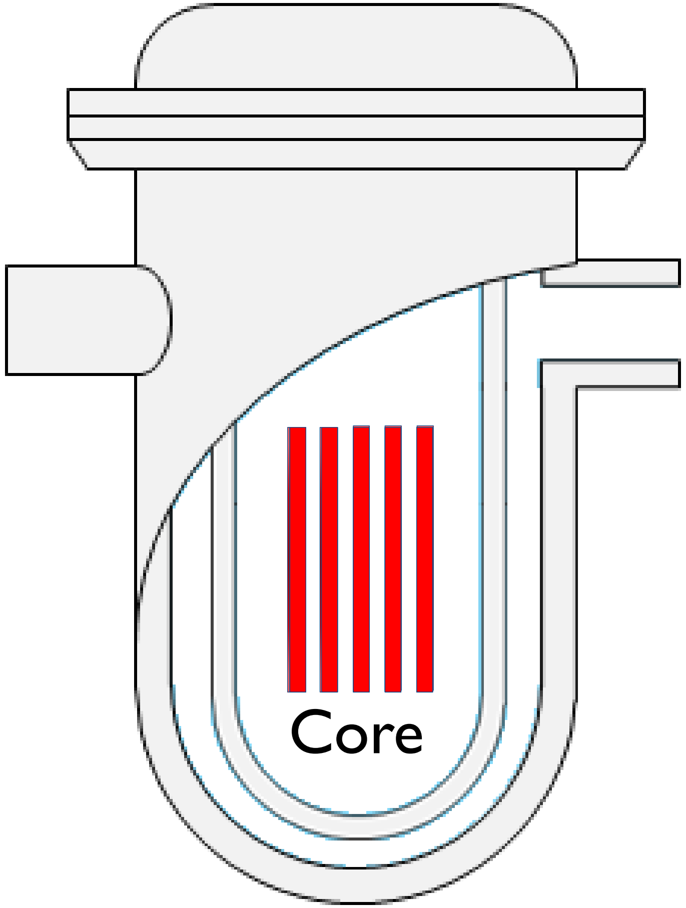

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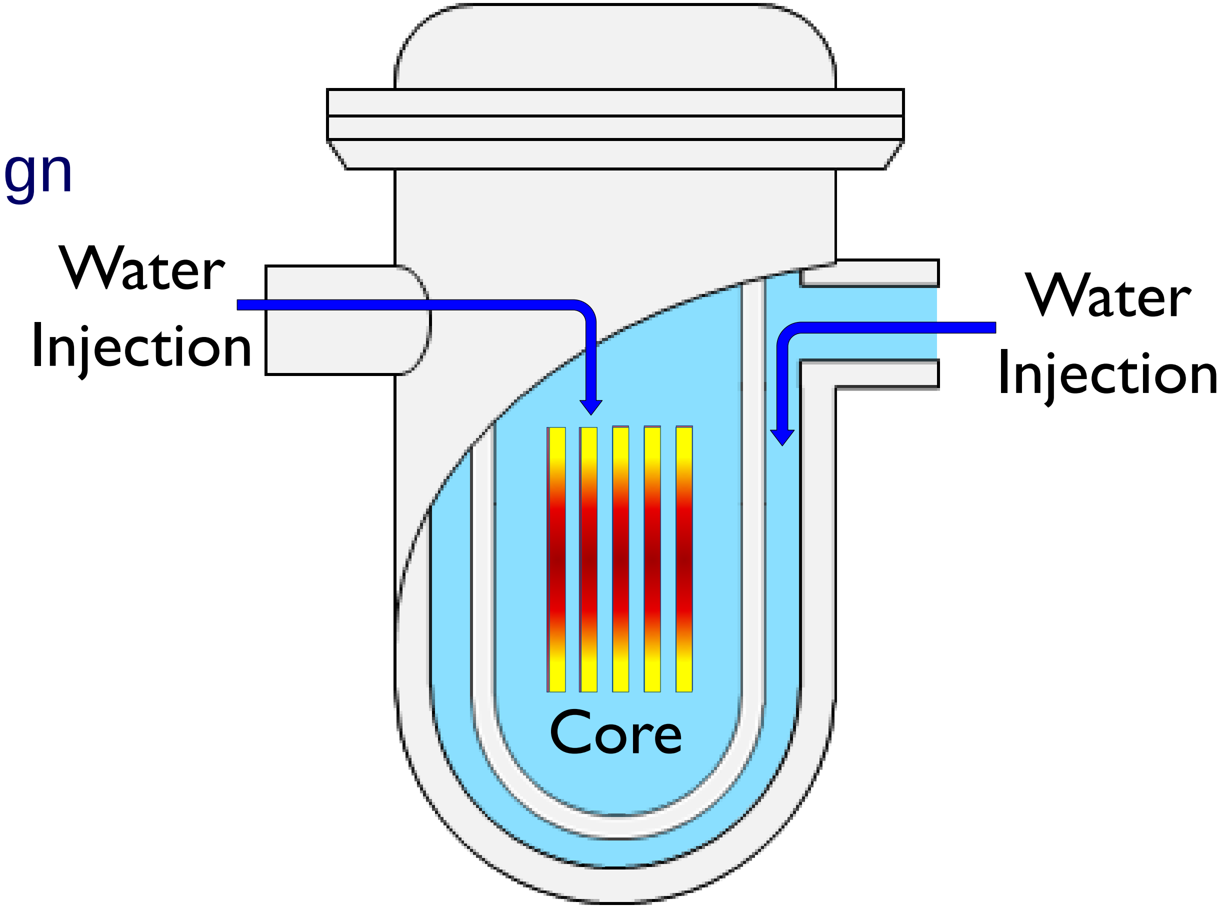

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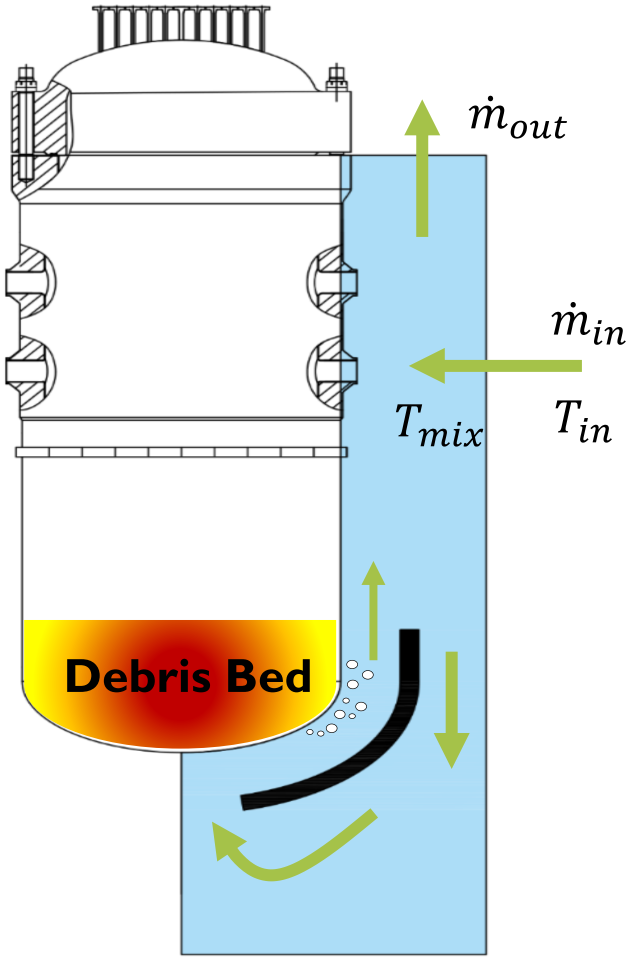
In-vessel retention of core melt (**IVMR**) by flooding reactor cavity



Schematic of IVMR Strategy

Design Extension Condition (DEC)

Strategies for Accident Management

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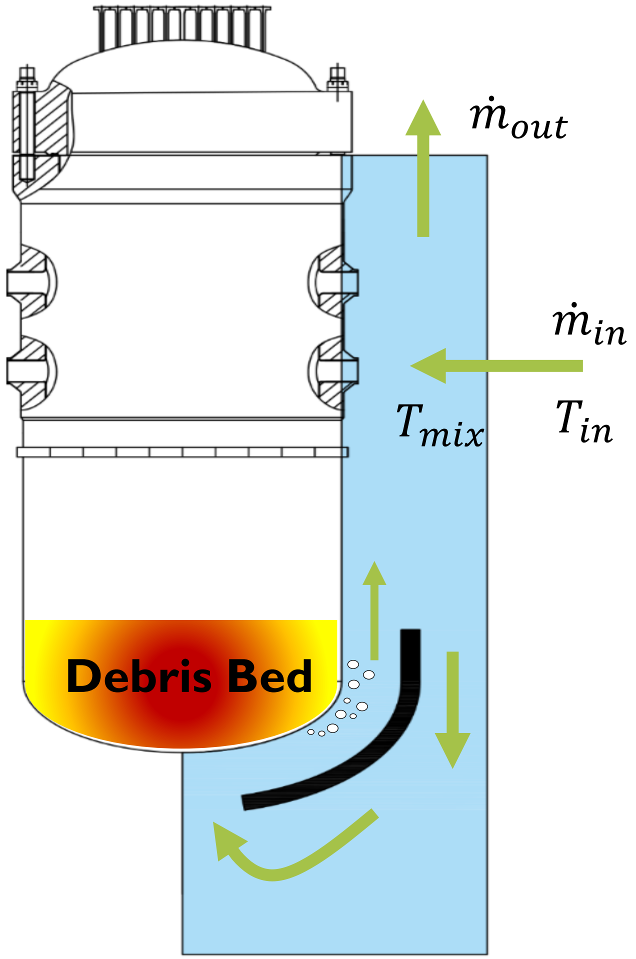
i. Cavity injection flow rates estimated



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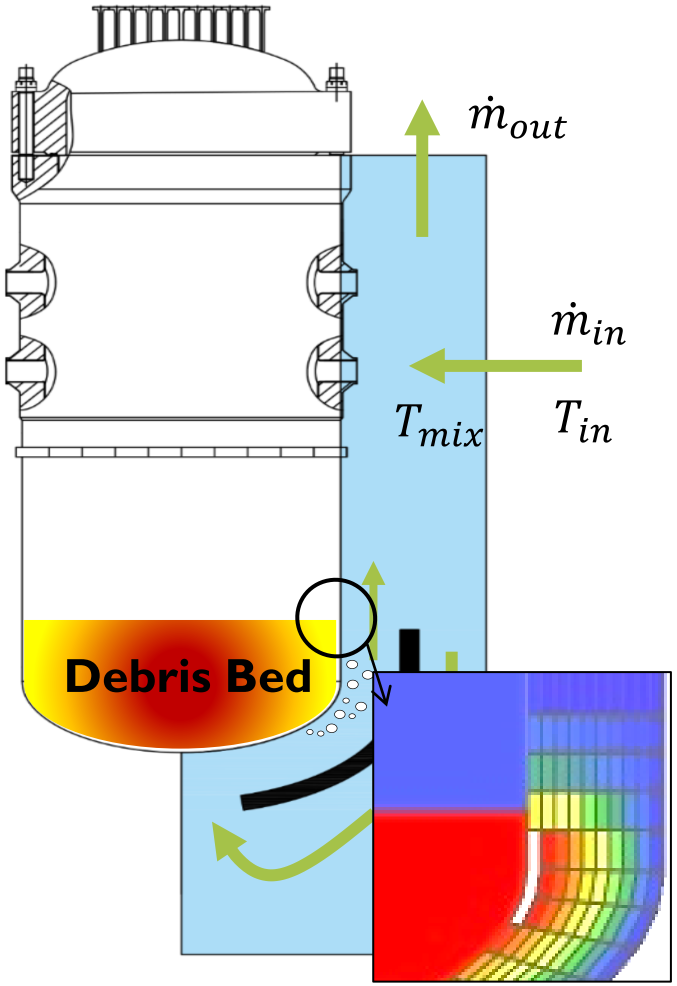
ii. Adequacy of Heat Removal ensured (ablation of RPV ~ 4 cm)



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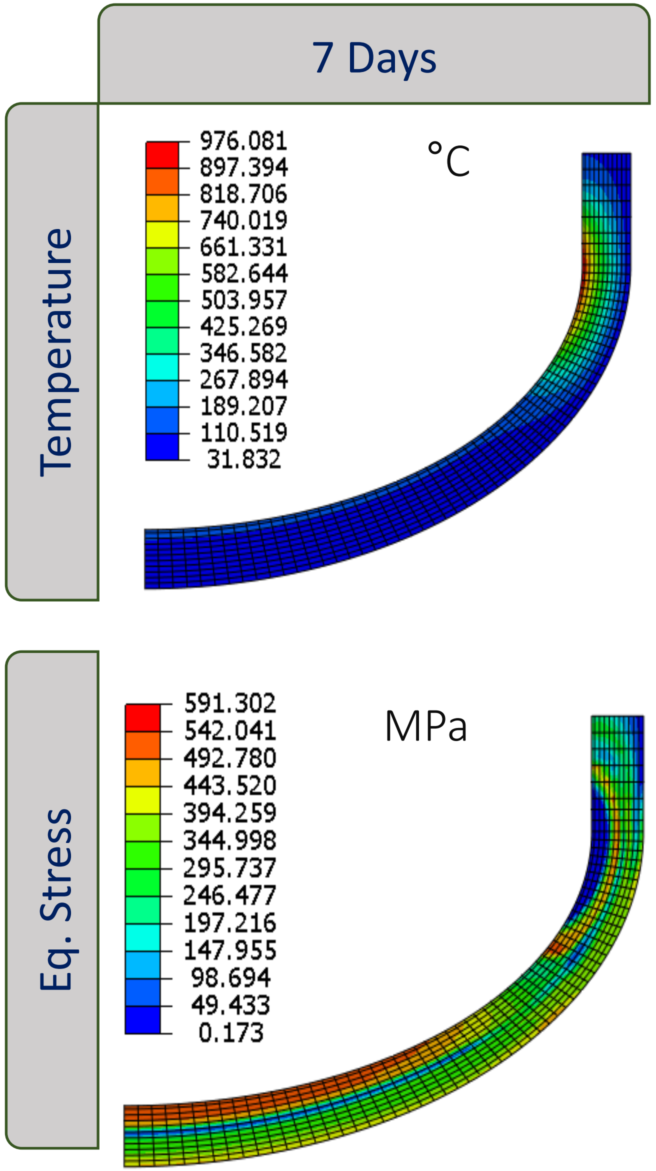
iii. Integrity of Lower Head ensured



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In-vessel retention of core melt (**IVMR**) by flooding reactor cavity

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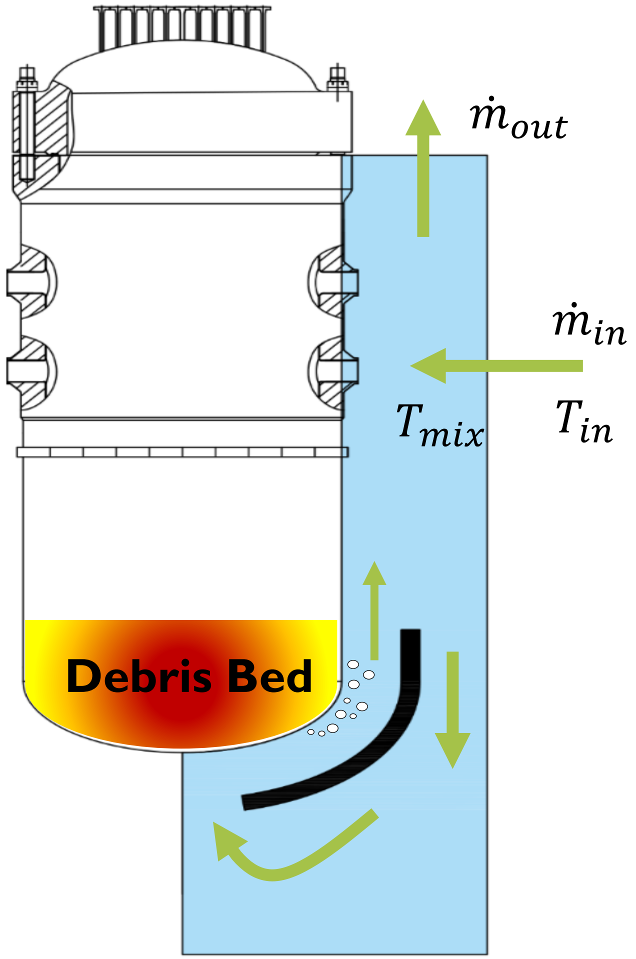
ii. Adequacy of Heat Removal ensured (ablation of RPV ~ 4 cm)

iii. Integrity of Lower Head ensured

iv. No re-criticality risk with the degraded core.

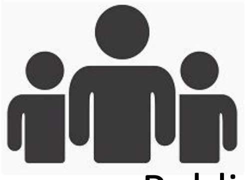


Schematic of IVMR Strategy

Public

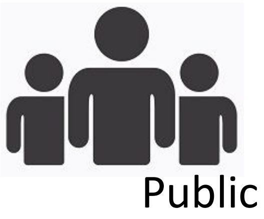
Key considerations

Preliminary Radiological Impact Assessment



Key considerations

1.Source Term



Key considerations

1. Source Term
2. Metrological + Site-specific Data

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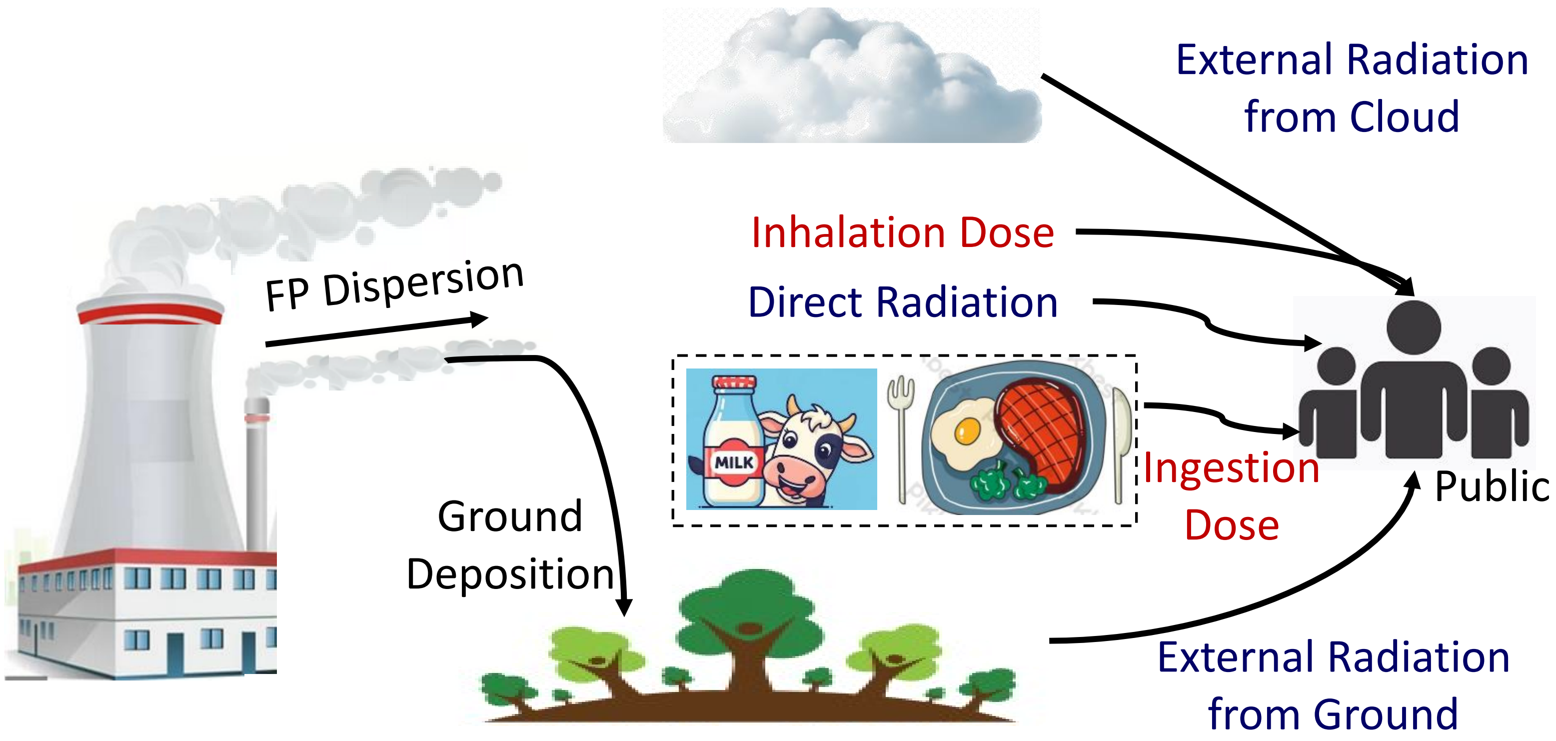

Key considerations

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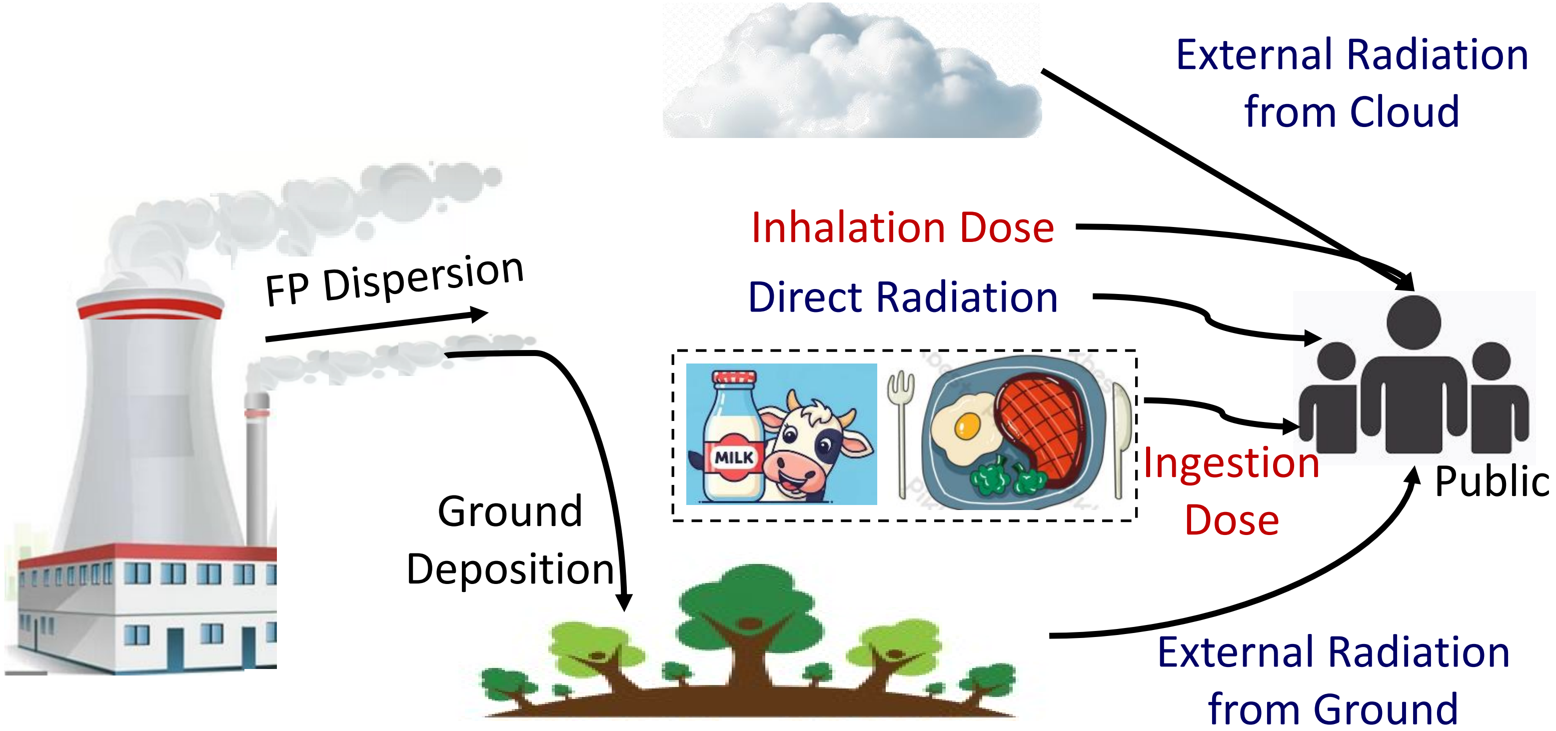
2.Metrological + Site-specific Data
3.Pathways of exposure

Preliminary Radiological Impact Assessment



Key considerations

1. Source Term
2. Metrological + Site-specific Data
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Plant State	Scenario	Total Annual Whole Body Effective Dose	
		Evaluated @ 500 m	Regulatory Limit @ EZ
DBA	LB-LOCA	$10\ \mu Sv/y$	***20 mSv***
DEC-A	MSLB + Single SG Tube Rupture	$2.35\ mSv/y$	***20 mSv***
DEC-B	LB-LOCA + Extended SBO	*No requirement of permanent relocation*	

Summary of Safety Assessment

1. Containment peak pressure of **1.73 kg/cm^2 (g)** in the event of **MSLB**
2. ECCS with **24 h of passive accumulators** to mitigate Severe Accidents
3. Accident Management during DEC condition
    - In-vessel Injection
    - Ex-vessel flooding: **Successful IVMR**
4. Preliminary Radiological Impact Assessment shows **robust design** ensuring **public safety**

Concluding Remarks: Benefits of BSMR-200

1. Development of PWR-based NPPs in India will be made possible

2. Enables demonstration of viability of PWR-based NPPs

3. Offers economically competitive PWR NPP than imported options

4. Will be a milestone in India’s nuclear energy mission, compliant to ‘make-in-India’

Thank you

For your attention !!!