Course on “SMR LWR technologies”

McSAFER Project

Cesar QUERAL / UPM

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

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McSAFER Project

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McSAFER Project. PROJECT OVERVIEW

- Interest on the development and deployment of Small Modular Reactors (SMR) has increased in the last years in Europe and worldwide. SMRs have a great potential for safe, flexible and CO2-free power generation, salt water desalination, and process heat generation.
- They are currently considered in various countries as an alternative to large nuclear power plants and as part of the future energy-mix to achieve the low-carbon power generation goals with low risk and cost in a competitive energy market.
- The new small core design, the integral concept, the innovative heat exchangers, and passive heat removal systems as well as the novel containment designs represent new challenges for the safety demonstration in the frame of a licensing process in the near term.
- The aim of the **McSAFER project** (High-Performance Advanced Methods and Experimental Investigations for the Safety Evaluation of Generic Small Modular Reactors) is to advance the safety research for SMRs by combining experimental investigations and numerical simulations. McSAFER has received funding from the Horizon 2020 Euratom Research and Training Programme.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Course objectives and schedule

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Course objectives

The course will focus on SMR LWR technologies (e.g. CAREM, NuScale, SMART …) including design peculiarities, safety systems and accident analysis. The lectures will cover the following topics for different SMR technologies:

1. Introduction to SMR LWR Technologies
2. Reactor and safety systems description
3. Core main features
4. Accident analysis

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Course schedule (1/2)

Time	Lecture's topics	Speaker
Monday		
14:30-14:45	Description of the objectives and contents of the Course.	UPM
14:45-15:30	Introduction to SMR LWR Technologies: Main Characteristics, Applications and Comparisons	John LILLINGTON (Jacobs)
15:30-15:40	*Questions*	
15:40-16:25	SMART SMR Reactor Core & Coolant System	Bart VERMEEREN and Simon VERDEBOUT (TBL)
16:25-16:35	*Questions*	
16:35-17:20	SMART reactor safety systems	Victor Hugo SANCHEZ-ESPINOZA (KIT)
17:20-17:30	*Questions*	

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Course schedule (2/2)

Time	Lecture's topics	Speaker
Tuesday		
14:30-15:15	NuScale reactor core and primary circuit	Ville VALTAVIRTA (VTT)
15:15-15:25	*Questions*	
15:25-16:10	Safety systems of NuScale reactor	Marek BENČÍK (UJV)
16:10-16:20	*Questions*	
16:20-17:05	NuScale reactor. Safety Analysis	Cesar QUERAL (UPM)
17:05-17:15	*Questions*	
Wednesday		
14:30-15:15	CAREM reactor main features	Darío DELMASTRO (CNEA)
15:15-15:25	*Questions*	
15:25-16:10	CAREM core main features	Edmundo LOPASSO (CNEA)
16:10-16:20	*Questions*	
16:20-16:30	End of the course.	KIT

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Next McSAFER Courses

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Next McSAFER Courses

You can find our McSAFER news, events and courses at:

- https://mcsafer-h2020.eu/news-and-events/

Next McSAFER courses:

- Training course on Core neutronics and thermal hydraulics of light water SMRs (February 2022)

  3 day course with lectures by experts from project partner organizations, hands-on training with the computational framework Kraken, and an experimental session at LUT.

- Multi-physics simulations applied to SMR (December 2022)

  MOOC Course.

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Comments

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Comments

- Questions from the audience to the speakers at the end of each presentation
- Turn off your video and microphone
- We have included a few follow up questions to test your understanding. Try to answer the questions included in each presentation.
- Enjoy the course
- Thank you for participating in this course

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Introduction to SMR LWR Technologies:

Main Characteristics, Applications and Comparisons

John Lillington / Jacobs

25.01.2021

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Scope of Presentation

My presentation will focus on following topics:

- What is a Small Modular Reactor (SMR)?
- Why SMRs
- Advantages and disadvantages of SMRs
- Include enhanced safety features
- Include potential economic benefits of SMRs
- Main Characteristics SMR LWRs
- Nuclear process heat applications
- Current Status of development
- Examples of different SMR technologies
- Specific examples of SMR LWR technologies
- Comparison table of typical current designs
- Presentation summary.

NB the term SMR is often used to imply small Light Water Reactors (LWRs) but many of the benefits or otherwise of modularity apply to latest Advanced Modular Reactors (AMRs) of other technologies.

Therefore although the focus and examples in this presentation will be on LWRs, some reference will also be made to AMRs to illustrate future directions of reactor development.

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What is an SMR?

- **Small**
  - IAEA typically recognise SMR < 300 (MWe)
  - But 'Small' also meaning 'Medium' size < 500 MWe – some vendors
  - 'Micro' ~10MWe
- **Modular**
  - Groups of multiple units
  - Units are similar
  - Manufacturing of units in a factory (as opposed to on-site)
- **Reactor**
  - Conventional isolated components (core, SGs, pressuriser, pumps, but also
  - Many designs are integral: above components in single pressure vessel
  - Mostly integral PWRs (whole primary circuit inside of vessel)
  - Passive decay heat removal is common feature

Ref: https://www.iaea.org/newscenter/news/nuclear-power-for-the-future-new-iaea-publication-highlights-status-of-smr-development

Nuclear Power for the Future: New IAEA Publication Highlights Status of SMR Development

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Why SMRs? - Advantages

- Single units of SMRs are easier and more open to Commercial finance as the size of an individual project is lower. Nuclear reactors have become much larger over time (now GW scale) to gain economies of scale but latest large new build reactors are difficult to finance:
  - Size of project > size of many utilities
  - Utilities have to pay high borrowing costs
  - Uncertainties (e.g. Change of government policy)
  - Government finance often unavailable

- Efficiencies of ‘Nth Of A Kind (NOAK) manufacture of SMR units
- Standardised factory units’ fabrication and assembly rather than on-site
- Better suitability for some energy grids (e.g. Sparsely populated areas)
- Supply chain efficiency competitive with the economies of scale of large reactors
- Improved safety (smaller units to manage)

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SMRs – Some Current Disadvantages

- Many Gen III reactors especially LWRs have a proven track record – so why change? SMRs have a lack of track record and operating experience compared with other Gen III reactors
- Licensing costs are largely a fixed cost- difficult to lever against a smaller revenue stream of small reactors in the first instance
- SMR commercial argument is based on building a lot of them- how many do we need to build to make a profit? Need to build a family of NOAK reactors!
- Supply chains for modular units need to be established and efficiency remains to be demonstrated

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LWR and SMR Applications

Process temperature	Up to 700 °C
Electricity Production	Rankine (steam) cycle
Utility Applications	Desalination District heating
Oil and chemical industry	Tar/oil sands and heavy oil recovery, Syncrude, Refinery and petrochemical. Production of hydrogen via steam electrolysis. Biomass ethanol production

LWRs produce heat at relatively low temperatures in relation to many industrial needs
In 2019 there were 79 nuclear reactors used for desalination, district heating, or process heat, with 750 reactor-years of experience in these, mostly in Russia and Ukraine.

Ref: https://www.world-nuclear.org/information-library/non-power-nuclear-applications/industry/nuclear-process-heat-for-https://www.world-nuclear.org/information-library/non-power-nuclear-applications/industry/nuclear-process-heat-for-industry.aspxindustry.aspx

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Higher Temperature Nuclear Process Heat Applications

Up to 900 °C	Up to 950 °C
Brayton (direct) cycle	
H_2 via steam reforming of methane or high-temperature electrolysis	Thermochemical H_2 production
Syngas for ammonia and methanol	Thermochemical H_2 production

At above 900 °C there are further possibilities, and above 950 °C an important future application to additional hydrogen production opens up.
Other than LWR reactor technologies are being considered for higher temperature applications e.g. high-temperature gas-cooled reactors (HTR).

Ref: https://www.world-nuclear.org/information-library/non-power-nuclear-applications/industry/nuclear-process-heat-for-https://www.world-nuclear.org/information-library/non-power-nuclear-applications/industry/nuclear-process-heat-for-industry.aspxindustry.aspx

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SMRs – Current Status

- Large number of designs currently in development
  - Light water reactors predominate (mainly PWRs, some BWRs)
    - Dominant reactor technology
    - Experience from naval reactors
    - Well suited to compact designs
  - High temperature reactors
    - Potential alternative to LWRs, designs available building on largely proven technology
  - Fast reactors
    - Some designs at varying levels of readiness level
    - PRISM proposal in UK
    - Russian experience with submarine reactors
- SMRs have not yet been deployed on a commercial scale
- This presentation will focus on a sample range of LWR designs currently under development but also make a brief mention of other SMR technologies under consideration.

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Examples of SMR Technologies (not exhaustive!)

- **LWR technologies, Gen III or III+ LWRs with targeted 10 year horizons for deployment e.g.**
  - **BWRX-300, CAREM, NuScale SMR, NUWARD, RR SMR, SMART, SMR-Holtec**

  (Focus for this presentation – the aim is to describe briefly a sample of the wide range of different SMR LWRs at advanced stages of development.)

- High Temperature Gas Reactors
  - HTR-PM, HTGR and Co-Generation
- Fast Reactors
  - GE-Hitachi (Na), Westinghouse (Pb), G4M(Pb-Bi), Moltex (Molten Salt)
- Micro Generation
  - U-Battery. 4S (Na)
- Barge Mounted Reactor
  - KLT-40S

Criteria for assessments:

- Technical maturity and viability
- Maturity of safety case and certification
- Key strengths and areas for development
- Programmes to address development needs
- Available resources – people, capability, facilities & funding
- Economic viability, including IP ownership

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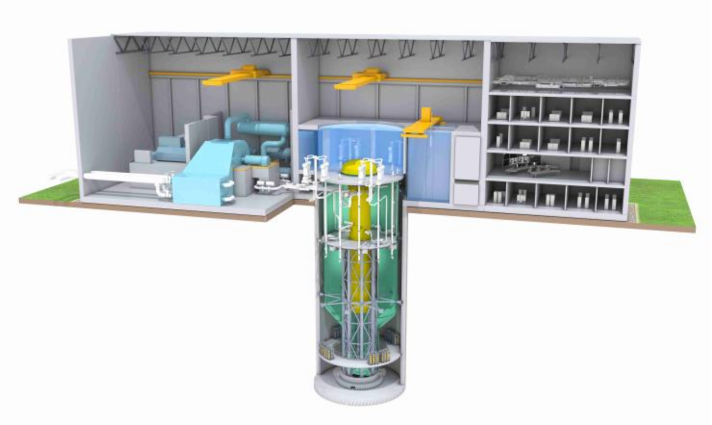
BWRX-300

- BWRX-300 is a ~300 MWe boiling water-cooled, natural circulation Small Modular Reactor (SMR) with passive safety systems.
- The BWRX-300 is based on the U.S. NRC-licensed, 1,520 MWe ESBWR and is designed to provide clean, flexible baseload electricity generation.
- Estimated to have lifecycle costs of typical natural gas combined-cycle plants targeting $2,250/kW capital cost for NOAK (nth of a kind) implementations.
- GE Hitachi Nuclear Energy is collaborating with various partners, including Dominion Energy, in proactively investing in the BWRX-300.
- The nuclear-related technology and components are based on well-established principles and the proposed construction technologies have already been utilised in the power industries.



https://nuclear.gepower.com/build-a-plant/products/nuclear-power-plants-overview/bwrx-300

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CAREM

- CAREM - *Central ARgentina de Elementos Modulares* - is Argentina's first domestically-designed and developed 25 MWe integral PWR nuclear power unit.
- The prototype of the small pressurised water reactor design is being built at a site adjacent to the Atucha nuclear power plant.
- First concrete was poured for the reactor in February 2014, marking the official start of its construction.
- At least 70% of the components and related services for CAREM-25 are to be sourced from Argentine companies.
- Agreement signed with Saudi Arabia in March 2015 targeting desalination activities

The Atucha site, near Lima, 110 km northwest of Buenos Aires (Image: NA-SA), 21 April 2020

https://world-nuclear-news.org/Articles/Argentinean-projects-to-resume-after-hiatus

https://www.world-nuclear.org/reactor/default.aspx/CAREM25

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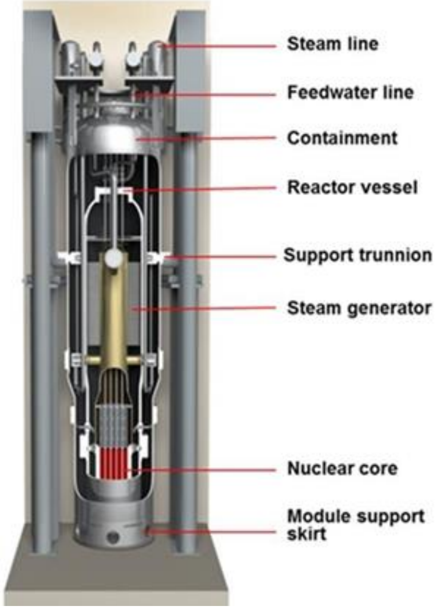
NuScale SMR

- NuScale Power LLC designed modular PWR plant with up to 12 units each of 45MWe
- Entirely natural circulation
- 24 month operating cycle
- Extensive USNRC engagement and programmes of work supported by Fluor and earlier USDoE award

https://www.nuscalepower.com/newsletter/nucleus-spring-2019/powering-the-next-generation-of-nuclear



www.nrc.gov/reactors/advanced/nuscale.html

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NUWARD

- French-developed SMR design unveiled 17 September 2019
- A new small modular reactor (SMR) design announced by the French Alternative Energies and Atomic Energy Commission (CEA), EDF, Naval Group and TechnicAtome.
- The NUWARD - with a capacity of 300-400 MWe - has been jointly developed using France's experience in pressurised water reactors (PWRs).
- Application - Electricity Generation
- Compact and simple with an integrated design, flexibility for construction and operation.
- Timescale: basic design between 2022 and 2025, "advanced concept phase" between 2025 and 2030 with demonstration plant.

https://world-nuclear-news.org/Articles/French-developed-SMR-design-unveiled

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RR SMR

- UK Consortium led by Rolls-Royce
- Provide 220MW to 440MW of power, depending on the configuration
- Being so compact (16 metres high and 4 metres in diameter) it can be transported by truck, train or even barge
- 60 year design life
- Aim to use off-the-shelf proven components

https://www.rolls-royce.com/~/media/Files/R/Rolls-Royce/documents/customers/nuclear/smr-booklet-28-sep.pdf

https://www.rolls-royce.com/media/press-releases/2019/23-07-2019-commitment-to-initial-funding-for-smr-welcomed-by-consortium.aspx

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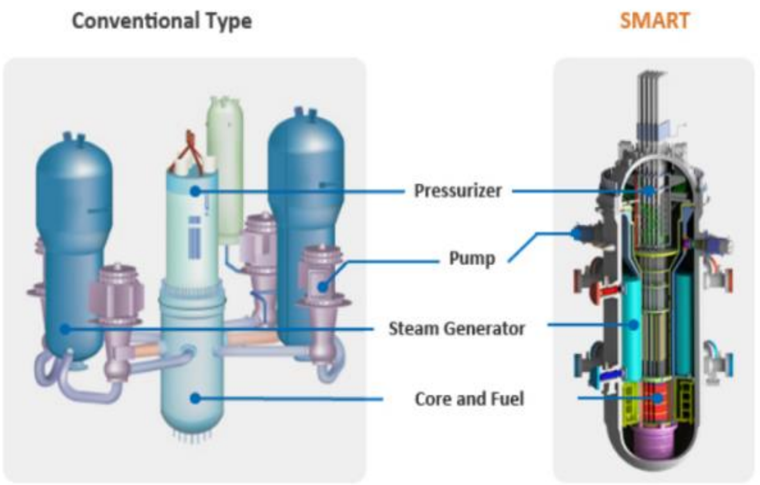
SMART (Korea)

- SMART, a 100MWe SMR LWR type unit, is designed for generating electricity and for thermal applications such as seawater desalination.
- Scientists in South Korea have been developing the technology for 22 years.
- Work has been going on in Saudi Arabia with its South Korean partners since 2011.
- Saudi Arabia has updated its agreement with South Korea to complete a 100 MWe SMR, to license it for use in that country and to offer it for export.
- This SMR design planned for Saudi Arabia is the Korea Atomic Research Institute’s “system-integrated modular advanced reactor.”

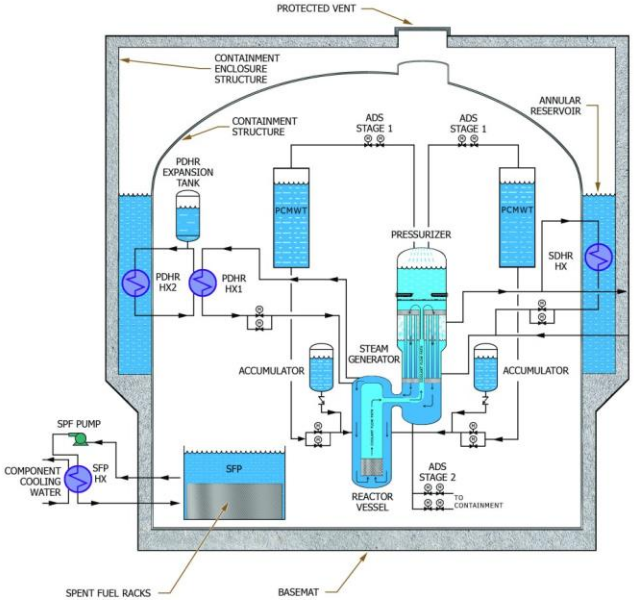


https://neutronbytes.com/2020/01/18/south-koreas-smart-smr-gets-new-life/

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SMR-160 (Holtec)

- Safe and Secure: passive features of its operation to ensure utmost safety and reliability, safety systems within containment, secure from external threats.
- Economical and Efficient: small footprint (4.5 acres), minuscule site boundary dose, large inventory of coolant in reactor vessel and its modularity.
- Reliable and Environment-Friendly: can operate using a standard cooling water system, or an air cooling system. Flexibility to use air cooled condensers, can be deployed in the most arid regions.
- Minimal Operation & Maintenance Cost (O&M): 80-year service life, On-site underground storage of used fuel for up to 100 years of operation.



https://holtecinternational.com/products-and-services/smr/

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Summary Comparison of SMR LWRs

Name	Type	Vendor	Output (MWe	Provenance	Application
BWRX	BWR	GE-Hitachi	300	USNRC licensed ESBWR	Baseload electricity generation
CAREM	Integral PWR	CAEM Argentina	25	Argentina	Power unit Desalination
NuScale	Modular PWR Natural circulation	NuScale Power LLC	45	US	Electricity & flexible power operations, H2 production, process heat, power for oil refineries, desalination.
NUWARD	PWR	CEA, EDF, Naval Group, Technatome	300 - 400	French PWR experience	Electricity generation
RR PWR	PWR	RR	220 - 440	UK Consortium	Electricity generation
SMART (Korea)	Integral PWR	KAERI	100	Korea	Electricity/thermal desalination
SMR-160 (Holtec)	PWR	Holtec	160	US	Electricity and process heat applications in arid regions

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Summary

Key points:

- Keen world-wide support for Small Modular Reactors in general, LWR main focus.
- Potential safety features and economic benefits of SMRs recognised.
- Large number of SMR designs are under development, at different stages of maturity and licensability.
- Different technologies for different energy applications: SMR LWR for lower temperature applications. Both PWR and BWR variants exist. More PWR than BWR.
- High inherent safety; low generation capacity, passive safety in some designs.
- Different generation capacities ranging from few tens of MWs to few hundreds of MWs.
- Economic case rests on building multiple units.

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Questions

Multiple Choice Questions (in each case one answer is the correct one!)

1. Which of the following characteristics is not a feature of an SMR?
   (a) Modular units
   (b) Largely built on its operational site
   (c) Factory production
2. What advantage does an SMR have compared with previous established reactors?
   (a) Many have been built previously
   (b) Multiple similar units can be manufactured
   (c) Supply chains are already in place.
3. Building a fleet of (Nth of a Kind) NOAK SMRs compared with a few large LWRs? Which statement is true?
   (a) Licensing costs will be much reduced
   (b) Easier financing of individual units built one at a time
   (c) Less flexibility regarding grid connection
4. Building SMR LWRs compared with other reactor technologies? Which statement is true?
   (a) Best technology for high temperature applications
   (b) Experience from naval reactors
   c) Not well suited to compact designs

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Acknowledgment

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Thank You

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SMART SMR Reactor Core & Coolant System

Bart VERMEEREN and Simon VERDEBOUT (Tractebel Engie)

25.01.2021

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Outline

- Introduction
- Main characteristics
- Vessel description
  - Integrated primary system
  - Reactor core and fuel assembly design
  - Pressurizer
  - Reactor coolant pumps
  - Steam generators
  - Primary flow mixer
  - Auxiliary systems
- In-core fuel management
- Summary
- Quiz

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Introduction

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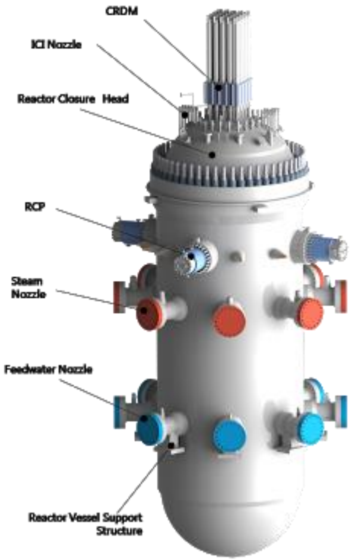
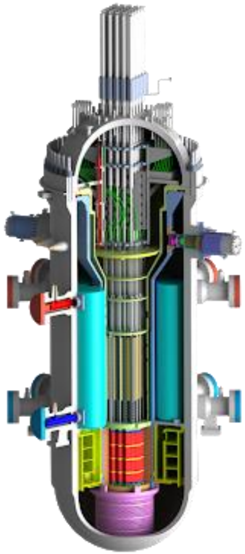
Introduction

SMART = **S**ystem-integrated **M**odular **A**dvanced **R**eac**T**or:

- Pressurized light Water Reactor → PWR
- Developed by KAERI (Korea Atomic Energy Research Institute)
- Integrated → all primary components in reactor vessel
- Modularization → ease field installation and maintenance
- Advanced → integrating Passive Safety Systems:
  - Passive Residual Heat Removal System (PRHRS)
  - Passive Hydrogen Removal System
  - Passive ex-vessel cooling (in-vessel core retention)
- SMART is designed for system simplification, minimizing components, and fully automated digital control and man-machine interface system (one-man operation under normal plant operation conditions)
- SMART Core Damage Frequency < 10^{-6} per reactor year
- SMART applications are electricity production, water desalination and district heating





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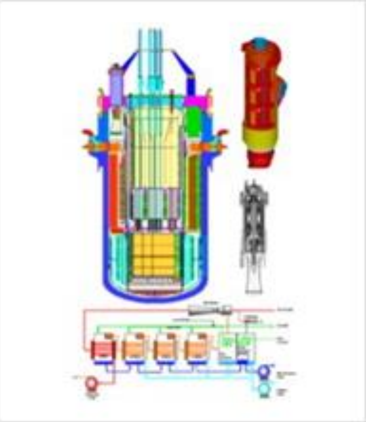
Introduction

SMART development history[1]:

- Conceptual design initiated in 1997, finalized in 1999
- Basic Design followed up to 2002
- From 2002 to 2006 - SMART-P pilot plant (1/5th scale)
- Design Optimization between 2006 and 2009 including pre-project phase
- SMART Standard Design approval obtained in 2012
- SMART Design improvement following post-Fukushima actions between 2012 and 2015
- Since 2015, agreement between KAERI and K.A.Care to build SMART reactors and develop human resources capability to run them in Riyadh, Saudi Arabia



[1]: SMART Power Co., ltd website (www.smart-nuclear.com)

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SMART Main characteristics

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Main characteristics

Extracted from https://www.youtube.com/watch?v=h7LKIauZ5UE

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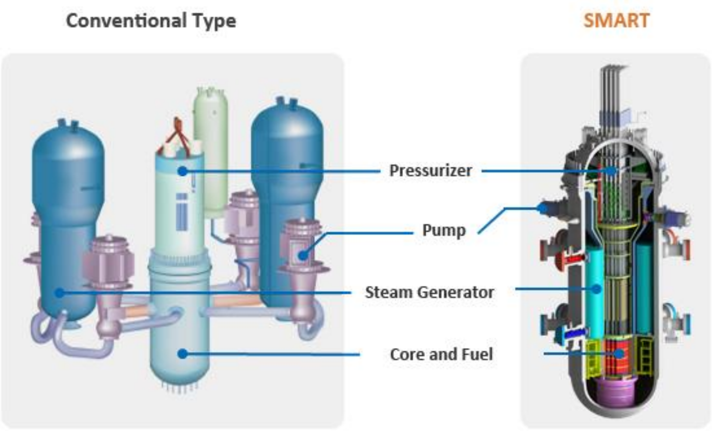
Main characteristics

Parameter	Value[1]
Core thermal power	330 MW
Electrical power	100 MW
Design life time	60 years
Fuel type	17x17 fuel assembly
Average core power density	62.62 MW/m³
Reactor operating pressure	15 MPa
Core coolant inlet temperature	295.7°C
Core coolant outlet temperature	323.0°C
Primary coolant flow rate	2090 kg/s



[1]: IAEA – Status report 77 – System-Integrated Modular Advanced Reactor (SMART).

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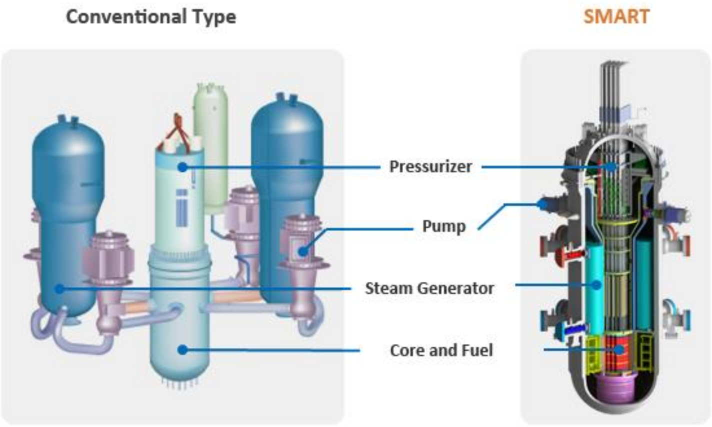
Main characteristics

Parameter	Value[1]
# Reactor coolant pumps	4
# Steam generators	8
Steam generator type	Once-through helically coiled tubes
Steam outlet condition	Superheated (30°C)
SG outlet pressure	5.2 MPa
SG water inlet temperature	200°C
SG mass flow	161 kg/s



[1]: IAEA – Status report 77 – System-Integrated Modular Advanced Reactor (SMART).

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SMART Vessel description

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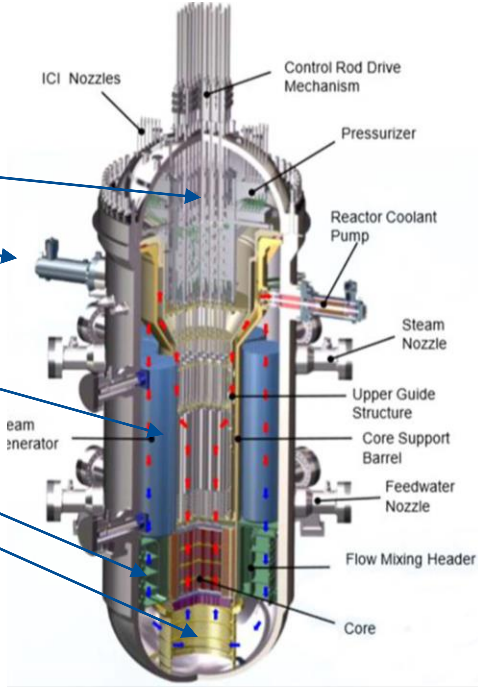
Integrated Primary Systems

SMART is an integrated design – all key components are within Reactor Pressure Vessel:

- Pressurizer located at vessel dome
- Primary Reactor Pumps (4 located above SGs)
- Steam Generators (8 located around riser)
- Flow Mixing Header Assembly
- Core Inlet mixer comparable to KONVOI reactor type



Integrated design leads to elimination of LB-LOCA accident from design basis events[1]

Some of that equipment is quite different than traditionally used in Gen 2 and Gen 3 PWR

[1]: IAEA – IAEA Advanced Reactors Information System (ARIS) - ADVANCES IN SMALL MODULAR REACTOR TECHNOLOGY DEVELOPMENTS

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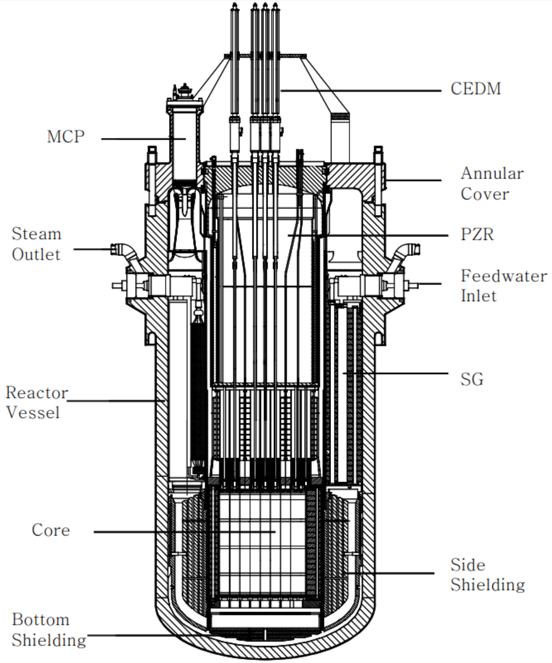
Integrated Primary Systems – Reactor core and fuel assembly

- 330 MWth reactor core with reduced active height of 2 m (about half of conventional PWR design), loaded with 57 fuel assemblies
- Core reactivity control for **boron-free design** using AIC control rods (49 CRDMs) with very fine-step manoeuvring capability (linear pulse motor driven, 4 mm/step[1]) to compensate for fuel depletion
- Core reactivity control for **soluble boron design** using AIC control rods (checkerboard pattern of 25 CRDMs) and soluble boron (less control rods needed to respect shutdown margin requirements because of reduced power defect[2])
- Core reactivity control using burnable poison material (Al_2O_3-B_4C and Gd_2O_3-UO_2 rods) with axial zoning for flattening radial and axial power profiles
- Inherently free from spatial xenon oscillation instability (because of reduced height)

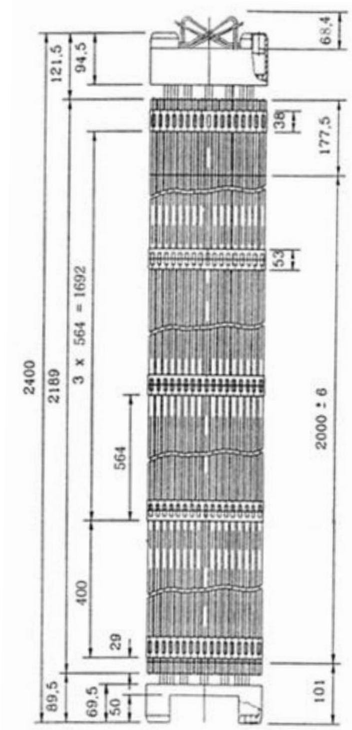


[1]: M.H. Chang et al., Advanced design features adopted in SMART, IAEA International Seminar Cairo, May 2001
[2]: Feasibility Study of SMART Core with Soluble Boron, KAERI, November 2000

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Integrated Primary Systems – Reactor core and fuel assembly

- 57 standard 17x17 square lattice UO_2 fuel assemblies (<5 w/o UO_2)[1]
  - based on KOFA (Korean Optimized Fuel Assembly) design from KAERI/Siemens-KWU used in 900 MWe Westinghouse type Korean PWRs
  - 264 fuel rods, 24 guide tubes for control rods, 1 instrumentation thimble
  - Fuel rod cladding made of Zircaloy-4
  - Top/bottom grids made of Inconel; 3 middle spacer grids made of Zircaloy
- Low core power density (62.6 kW/l); about 60% of typical PWRs[2], ensures adequate thermal margin of more than 15% regarding critical heat flux
- Core region shrouded with shielding materials of several sheets of stainless-steel plates (reduces fast neutron fluence to about 2 x 10^{16} n/cm^2 after 60 years RPV lifetime[2])



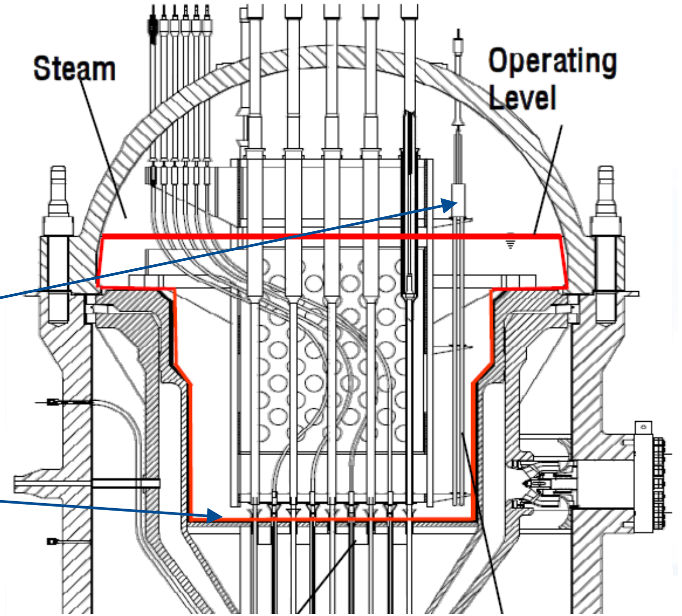
[1]: Optimization of The Coupling of Nuclear Reactors and Desalination Systems, IAEA-TECDOC-1444, 2005
[2]: M.H. Chang et al., Advanced design features adopted in SMART, IAEA International Seminar Cairo, May 2001

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Integrated Primary Systems – Pressurizer

SMART Pressurizer is designed to control system pressure at nearly constant level :

- 2 Safety Valves to protect reactor against overpressure – installed on top of reactor head
- Large steam volume[1] (40m³) in comparison with Gen2/Gen2+ PWR (~18m³)
- Pressure controlled system is labelled as “semi-passive” as it is composed of Heaters only
- No spray system is implemented within SMART
- Wet Thermal Insulator is installed between PZR and primary system to reduce conductive heat transfer[2]



[1]: H. S. Park et al., Basic Design Of An Integral Effect Test Facility, Smart-itl For An Integral Type Reactor, Korea Atomic Energy Research Institute, May 2013
[2]: J. Yoon et al., Plant System Design Features of SMART, GENES4/ANP2003, Sep. 15-19, 2003, Kyoto, JAPAN

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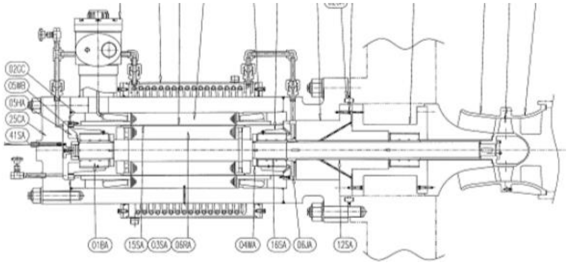
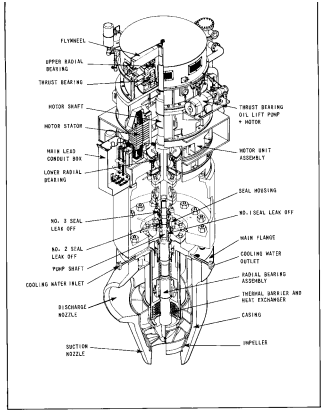
Integrated Primary Systems – Reactor Coolant Pumps

SMART Reactor Coolant Pumps (RCP) are canned pumps[1]:

- Canned pump → no pump seals required
- Elimination of SB-LOCA event associated with pump seal failure
- Asynchronous electric motor (3600 rpm)
- Cooling of motor is accomplished by Component Cooling (CC) water
- No Flywheel on pump → limited slow coast-down of primary flow in case of Loss Of Onsite Power
- SMART Pumps see core outlet temperature – while in conventional PWR RCPs see cold leg temperature





Westinghouse REACTOR COOLANT PUMP

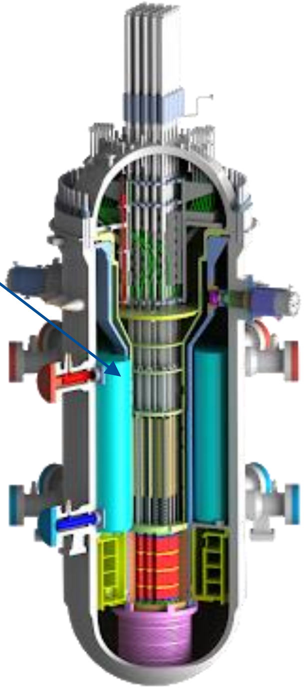
[1]: IAEA – Status report 77 – System-Integrated Modular Advanced Reactor (SMART).

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

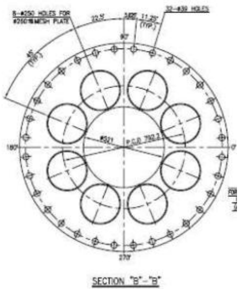
Integrated Primary Systems – Steam Generator

SMART Steam Generators (SG) are cassettes:

- Eight identical SG cassettes located on annulus between RPV and core support barrel
- Once-through SG helicoid cassette composed of:
  - 375 tubes of INCONEL 690
  - Heat exchange area: 500 m²
  - Tube outside diameter: 17 mm
  - Small mass inventory compared to SG with U-tubes
- Feedwater subcooled flow: 160.8 kg/s (subcooling of ~66°C)
- Steam outflow is superheated: 298°C (30°C of superheating)
- Secondary nominal pressure: 5.2 MPa



Side view



Top view

[1]: H. S. Park et al., Basic Design Of An Integral Effect Test Facility, Smart-itl For An Integral Type Reactor, Korea Atomic Energy Research Institute, May 2013
[2]: J. Yoon et al., Plant System Design Features of SMART, GENES4/ANP2003, Sep. 15-19, 2003, Kyoto, JAPAN

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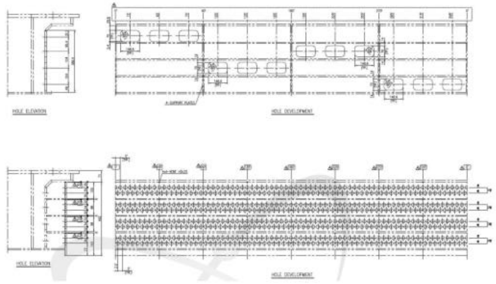
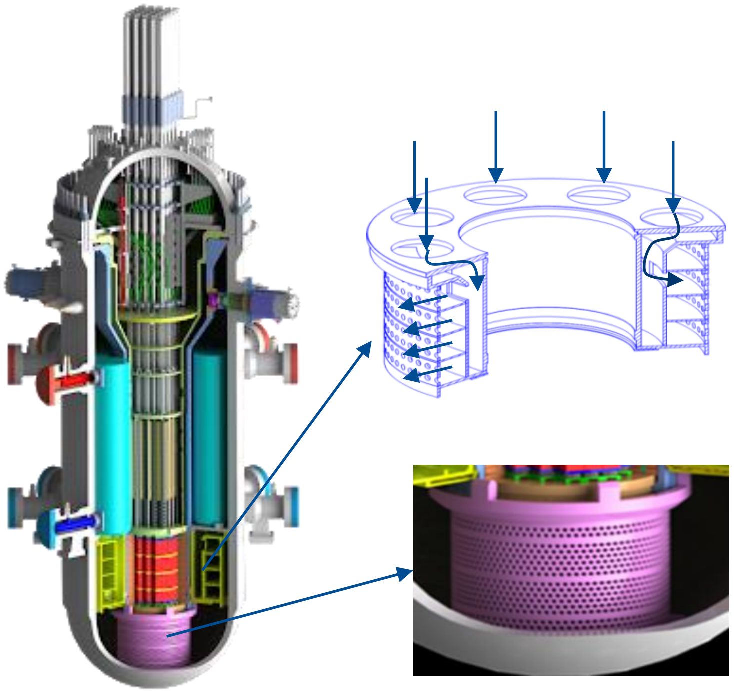
Integrated Primary Systems – Flow Mixing Assemblies

SMART Flow mixing assemblies:

- Aim of mixer is to blend outlet of 8 SG cassettes
- Flow mixing header assembly located in downcomer at same level as core. Complex flow structure (cf. drawing) - “larger” openings are located at different heights[1]



- Flow skirt[1] located at lower plenum comparable to flow mixer present within KONVOI reactor



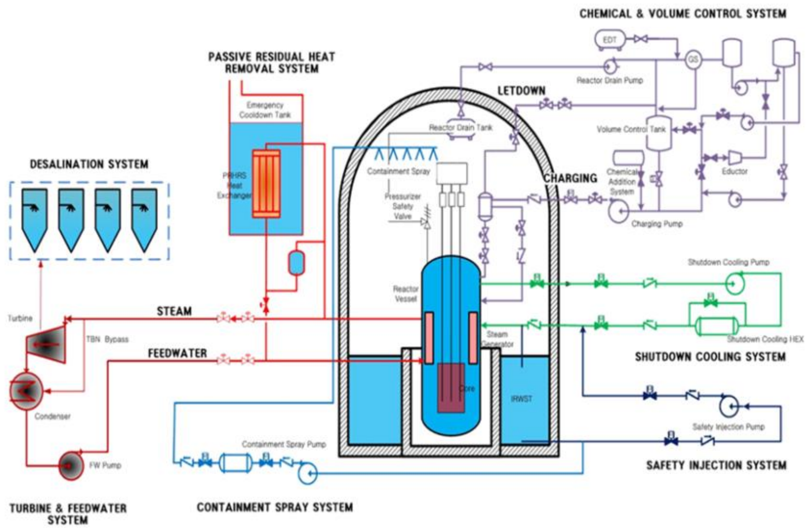
[1]: Euh D.J., Youn Y.J., Chu I.C., Test Facility for SMART Reactor Flow Distribution, KAERI/TR-4239/2010

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Integrated Primary Systems – Auxiliary & Safety systems

SMART Auxiliary & Safety systems[1]:

- Chemical and Volumetric Control System: to maintain primary water properties and control water level within reactor – SMART follows constant primary average program to limit water level modification within PRZ
- Shutdown Cooling System (SCS) keeps reactor cooled from Hot shutdown condition (~200°C) to Cold shutdown condition (~50°C). This system is used in normal conditions
- Passive Residual Heat Removal System (PRHRS) has capability to keep core undamaged for 72 h via natural circulation using Emergency Cooldown Tank as heat sink



[1]: Keung Koo Kim et al., SMART: The First Licensed Advanced Integral Reactor, Journal of Energy and Power Engineering 8 (2014) 94-102

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SMART In-core fuel management

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

In-core fuel management – Boron-free core design

- SMART adopted boron-free operation concept throughout its development[2]
    - Increased moderator temperature coefficient (inherently more stable, but potentially unfavorable with respect to overcooling accidents such as steam line break accident)
    - Elimination of boron dilution accident
    - Elimination of boric acid induced corrosion
    - Reduction of radioactive waste (90% of tritium due to neutron activation with soluble boron)
    - Reduced requirements of chemical volume control system (CVCS); emergency boron storage tank remains
- Low core power density (62.6 kW/l) makes boron-free core design possible as it is easier to accommodate higher associated power peaking factors
- Burnable poison material (Al_2O_3-B_4C and Gd_2O_3-UO_2 rods) with axial zoning for flattening radial and axial power profiles: to compensate for negatively skewed axial power profile because of deeply inserted control rods and very negative Moderator Temperature Coefficient (−72 pcm/°C, most negative at HFP BOC, not EOC[1])
- Increased number of control rods to compensate for fuel burnup (increasing individual rod worth instead would be penalizing for rod ejection accident criteria)

[1]: M.H. Chang et al., Advanced design features adopted in SMART, IAEA International Seminar Cairo, May 2001
[2]: Y. Alzaben et al., Core neutronics and safety characteristics of a boron-free core for Small Modular Reactors

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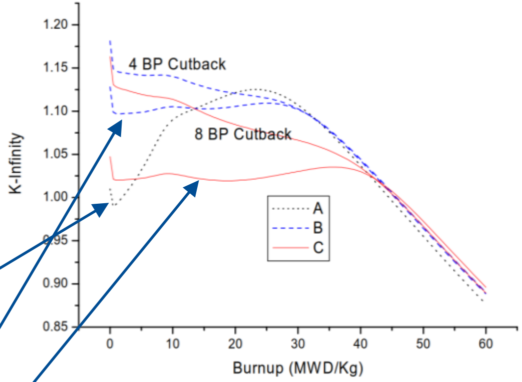
In-core fuel management – Basic design loading pattern (boron free)



	J	H	G	F	E	D	C		
1				A	B	A			
2			B	B	B	B	B		
3		B	B	C	C	C	B	B	
4	A	B	C	C	C	C	C	B	A
5	B	B	C	C	C	C	C	B	B
6	A	B	C	C	C	C	C	B	A
7		B	B	C	C	C	B	B	
8			B	B	B	B	B		
9				A	B	A			

A	4.95w/o U-235 / 28 Al_2O_3-B_4C Shim / 12 Gd_2O_3-UO_2
B	4.95w/o U-235 / 20 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2
C	4.95w/o U-235 / 24 Al_2O_3-B_4C Shim / 4 Gd_2O_3-UO_2

- Target cycle length of >3 years or about 990 Equivalent Full Power Days (capacity factor ~ 90%)
- Multiple reload schemes possible: single batch, 1.5 batch or double batch
- Typical enrichments: 4.80 to 4.95w/o U-235
- Typical end-of-cycle core burnup[1]: 26.6 to 31 MWd/kgU
- Typical average discharge burnup[2]: 36.1 MWd/kgU (double batch scheme)
- Burnable absorber rods distribution to flatten reactivity evolution: minimize rod movement for burnup compensation:
  - Type A has high burnable absorber load to reduce initial k-inf: ensures shutdown margin at BOC for positions without control rods
  - Type B and C have some burnable poison rods cutback at top to compensate for control rod insertion and reduce local power peaking

[1]: C. Lee et al., Nuclear and Thermal Hydraulic Design Characteristics of the SMART Core, 2003
[2]: IAEA – Status report 77 – System-Integrated Modular Advanced Reactor (SMART)

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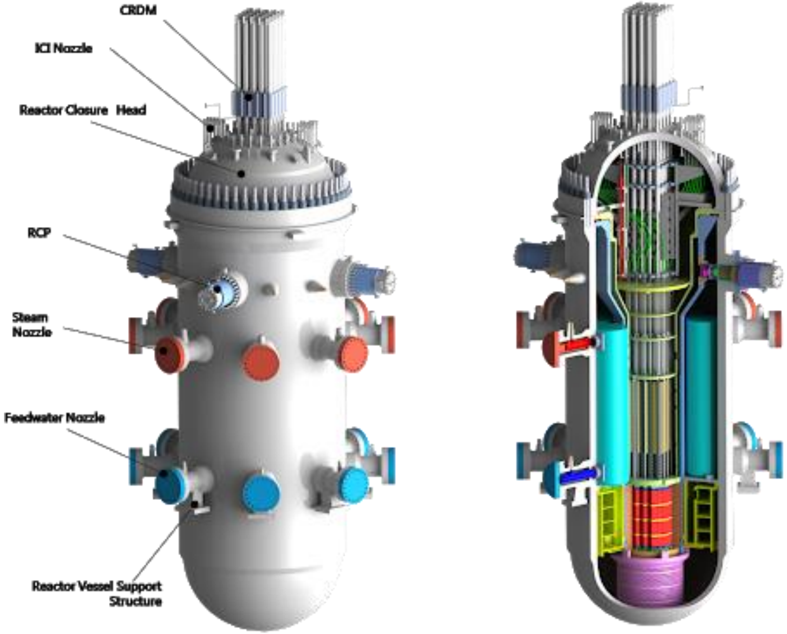
Summary

Summary

- SMART is a small-sized integral type PWR with a thermal power of 330MW
- SMART is a mixture of new innovative design features combined with proven technologies with the aim of enhancing safety and improving economics
- SMART is the first licensed Advanced Integral Reactor (since 4th July 2012 by the Korea Nuclear Safety and Security Commission)
- SMART proposes 3 main applications: electricity production, water desalination and district heating



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Quiz

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Q1

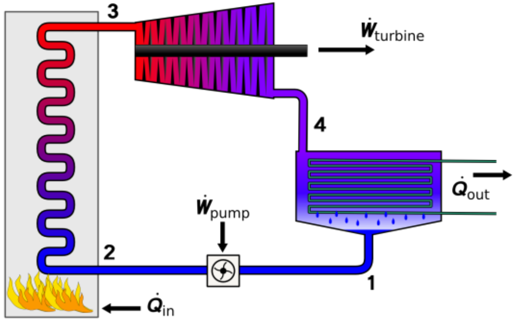
Assuming that SMART is deployed to generate electricity. Its secondary cycle is, as for a usual PWR, a Rankine Cycle. Conventional PWR and SMART present quite different steam conditions. Is there any gain from this modification?

A. Dry steam (PWR) or superheating steam (SMART) does not affect the secondary cycle

B. To reach the same efficiency, dry steam based secondary cycle requires the simplest system

C. The presence of superheating steam is more likely to damage the turbine blades during the expansion

D. The presence of superheating simplifies the secondary system since no water droplets are generated during the expansion process



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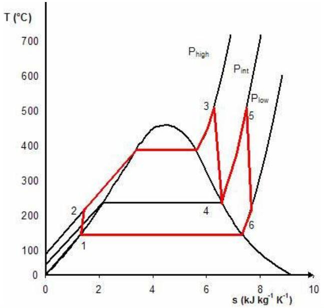

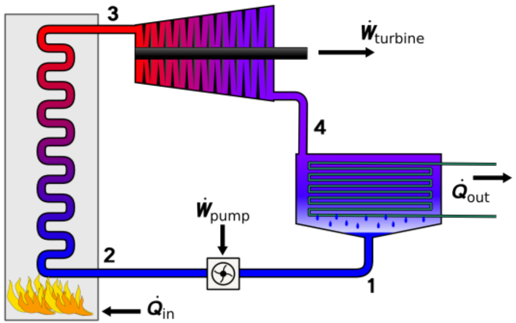

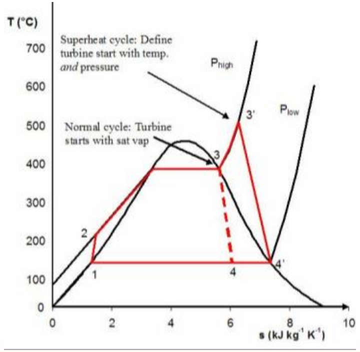

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Q2

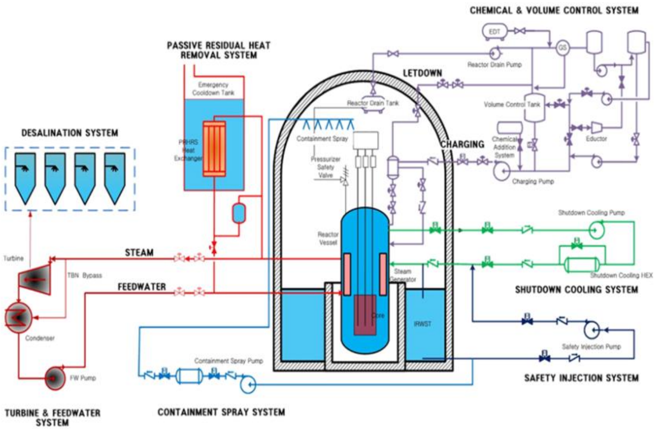
A boron-free core design certainly has some advantages. However, which ones would **NOT** be one of them (multiple answers possible)?

A. Elimination of boron dilution accident

B. No need for chemical volume control system (CVCS)

C. Inherent stability because of increased (more negative) moderator temperature coefficient

D. Elimination of overcooling accidents (steam line break)



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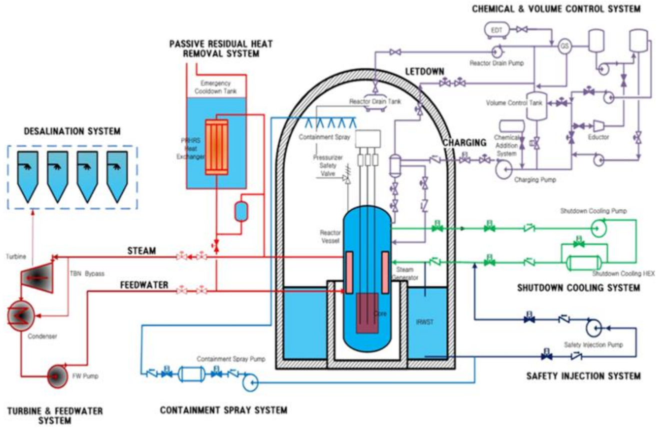

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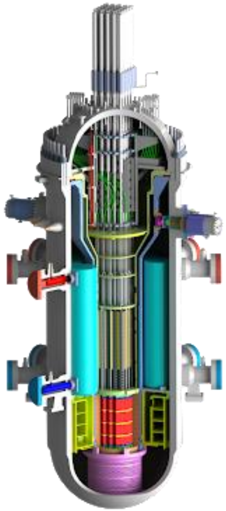
Q3

Do you think that a secondary break affecting 1 SG cassette could lead to any asymmetry within the primary core?

A. No asymmetry is possible as each SG cassette affects the primary globally
B. The asymmetry would be as important as within a conventional PWR
C. The asymmetry would be less important than within a conventional PWR thanks to the small SG secondary inventory and the primary mixing induced by the flow mixing header and flow skirt



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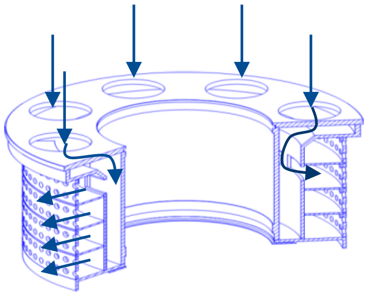

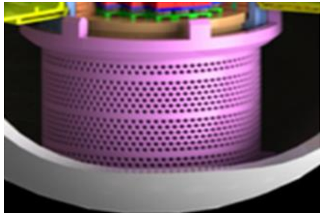

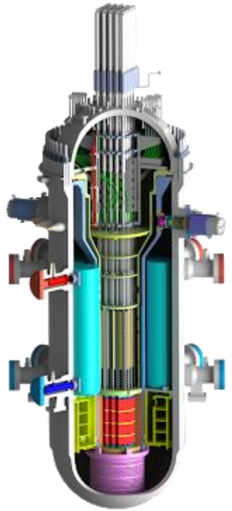

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SMART SAFETY FEATURES AND ACCIDENT ANALYSIS

McSAFER SMR First Course

V. Sanchez / KIT

25-27.01.2021

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Content

- **General safety features**
- **Safety systems**
- **Deterministic safety analysis**
- **SMART validation matrix**
- **Developing a NPP-Models for Safety analysis**

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General Safety Features

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SMART Safety Features compared to LWR Gen-II

- Combination of proven commercial reactor technologies with new and advanced technologies e.g. for fuel, self-pressurizing PZR, helically coiled once-through SG, new control concepts

- Implementation of DiD concept to assure enhanced safety and reliability
  - Inherent safety features such as low core power density
  - Elimination of large break loss of coolant accident, etc.
  - No core-uncover under during SBLOCA

- Accident prevention:
  - Passive engineered safety features

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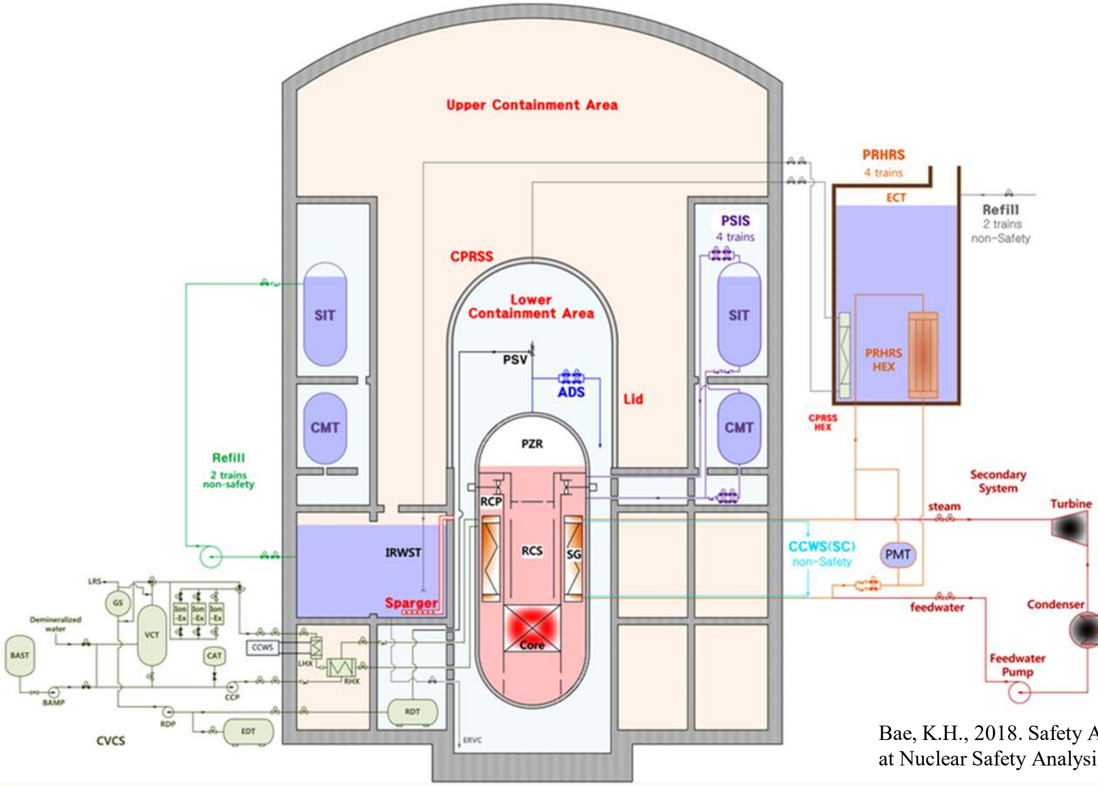
SMART: Engineering Safety Systems: Overview



- Passive gravity driven Reactor Shutdown System (RSS)
- Passive safety injection system (PSIS)
- Passive residual heat removal system (PRHRS)
- Reactor Over-pressure control System (ROPS)
- Containment Pressure and Radioactivity Suppression System (CPRSS)

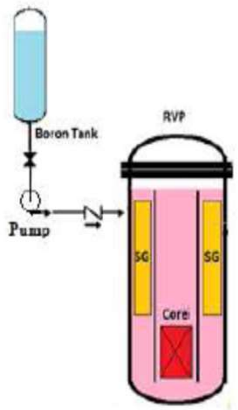
Bae, K.H., 2018. Safety Analysis for the Major DBAs in Passive PWR SMART [Presentation]. Presented at Nuclear Safety Analysis Symposium of KAERI-KACARE, 28-29 June, Daecheon, South Korea

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

How to Control the Sub-criticality?

- Passive gravity driven Reactor Shutdown System (RSS):
  - The shutdown signal de-energizes the control rod drive mechanism and then the control rods drop into the reactor core by the force of gravity.
  - Sufficient shutdown margin to bring the reactor from hot full power to hot shutdown condition, assuming “stuck rod” condition
  - It consists of the **nine shutdown banks** with control elements of B_4C

- Emergency boron injection system (backup)
  - Two trains: **2 X 100 %**
  - **One train** is able to bring the reactor to the subcritical state
  - Each train consists of a 6 m³ tanks filled with 30 g of boric acid per 1 kg of water
  - Active system working with a pump



IAEA TE-1785.2016

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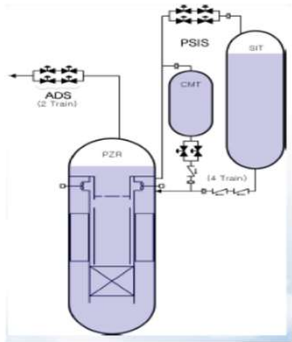
Emergency Core Cooling Systems (ECCS)

- A passive safety injection system (PSIS) **provides emergency core cooling** after a postulated DBA
  - Passive injection from CMTs and SITs
  - CMTs filled with borated water
    - provide makeup and **borated water to RCS** during early stage of a SBLOCA or non-LOCA events
- Redundancy:
  - 100% x 4 trains
  - Electrically independent: 2 trains,
  - Mechanically independent: 4 trains
- Safety requirement for DBA:
  - The **safety injection function of the PSIS** is maintained for long term since SITs are refilled periodically
  - Keep safe shutdown condition for 36 h
  - Prevent core damage for 72 h without AC-power and operator action



SMART ECCS: PSIS

- CMT
  - Low PZR-pressure < 10 MPa)
  - Low MSL-pressure
  - PRHRS actuation signal
- SIT
  - Low pressure

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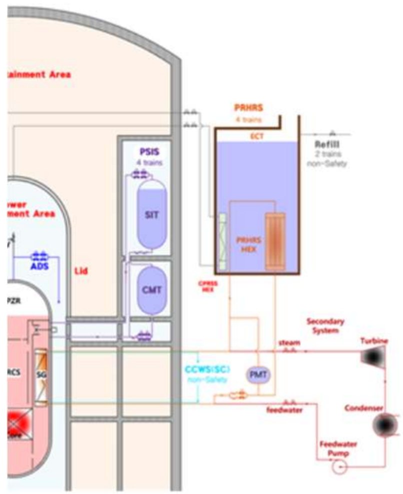
How to control long term core coolability?

- Passive Residual Heat Removal System (**PRHRS**)
  - Removes the **decay** and **sensible** heat by natural circulation in case of situations e.g. steam extraction, loss of FW, and SBO
- Key features:
  - 50% x 4 trains
  - 36 h operation without refill
  - Single train failure
- Actuation:
  - After the reactor is shutdown, if main heat sink is not operable for any reason
    - Safe shutdown condition of RCS within 36 h after accident initiation and
    - maintain the “safe shutdown condition” for at least additional 36 h
- Safety requirement
  - Keep core coolability for 72 h without any operator actions



SMART ECCS: PRHRS

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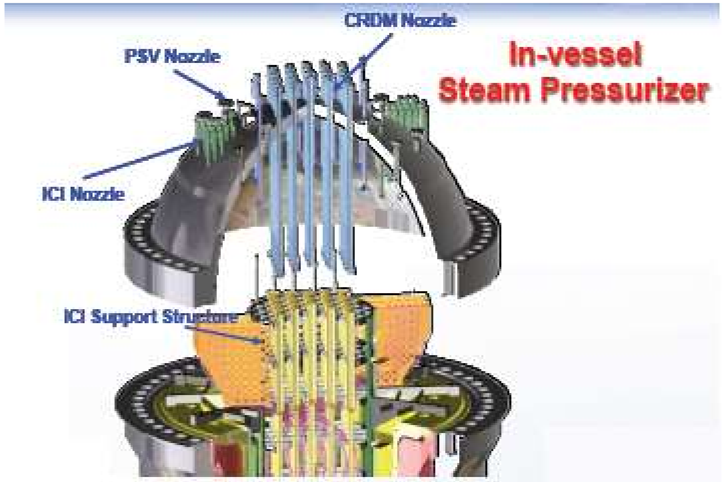
Reactor Over-pressure Protection System (ROPS)

- Function:
  - Reduce reactor pressure during the postulated design basis accident related to a control system failure.
- Safety Depressurization system (SDS)
  - Two pressurizer safety valves (PSVs)
  - Two trains are controlled by operators
    - PSV steam discharge lines are combined to a single pipe and connected to the containment atmosphere through the **Reactor Drain Tank** (RDT)
  - If RPV-pressure increases, PSVs open according to set-points discharging steam into the RDT
- Automatic Depressurization System (ADS)
  - In case of DBA, automatic actuation of valves (2 steps)
  - Two trains



SMART: In-Vessel pressurizer

Ji-Han Chun, J.S. Song, H. O. Kang, K. K. Kim; SMART Development Status and Collaboration with KSA for the Deployment. INPRO dialog Forum. Vienna October 18-21.2016.

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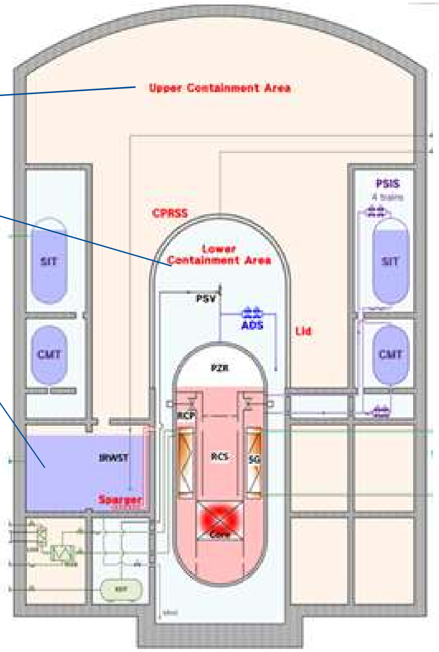
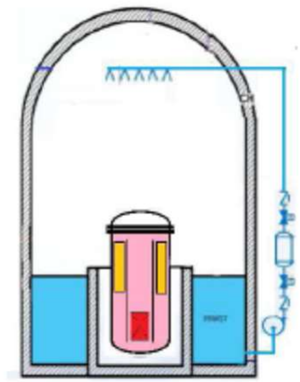
Containment systems (1/2)

- Containment design
  - Upper containment area (UCA)
  - Lower containment area (LCA)
  - **I**n-containment **R**efueling **W**ater **S**torage **T**ank (IRWST)
- Containment Pressure Control
  - Active CSS (**C**ontainment **S**pray **S**ystem)
  - Two trains: 100% x 2 capacity
- **Hydrogen risk**: PARs located inside the LCA and UCA



Large dry containment



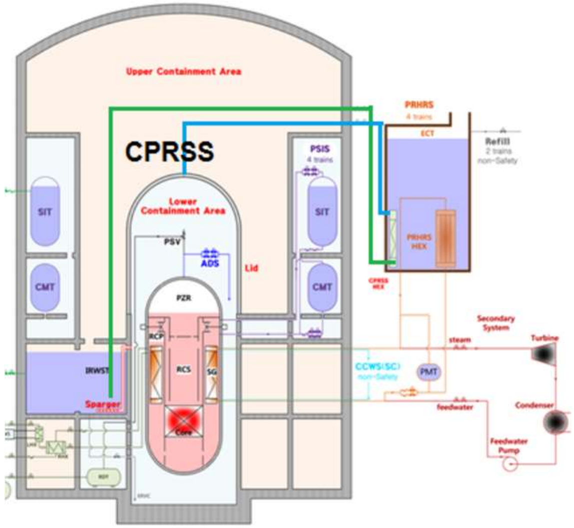
Containment with one Spray Line

IAEA TE-1785. 2016

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Containment systems (2/2)

- Passive **Containment Pressure and Radioactivity Suppression S**ystem (CPRSS)
- **Confine** radioactive fission products within the containment building
- Protect the environment against primary coolant leakage
  - Fission products are scrubbed in the IRWST water
- **C**ontainment **H**eat **R**emoval **S**ystem (CHRS): In case of MSLB or LOCA
  - Remove part of the released energy to IRWST
  - Remove part of the released energy to environment



SMART: Passive CPRSS
- Green line: IRWST-connection to HX-top inside the ECT
- Blue line: LCA-connection to the HY-lower part inside the ECT

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

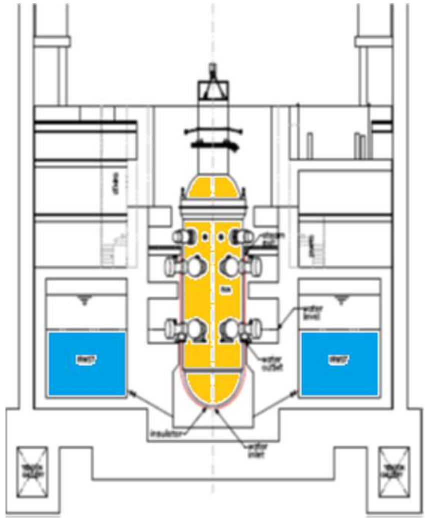
Other Safety Systems:

- **Emergency power supply system**
  - Active safety system
  - 100% x 2 trains
- **Shutdown Cooling System (SCS)**
  - Active safety system
  - 100% x 2 trains

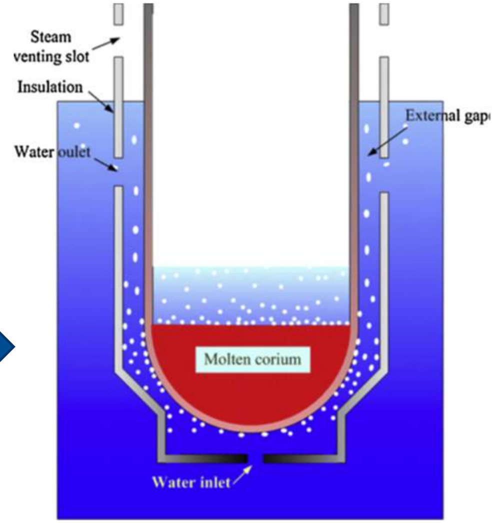
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

How to control Severe Accidents?

- The Severe Accident Mitigation system (SAMS) prevents the release of molten corium resulting from a SA outside the containment.
- How to control RPV failure?:
  - IVR-ERVC: **I**n-**V**essel corium **R**etention through **E**xternal **R**eactor **V**essel **C**ooling
  - Passive flooding of reactor cavity with water from the IRWST
- How to control hydrogen explosions?
  - PARs located in containment
  - 12 units



Scheme of the containment of SMR



IVR-ERVC Concept

Rae-Joon Park, Jae Ryong Lee, Kwang Soon Ha, Hwan Yeol Kim; Evaluation of in-vessel corium retention through external reactorvessel cooling for small integral reactor. NED 262 (2013) 571-578

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

ACCIDENT ANALYSIS

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SMART Preliminary SAR: Event Categorization

- Physical phenomena: Korean Regulations (RG. 1.70, SRP, SRG)
  - **(1) 15.1 Increase in heat removal by the secondary system**
  - **(2) 15.2 Decrease in heat removal by the secondary system**
  - **(3) 15.3 Decrease in reactor coolant flow rate**
  - **(4) 15.4 Reactivity and power distribution anomalies**
  - (5) 15.5 Increase in RCS inventory
  - **(6) 15.6 Decrease in RCS inventory**
  - (7) 15.7 Release of Radioactive Material from a subsystem or component
  - (8) 15.8 Anticipated transients without scram (ATWS)

Chapter	Initiating event	Frequency
15.1.6	Improper operation of a Passive Residual Heat Removal System	AOO
15.1.7	Inadvertent opening of a safety relief valve of the PRHRS	AOO
15.6.1	Inadvertent opening of a pressurizer safety valve Inadvertent operation of automatic depressurization system	PA

- AOO: Anticipated Operational Occurrence
- PA: Postulated Accidents

Bae, K.H., 2018. Safety Analysis for the Major DBAs in Passive PWR SMART [Presentation]. Presented at Nuclear Safety Analysis Symposium of KAERI-KACARE, 28-29 June, Daecheon, South Korea

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SMART SAFETY ASSESSMENT

- Based on PSA Level 1, the following 10 scenarios are considered:
  - General transients
  - Loss of feed-water
  - Loss of offsite power
  - SBLOCA
  - **SLB (steam line break)**
  - SGTR (steam generator tube rupture),
  - Large secondary side break,
  - **Control rod ejection (REA)**
  - **ATWS**
  - Control rod BWA (bank Withdrawal)

Scenarios to be investigated in McSAFER project for a generic SMART plant

S. K. Zee, Design Report for SMART Reactor System Development, KAERI/TR-2846/2007, KAERI, Taejon, 2007.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Deterministic Safety Analysis

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Deterministic Approach for Safety Analysis

Option	Quality of computer code	Availability of safety systems	Initial and boundary conditions	Remark
Conservative	Simplified code, conservative model parameters	Conservative: usually repair case plus single failure	Conservative: aggravating initial and boundary conditions	Historical approach: • Point neutron kinetics • Global feedback coefficients.
Combined	Best estimate	Conservative: usually repair case plus single failure	Conservative: aggravating initial and boundary conditions	Current approach: • Initial power 106% • Conservative decay heat curve
Best estimate	Best-estimate	Conservative: usually repair case plus single failure	Realistic I. and B. C. considering the uncertainties	Current approach: uncertainty analysis of model, input, and output parameters
Risk informed	Best-estimated + Uncertainty quantification	Derived from PSA	Realistic + uncertainties	Future approach

IAEA: SSG-2

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

McSAFER Safety Analysis Methods

- **Core analysis (Static /transient)**
  - Traditional with system code and point kinetics
    - **RELAP5, ATHLET**
  - 1D system code + 3D nodal diffusion
    - **TRACE/PANTHER**
    - **TRACE/ANTS**
  - Low order transport + subchannel codes
    - **PARCS-SP3/SCF**
    - **APOLLO3/FLICA**
    - **WIMS/ARTHUR**
    - **DYN3D-SP3/SCF**
  - High-fidelity MC + subchannel codes
    - **SERPENT2/SCF**

- **Multiscale SMR RPV Analysis**
  - 1D system TH code + PK
    - **TRACE**
    - **ATHLET**
  - 3D system TH-code + Subchannel code
    - **TRACE/SCF**
    - **TRACE/ARTHUR**
  - 3D system TH code + 3D nodal diffusion + CFD code
    - **TRACE/OpenFOAM**
    - **ATHLET/OpenFOAM**
    - **RELAP5/FLUENT**

- **Multiscale/multiphysics SMR Plant Analysis**
  - 1D system TH code + 3D nodal diffusion
    - **TRACE/PARCS**
    - **TRACE/PANTHER**
    - **ATHLET-DYN3D**
  - 3D system TH-code + Subchannel code + 3D nodal diffusion
    - **TRACE/PARCS/SCF**
    - **TRACE/WIMS/ARTHUR**
  - 3D system TH code + 3D nodal diffusion + CFD code
    - **TRACE/PARCS/OpenFOAM**
    - **ATHLET/DYN3D/OpenFOAM**
    - **TRACE/ANTS/OpenFOAM**

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Numerical Simulation Tools

- Numerical safety analysis tools
  - Based on “State-of-the-Art” physical models
  - Extensive validation and qualification needed for used in licensing processes
    - Code-to-code comparison
    - Numerical benchmarks
    - Experimental data (Separate effects, bundle and integral tests
    - Plant data
- Quantification of code‘s uncertainties is required in licensing
- Nuclear power plant is complex system involving different sciences and engineering branches

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Requirements on Numerical Simulations Tools

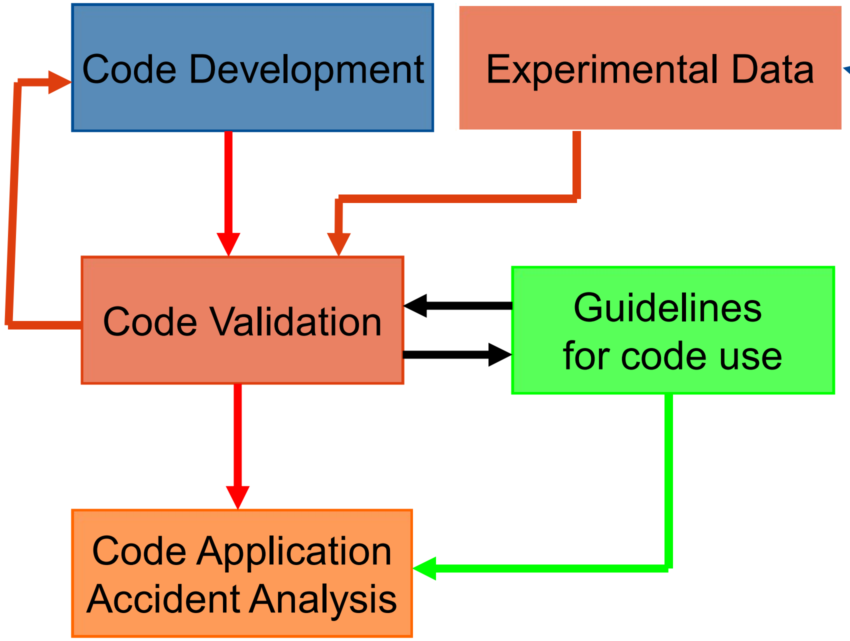
- Sufficient documentation of all models and correlations
  - To make sure that models are relevant to safety-related phenomena in Nuclear Power Plants
  - To assure that models are being applied inside the foreseen application range

- Availability of Guidelines and User Manuals
  - Recommendations for nasalization
  - Recommendations for the choice of model parameters
  - Quantification of uncertainties of results

- Wide verification, validation and qualification for the relevant phenomena of a reactor type under consideration (specific validation matrix)

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Numerical Tools: Development, Validation, Application



Framework for the validation of codes for safety demonstration

- Single effect tests
- Integral tests
- Plant data

Experimental data must representative for each reactor design regarding safety-relevant phenomena

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SMART Validation Matrix

This project has received funding from the Euratom research and
training programme 2019-2020 under grant agreement No 945063.

SMART Experimental Validation Program

- 10 safety tests:
  - Core CHF tests
    - CHF Water and Freon
  - Separate effect
    - Safety injection
    - Helical HX HT
    - Condensation HT in helical HX
  - Integral effect tests
    - VISTA SBLOCA
    - SMART-ITL

- 12 performance tests
  - Fuel assembly out-of-pile test
    - Out-of-pile mechanical tests
    - Out-of-pile hydraulic tests
  - RPV thermal-hydraulics tests
    - Flow distribution
    - Flow mixing Header assembly
    - PZR steam
    - PZR level measurement
  - Component tests
    - RCP hydrodynamics
    - RPV internal dynamics
    - SG-tube irradiation
    - Helical SG ISI
    - In-core instrumentation

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SMART Validation Matrix

- Thermal hydraulic tests - IET and SET -selected based on PIRT
- Separate effect tests (SET) for SMART
  - SCOP tests: verify the reactor internal flow and pressure distributions
  - SWAT test: to test the ECC (Emergency Core Coolant) performance and to verify the safety injection performance and validate the thermal-hydraulic model used in the safety analysis code
  - CHF Freon tests done at the FTHEL facility [6] to construct a database from a 5x5 rod bundle Freon CHF tests and to verify the DNBR (Departure from Nucleate Boiling Ratio) model in a safety analysis and core design codes

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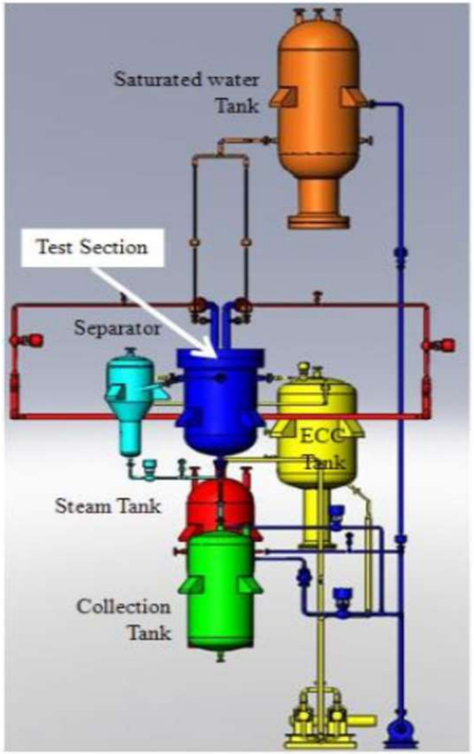
SET SWAT FACILITY

- Goals:
  - to test the ECC performance
  - to verify the safety injection performance and
  - To validate the TH-model of safety analysis code

S. J. Yi1, H. S. Park1, T. S. Kwon1, S. K. Moon1, S. Cho1, D. J. Euh1, Y. J. Ko1, Y. I. Cho1, B. Y. Min1, Y. C. Shin1, Y. J. Chung1, and W. J. Lee; Major results of thermal-hydraulic validation tests for the Smart design licensing. NURETH15. 2013 Pisa Italy.

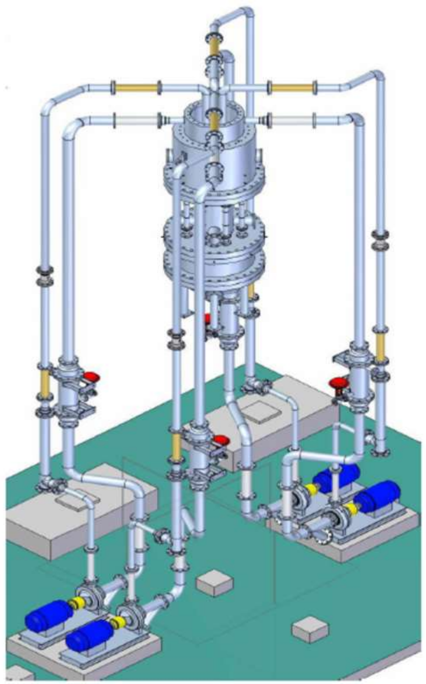


SWAT Facility: Schematic view

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SET SMART Core flow distribution and Pressure drop test facility (SCOP)

- Goals:
  - Verify design quantification tests for the flow and pressure distribution with a preservation of flow geometry
- SCOP facility
  - Scaled down facility
  - Core (57 FAs()
  - Steam generators (8)



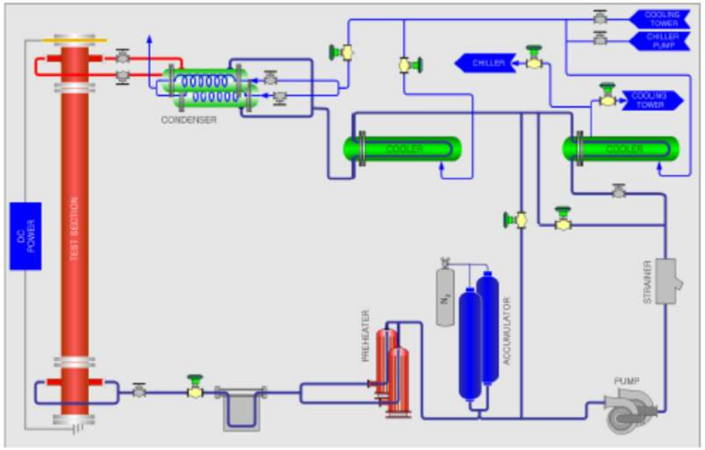
SCOP Facility: Schematic view

S. J. Yi1, H. S. Park1, T. S. Kwon1, S. K. Moon1, S. Cho1, D. J. Euh1, Y. J. Ko1, Y. I. Cho1, B. Y. Min1, Y. C. Shin1, Y. J. Chung1, and W. J. Lee; Major results of thermal-hydraulic validation tests for the Smart design licensing. NURETH15. 2013 Pisa Italy.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SET FTHEL Facility: CHF Freon Tests

- Goals:
  - to assess the conservatism of the CHF correlation, which would be used for the design and licensing of the SMART reactor,
  - to obtain some insights on the CHF characteristics of the SMART to reflect the Freon CHF test results on the water CHF test.
- Performed tests:
  - Type 1: Typical core without unheated rods (with the same configuration with SMART)
  - - Type 2: Thimble core with one unheated rod (simulating guide tube)
  - - Type 3: Typical core without unheated rods (with different spacer grid configuration))



SET FTHEL: Schematic layout

S. J. Yi1, H. S. Park1, T. S. Kwon1, S. K. Moon1, S. Cho1, D. J. Euh1, Y. J. Ko1, Y. I. Cho1, B. Y. Min1, Y. C. Shin1, Y. J. Chung1, and W. J. Lee; Major results of thermal-hydraulic validation tests for the Smart design licensing. NURETH15. 2013 Pisa Italy.

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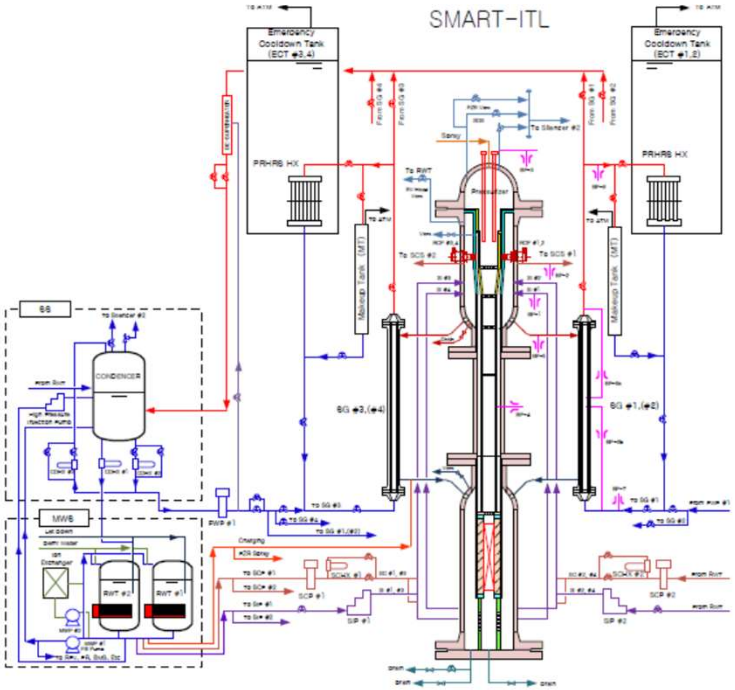
SMART-ITL Facility

- to investigate the integral performance of the inter-connected components and possible thermal hydraulic phenomena occurring in the SMART design, and to validate its safety for various design basis events
- Includes: primary circuit, 4 SG, secondary system, 4 trains of PRHRS, 4 trains of safety injection system (SIS), 2 trains of shutdown cooling system (SCS), break simulator (BS), Break-flow measuring system (BMS)
- Performed tests:
  - SBLOCA, break at SIS-piping nozzle

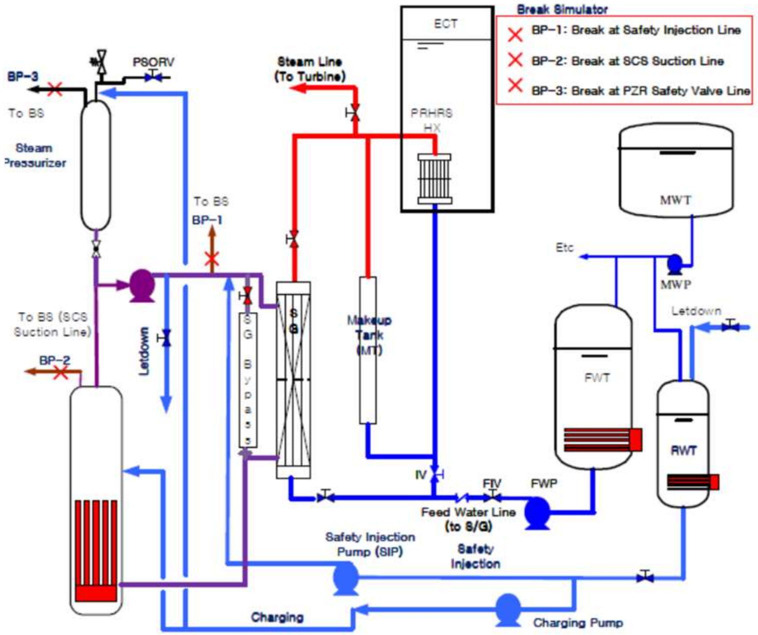
S. J. Yi1, H. S. Park1, T. S. Kwon1, S. K. Moon1, S. Cho1, D. J. Euh1, Y. J. Ko1, Y. I. Cho1, B. Y. Min1, Y. C. Shin1, Y. J. Chung1, and W. J. Lee; Major results of thermal-hydraulic validation tests for the Smart design licensing. NURETH15. 2013 Pisa Italy.



SMART-ITL Facility: Schematic view

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

VISTA-ITL Facility



VISTA-ITL Facility: Schematic view

- Goals:
  - to validate the TASS/SMR-S safety analysis code [3]
  - to support standard design licensing, and
  - to construct a database for the SMART design optimization and foreign export
- Performed tests: SBLOCA, complete loss of primary coolant mass flowrate (CLOF), PRHRS-performance
- Break type: guillotine break
- Break location:
  - at the safety injection system (SIS) line (nozzle part of the RCP discharge),
  - at the suction line of the shutdown cooling system (SCS) (nozzle part of the RCP suction)
  - At Pressurizer safety valve (PSV) line connected to the pressurizer top

S. J. Yi[1], H. S. Park[1], T. S. Kwon[1], S. K. Moon[1], S. Cho[1], D. J. Euh[1], Y. J. Ko[1], Y. I. Cho[1], B. Y. Min[1], Y. C. Shin[1], Y. J. Chung[1], and W. J. Lee; Major results of thermal-hydraulic validation tests for thesmart design licensing. NURETH-15, 2013. Pisa Italy.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Developing a Nuclear Power Plant Model for Safety Analysis

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Key-Steps in Modelling Nuclear Power Plants

Integral Nuclear Power Plant Models for Accident Analysis

NPP Data
- Reactor type
- Geometrical data
- Material data
- Heat removal systems
- Control systems
- Safety systems
- Operational data

Key-Physical Phenomena
- Fluid dynamics
- Heat transfer phenomena
- 2D/ 3D phenomena (location)
- Safety-relevant phenomena

Numerical Tool
- Selection of numerical tool
- Modeling capabilites
- Modeling scope
- Component models
- I&C components

Accidental Sequences
- Choice of accidental sequence (PSA)
- Sequence of events
- Involved safety systems
- Assumption
- Operator actions

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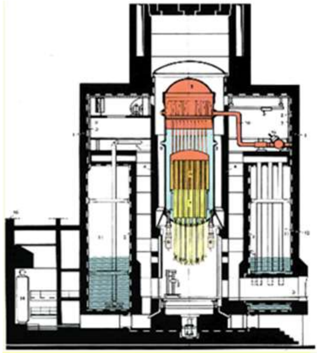
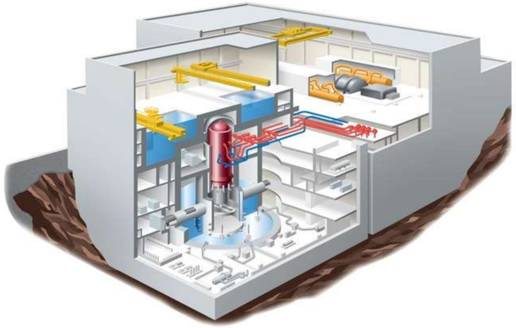
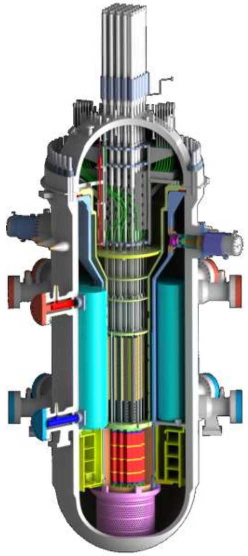
Plant Data: Nuclear Power Plant Designs



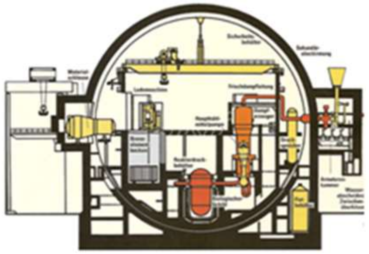
GEN-II BWR



GEN-III BWR: ESBWR



SMR



GEN-II PWR

GEN-III PWR: AP-1000

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Classification of Codes for Design Basis Accidents

Thermal Hydraulics:

Best-estimate codes system TH codes

- 1D, 3D models,
- Six equation models
- Vertical and horizontal flow regimes
- Full boiling curve

- RELAP5
- ATHLET
- TRACE
- APROS
- CATHARE

Neutronics:

Reactor dynamic codes

- 3D nodal diffusion
- 3D transport
- Dynamic Monte Carlo

- PARCS, DYN3D
- MPACT, DeCART, DYN3D-SP3
- SERPENT2

N/TH Coupled Codes:

- TRACE/PARCS, ATHLET/DYN3D, CATHARE/CRONOS
- MCPACT-CTF, DYNSUB5.0, PARCS/SubChanFlow
- SEPENT2/SubChanFlow, TRIPOLI4/SubChanFlow

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Classification of Severe Accident Codes

Category	Characteristics	Codes
Mechanistic codes	• Mechanistic model • CPU-intensive • Single /two phase flow	• SCDAP-R5 • ATHLET-CD • SAMPSON • SOCRAT
Integral Codes	• Lamped parameter • Fast running • 0D/1D discretization	• ASTEC • MELCOR • MAAP • THALES • AC2
Dedicaded codes	• Detailed models • 2D/3D modes • Specific phenomena	• GOTHIC • GASFLOW • CFD • COM3D

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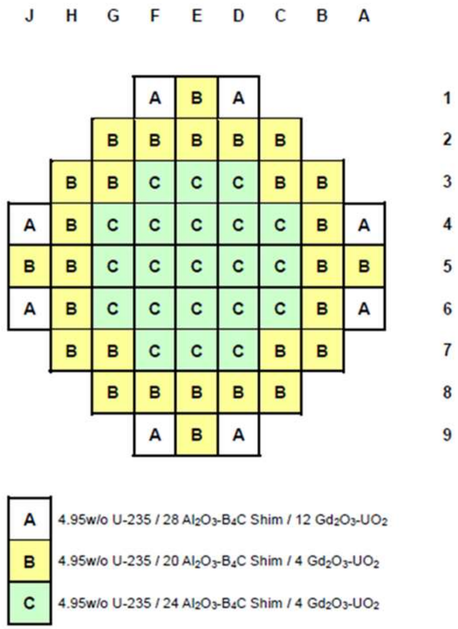
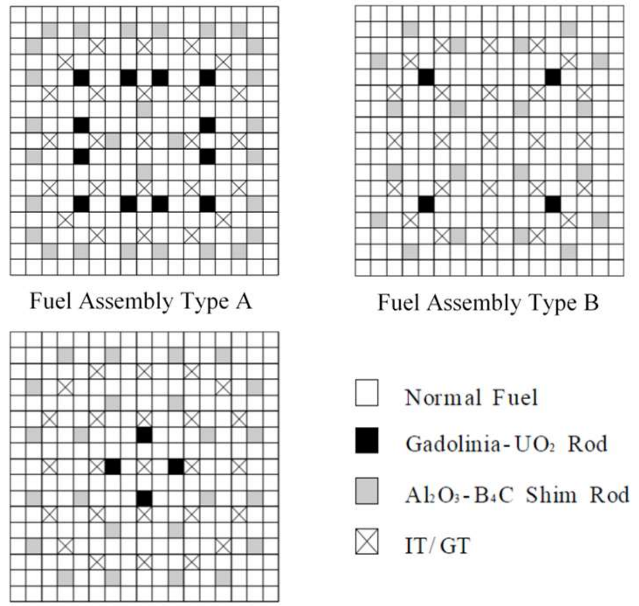
SMART Core Characteristics





Fuel Assembly Type A

Fuel Assembly Type B

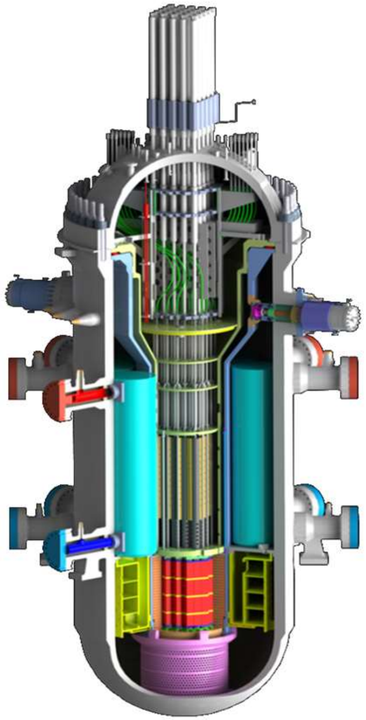
Fuel Assembly Type C

- 1) Strategic Insights Inc., “2015 Small Modular Reactor (SMR) Market Outlook Report”. (2015).
- 2) IAEA-TECDOC-1444, “Optimization of the Coupling of Nuclear Reactors and Desalination Systems”, (2015).

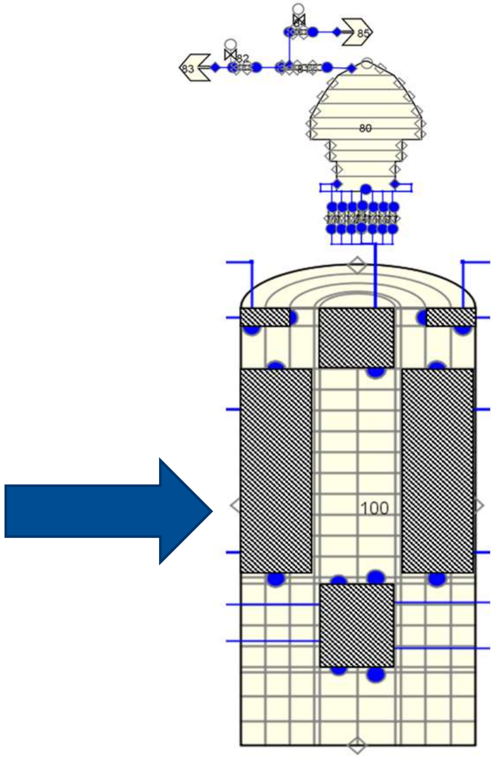
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

TRACE: RPV Model of SMART

- Core model (3D Cartesian)
- RPV (3D Cylindrical model)
  - Downcomer
  - Mixer
  - Lower plenum
  - Riser
  - 8 Helical HX
  - 4 canned MCP
  - Inlet/outlet nozzles
- Pressurizer



SMART

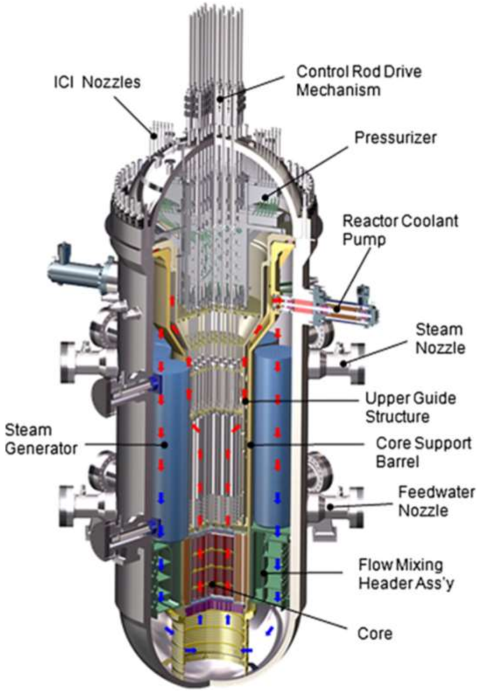


TRACE Model

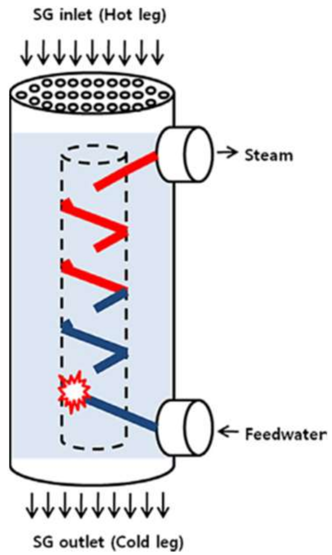
Ref: Keun Bae Park, "*SMART An Early Deployable Reactor for Multi-purpose Applications*", KAERI, October 2011.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SMART RPV and Helical SG



SMART RPV: SG

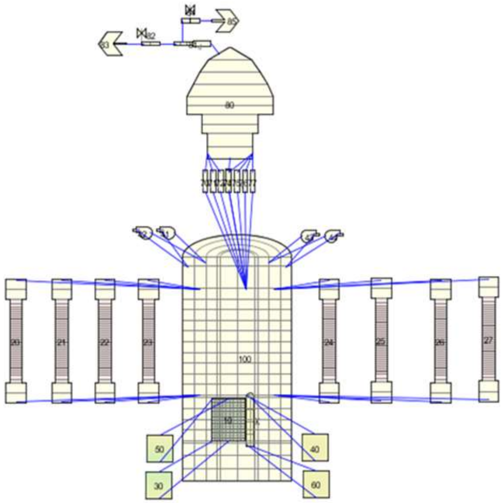


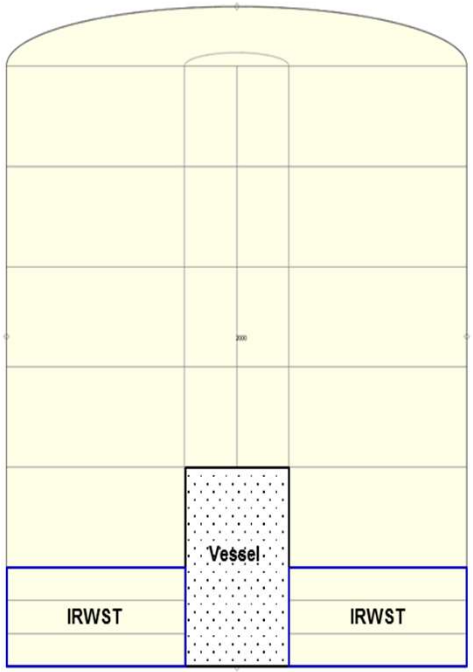
Scheme Helical SG

Hee-Kyung Kim, Soo Hyoung Kim, Young-Jong Chung, Hyeon-Soo Kim; Thermal-hydraulic analysis of SMART steam generator tube rupture using TASS/SMR-S code. ANE 55 (2013)331-340

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

TRACE: Developing an Integral Plant Model for SMART (1/2)





Overall form	Adopted from Reference (24)	-	Cylindrical
Diameter	Adopted from Reference (32)	m	36
Height	Adopted from Reference (32)	m	68,4
Wall Thickness	Adopted from Reference (32)	m	1,4
Volume	Adopted from Reference (32)	m^3	56390
Design Pressure	Adopted from Reference (24)	MPa	0,42

- This component is not interacting in the steady state calculation, its intended to be used in future transients

TRACE: Integral SMART model(Primary Circuit)

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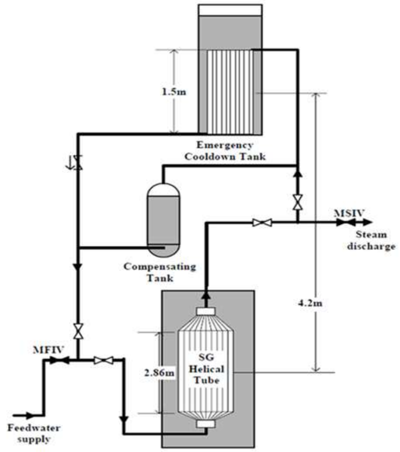
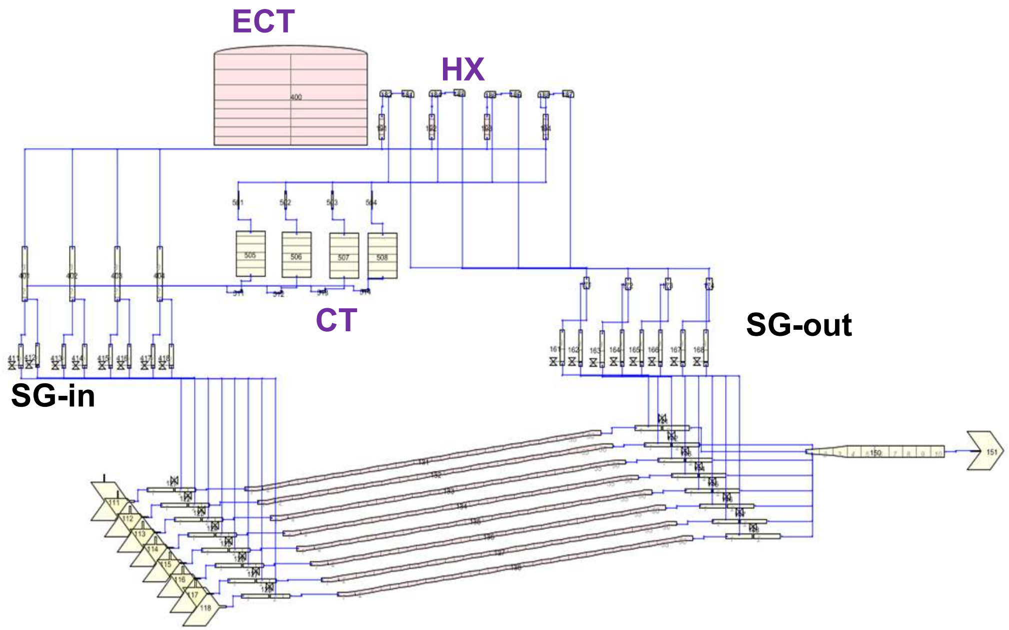
TRACE: Developing an Integral Plant Model for SMART (2/2)





- PRHRS
  - ECT: Emergency Cooldown Tank
  - CT: Compensating Tank
  - HX: SG Helical tube
- Secondary side: Inlet, Outlet, Helical tubes

TRACE: Integral SMART model (Secondary Circuit)

Y. J. Chung, G. H. Lee, H. C. Kim, K. K. Kim and S. Q. Zee, "*Parameters which effect the mass flow in the PRHRS under a natural convection condition*", KAERI, May 2004.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

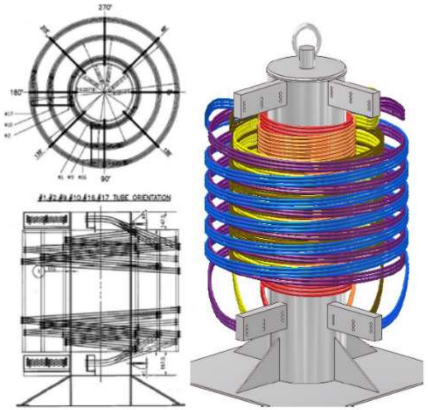
SMART: Peculiarities- Helical Steam Generators

- SG Primary side:
  - Cassettes
- SG Secondary side:
  - number of curved tubes

Parameter	Data Status	Unit	Value	Remark
Quantity	Adopted from Reference (1)	-	8	Number of Steam Generators
Length	Adopted from Reference (31)	m	6,0	-
Heat Transfer Height	Adopted from Reference (23)	m	3,8	-
Axial Pitch	Calculated from Reference (21)	-	1,176470588	Tubes Axial Pitch/Tubes Outer Diameter
Radial Pitch	Calculated from Reference (21)	-	1,323529412	Tubes Radial Pitch/Tubes Outer Diameter
Tube Rows	Adopted from Reference (21)	-	17	Same for every cell

Data from *SMART Cylindrical Vessel + Cartesian Core.xlsx* database

- TRACE helically coiled SG: implementation technique from J. W. Spore and W. Arcieri is followed



IAEA-TECDOC-1444, “Optimization of The Coupling of Nuclear Reactors and Desalination Systems”, June 2005

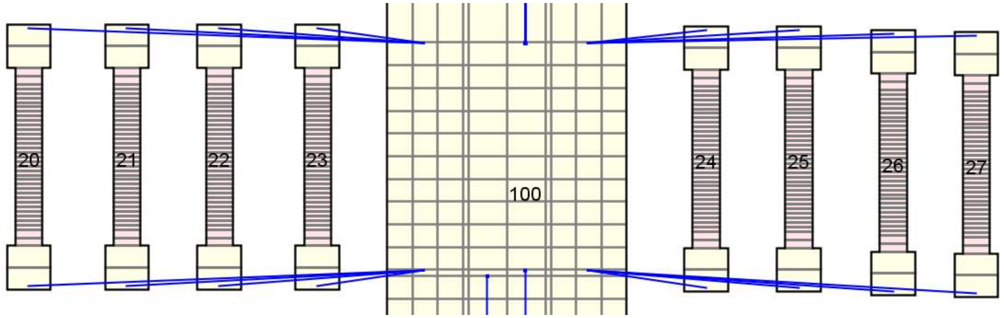
Hwang Bae a∗, Hyun-Sik Parka, Sung-Jae Yi a, Sang-Ki Moona, "Design of Steam Generator for SMART ITL", Korea Atomic Energy Research Institute, May 2010.

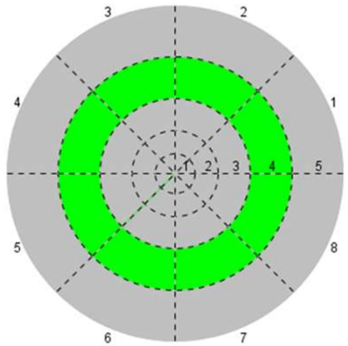
Han-Ok Kang, Hun Sik Han, Young-In Kim, Keung Koo Kim, "Thermal Sizing of Printed Circuit Steam Generator for Integral Reactor", Korea Atomic Energy Research Institute, May 2014.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

SMART: Peculiarities- Steam Generator Primary Side

- Cassettes modeling
  - PIPE TRACE component
  - use TUBE BANK CROSSFLOW option
- Cassettes connected directly to the 3D VESSEL
  - Connection at the middle cell of the downcomer for each azimuthal sector
  - Inlet connection at level 20 (negative)
  - Outlet connection at level 9 (negative)
- The nodalization follows the one chosen at the helical coils (secondary side)
- Volume data
  - Different values in different papers
  - Adjusted to obtain proper heat transfer
- K-Factor for the descendent flow against tube bundles = 0,7 (PWR Modelling Guidance)





Inlet and outlet azimuthal distribution

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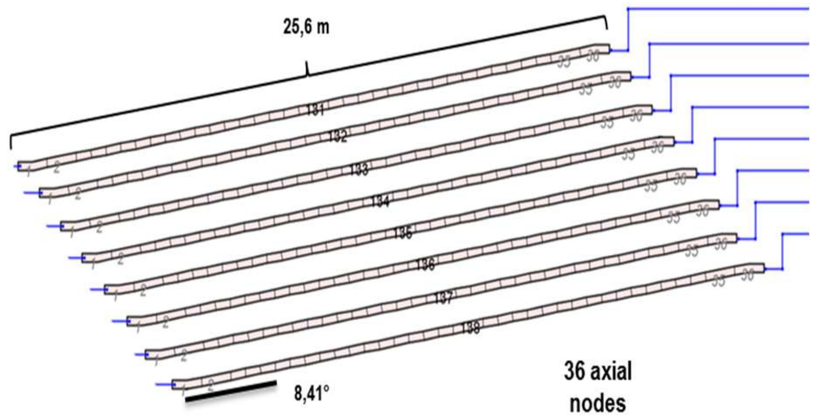
SMART: Peculiarities- Steam Generator: Secondary Side

- Representation of **helical tubes**
  - PIPE TRACE i.e. CURVED PIPE option
  - Diameter of 12 mm
  - Adjust K-Factors to get right ∆pressure
- The **connection** between **coils and cassettes**
  - eight HEATSTR
- Decide about the nodalization:
  - **no standard procedure**
  - changed many times
  - Needed optimization process

  until **steady state has been reached**
- Number of pipes
  - from bibliographical data
- Cassette volume
  - adjusted to ensure HT over SG



- Pitch-to-diameter ratio = 1,32 (PWR Modelling Guidance)
- Surface multiplier = number of tubes = 500
- Activation of axial conduction on heat structures and establish heat flux on AECL_IPPE w/ Biasi Correlations (PWR Modelling Guidance)

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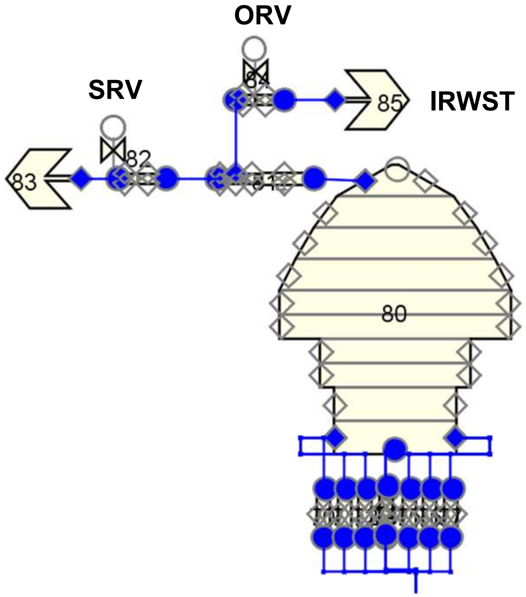
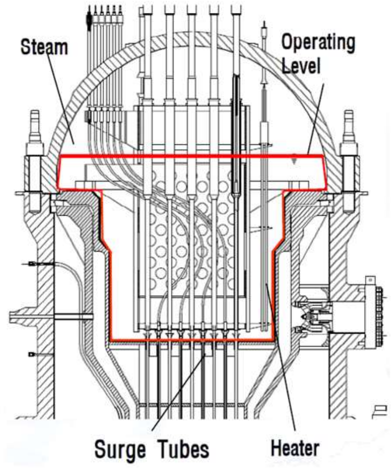
SMART: Peculiarities-Pressurizer

- Check constructive details
- Check if public data base includes all information needed
  - Heater length = 2m
  - Etc.
- Select best TRACE component to represent it: e.g. PIPE TRACE component with the type PRESSURIZER
- Introduce approximations if not precise info about shape
  - number of surge tubes
  - geometrical details
- Modeling issues e.g.
  - K-Factors are added to the surge lines using PWR Modelling Guidance
- Consider PZR SRV and a ORV
- Connections to IRWST (ADS)



TRACE



SMART

Ref: Keun Bae Park, "*SMART An Early Deployable Reactor for Multi-purpose Applications*", KAERI, October 2011

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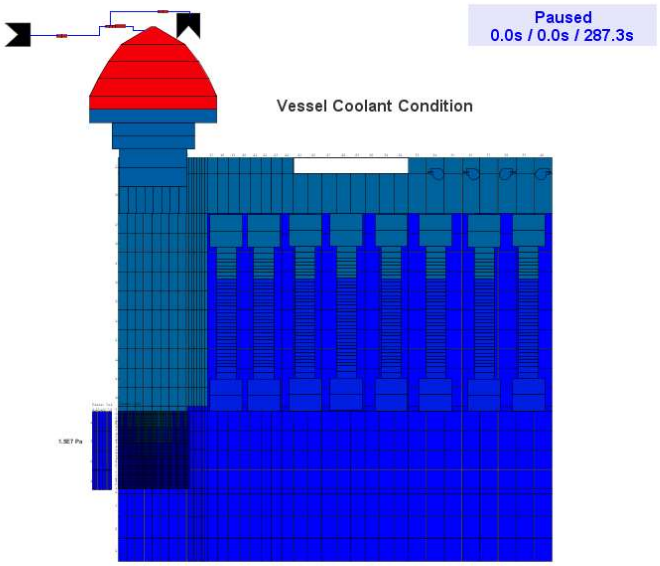
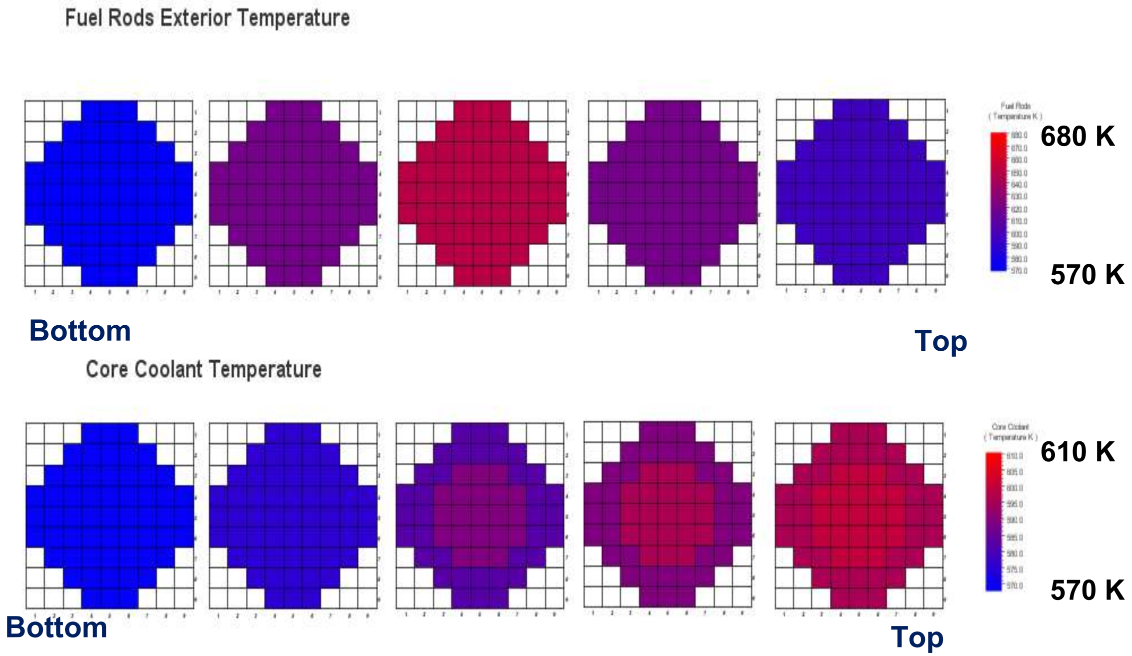
Karlsruhe Institute of Technology

TRACE: Visualization of Results with SNAP (1/2)



- **TRACE: RPV Coolant temperature distribution**



TRACE: 2D Coolant temperature (bottom) and Cladding temperature (top) in the SMART core

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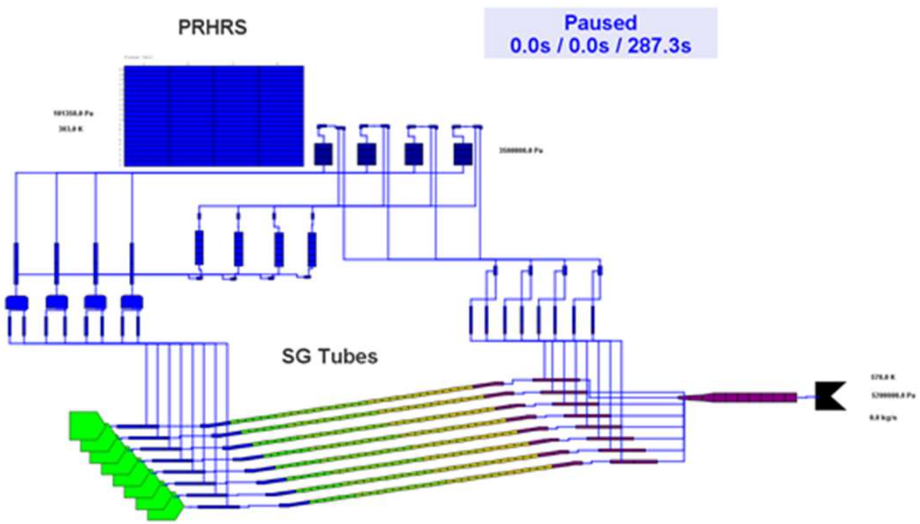
TRACE: Visualization of Results with SNAP (2/2)



Parameters	TRACE	Reference	Difference %
Inlet Core Temp. (K)	570.30	568.85	0,25
Outlet Core Temp. (K)	597,33	596,15	0,19
Outlet SG Temp. (K)	567,85	571 (theoretical)	-0,55
Steam Mass Flow (kg/s)	160,8	160,8	0
Steam Pressure (Pa)	5,198E06	5,2E06	-0,0038
Reactor Coolant Flow (kg/s)	2090	2090	0
Bypass Mass Flow (kg/s)	46,53	45,98	1,1
Core Mass Flow (kg/s)	2043,47	2044,02	-0,0026
SG Mass Flow (kg/s)	261,25	261	0,0095
Pump Impeller Speed (rad/s)	178,22	179	-0,43
Core Pressure Drop (KPa)	35	(values between 5 & 45)	-
SG Cassette Pressure Drop (KPa)	95	55	49
SG Coil Pressure Drop (KPa)	180	170	14

- **SMART Secondary Side**

TRACE: Comparison of predicted and reference data for nominal conditions

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

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Backup Slides

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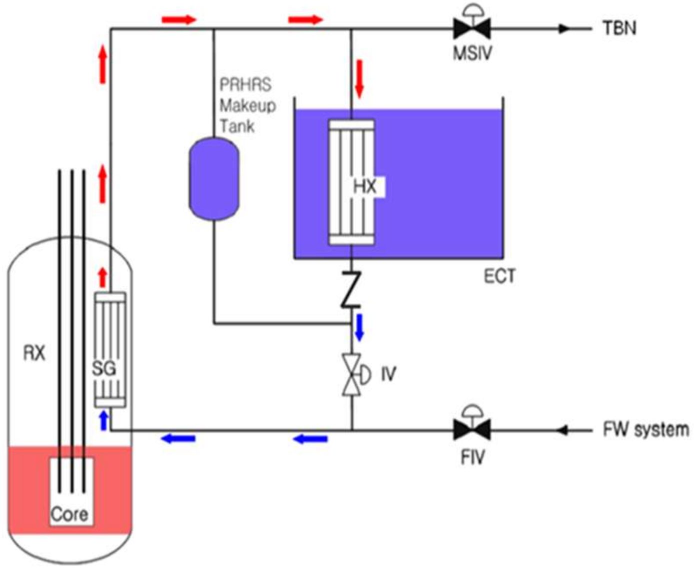
Long Term Coolability: PRHRS

- Each train has a heat exchanger submerged in an emergency cooldown tank (ECT), a makeup tank, valves, and pipes:
  - **Normal operation**: PRHRS is deactivated through closing all isolation valves connected to the secondary-side
  - **Accidental conditions**: the main steam isolation valves (MSIVs) and feedwater isolation valves (FIVs) are closed
    - generated steam from helical SGs flow into the submerged heat exchangers within the ECT
- Heat is transferred from the steam to the water of the ECT through condensation



PRHRS Layout connected to the SG-secondary-side

Ref-1; Park, K.B., 2011. SMART: An Early Deployable Integral Reactor for Multi-purpose Applications [Presentation], in: INPRO Dialogue Forum on Nuclear Energy Innovations: CUC for Small & Medium-Sized Nuclear Power Reactors. Vienna, Austria. October 10-14.

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NuScale reactor core and primary circuit

Course on „SMR LWR Technologies“

Ville Valtavirta / VTT Technical Research Centre of Finland Ltd

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

- General overview of the NuScale reactor core.
- Primary coolant loop.
- Reactor core.
- Reactivity control.

See for yourself:
https://www.nrc.gov/reactors/new-reactors/smr/nuscale/documents.html

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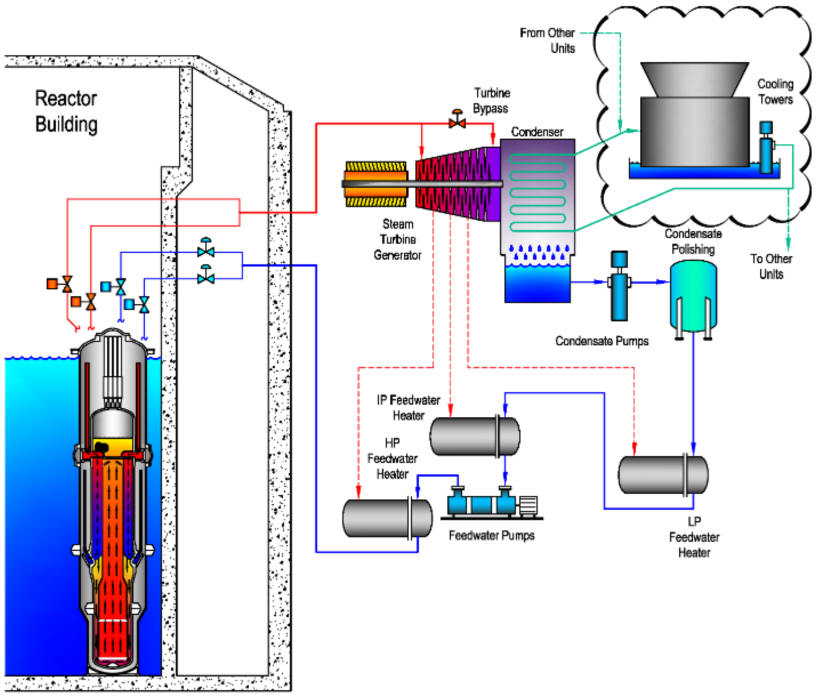
Figure 1.2-3: Schematic of a Single NuScale Power Module and Associated Secondary Equipment



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General overview

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General overview / operating parameters

Reactor parameter	Value (NuScale)
Number of fuel assemblies	37
Uranium mass	Approximately 8800 kg
Core active height	78.74 in (200 cm)
Core thermal power	160 MW ~4.3 MW/assembly ~2.15 MW/assembly/metre
Average fuel power density	~18 kW/kgU
System pressure	1850 psia (12.76 MPa)
Inlet temperature	497 °F (531 K)
Best estimate (BE) total flowrate	4.66e6 lb/h (587 kg/s) (7.3 % in bypass based on BE) ~15 kg/assembly/second
Core average coolant velocity	2.7 ft/s (0.8 m/s)

Chapter 4 Reactor – Table 4.1-1 NuScale Reactor Design Parameters

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

General overview / operating parameters – contrast

https://github.com/mit-crpg/BEAVRS/

Reactor parameter	Value (NuScale)	Value (BEAVRS)
Number of fuel assemblies	37	193
Uranium mass	Approximately 8800 kg	81800 kg
Core active height	78.74 in (200 cm)	365.76 cm
Core thermal power	160 MW ~4.3 MW/assembly ~2.15 MW/assembly/metre	3411 MW ~17.7 MW / assembly ~4.8 MW/assembly/metre
Average fuel power density	~18 kW/kgU	~42 kW/kgU
System pressure	1850 psia (12.76 MPa)	2250 psia (15.51 Mpa)
Inlet temperature	497 °F (531 K)	560 °F (566 K)
Best estimate (BE) total flowrate	4.66e6 lb/h (587 kg/s) (7.3 % in bypass based on BE) ~15 kg/assembly/second	61.5e6 kg/h (17 000 kg/s) (5 % in bypass) ~89 kg/assembly/second
Core average coolant velocity	2.7 ft/s (0.8 m/s)	~4.4 m/s in inlet

Chapter 4 Reactor – Table 4.1-1 NuScale Reactor Design Parameters

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Primary coolant loop

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General overview / operating parameters – contrast to BEAVRS

Unlike traditional PWRs the coolant flow in NuScale is driven by natural circulation.

Coolant flowrate (per assembly) around 1/6 of that in ref. PWR.

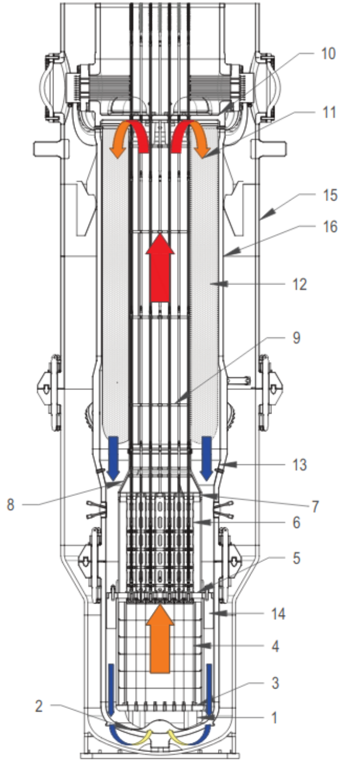
- Greater increase in coolant specific enthalpy. (increases driving forces of natural circulation)
- Slower coolant velocity (decreases flow resistance, pressure losses)

Note large height difference between core lower plenum (nr. 2 in figure) and upper riser outlet (nr. 11 in figure)

- Natural circulation driven by differences in coolant density.
  - Temperature & pressure.

Cool and dense water flows to core from below, absorbs heat from the core and gets less dense due to increase in temperature and due to decrease in hydrostatic pressure in riser. At the top of the riser, the flow turns and moves downwards along the steam generators getting denser due to cooling and hydrostatic pressure.

Figure 5.1-3: Reactor Coolant System Schematic Flow Diagram



No.	Stage
1	Core support blocks in downcomer
2	Downcomer to lower plenum turn
3	Lower core plate
4	Core
5	Upper core plate
6	Control rod assembly guide tubes
7	Control rod assembly guide tube support plate
8	Riser transition
9	Control rod drive shaft support
10	Pressurizer baffle
11	Upper riser turn to annulus
12	Downcomer through steam generator
13	Downcomer transition
14	Upper core support blocks
15	Containment vessel
16	Reactor vessel

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Steam generators

In the integrated PWR design, steam generators (SGs) are moved inside the pressure vessel.

Design challenge between pressure vessel (and containment vessel) size and heat transfer capability between primary and secondary circuits:

- Larger heat transfer capability can be achieved with larger steam generator surface area, but larger steam generators require a larger pressure vessel, which requires a larger containment vessel, which increases the size of the whole module.

NuScale uses unique helical coil steam generators, which consist of a large number of helically coiled SG tubes.

Figure 5.4-1: Steam Generator Helical Tube Bundle

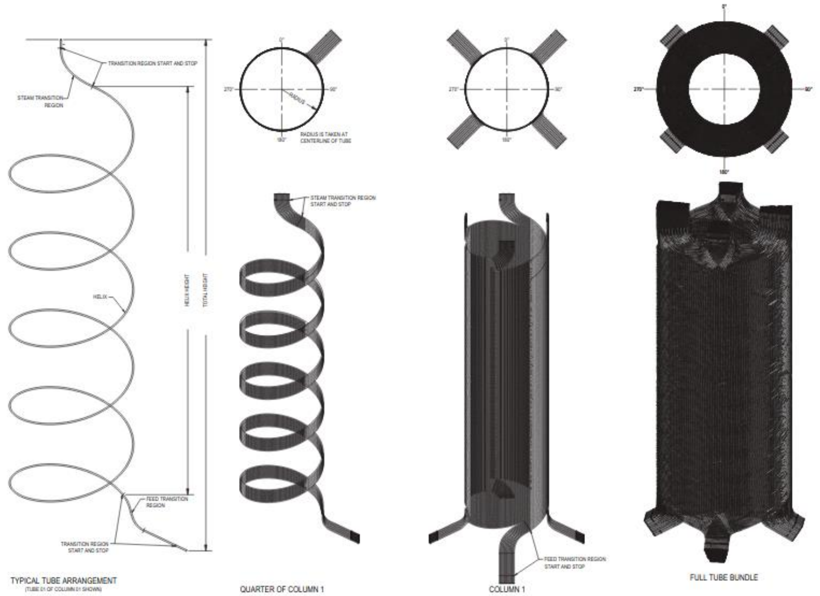

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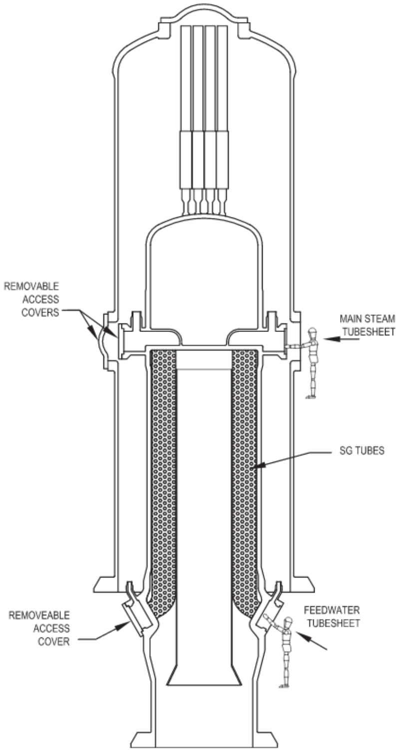
Figure 5.4-2: Configuration of Steam Generators in Upper Reactor Pressure Vessel Section



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Pressurizer

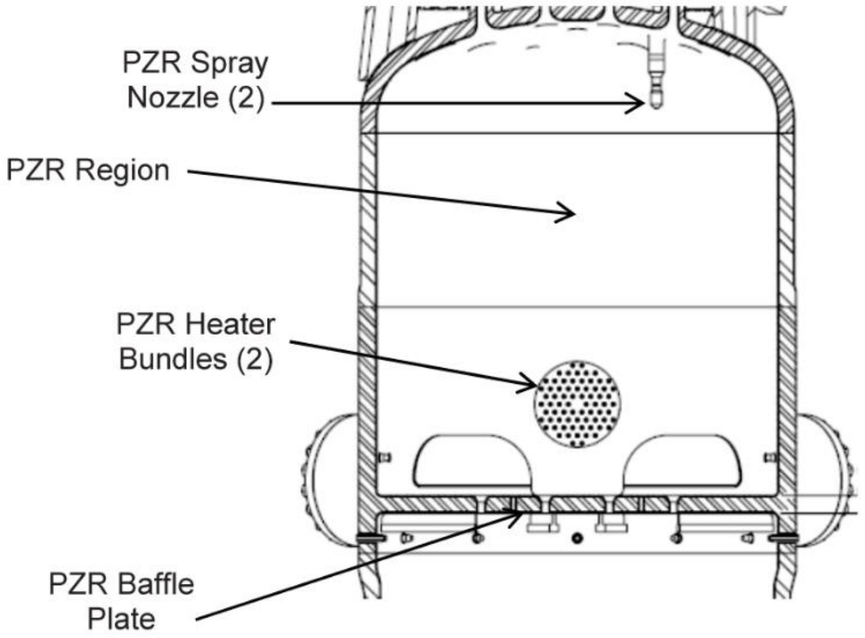
In the integrated PWR design, also the pressurizer is located inside the RPV, above the steam generators.

The pressurizer is located at the top of the RPV, separated from the hot side of reactor coolant system by the pressurizer baffle plate.

The baffle plate contains holes to allow pressure equalization and fluid flow between the pressurizer and the primary coolant loop.

In reactor startup, a low pressure is first introduced with nitrogen after which the pressurization to HZP conditions will be done via a steam bubble formed with the pressurizer heater bundles.

Figure 5.4-17: Pressurizer Region of Reactor Vessel



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Reactor core

Reactor core – overview

37 fuel assemblies

8.466 in assembly pitch (21.504 cm)

200 cm active height

Active core perimeter (cm) / Active core horizontal area (cm2) ≈ 0.035 1/cm

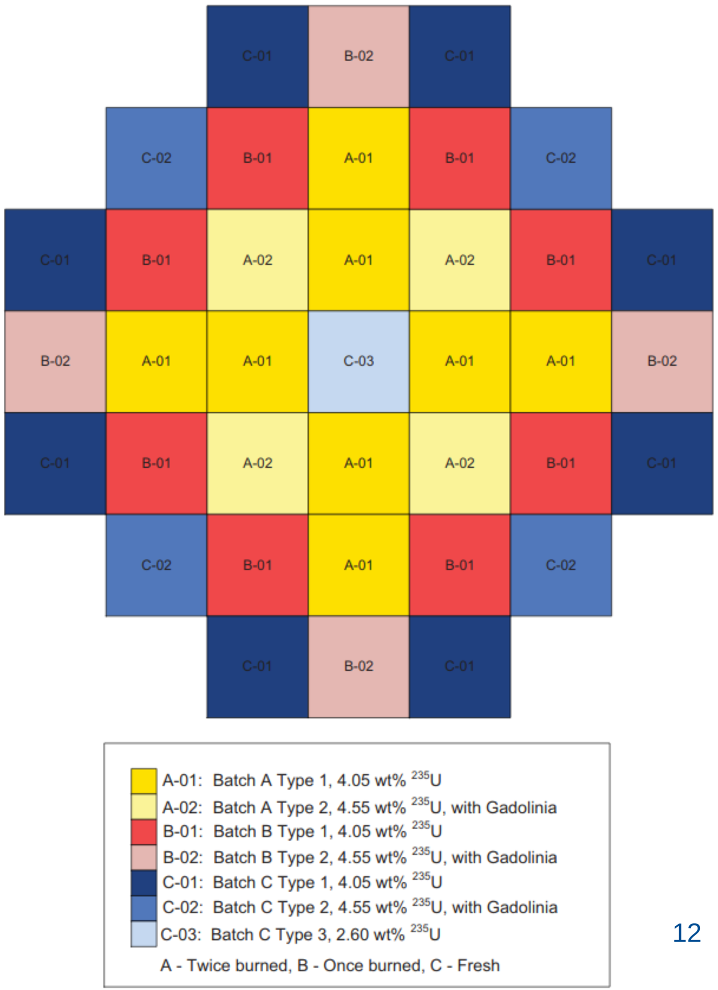
- Contrast to BEAVRS (≈ 0.014 1/cm)
- For perfect circle $\frac{2\pi r}{\pi r^2} = \frac{2}{r}$
- Will affect neutron economy.

Core surrounded radially by a stainless steel heavy reflector.

- Improved neutron balance
- Reduced neutron load on ex-core structures.

Equilibrium cycle based on three batch loading. Cycle length of two years and 12 MWd/kgU.

Figure 4.3-1: Loading Pattern for Reference Equilibrium Cycle



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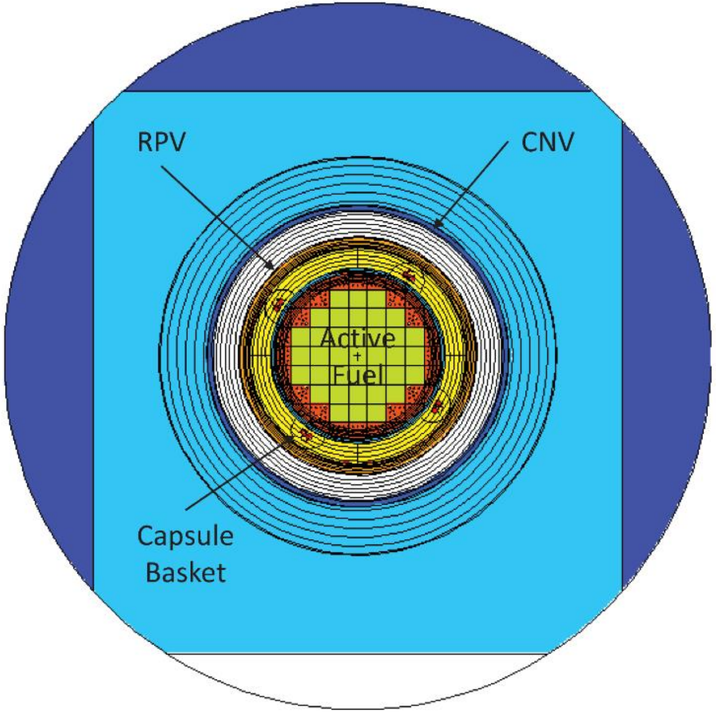
Equilibrium cycle based on three batch loading. Cycle length of two years and 12 MWd/kgU.

Figure 4.3-25: Cross-section view of MCNP6 Model for Vessel Irradiation Flux Calculation



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Reactor core – overview

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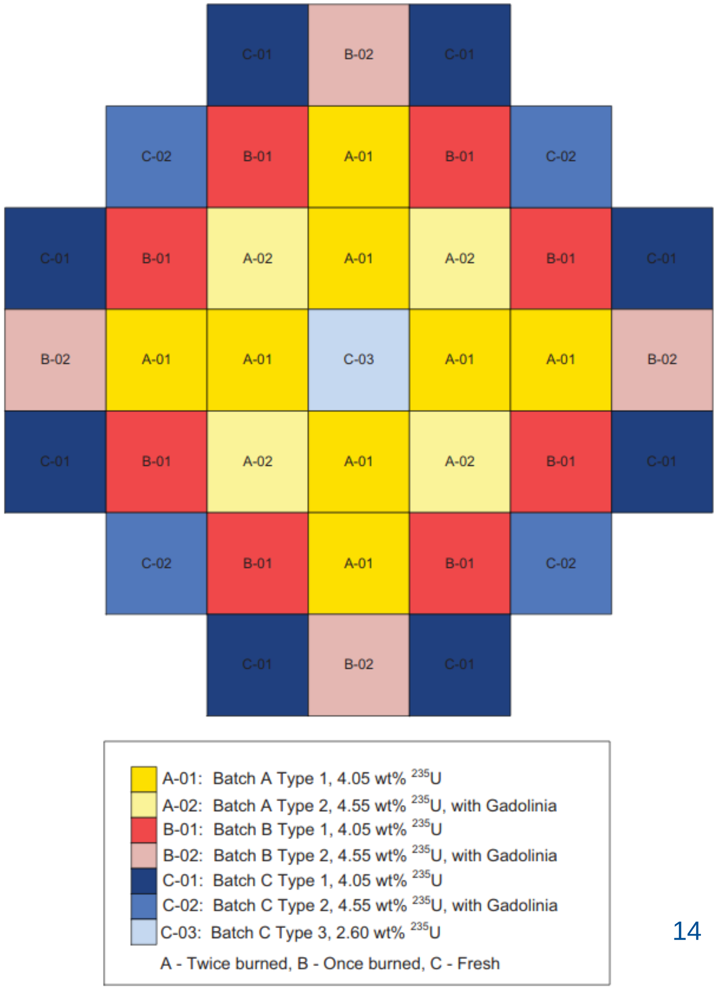

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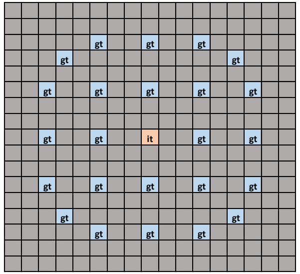
Fuel assemblies– overview

37 fuel assemblies

- 264 fuel rods in a 17x17 lattice
- 24 control rod assembly guide tubes
- 1 central instrumentation tube
- 5 spacer grids along the axial length

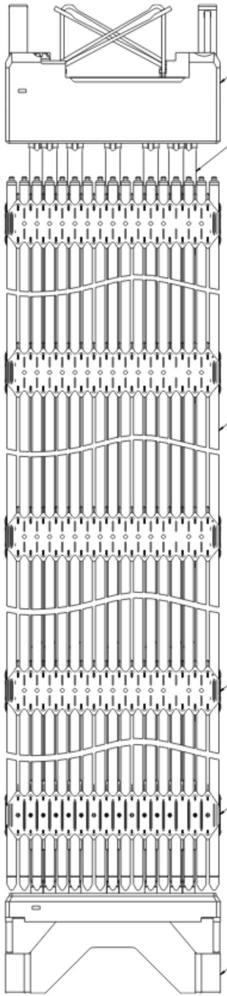


200 cm active length

- Includes axial blankets (lower uranium enrichment).
- May use gadolinia absorbers (Gd_2O_3 mixed with UO_2).

Superficially not too different compared to traditional PWR fuel assemblies except for reduced height and axial profiling.

Figure 4.2-1: Fuel Assembly General Arrangement



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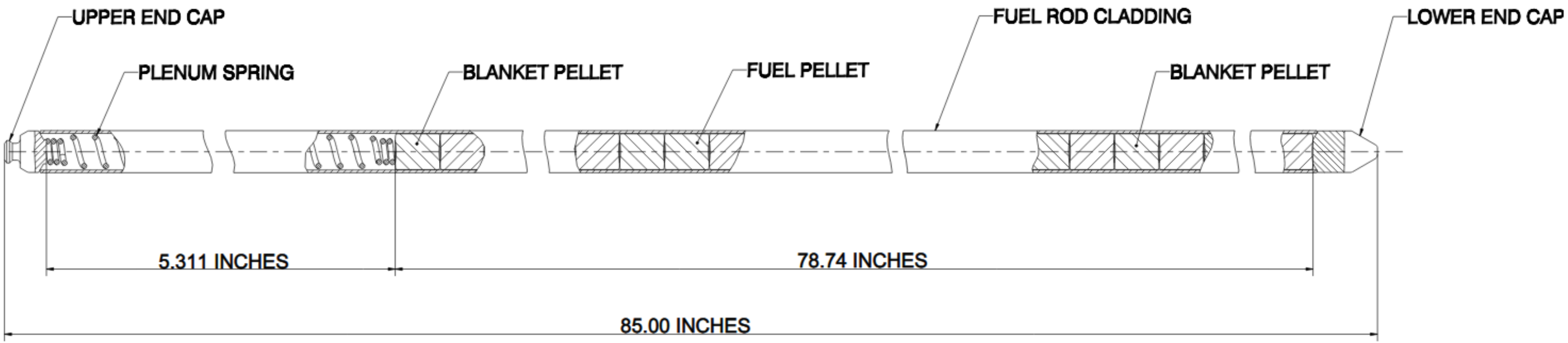
Fuel rod – overview

Again, superficially not too different from traditional PWR fuel.

UO_2 pellets (may have Gd_2O_3) at max. 4.95 % enrichment, axial blankets with reduced enrichment.

M5® cladding

Figure 4.2-10: Fuel Rod Assembly
(nominal dimensions)



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Reactivity and power control

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Reactivity control (general nuclear engineering)

Normal operation of nuclear reactor includes many changes in core reactivity:

- After reloading, core must have excess reactivity so that core is still critical at the end of the next operating cycle.
    - Excess reactivity must be countered. Burnable absorbers in fuel help.
- Moving from reloading to power operation means that:
    - Reactivity loss to heating of coolant (reactor design relies on negative feedback).
    - Reactivity loss to heating of fuel (reactor design relies on negative feedback).
    - Reactivity loss due to production of neutron absorbing fission products and actinides.
    - Reactivity loss due to removal of fissile content.
- The core total reactivity:

Initial excess reactivity – burnable absorbers – temperature changes – accumulation of absorbers – loss of fissile material

must be kept at zero for steady state operation through chosen reactivity control approaches.

- The NuScale concept relies on two approaches (during normal operation):
    - Soluble boron in coolant.
    - Regulating control rods (can also be used to shape core power distribution).

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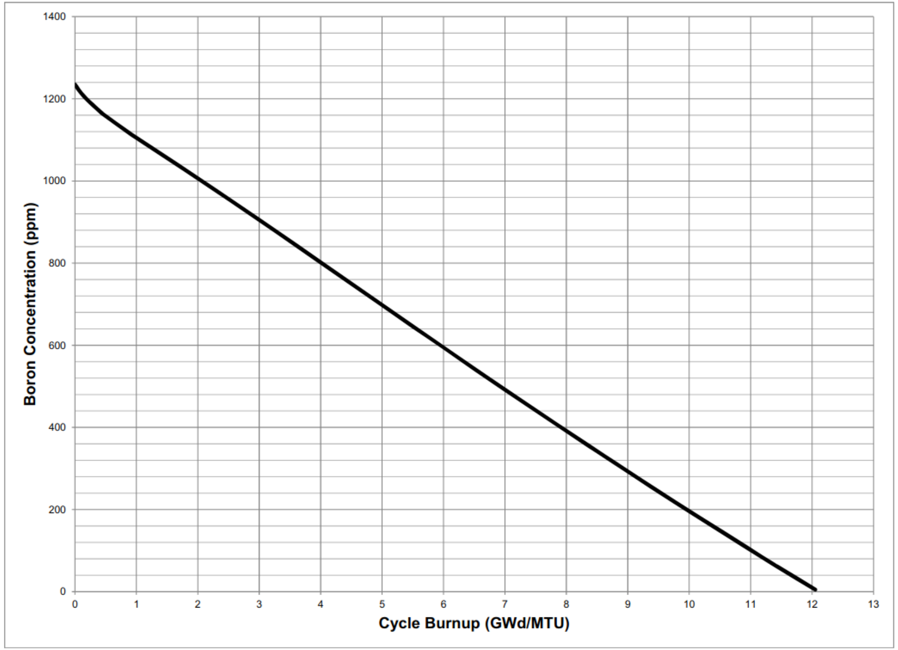
Reactivity control (soluble boron)

Boron concentration controlled by the chemical and volume control system (CVCS).

Soluble boron used to counter long term reactivity effects related to changes in fuel nuclide composition during operation.

- Decrease in fissile content.
- Increase in presence of neutron absorbing fission products and actinides.

Figure 4.3-17: Boron Letdown Curve for Equilibrium Cycle



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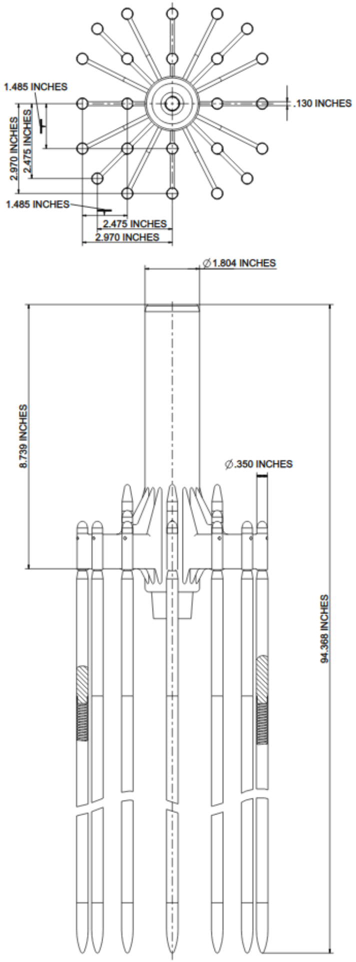
Control rods

Finger type control rods with 24 rods per control rod assembly (CRA).

Contains three axial absorber regions (from top to bottom):

- Boron carbide (B4C) part..
- Solid Ag-In-Cd (AIC) part.
- Annular Ag-In-Cd part.

Figure 4.2-11: Control Rod Assembly General Arrangement
(nominal dimensions)



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Control rods

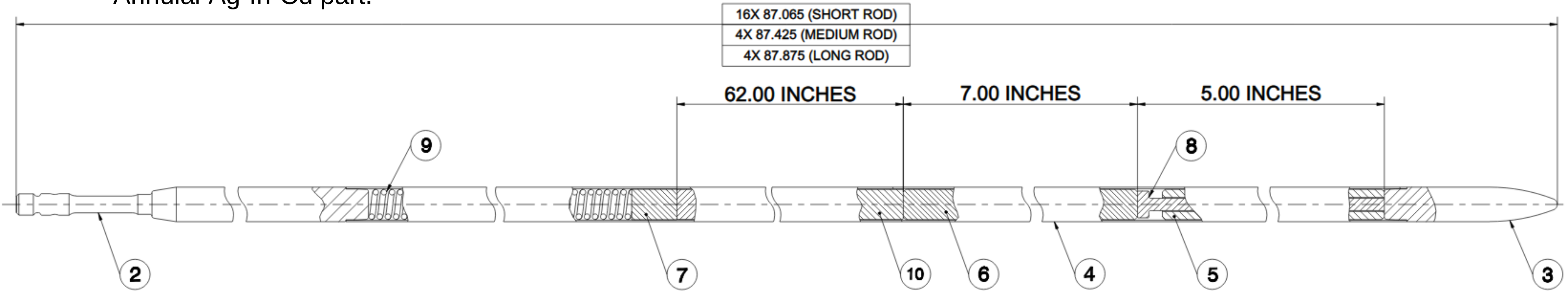
Figure 4.2-13: NuScale Control Rod Assembly Design
(nominal dimensions)

Contains three axial absorber regions

(from top to bottom):

- Boron carbide (B4C) part.
- Solid Ag-In-Cd (AIC) part.
- Annular Ag-In-Cd part.

ITEM	DESCRIPTION
1	CONTROL ROD
2	UPPER END PLUG
3	LOWER END PLUG
4	CLADDING
5	ANNULAR AIC ABSORBER
6	AIC ABSORBER ROD
7	SOLID SPACER
8	STACK SUPPORT
9	PLENUM SPRING
10	B4C PELLET



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Control rods

Finger type control rods with 24 rods per control rod assembly (CRA).

16 CRA in core.

Divided into four groups.

2 regulating groups (together the regulating bank).

2 shutdown groups (together the shutdown bank).

Shutdown banks used for normal reactor shutdown and reactor trips.

Regulating banks used for reactivity control during normal operation and to control axial power shape.

- Insertion of regulating banks limited by power dependent insertion limits.

Figure 4.3-18: Control Rod and Incore Instrument Locations

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Control rods

Finger type control rods with 24 rods per control rod assembly (CRA).

16 CRA in core.

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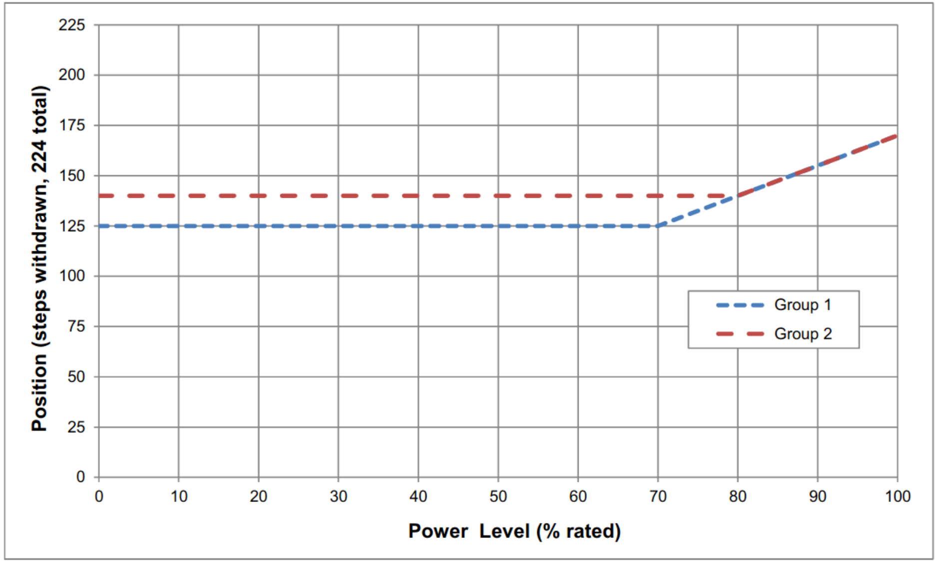
Shutdown banks used for normal reactor shutdown and reactor trips.

Regulating banks used for reactivity control during normal operation and to control axial power shape.

- Insertion of regulating banks limited by power dependent insertion limits.

Figure 4.3-2: Power Dependent Insertion Limits



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Questions

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Questions

How does the use of natural circulation for primary coolant flow show in the general operating parameters of the NuScale reactor?

What is NuScale's approach for providing a large heat transfer area between the primary and secondary circuits in the limited volume of the integrated PWR?

Small reactor cores have a larger surface area compared to their volume, which naturally increases neutron leakage and makes the neutron economy of the core worse. How is this alleviated in the NuScale design?

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Acknowledgments

We want to thank NuScale Power for granting us permission to use the figures in this presentation. Any errors that may be included in this presentation are the responsibility of the authors.

Safety Systems of NuScale Reactor

Course on “SMR LWR technologies”

Marek Benčík Ph.D. / ÚJV a.s.

26.1.2021

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Content

– Introduction

– Defence in Depth

– Safety Systems

– Safety Systems of NuScale Reactor

– Emergency Core Cooling System (ECCS)

– Decay Heat Removal System (DHRS)

– Conclusion

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Introduction

Topical safety objectives and principles (in IAEA approach, Ref.1)

- Fundamental safety objective - protect people and the environment from harmful effects of ionizing radiation
- Fundamental safety principles
    - Principle 1: Responsibility for safety
    - Principle 2: Role of government
    - Principle 3: Leadership and management for safety
    - Principle 4: Justification of facilities and activities
    - Principle 5: Optimization of protection
    - Principle 6: Limitation of risks to individuals
    - Principle 7: Protection of present and future generations
    - Principle 8: Prevention of accidents
    - Principle 9: Emergency preparedness and response
    - Principle 10: Protective actions to reduce existing or unregulated radiation risks

Ref. 1: IAEA, Fundamental Safety Principles, SF-1 (2006)

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Introduction

Technical safety concepts:

- Defence-in-depth (DiD) concept
- Multi-barrier confinement of radioactive inventory (as separate concept of part of DiD)
- Critical safety functions (reactivity, heat removal, integrity of barriers)
- Protection against external and internal hazards
- Radiological safety objectives (ALARA + limits for each level)
- Further design principles like redundancy, diversity, independence, physical separation, conservative design, single failure criterion, design margins etc.

Organizational safety concepts:

- Prime responsibility of the operator
- Oversight by independent regulator (role of government)
- Integrated management system (IMS) including quality control
- Safety culture, graded approach etc.

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Defense in depth

Defence in depth (DiD) is a design philosophy focused on ensuring security that does not depend on any single feature. It is a general defence strategy used in many fields. DiD as applied to NPP safety is an overall safety philosophy that encompasses all safety activities, including the siting, design, manufacture, construction, commissioning, operation and decommissioning of nuclear power plants.

DiD employs successive levels of redundant and diverse safety functions in design, construction and operation to provide appropriate barriers, controls and personnel to prevent, contain and mitigate accidents and exposure to radioactive substances.

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Defence in depth

Level of defence	Objective	Essential design means	Essential operational means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction of normal operation systems, including monitoring and control systems	Operational rules and normal operating procedures
Level 2	Control of abnormal operation and detection of failures	Limitation and protection systems and other surveillance features	Abnormal operating procedures / emergency operating procedures
Level 3 — 3a	Control of design basis accidents	Engineered safety features (safety systems)	Emergency operating procedures
Level 3 — 3b	Control of design extension conditions to prevent core melt	Safety features for design extension conditions without core melt	Emergency operating procedures
Level 4	Control of design extension conditions to mitigate the consequences of severe accidents	Safety features for design extension conditions with core melt. Technical Support Centre	Complementary emergency operating procedures / severe accident management guidelines
Level 5	Mitigation of radiological consequences of significant release of radioactive materials	On-site and off-site emergency response facilities	On-site and off-site emergency plans

Ref. 2: IAEA, Considerations for the application of the IAEA Safety Requirements on Design, TECDOC-1791 (2016)

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Safety systems

Safety systems of any nuclear power plant are designed to prevent accidents and limit their impact if they occur. These systems must provide three fundamental safety functions (Ref.3):

- to control reactivity
- to cool the fuel
- to contain radioactive substances

Active safety system - they involve electrical or mechanical operation on command.

Passive safety system - depends only on physical phenomena such as convection or gravity, not on functioning of engineered components.

Most of the currently operated nuclear power plants use a combination of passive and active safety systems.

Ref. 3: IAEA Safety Standards, No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design, 2016

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Safety systems of NuScale reactor

Safety Systems and Components Required to Protect the Reactor Core - NuScale Comparison with Typical PWR

Safety System or Component	Typical PWR/VVER	NuScale
Reactor Pressure Vessel	YES	YES
Containment	YES	YES
Reactor Coolant System	YES	YES
Decay Heat Removal System	YES	YES
Emergency Core Cooling System	YES	YES
Control Rod Drive System	YES	YES
Containmnent Isolation System	YES	YES
Ultimate Heat Sink	YES	YES
Residual Heat Removal System	YES	NO
Safety Injection System	YES	NO
Refueling Water Storage Tank	YES	NO
Condensate Storage Tank	YES	NO
Auxiliary Feed Water System	YES	NO
Emergency Feed Water System	YES	NO
Hydrogen Recombiner or Ignition System	YES	NO
Containment Spray System	YES	NO
Reactor Coolant Pumps	YES	NO
Safety-Related Electrical Distribution System	YES	NO
Alternative Off-Site Power	YES	NO
Emergency Diesel Generator	YES	NO
Safety-Related Batery System	YES	NO

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Safety systems of NuScale reactor

A NuScale Power Module (NPM) includes the reactor vessel, pressurizer, steam generators, and containment in an integral package - simple design that eliminates reactor coolant pumps, most pipings and other components and systems found in conventional PWR.

Simplified design, inherent cooling capabilities, passive engineered safety features, lower core power density ⇒ <u>possibility of core damage significantly reduced</u>.

NuScale safety systems:

- **Emergency Core Cooling System (ECCS)**
- **Decay Heat Removal System (DHRS)**

NuScale design eliminated the need for DC power to actuate DHRS and ECCS and place the plant in a safe cool-down condition following an extreme event.

Only a handful of safety valves need to be opened/closed in the event of an accident to ensure actuation of the ECCS or DHRS.

No pumps or additional water are required to provide core cooling for an indefinite period of time.

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Emergency Core Cooling System (ECCS)

ECCS is a passive system that provides a sufficient supply of coolant to keep the core always covered and cooled.

ECCS provides core cooling during and after anticipated operational occurrences (AOOs) and postulated accidents including loss of coolant accidents (LOCA) when it cannot be cooled by other means.

In conjunction with the containment heat removal function the ECCS provides core decay heat removal in the event of a loss of coolant that exceeds makeup capability.

The ECCS provides also low temperature overpressure protection for reactor pressure vessel.

Ref. 4: NuScale Power-LLC, NuScale Standard Plant: Design Certification Application: CHAPTER 6: Engineered Safety Features, Rev. 5, July 2020

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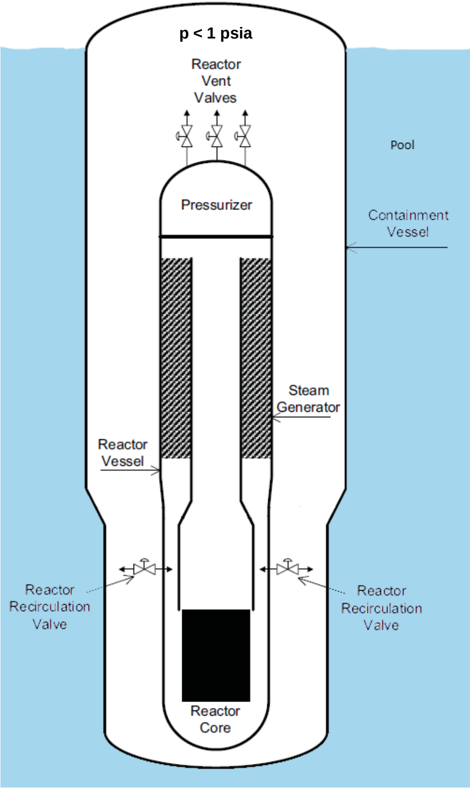
Emergency Core Cooling System (ECCS)



The ECCS is passive system that allows recirculation of the reactor coolant between the reactor pressure vessel and the containment vessel.

The ECCS consists of:

- Three reactor vent valves (RVVs) on the upper head of the reactor pressure vessel
- Two reactor recirculation valves (RRVs) on the side of the reactor pressure vessel

The ECCS can provide adequate core cooling with two RVVs and one RRV in the open position.

The ECCS valves are designed to actuate by stored energy and have no reliance on power or nonsafety-related support system.

Figure modified from NuScale DCA. Chapter 6. Figure 6.3-2

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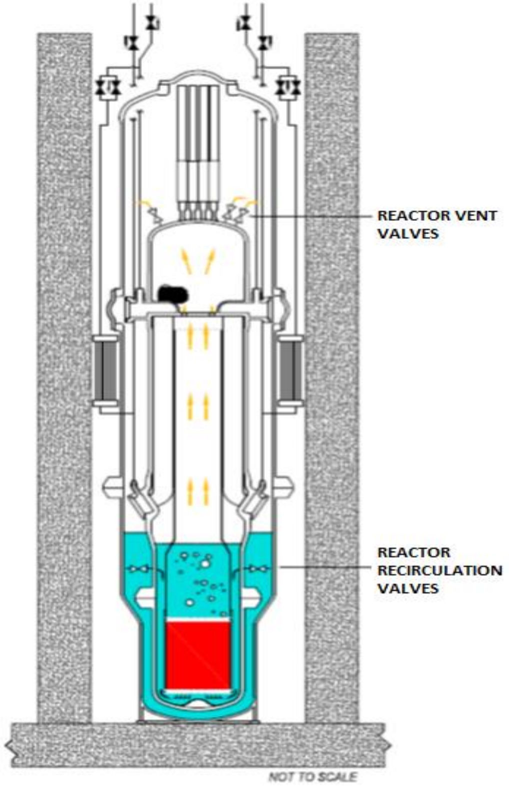
Emergency Core Cooling System (ECCS)



The ECCS valves are closed during normal plant operation.

Actuation signal upon predetermined event conditions:

- High containment level
- Low RCS pressure
- Low voltage AC electrical distribution system (24-hour timer)

Depressurization of reactor pressure vessel allows flow of reactor coolant between reactor pressure vessel and containment vessel.

In events that results in rapid equalization of pressure between the reactor coolant system (RCS) and containment vessel (CNV) such as inadvertent RVV opening, ECCS valves can open on low differential pressure without ECCS actuation signals.

Coolant released during LOCA is collected within CNV with precludes the requirement to provide the makeup capacity to replace lost coolant inventory.

Figure modified from NuScale DCA. Chapter 1. Figure 1.2-9

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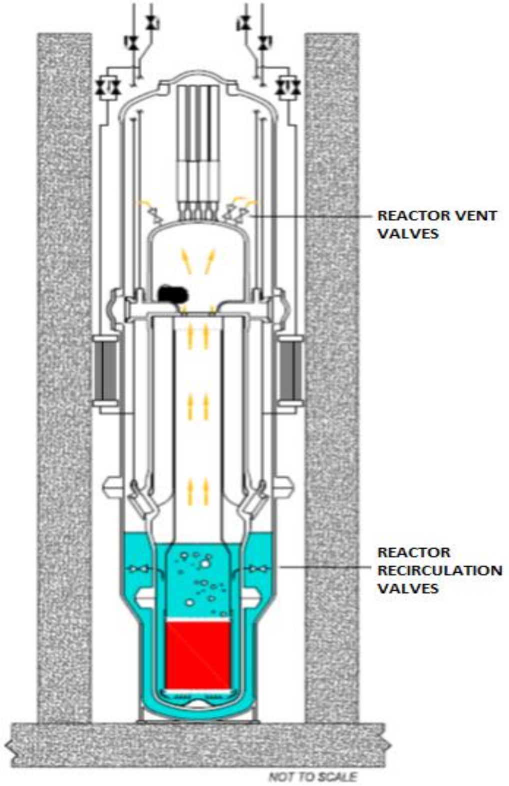
Emergency Core Cooling System (ECCS)



Coolant inventory released from the reactor vessel is collected and maintained within the CNV.

After the ECCS valves open, the collected coolant is returned to the reactor vessel by natural circulation.

The ECCS passively transfers heat from RCS to the reactor pool though CNV wall.

The ECCS ensures that the core always remains covered during ECCS operation (water level in RPV and CNV) stabilize above the reactor core.

The ECCS provides core cooling at a rate such that cladding water reactions are limited to negligible amounts, and fuel and cladding damage is prevented.

The ECCS is designed such that no single failure event prevents the ECCS from performing its safety functions.

Figure modified from NuScale DCA. Chapter 1. Figure 1.2-9

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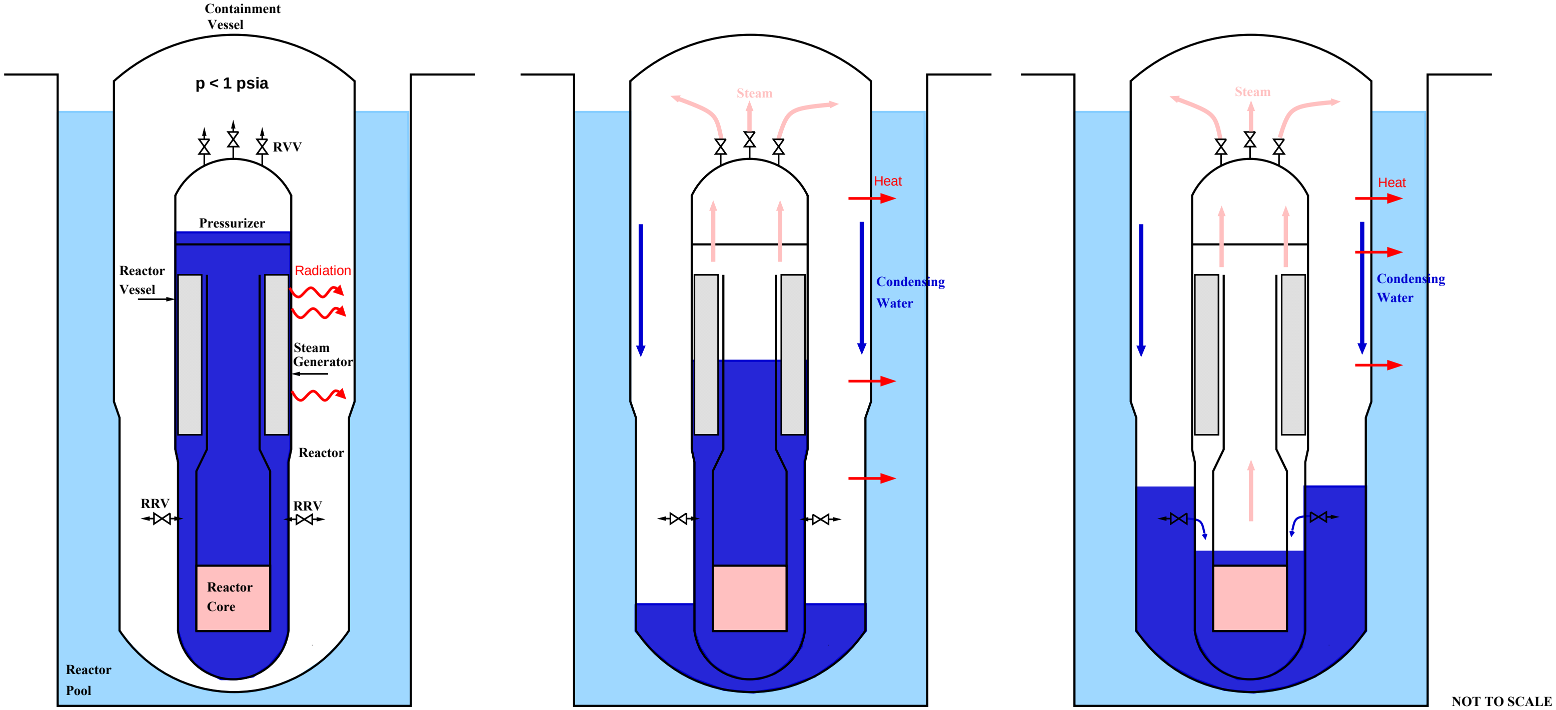
Emergency Core Cooling System (ECCS)



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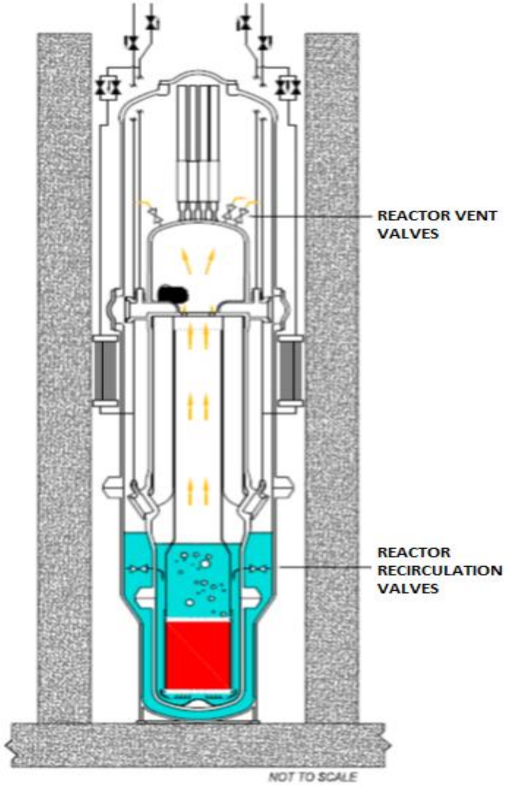
Emergency Core Cooling System (ECCS)



Independent experts found a potential flaw in ECCS scheme.

ECCS system does not provide reactivity control function (without the addition of boron).

Steam does not transport boron, so the element will be missing from the water condensing in containment. When the boron-poor water reenters the core, it could conceivably revive the chain reaction.

NuScale modified its design to ensure that more boron would spread to the returning water.

Through a conservative analytical approach has been demonstrated that the design modifications maintain the boron concentration in the downcomer above the critical boron concentration level necessary to prevent recriticality and a return to power.

Figure modified from NuScale DCA. Chapter 1. Figure 1.2-9

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Decay Heat Removal System (DHRS)

DHRS provides cooling for non-LOCA design basis events when normal secondary-side cooling is not available.

The system remove residual and core decay heat and provide transition to safe shutdown conditions without relying on external power.

The DHRS heat removal function is independent on ECCS (both systems can operate at the same time).

The DHRS is connected to the secondary system without direct connection to reactor cooling pressure boundary.

Ref. 5: NuScale Power-LLC, NuScale Standard Plant: Design Certification Application: CHAPTER 5: Reactor Coolant System and Connection systems, Rev. 5, July 2020

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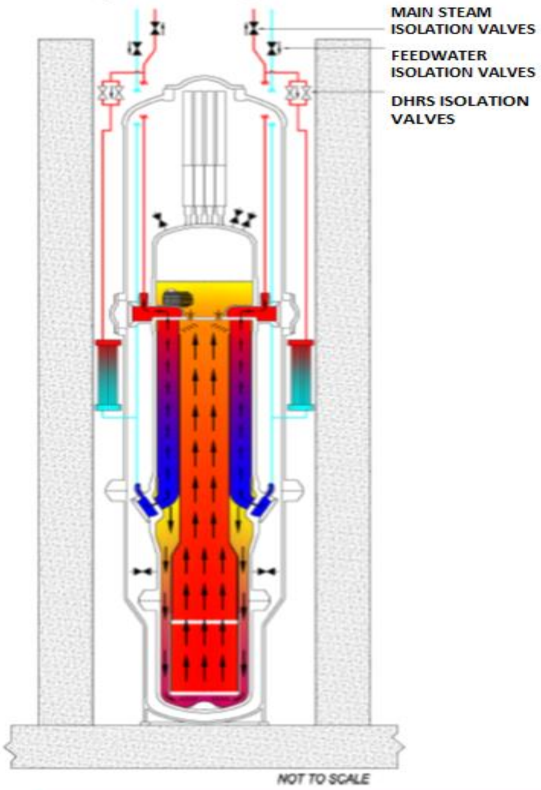
Decay Heat Removal System (DHRS)



DHRS is a passive closed system that uses natural circulation from SG to dissipate residual heat from the core to the reactor pool.

System consist of two independent trains connected to the main steam and feedwater lines with condensers located on the outer wall of the containment vessel.

During normal power operations, the DHRS is in a standby configuration - isolated from the main steam lines.

DHRS isolation valves (two parallel in each line) are designed to open upon interruption of control power (due to control system actuation or loss of power).

Each DHRS train is capable of performing the system safety function in the event of a single failure.

Figure modified from NuScale DCA. Chapter 1. Figure 1.2-7

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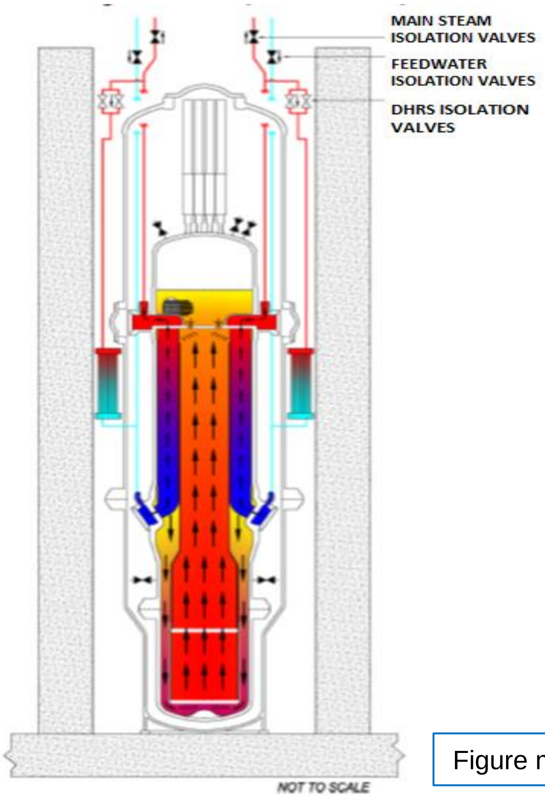
Decay Heat Removal System (DHRS)



Upon actuation, the main steam isolation valves (MSIVs) and feedwater isolation valves (FWIVs) are closed and the DHRS isolation valves are open.

DHRS is activated in case:

- High pressurizer pressure
- High RCS temperature
- High main steam pressure
- Loss of power

The DHRS function depends on the closure of MSIVs and FWIVs - in case of failure, the valves are backed up.

Accumulation of noncondensable gases in the DHRS can degrade DHRS performance.

DHRS can provide 3-10 days of decay heat removal.

Figure modified from NuScale DCA. Chapter 1. Figure 1.2-7

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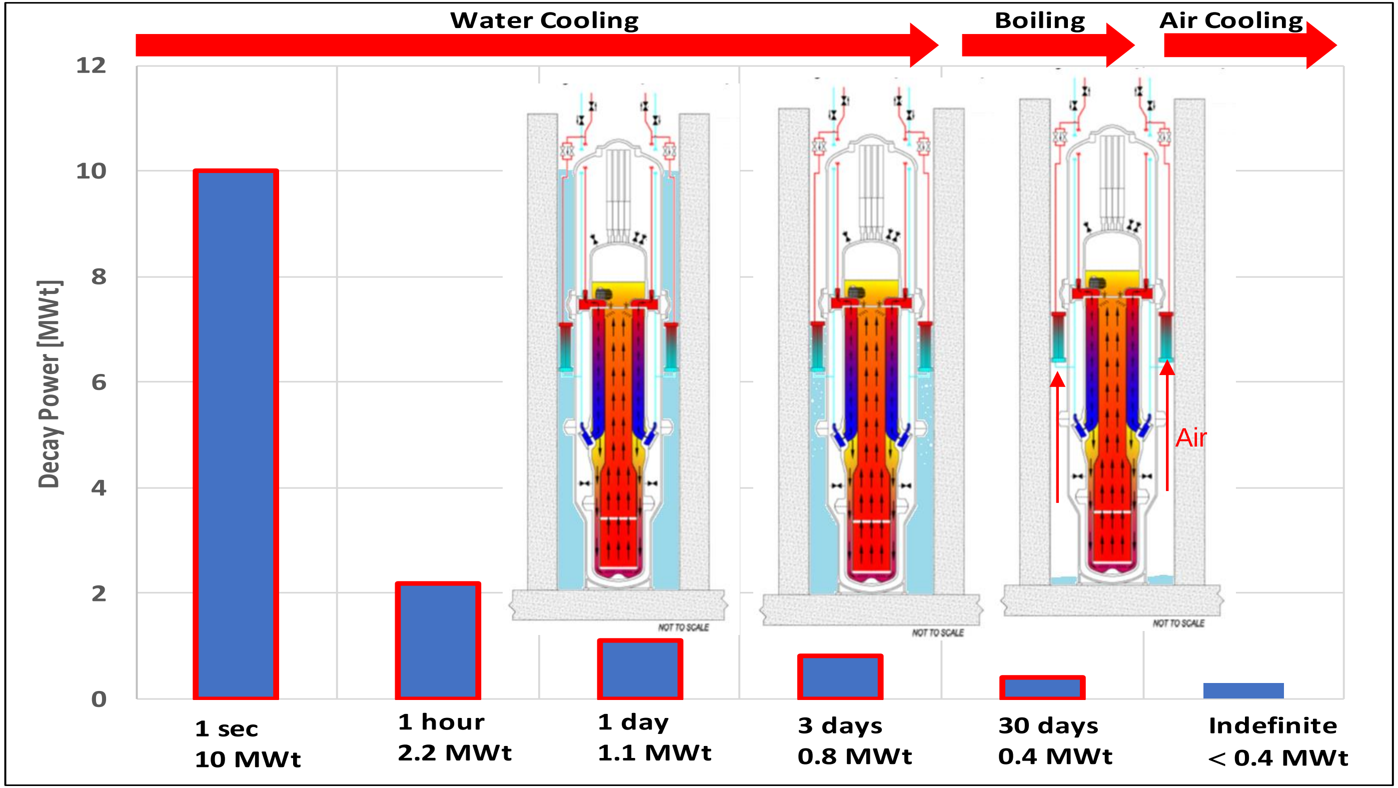
Decay Heat Removal System (DHRS)



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Conclusion

NuScale's design is based on well-established principles with a focus on integration of components and simplification.

Reliable, passive safety systems are simple in design and operation.

No additional water are required to provide core cooling for an indefinite period of time.

NuScale is the first ever small modular reactor (SMR) to receive U.S. Nuclear Regulatory Commission (NRC) design approval (in August 2020).

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Questions

1. Explain the boron dilution issue in NuScale design during Emergency Core Cooling System operation.

2. Describe the basic principles of NuScale’s Emergency Core Cooling System.

3. Describe the basic principles of NuScale’s Decay Heat Removal System.

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NuScale reactor. Safety Analysis

Course on “SMR LWR technologies”

Cesar QUERAL / UPM, Jorge SANCHEZ / UPM

26-01-2021

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Content

- NuScale Safety Analysis
- LOCA Sequences
- SLB Sequences
- Boron dilution sequences

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NuScale Safety Analysis

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NuScale. Deterministic Safety Analysis (DSA).

What do we need to understand the DSA?

- Mainly, Chapter 15: Selection of the bounding sequence for each initiator (e.g. LOOP/No-LOOP);Conservative Hypotheses; Single Failure Criteria; Acceptance criteria (Tables 15-0.2 to 15-0.5); Initial conditions and uncertainties (Table 15.0-6)
- Reactor description: Chapter 4
- RCS description: Chapter 5
- Engineered Safety Feature Actuation Systems (ESFAS) description (Chapter 6) and Setpoints (Tables 7.1-4, 6.3-1). The containment pressure and temperature response following mass and energy releases inside containment (e.g., LOCA, main steam or FWLBs) is analyzed in Section 6.2.1.
- Instrumentation: Chapter 7. Including: Reactor Trip Setpoints (Table 7.1-3); Actuation Delays (Table 7.1-6); Interlocks (Table 7.1-5).
- Electric Power: Chapter 8. Including SBO analysis
- Auxiliary Systems Description: Chapter 9
- Steam and Power Conversion System Description: Chapter 10
- Valves: Opening/closure times, setpoints, control systems, massflow rate

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NuScale. Deterministic Safety Analysis (DSA). Acceptance criteria

Classification	Fuel Clad	RCS Pressure	Main Steam System Pressure	Containment	Event Progression
AOO	Fuel cladding integrity shall be maintained by ensuring that Specified Acceptable Fuel Design Limits (SAFDLs) are met. SAFDLs are met by assuring that Minimal Critical Heat Flux Ratio (MCHFR) is maintained above the 95/95 limit (1.284, Table 15.1-15).	Peak pressure ≤ 110% of system design pressure	Peak pressure ≤ 110% of system design pressure	Peak pressure ≤ design pressure	An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant
IE		Peak pressure ≤ 120% of system design pressure	Peak pressure ≤ 120% of system design pressure		Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Postulated Accidents					Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Special Event (SBO)	Core cooling		N/A	N/A	N/A

From: NuScale DCA.. Table 15.0-2.

LOCA acceptance criteria (see Table 15.0-4 for more details):

- MCHFR > 1.29 and collapsed water level above the Top of Active Fuel (TAF).
- Long term cooling with core temperature maintained in an acceptable low value

See SRP Chapter 15 and NuScale DCA Tables 15.0-3 and 15.0-5 for other sequences criteria

Boron dilution criteria (see SRP section 15.4.6 for more details):

- If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost: A. During refueling: 30 minutes. B. During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.

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NuScale. Deterministic Safety Analysis (DSA)

- **Increase in Heat Removal by the Secondary System**
  1. Decrease in Feedwater Temperature
  2. SLB Inside CNV
  3. **SLB Outside CNV**
- **Decrease in Heat Removal by the Secondary System**
  1. FWLB Inside CNV
  2. FWLB Outside CNV
- **Reactivity and Power Distribution Anomalies**
  1. Uncontrolled Control Rod Assembly Withdrawal
  2. Control Rod Assembly misalignments
  3. Control Rod Assembly drop
  4. **Inadvertent Decrease in Boron Concentration in the RCS**
  5. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
  6. Spectrum of REA
- **Increase in Reactor Coolant Inventory**
  1. CVCS malfunction
- **Decrease in Reactor Coolant Inventory**
  1. Inadvertent Opening of a Reactor Safety Valve
  2. Failure of Small Lines Carrying Primary Coolant Outside Containment: CVCS line break SGTF
  3. **LOCA**
  4. Inadvertent operation of ECCS
- **Radioactive Release from a Subsystem or Component**
- **ATWS** (Not included due to ATWS contribution to CDF significantly below of 1E-5/year)
- **Stability**

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NuScale. Probabilistic Risk Assessment (PRA)

Level 1 Internal Probabilistic Risk Assessment Initiating Events, CDF and LRF

Initiator	Frequency (1/y·r)	CDF (%)	LRF(%)
CVCS Pipe Break Outside Containment - Charging Line	2.80E-04	6	93
CVCS Pipe Break Outside Containment - Letdown Line	1.40E-04		6
CVCS LOCA Inside Containment - Charging Line	1.40E-04	3	
RCS LOCA Inside Containment	2.00E-03	22	
Spurious Opening of an ECCS Valve	1.10E-05		
Steam Generator Tube Failure (SGTF)	4.50E-05		1
Secondary Side Line Break	4.40E-05		
Loss of Offsite Power (Loss of Normal AC Power)	3.10E-02	22	
Loss of DC Power	4.70E-05	16	
General reactor trip	1.30E+00	12	
Loss of support systems	1.60E-02	18	
CDF (1/y·r)	2.7E-10		
LRF (1/y·r)	1.8E-11		

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LOCA Sequences

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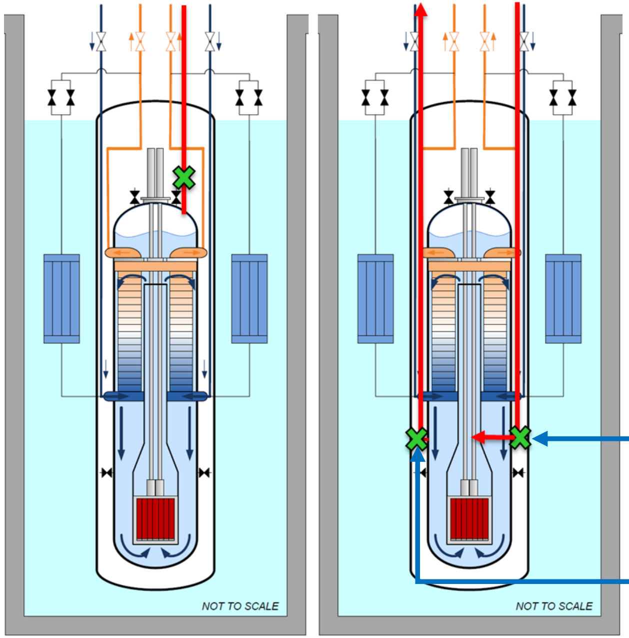
LOCA Sequences

steam space breaks liquid space breaks

Steam space breaks:

- high point vent line and
- PZR spray supply line

Generally, ECCS is activated on low RCS pressure



Liquid space breaks:

- CVCS injection line and
- discharge line

Generally, ECCS is activated on high CNV level

The **CVCS injection line** discharges inside the riser above the core.

The **CVCS discharge line** takes suction from a RPV penetration located just below the SG region.

Figure modified from: NUSCALE PLANT DESIGN OVERVIEW. RP-1114-9375. 2014. Figure 3-4

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LOCA Sequences. Hypotheses

Hypotheses (see Section 15.6.5.3.2 for more details):

- The Module Protection System (MPS) is credited to
  - initiate the reactor trip,
  - isolate the containment, and
  - initiate the DHRS, Secondary System Isolation (SSI) and ECCS.
- However, DHRS is not credited for cooling following a LOCA.
- The plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty.
- No operator action is credited in this event analysis.

Single failure: failure of one ECCS division (one Reactor Vent Valve and one Reactor Recirculation Valve) to open

Initial Conditions (see Section 15.6.5.3.2 for more details):

- P_0^{RCS} = (1850 – 70 bias) psia = 1780 psia = **122.7 bar**
- P_0^{CNV} = 3 psia = **0.21 bar**

Limiting scenario: an equivalent 5% cross-sectional area break of the **RCS injection line** with a loss of normal AC

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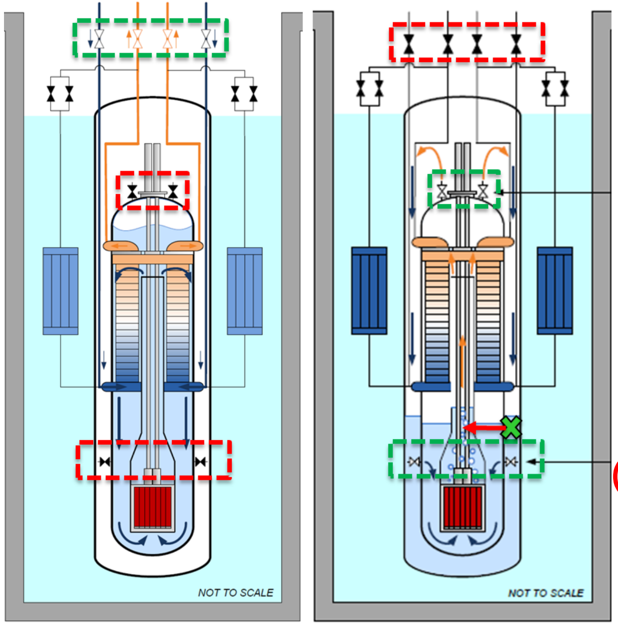
LOCA Sequences. ECCS

- The ECCS design does not require alternating current (AC) or direct current (DC) power to effectively cool the core (NuScale FSAR Section 6.3.1.).
- The ECCS setpoints are chosen to ensure automatic actuation of ECCS valves in response to design basis LOCA events or 24 hours after a loss of AC power (NuScale FSAR Section 6.3.1.):
  - **High CNV level**: 252 inches above reactor pool floor
  - Low RCS pressure: 800 psia
  - RPV low temperature & high pressure (LTOP)
- The ECCS valves also open when electric power is lost.
- These signals are bypassed when:
  - RCS temperature is below the T-3 interlock (RCS Hot Temperature < 350° F) **AND**
  - pressurizer level is above the L-2 interlock (PZR level> 20%)
- If the differential pressure across the valves is greater than the inadvertent actuation block (IAB) threshold when the ECCS signal actuates, then the valves stay closed until the differential pressure decreases to below the IAB release pressure.
- In events that result in rapid equalization of pressure between the RCS and the CNV, such as an inadvertent RVV opening, the ECCS valves can open on low differential pressure without an ECCS actuation signal.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

LOCA Sequences. ECCS

Normal Operation **LOCA Operation** (DHRS is not credited)



If the differential pressure across the valves is > 900 psid (62.1 bar) when the ECCS signal actuates, then the valves stay closed until the pressure differential decreases to below the IAB release pressure.

RVV (3 valves)

Reactor vent valves (**RVV**) open on safety signal

Reactor Recirculation Valves (**RRV**) open when the containment liquid level rises above the top of the recirculation valves.
The RRV penetrations are located six feet above the top of the reactor core (DCA Section 6.3).

RRV (2 valves)

Decay heat removed by means of:

- condensing steam on inside surface of CNV
- convection and conduction through liquid and both vessel walls

Nuscale Plant Design Overview. 2014. Figures 3-4 and 4.2

https://www.youtube.com/watch?v=I7-GNwe3WEQ&feature=emb_logo&ab_channel=NuScalePower

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LOCA Sequences. Selection of the bounding sequence

The LOCA analysis is performed for a spectrum of break sizes and break locations to determine the location and size of the break that is limiting for maintaining the collapsed liquid level above the core.

Break Area (%)	Break Area (in^2)	Equivalent Diameter (in)	Min Collapsed Level above TAF (ft)			
			Discharge Line	Injection Line	High Point Vent	PZR Spray Line
100	2.23	1.69	7.9	5.4	9.7	-
75	1.67	1.46	7.7	5.3	9.7	-
50	1.12	1.19	7.2	3.8	9.7	-
35	0.78	1.00	6.8	3.7	9.7	9.7
20	0.45	0.75	5.9	3.4	9.7	-
10	0.22	0.53	4.4	**1.7**	8.6	-
5	0.11	0.38	4.0	**1.7**	6.3	-
2.2	0.05	0.25	4.4	3.1	5.8	-

NuScale DCA. Tables 15.6-18 to 15.6-22: Loss-of-Coolant Analysis - Summary of Break Spectrum and minimum level above TAF

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LOCA Sequences. Selection of the bounding sequence

- From these results, the **5% cross-sectional area break of the RCS injection line** has the minimum collapsed level above the top of active fuel (TAF), but also the lowest MCHFR.

Break Size (%)	Time of RTS (s)	Time of ECCS Valves Opening (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level above TAF (ft)
100	7.6	908	1.81	892	5.4
75	7.9	980	1.81	893	5.3
50	8.8	1386	1.81	862	3.8
35	10.2	1629	1.79	847	3.7
20	13.4	2893	1.76	759	3.4
10	12.4	6859	1.75	541	**1.7**
5	11.9	13547	**1.74**	470	**1.7**
2.2	11.6	24325	1.74	404	3.1

NuScale DCA. Table 15.6-19: Loss-of-Coolant Analysis - Discharge Line Break Spectrum with Loss of AC Power

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LOCA Sequences. Selection of the bounding sequence

- Then, the 5% injection line break was then analyzed for **different power scenarios**

Case	Time of RTS (s)	Time of ECCS Valve Opening (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level Above TAF (ft)
All power available	243	13359	1.82	443	1.8
Loss of Normal AC Power	11.9	13547	1.74	470	**1.7**
Loss of normal AC and DC power	2	13375	1.81	459	1.8

NuScale DCA. Table 15.6-23: Loss-of-Coolant Analysis - Five-percent Injection Line Break with Evaluation of Electric Power Available

- Therefore, the limiting power scenario is the **loss of normal AC power**

- Finally, this scenario is then analyzed for **different single failure scenarios**

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LOCA Sequences. Selection of the bounding sequence

Failure of one ECCS division

Scenario	Time of RTS (s)	Time of ECCS Valves Opening (s)	MCHFR	Peak CNV Pressure (psia)	Min Collapsed Level Above TAF (ft)
No failure	11.9	13547	1.74	470	1.7
Failure of one RVV to open	11.9	13547	1.74	456	1.6
Failure of one RRV to open	11.9	13547	1.74	484	1.6
Failure of one RVV and RRV to open	11.9	13547	**1.74**	461	**1.5**

NuScale DCA. Table 15.6-24: LOCA Analysis – 5% Injection Line Break with Loss of Normal AC Power Evaluation of Single Failure

Thus, the limiting scenario begins with an equivalent 5 % cross-sectional area break of the RCS injection line with a loss of normal AC and single failure of one Reactor Recirculation Valve and one Reactor Vent Valve (one ECCS division)

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LOCA Sequences. 5% Injection Line Break

Event	Time (s)
Line break and a coincident Loss of normal AC (hypothesis). *A loss of normal AC power stops FW flow →* ***PZR pressure*** ↑	0
High PZR pressure (2000 psia= 137.9 bar): **SCRAM setpoint, isolation of the SGs and** DHRS actuation signal (**not credited**)	8
SCRAM signal (High PZR pressure + 2s delay) and isolation signal of the SGs (by closing the MSIVs and the feedwater isolation valves)	10
Control rods fully inserted (SCRAM signal + bounding CR drop-time)	12
High containment pressure (9.5 psia = 0.65 bar): **Containment and CVCS isolation setpoints**	16
Containment isolation signal (high containment pressure + 2s delay)	18
Containment isolation (containment isolation signal + 2s)	20
Low pressurizer level (35%): SCRAM setpoint and DWS isolation setpoint	1011
Low Low pressurizer level (20%): Containment, CVCS and SG isolation setpoints	1750
High CNV water level ECCS actuation limit (252 inches above reactor pool floor = 21 ft = 6.4 m): **ECCS signal.** *If the differential pressure across the valves is greater than the inadvertent actuation block (IAB) threshold when the ECCS signal actuates, then the valves stay closed until the differential pressure decreases to below the IAB release pressure*.	7202
ECCS actuates (RPV/CNV differential pressure below IAB threshold = 900 psid = 62.1 bar)	13547

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LOCA Sequences. 5% Injection Line Break

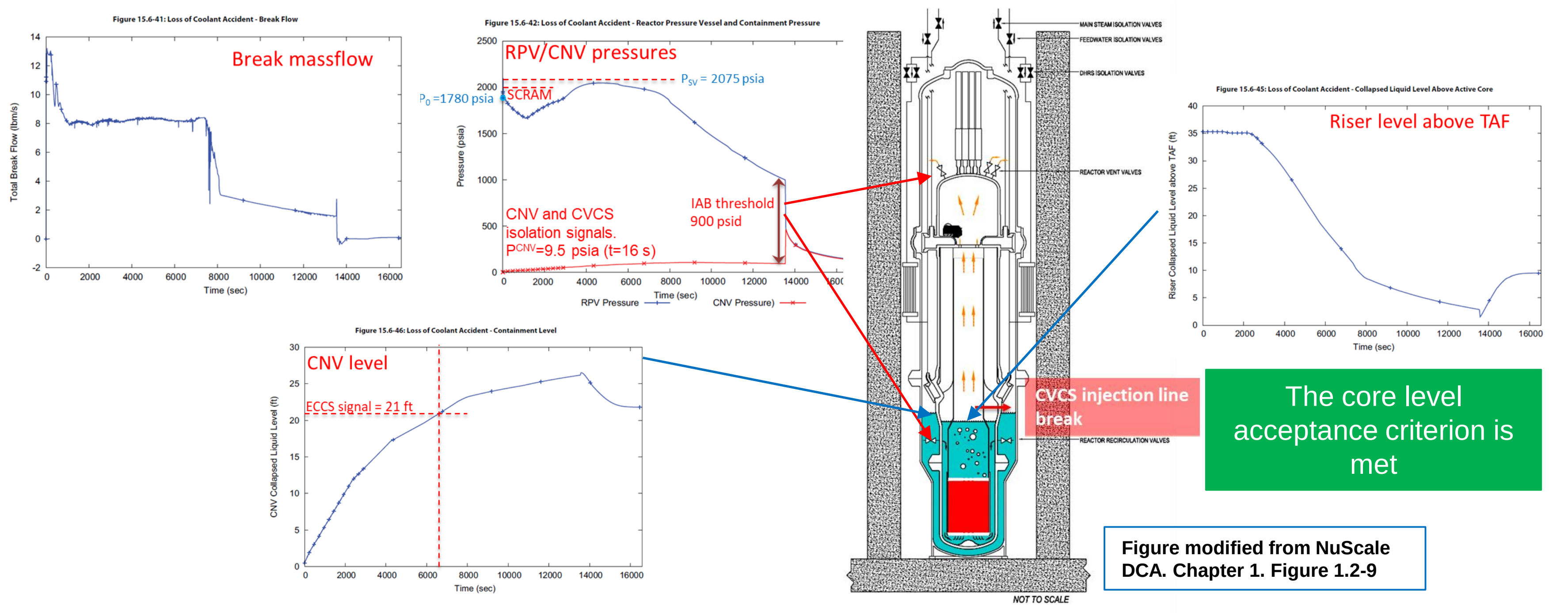

Figure modified from NuScale DCA. Chapter 1. Figure 1.2-9

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LOCA Sequences. 5% Injection Line Break

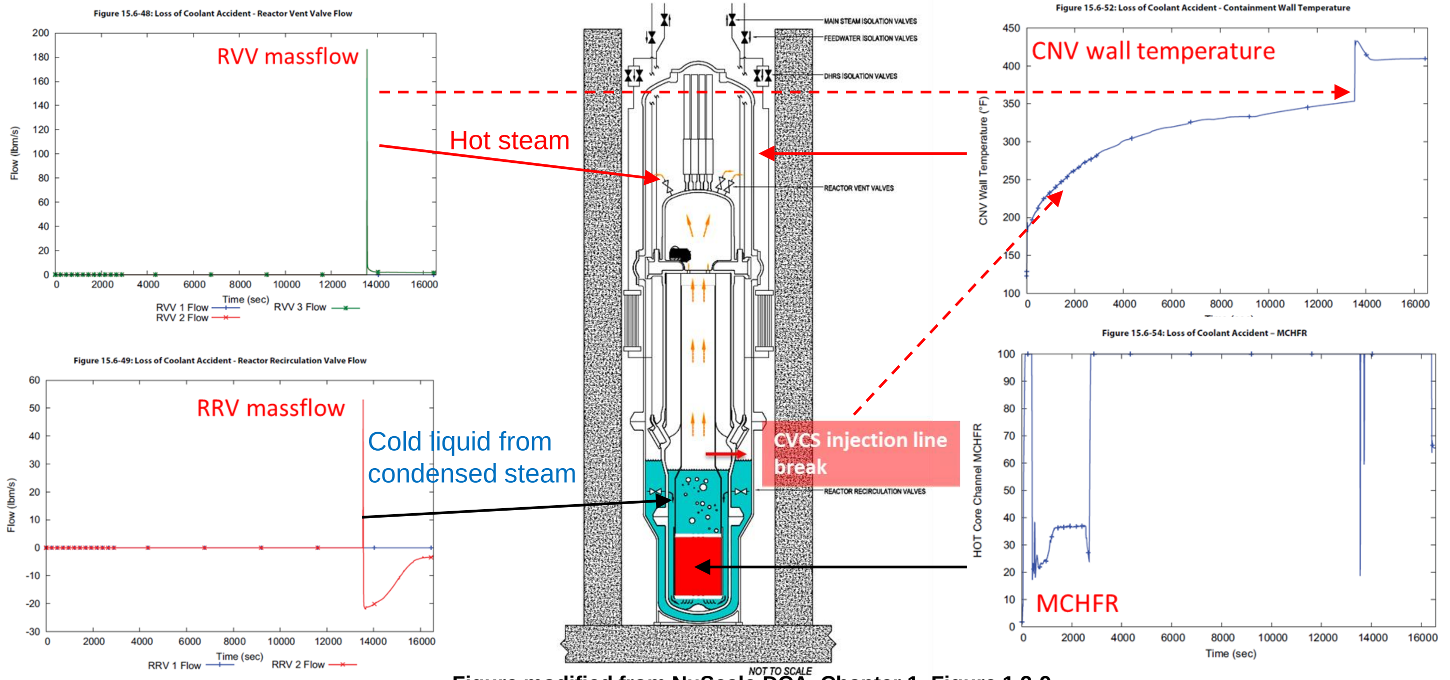

Figure modified from NuScale DCA. Chapter 1. Figure 1.2-9

CNV pressure and temperature reaches a maximum after the ECCS valves open and then decreases as thermal energy is transferred to the reactor pool through the containment wall.

The MCHFR acceptance criterion is met

LOCA acceptance criteria are met

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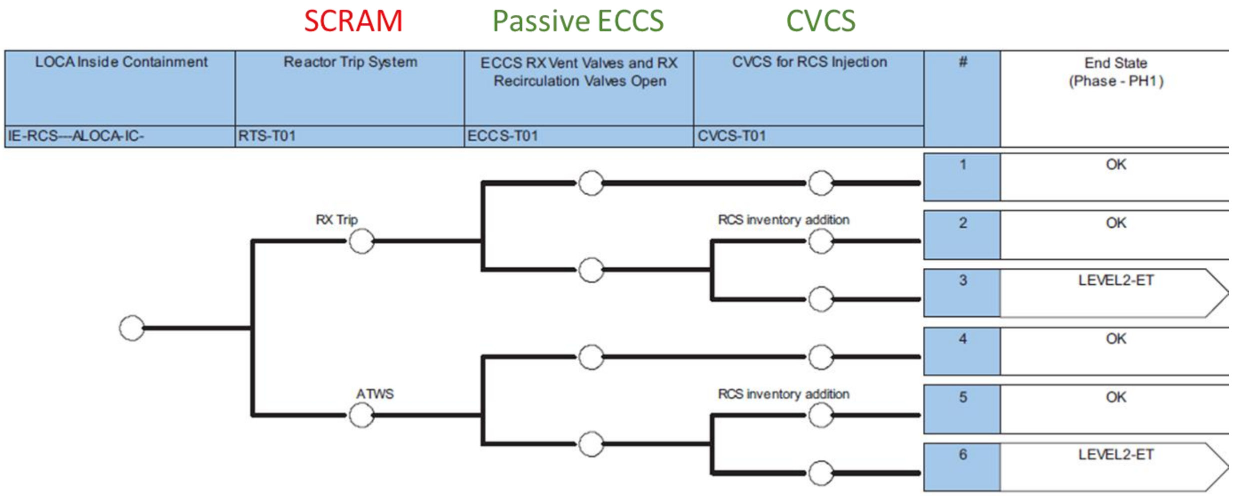
LOCA Sequences. PRA: Event Tree



NuScale FSAR. Rev5. Figure 19.1-5: Event Tree for Reactor Coolant System Loss-of-Coolant Accident Inside Containment

NuScale PRA also includes three specific CVCS breaks Event Trees: CVCS Charging Line Pipe Break Outside CNV; CVCS Letdown Line Pipe Break Outside CNV; CVCS Charging Line LOCA Inside CNV

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SLB Sequences

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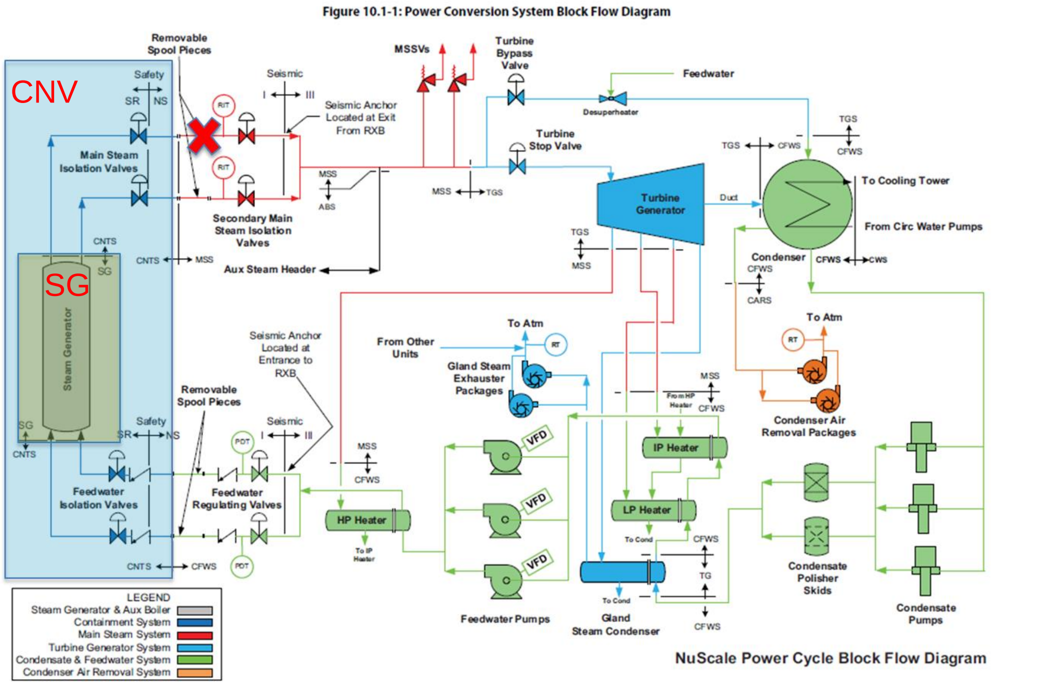
SLB Resulting in the Limiting MCHFR

There are separate SLB cases outside of containment that are limiting with respect to: primary pressure, SGS pressure, MCHFR, and radiological consequences

SLB: 2.5 % split break in the MS piping just outside containment between the primary and secondary MSIVs (102% power)



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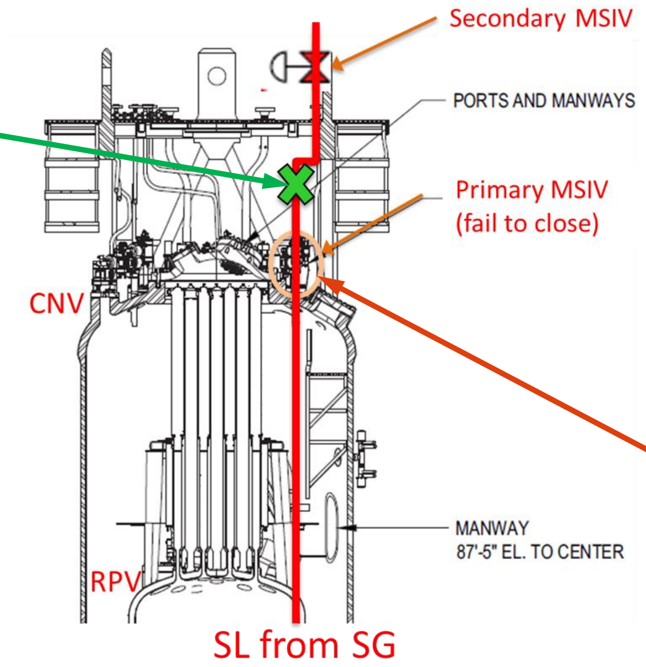
SLB Resulting in the Limiting MCHFR

SLB: 2.5 % split break in the MS piping just outside CNV between the primary and secondary MSIVs (102% power)



The limiting **single failure** is the failure of the primary MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation

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SLB Resulting in the Limiting MCHFR. Hypotheses.

Hypotheses (see Sections 15.1.5.3.2 and 15.1.5.3.3, Table 15.1-13 for more details):

- Initial power level is assumed to be 102 % RTP with BOC core exposure
- Normal AC power is assumed to be available.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- A 30 % uncertainty is added to the SG heat transfer to maximize the cooldown event.
- The FW pump flow rate is allowed to increase with the steam flow to maximize the overcooling effect.

Initial conditions:

- P_0^{RCS} = (1850 + 70 bias) psia = 1920 psia = 132.38 bar (Table 15.1-13)
- Initial RCS flow rate low (DCA Table 15.1-13): 1155 lbm/s = 523.90 kg/s
- SG pressure = 500+35 psia = 535 psia = 36.89 bar

Single failure: the failure of the primary MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation.

Limiting MCHFR scenario: 2.5 % split break in the MS pipe between the primary and secondary MSIVs.

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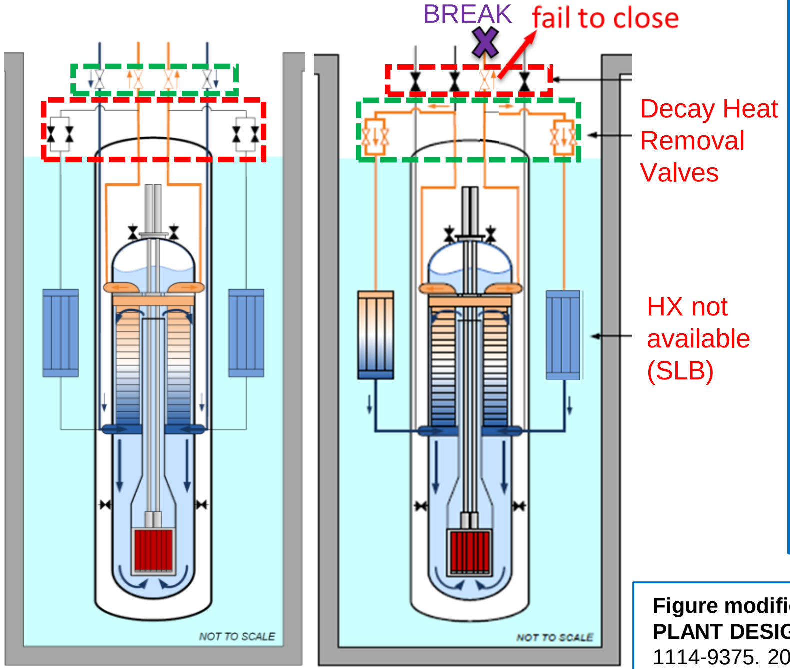
SLB Resulting in the Limiting MCHFR. DHRS.

Decay Heat Removal System actuation setpoints:

- High PZR Pressure > 2000 psia
- High RCS Hot Temp. > 610°F
- High Main Steam Pres > 800 psia
- Low AC Voltage to Battery Chargers < 80%

Initially, the DHRS is filled with water. DCA Chapter 5 (pp 5.4-18)

See also https://www.youtube.com/watch?v=I7-GNwe3WEQ&feature=emb_logo&ab_channel=NuScalePower



System Automated Function:

- Removes electrical power to the trip solenoids of the **decay heat removal valves**
- Removes electrical power to the trip solenoids of the following valves in the CNV, main steam, and FW systems:
  - MSIVs;
  - MSI bypass valves (MSIBV);
  - secondary MSIVs;
  - secondary MSIBVs;
  - FW isolation valves;
  - FW regulating valves

Figure modified from: NUSCALE PLANT DESIGN OVERVIEW. RP-1114-9375. 2014. Figures 3-4 and 4.2

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SLB Resulting in the Limiting MCHFR.

Event	Time [s]
SLB occurs (P_0^{RCS} = 1920 psia = 132.38 bar). RCS cooling → ***Positive reactivity feedback*** ↑ → **Power** ↑	0
MCHFR occurs (MCHFR= 1.866 > 1.284 SL-MCHFR, Table 15.1-15). The MCHFR acceptance criterion is met.	66
High reactor power limit is reached (120% RTP= 192 MW): **SCRAM setpoint**	67
Peak reactor power reached	69
Reactor Trip (120% RTP + 2s)	69
High pressurizer pressure limit is reached (2000 psia): *The overcooling effect initially lowers RCS pressure. However, as core power rises, pressure increases until the reactor trip.*	71
Control rods fully inserted (SCRAM signal + bounding CR drop-time)	72
Secondary System Isolation Signal (High pressurizer pressure + 2s): **FWRV, FWIVs and MSIVs close signal**	73
DHRS Actuation (High PZR pressure + 2s). *The intact SG pressure increases following SG isolation and DHRS actuation*	73
Peak RCS pressure reached (2040 psia=140,65 bar < 120% design pressure & Safety Valve setpoint (2075 psia= 143.1 bar))	75
Peak MSS pressure reached (not-available)	122

Modified from: NuScale DCA. Table 15.1-11

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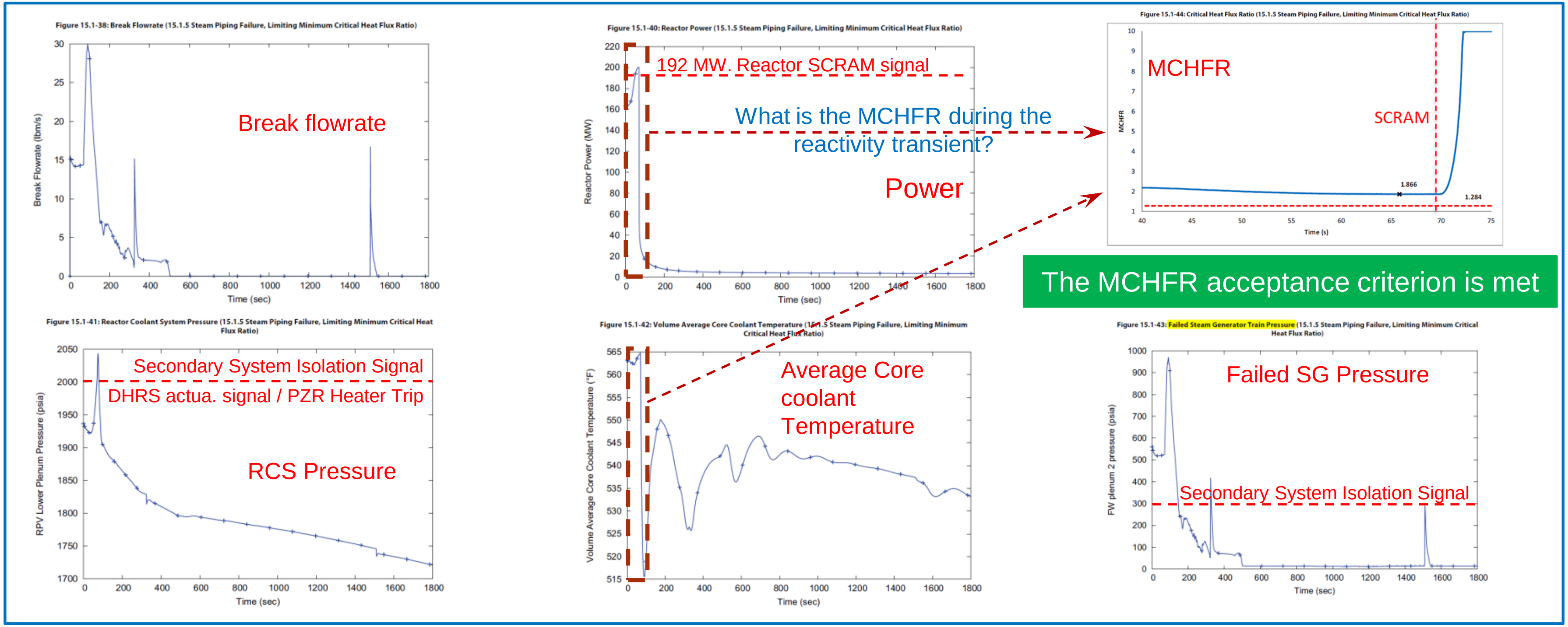
SLB Resulting in the Limiting MCHFR



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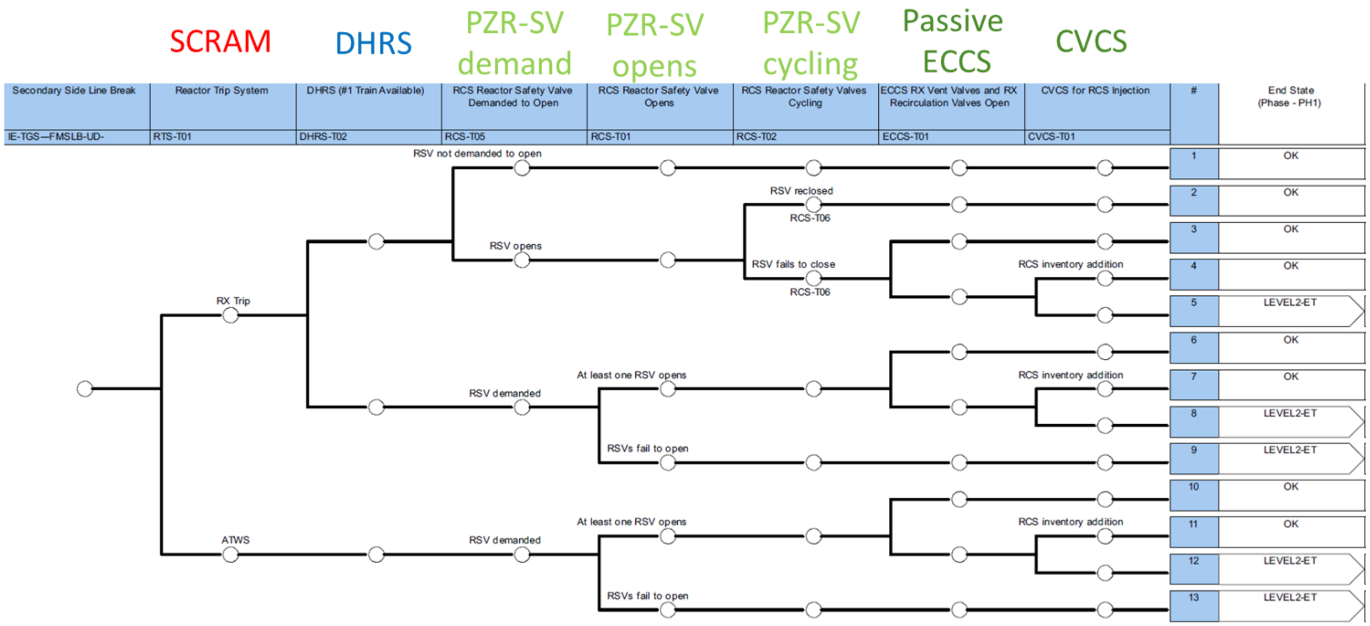
SLB Sequences. PSA: Event Tree



NuScale FSAR. Rev5. Figure 19.1-8: Event Tree for Secondary Line Break

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Boron Dilution Sequences

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Boron Dilution Sequences

- As a result of a malfunction in the CVCS or an operator error, a Boron dilution transient could take place within the RCS of the NPM.
- **The limiting CVCS dilution source is the demineralized water system (DWS) supply**.
- The **unborated water is injected to the RCS by means of the CVCS makeup pumps**.
- The Boron dilution is governed by controls that establish the limits on the rate and the duration of dilution.
- The **decrease in Boron concentration** within the RCS **increases the total reactivity** of the core and **reduces** significantly **the shutdown margin**.
- The Module Protection System automatically isolates the unborated water source on high subcritical multiplication, low RCS flow, and any reactor trip system actuation.
- The Boron dilution transient is classified as an AOO

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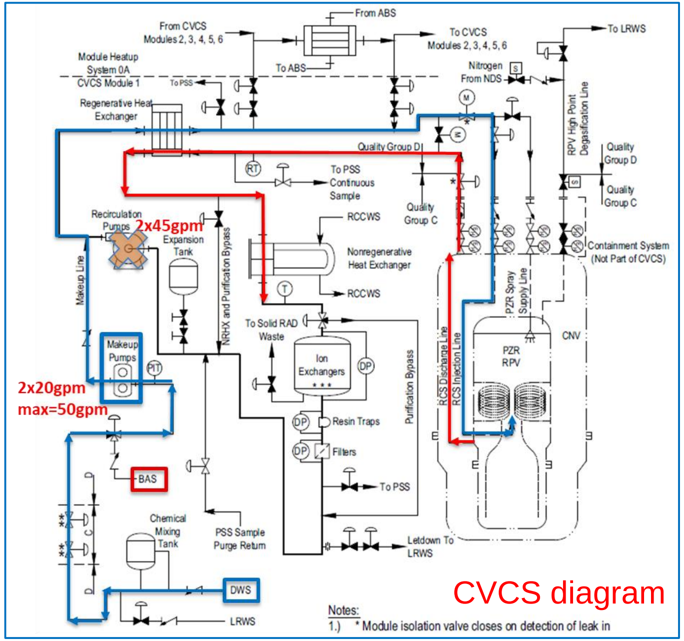
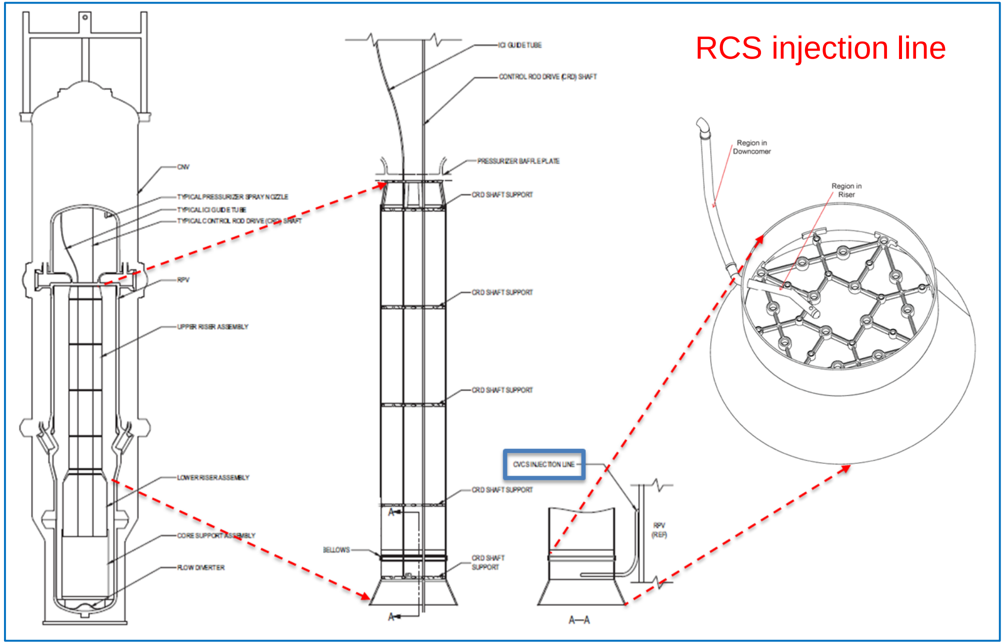
Boron Dilution Sequences





Modified from: NuScale FSAR. Rev5. Figures 9.3.4-1, 3.9-1, 3.9-2 and NuScale Comprehensive Vibration Assessment Program. Figure 2-21.

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Boron dilution sequences. Hypotheses.

Hypotheses (see Sections 15.4.6.2 and 15.4.6.3.3, Table 15.4-13 for more details):

- Initial power level is the associated with the HFP condition 100 % RTP. **HFP is considered as a limiting case**.
- **Normal AC power is assumed to be available**.
- The plant control systems and safety features perform as designed, with allowances for instrument accuracy.
- No human actions are credited to mitigate the effects of the transient.
- **The regulating CRA bank is not credited with mitigating the reactivity insertion associated with a boron dilution of the RCS**.

Initial conditions:

- Makeup flow rate: 50 gpm = **3.15 kg/s**; makeup temperature = **278 K**
- The letdown flow rate is assumed to be equal to the makeup flow rate.
- The RCS flow rate is at the minimum value, **535.24 kg/s**.
- Initial boron concentration = **1600 ppm**; and initial boron reactivity coefficient = **-10 pcm/ppm**.

Single failure: There are no single failures that could occur during a boron dilution of the RCS that result in a more severe outcome for the limiting cases.

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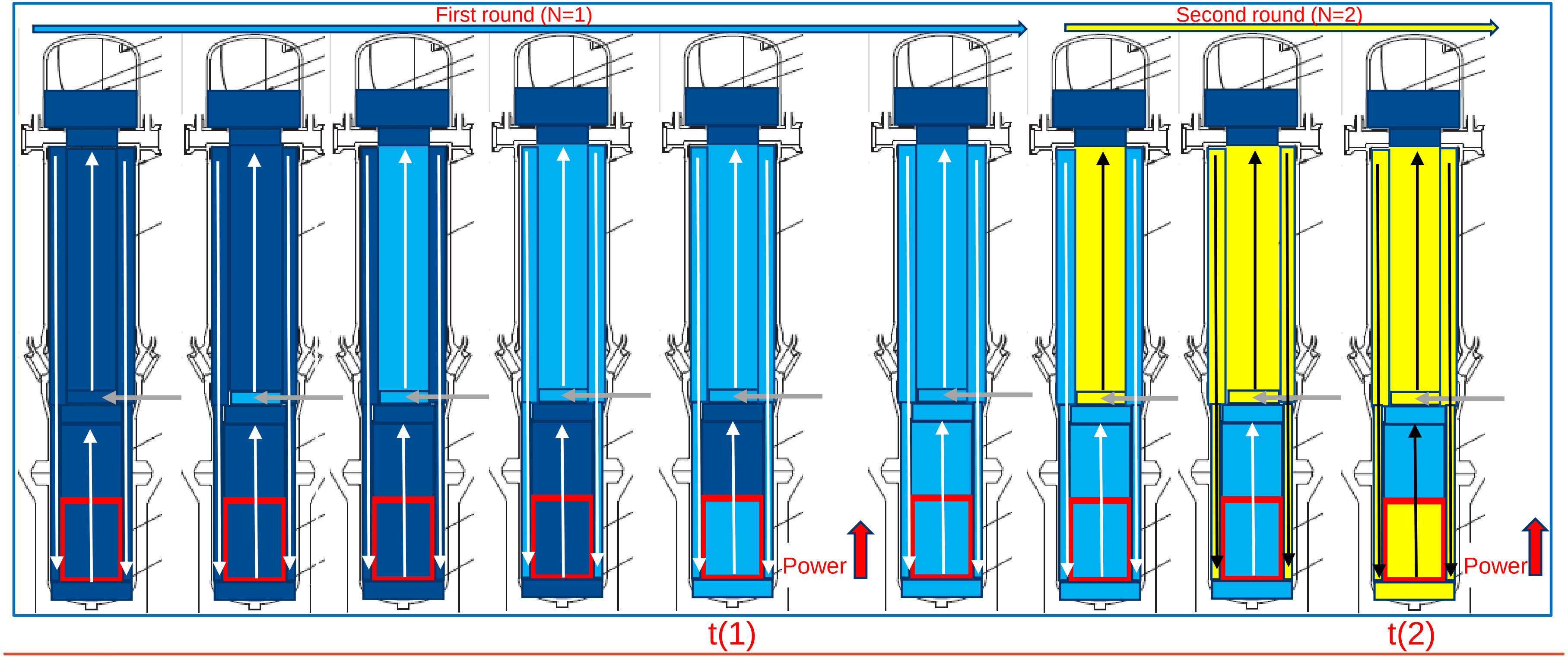
Boron dilution sequences. Slug model



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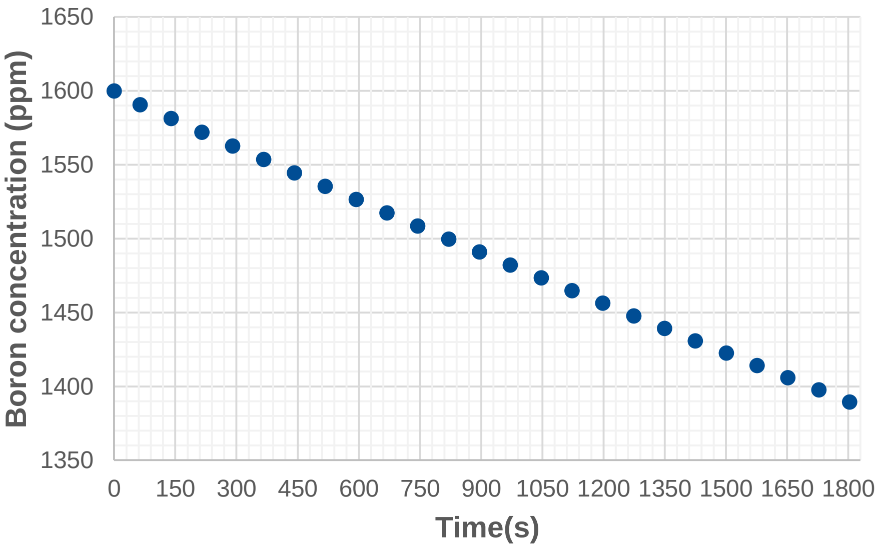
Boron dilution sequences. Calculation techniques.

Boron dilution assuming Dilution Front or Slug Flow model

$$C(N) = C_0 \left[\frac{W_{Nc}}{W_D + W_{NC}}\right]^N$$

$$t(N) = \frac{M_{RCSI}}{W_D + W_{NC}} + (N-1)\frac{M_{RCS}}{W_D + W_{NC}}$$

C(N)	The Nth front Boron concentration (ppm)
N	number of times the wave front passes through the core
t(N)	time at which the wave front passes through the core
W_{NC}	Natural circulation mass flow rate
W_D	Dilution mass flow rate
M_{RCS}	RCS fluid mass (w/o PZR)
M_{RCSI}	mass between the CVCS injection point to core inlet



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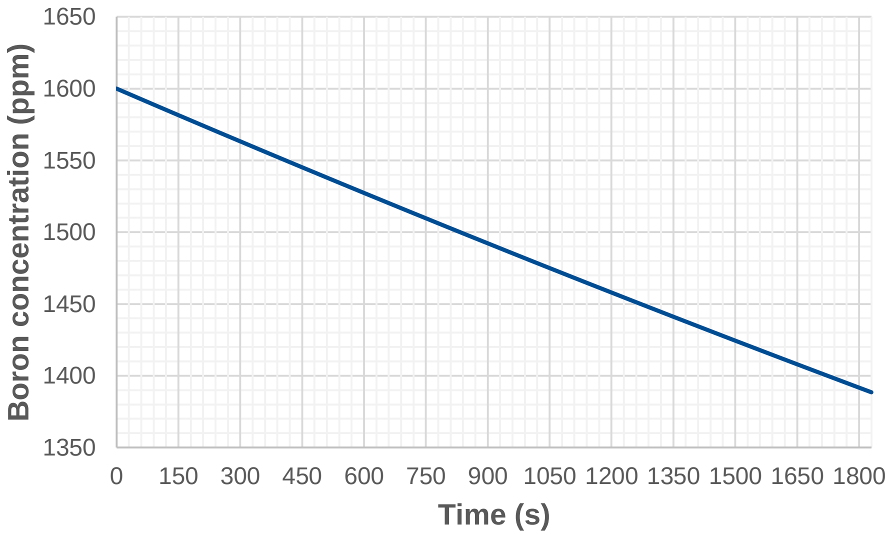
Boron dilution assuming perfect mixing

$$\frac{dC(t)}{dt} = -\frac{Q_{in}\rho_{in}}{V_r\rho_r} \cdot C(t) \rightarrow \begin{cases} \dfrac{1}{\tau_B} = \dfrac{Q_{in}\rho_{in}}{V_r\rho_r} \\ C(t) = C_0 \cdot e^{-\frac{t}{\tau_B}} \end{cases}$$

$$\frac{dR(t)}{dt} = \alpha_B \cdot \frac{dC(t)}{dt}$$

$C(t)$	Boron concentration (ppm)
$R(t)$	Reactivity (pcm)
Q_{in}	Makeup volumetric flow rate, with [B]=0
ρ_{in}	Density (1.01 bar, 277.59 K)
$V_r\rho_r$	RCS fluid mass (w/o PZR)
α_B	Differential Boron worth (pcm/ppm) = -10 pcm/ppm



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Boron dilution sequences. Results.

Parameter	Value
Dilution rate (2 pumps)	50 gpm = 3.15 kg/s
Initial reactivity insertion rate (complete mixing model)	1.11 pcm/second
Initial reactivity insertion rate (wave front model)	34.35 pcm/second
Time to Loss of SDM – Complete Mixing Model	30.5 minutes
Time to Loss of SDM – Wave front model (N=25)	31.3 minutes

NuScale DCA. Table 15.4-14

The time to loss SDM acceptance criterion (t> 15 min) is met. See SRP section 15.4.6

The sequence evolution until reactor trip is bounded by the "Uncontrolled Control Rod Assembly Withdrawal" sequence at Power (NuScale DCA. Section 15.4.2). ***Similar to SLB***

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Uncontrolled Control Rod Assembly Withdrawal (75% TRP)

Event	Time [s]
CRA bank begins to withdraw	0
High RCS hot temperature limit reached (610 F = 321.11 C): **SCRAM setpoint, DHRS setpoint Secondary System Isolation and PZR Heaters Trip**	144
High pressurizer pressure limit reached (2000 psia): SCRAM setpoint, DHRS setpoint Secondary System Isolation and PZR Heaters Trip	150
MCHFR occurs (**1.499 > 1.284 SL-MCHFR, Table 15.1-15**). **The MCHFR acceptance criterion is met.**	151
Reactor trip actuated, DHRS actuation signal, secondary system isolation and PZR Heaters Trip (High RCS hot temperature limit reached + 8s delay)	152
Maximum RCS pressure occurs and control rods fully inserted (SCRAM signal + bounding CR drop-time)	154
DHRS valves fully open (DHRS actuation signal + 30s opening time)	182

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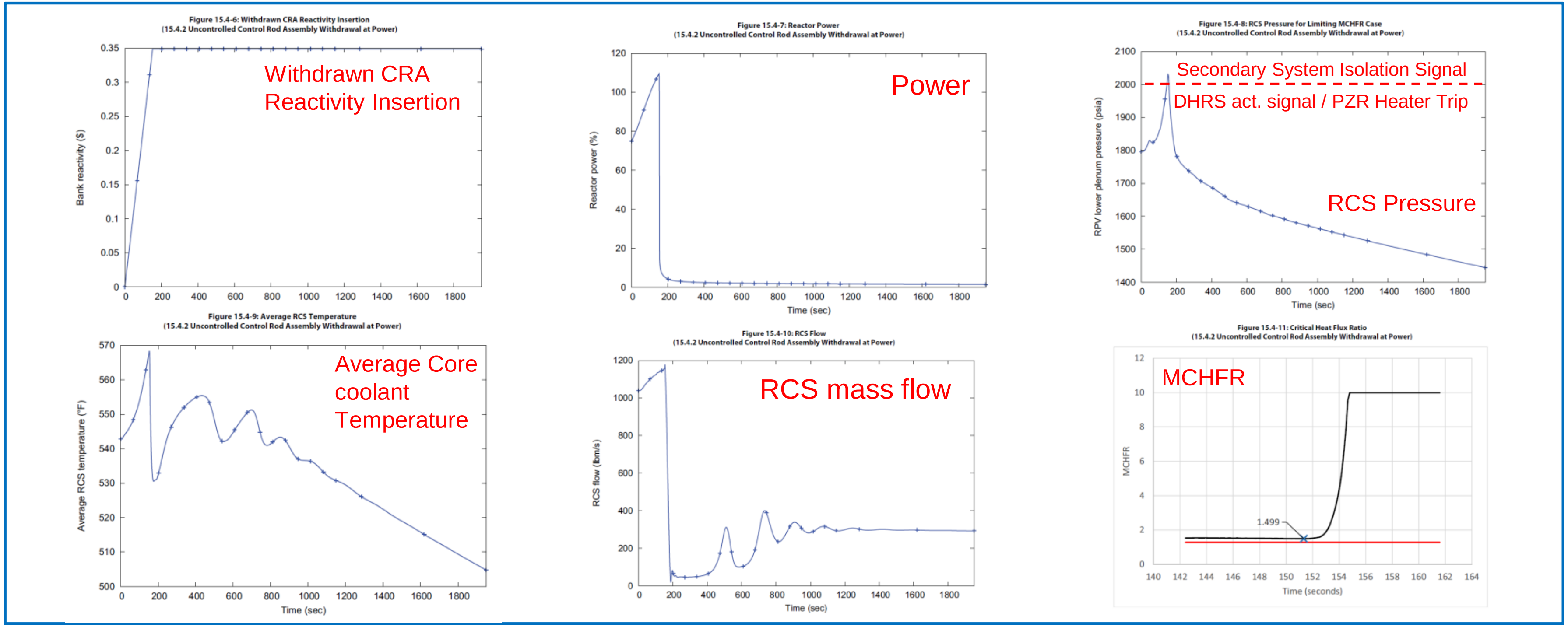
Uncontrolled Control Rod Assembly Withdrawal (75% TRP)



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Questions

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Questions

- What are the differences between the success criteria of NuScale and standard PWR in US regulation?
- If the ECCS were unavailable, could the CVCS system mitigate the consequences of a LOCA transient?
- If necessary, could the ECCS and DHRS work simultaneously during the SLB sequence? And during the LOCA sequence?
- What is the main source of unborated water in the NuScale design?

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Acknowledgments

We want to thank NuScale Power for granting us permission to use the figures in this presentation.

Any errors that may be included in this presentation are the responsibility of the authors.

Appendix

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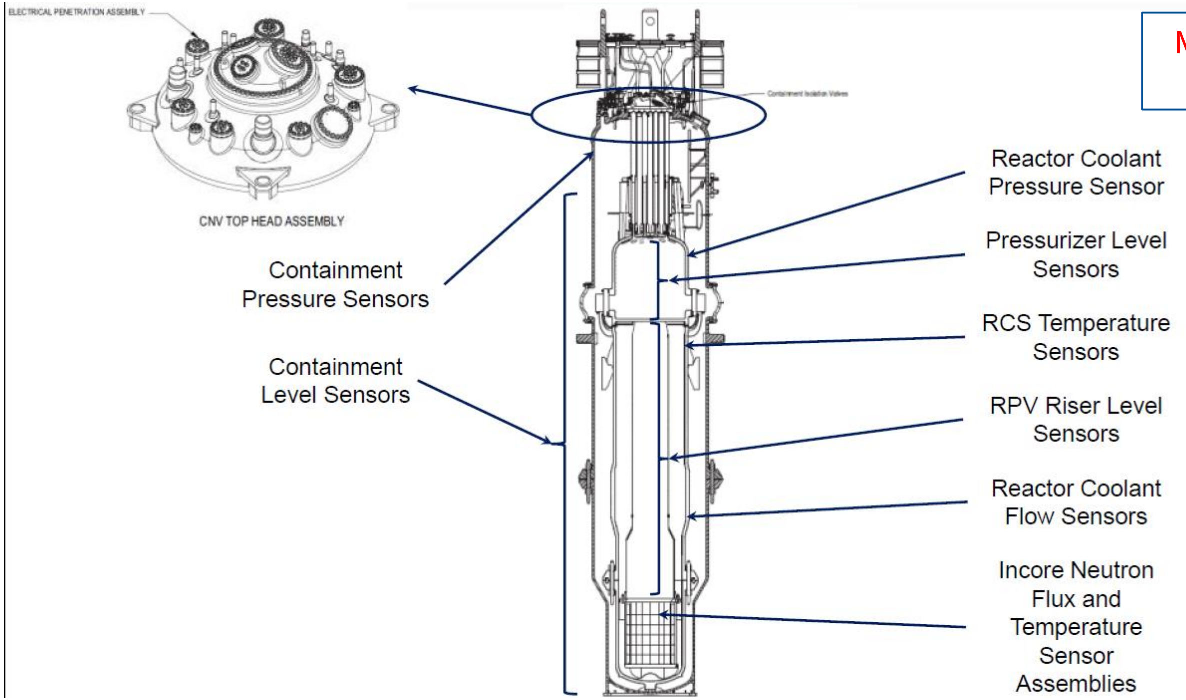
Instrumentation. Measurement Uncertainties

Measurement uncertainties are taken into account in the safety analysis



Advancements in Technology Applications for the NuScale Power I&C Design. Brian Arnholt, NuScale Power. June 5, 2018

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NuScale. Initial Conditions for DSA

Plant Parameter	Value	Uncertainty
Design core power	100% RTP = 160 MW	+2%
RCS Tavg at 100% RTP	545 °F (285 °C / **558,15 K**)	± 10 °F (**± 5.56 K**)
Pressurizer Pressure	1850 psia (**127.55 bar**)	± 70 psi (**± 4.83 bar**)
Pressurizer level at core power >15% RTP	60%	± 8%
Containment pressure	0 – 3 psia (**0.21 bar**) Nominal pressure < 1 psia (**0.07 bar**)	3.0 psia (**0.21 bar**) bounds the initial pressure
Main steam pressure at 100% RTP	500 psia (**34.47 bar**)	± 35 psi (**± 2.41 bar**)
Feedwater temperature at 100% RTP	300 °F (148.89 °C / **422.04 K**)	± 10 °F (**± 5.56 K**)
RCS flow at 100% RTP	1180 – 1480 lbm/s (**535.24 – 671.32 kg/s**)	N/A

From: NuScale DCA. Chapter 15. Table 15.0-6: Module Initial Conditions Ranges for DBE Evaluation

The worst initial conditions (bias) must be selected for each DSA sequence

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Reactor Trip Signals

Parameter	Limit	Reference units	Limit	SI Units
High Power Range Linear Power	High-1 = 25%	RTP		
	High-2 = 120%	RTP		
High Intermediate Range Log Power Rate	3	dpm		
High Power Range Positive and Negative Rate	± 15%	RTP/minute		
High Source Range Count Rate	5E+5	cps		
High Source Range Log Power Rate	3	dpm		
High Narrow Range RCS Hot Temperature	610	F	321.11	C
High Narrow Range Containment Pressure	9.5	psia	0.65	bar
High Pressurizer Pressure	2000	psia	137.89	bar
Low Pressurizer Pressure	1720	psia	118.59	bar
Low Low Pressurizer Pressure	1600	psia	110.32	bar
High Pressurizer Level	80	%		
Low Pressurizer Level	35	%		
High Main Steam Pressure	800	psia	55.16	bar
Low Main Steam Pressure (≥15% RTP)	300	psia	20.68	bar
Low Low Main Steam Pressure	20	psia	1.38	bar
High Main Steam Superheat	150	F	65.56	C
Low Main Steam Superheat	0	F	0.00	C
Low Low RCS Flow	0	ft^3/s	0.00	m^3/h
Low AC Voltage to Battery Chargers	80%	of normal ELVS voltage		
High Under-the-Bioshield Temperature	250	F	121.11	C

NuScale DCA Rev 5. Table 7.1-3: Reactor Trip Functions

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Engineered Safety Feature Actuation System (ESFAS) Functions

ESF Function	Process Variable	Limit	Reference units	Limit	SI Units	System Automated Function
Emergency Core Cooling System (ECCS)	High Containment Water Level	240-260	in	6.096-6.604	m	Removes Electrical Power to the trip solenoids of the reactor vent valves. Removes electrical power to the trip solenoids of the reactor recirculation valves
	Low WR RCS Pressure	800	psia	55.16	bar	
	Low ELVS voltage 24-hour Timer	24	hour	24	hour	
Decay Heat Removal System (DHRS)	High Pressurizer Pressure	2000	psia	137.90	bar	Removes electrical power to the trip solenoids of the decay heat removal valves Removes electrical power to the trip solenoids of the following valves in the containment, main steam, and feedwater systems: • main steam isolation valves • main steam isolation bypass valves • secondary main steam isolation valves • secondary main steam isolation valve bypass valves • feedwater isolation valves • feedwater regulating valves
	High Narrow Range RCS Hot Temperature	610	F	321.11	C	
	High Main Steam Pressure	800	psia	55.16	bar	
	Low AC Voltage to Battery Chargers	80% of normal ELVS voltage				

NuScale DCA Rev 5. Chapter Seven Instrumentation and Controls. Table 7.1-4: Engineered Safety Feature Actuation System Functions

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Engineered Safety Feature Actuation System (ESFAS) Functions

ESF Function	Process Variable	Limit	Reference units	Limit	SI Units	System Automated Function
Secondary System Isolation	High PZR Pressure	2000	psia	137.90	bar	Removes electrical power to the trip solenoids of the following valves: • MSIVs • MSI bypass valves (MSIBV) • secondary MSIV • secondary MSIBV • feedwater isolation valves • feedwater regulating valves
	High Narrow Range RCS Hot Temperature	610	F	321.11	C	
	Low Main Steam Pressure (≥15% RTP)	300	psia	20.68	bar	
	Low Low Main Steam Pressure	20	psia	1.38	bar	
	High Main Steam Pressure	800	psia	55.16	bar	
	Low Main Steam Superheat	0	F	0.0	C	
	High Main Steam Superheat	150	F	65.556	C	
	High Narrow Range Containment Pressure	9.5	psia	0.66	bar	
	Low Low PZR Pressure	1600	psia	110.32	bar	
	Low Low PZR Level	20	%	20	%	
	Low ELVS 480VAC to EDSS Battery Chargers	80%	of normal ELVS voltage	80%	of normal ELVS voltage	
	High Under-the-Bioshield Temperature	250	F	121.11	C	
CVCS Isolation (CVCSI)	High PZR Level	80	%	80	%	Removes electrical power to the trip solenoids of the following valves: • RCS injection valves • RCS discharge valves • PZR spray valves • RCS high point degasification valves
	High Narrow Range Containment Pressure	9.5	psia	0.66	bar	
	Low Low PZR Pressure	1600	psia	110.32	bar	
	Low Low PZR Level	20	%	20	%	
	Low Low RCS Flow	0	ft^3/s	0.00	m^3/s	
	Low AC Voltage to Battery Chargers	80%	of normal ELVS voltage	80%	of normal ELVS voltage	
	High Under-the-Bioshield Temperature	250	F	121.11	C	

NuScale DCA Rev 5. Table 7.1-4: Engineered Safety Feature Actuation System Functions

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

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Engineered Safety Feature Actuation System (ESFAS) Functions

ESF Function	Process Variable	Limit	Reference units	Limit	SI Units	System Automated Function
Containment System Isolation (CSI) Signal	High Narrow Range Containment Pressure	9.5	psia	0.066	MPa	Removes electrical power to the trip solenoids of the following valves: • RCS injection valves • RCS discharge valves • PZR spray valves • RPV high point degasification line valves • feedwater isolation valves • feedwater regulating valves • MSIVs • MSIBVs • secondary MSIVs • secondary MSIBVs • containment evacuation system valves • reactor CCWS supply and return valves • containment flooding and drain system valves
	Low Low PZR Level	20	%	20	%	
	Low AC Voltage to Battery Chargers	80%	of normal ELVS voltage	80%	of normal ELVS voltage	
	High Under-the-Bioshield Temperature	250	F	121.11	C	
Pressurizer Heater Trip	Low PZR Level	35	%	35	%	Removes electrical power to the PZR heaters
	High PZR Pressure	2000	psia	137.90	bar	
	High Narrow Range RCS Hot Temperature	610	F	321.11	C	
	High Main Steam Pressure	800	psia	55.16	bar	
	Low AC Voltage to Battery Chargers	80%	of normal ELVS voltage	80%	of normal ELVS voltage	

NuScale DCA Rev 5. Table 7.1-4: Engineered Safety Feature Actuation System Functions

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Engineered Safety Feature Actuation System (ESFAS) Functions

ESF Function	Process Variable	Limit	Reference units	Limit	SI Units	Automated Function
Demineralized Water System Isolation (DWSI)	High Power Range Linear Power	High-1 = 25% High-2 = 120%	RTP			Removes electrical power to the trip solenoids of the demineralized water supply valves
	High Intermediate Range Log Power Rate	3	dpm			
	High Power Range Positive and Negative Rate	± 15%	RTP/minute			
	High Source Range Count Rate	5E+5	cps			
	High Source Range Log Power Rate	3	dpm			
	High Narrow Range RCS Hot Temperature	610	F	321.11	C	
	High Narrow Range Containment Pressure	9.5	psia	0.66	bar	
	High PZR Pressure	2000	psia	137.90	bar	
	Low PZR Pressure	1720	psia	118.59	bar	
	Low Low PZR Pressure	1600	psia	110.32	bar	
	High PZR Level	80	%			
	Low PZR Level	35	%			
	High Main Steam Pressure	800	psia	55.16	bar	
	Low Main Steam Pressure (≥15% RTP)	300	psia	20.68	bar	
	Low Low Main Steam Pressure	20	psia	1.38	bar	
	High Main Steam Superheat	150	F	65.56	C	
	Low Main Steam Superheat	0	F	0.0	C	
	Low RCS Flow	1.7	ft^3/s	0.048	m^3/s	
	Low Low RCS Flow	0	ft^3/s	0.000	m^3/s	
	Low AC Voltage to Battery Chargers	80%	of normal ELVS voltage			
	High Under-the-Bioshield Temperature	250	F	121.11	C	
	High Subcritical Multiplication (SCM)	3.2				

NuScale DCA Rev 5. Table 7.1-4: Engineered Safety Feature Actuation System Functions

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Actuation Delays Assumed in the Plant Safety Analysis

Signal	Actuation Delay	Signal	Actuation Delay
High Power Range Linear Power	2,0s	Low Low PZR Level	3,0s
SR and IR Log Power Rate	Variable	Low RCS Pressure	2,0s
High Power Range Rate	2,0s	Low Main Steam Pressure	2,0s
High Source Range Count Rate	3,0s	Low Low Main Steam Pressure	2,0s
High Subcritical Multiplication	150,0s	High Main Steam Pressure	2,0s
High Narrow Range RCS Hot Temperature	8,0s	Low Main Steam Superheat	8,0s
High Narrow Containment Pressure	2,0s	High Main Steam Superheat	8,0s
High PZR Pressure	2,0s	Low RCS Flow	6,0s
High PZR Level	3,0s	Low Low RCS Flow	6,0s
Low PZR Pressure	2,0s	High Containment Water Level	3,0s
Low Low PZR Pressure	2,0s	Low AC Voltage to the Battery Chargers	60,0s
Low PZR Level	3,0s	High Under-the-Bioshield Temperature	8,0s

NuScale DCA Rev 5. Chapter Seven Instrumentation and Controls. Table 7.1-6: Design Basis Event Actuation Delays Assumed in the Plant Safety Analysis

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Valves

Parameter	Value	British Imperial units	Value	SI Units	Uncertainty
RSV setpoint (Valve 1)	2075	psia	143.1	bar	± 1%
RSV setpoint (Valve 2)	2100	psia	144.8	bar	± 1%
Primary and Secondary MSIVs. Closure time	7	s	7	s	
FW Isolation Valves (FWIVs). Closure time	7	s	7	s	
DHRS actuation valves. Opening/closure time	30	s	30	s	

NuScale DCA Rev 5. Table 7.1-4 (Engineered Safety Feature Actuation System Functions), Chapter 6 (pp. 6.2-42) and Chapter 5 (pp. 5.4-21)

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

CAREM reactor main features

Course on “SMR LWR technologies”

Darío DELMASTRO / CNEA

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

– CAREM
– COMPARISON: CLASSIC PWR / CAREM
– CAREM-25 main features
– Integrated Primary System
– Self-pressurization and Natural Circulation

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

CAREM

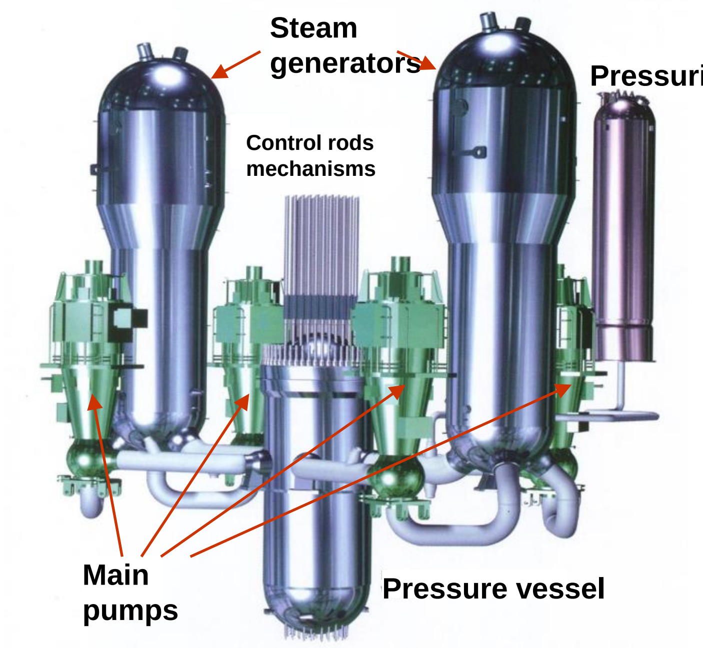
CAREM is an Argentine project aiming to achieve the development, design and construction of an innovative, simple and small nuclear power plant.

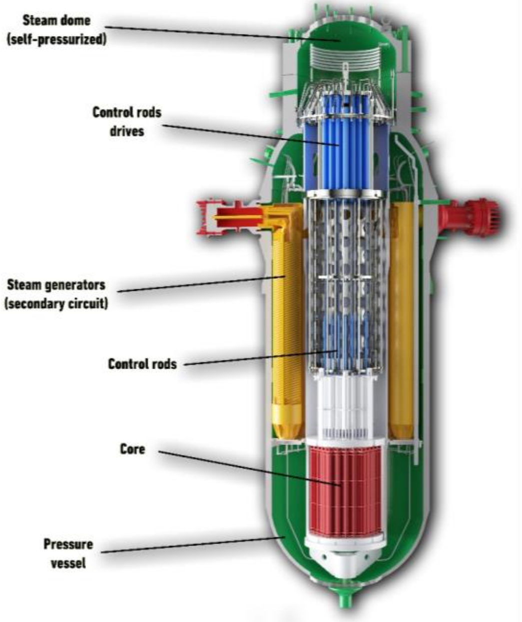
The first step of this project is the construction of the prototype CAREM-25.

The construction permits were obtained and the construction is ongoing at Atucha site.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

COMPARISON: CLASSIC PWR / CAREM

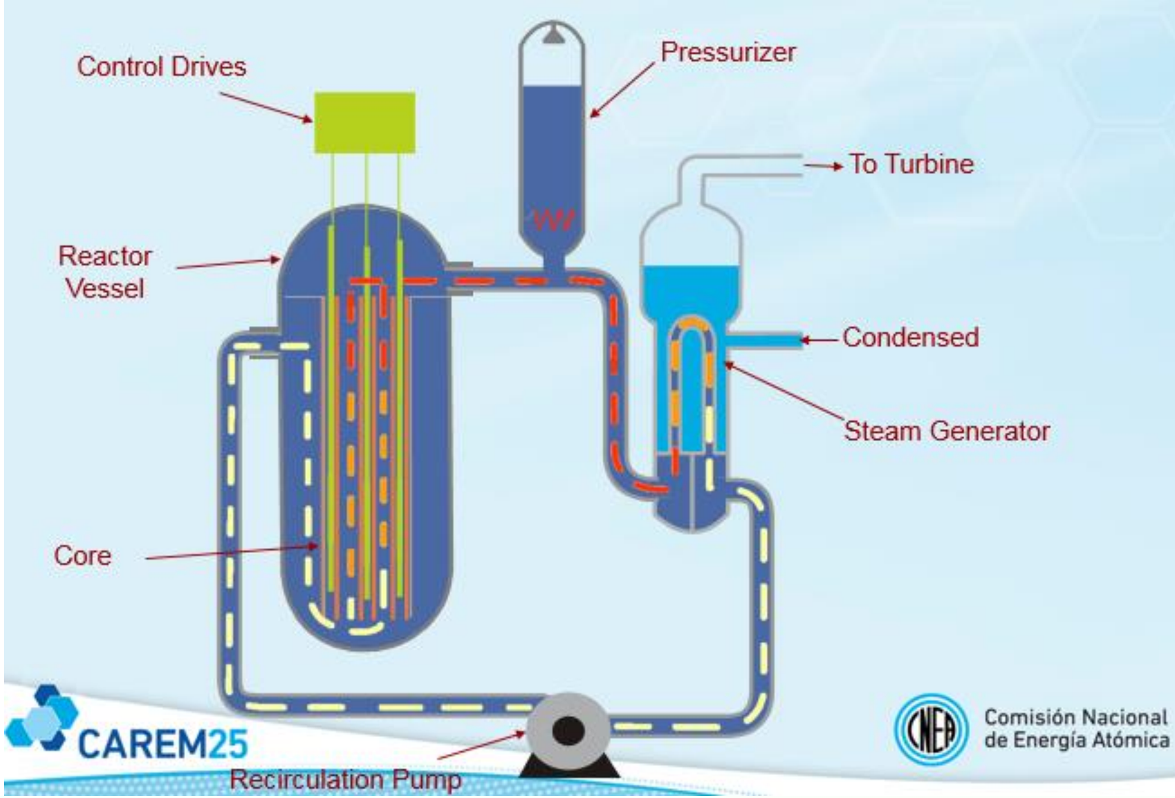




CLASSIC PWR | **CAREM**

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

COMPARISON: CLASSIC PWR / CAREM



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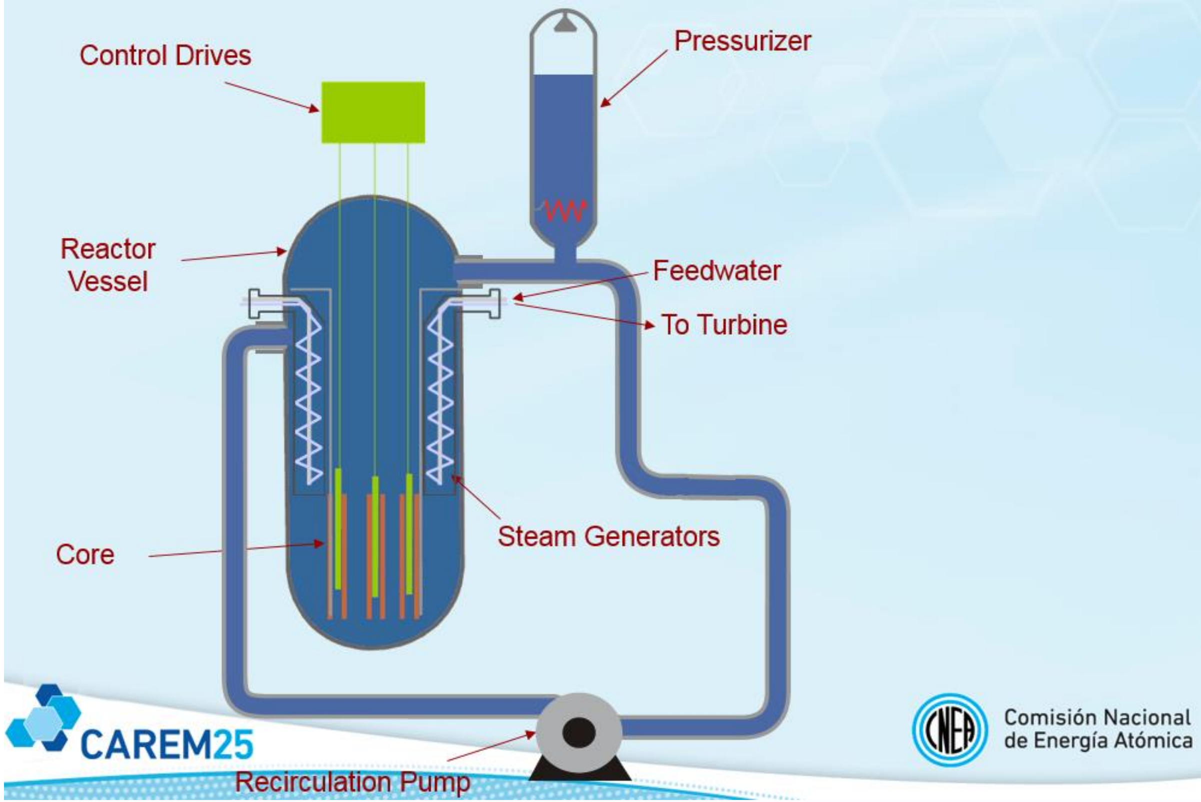
TRANSFORMATION: INTEGRATION OF THE SGs



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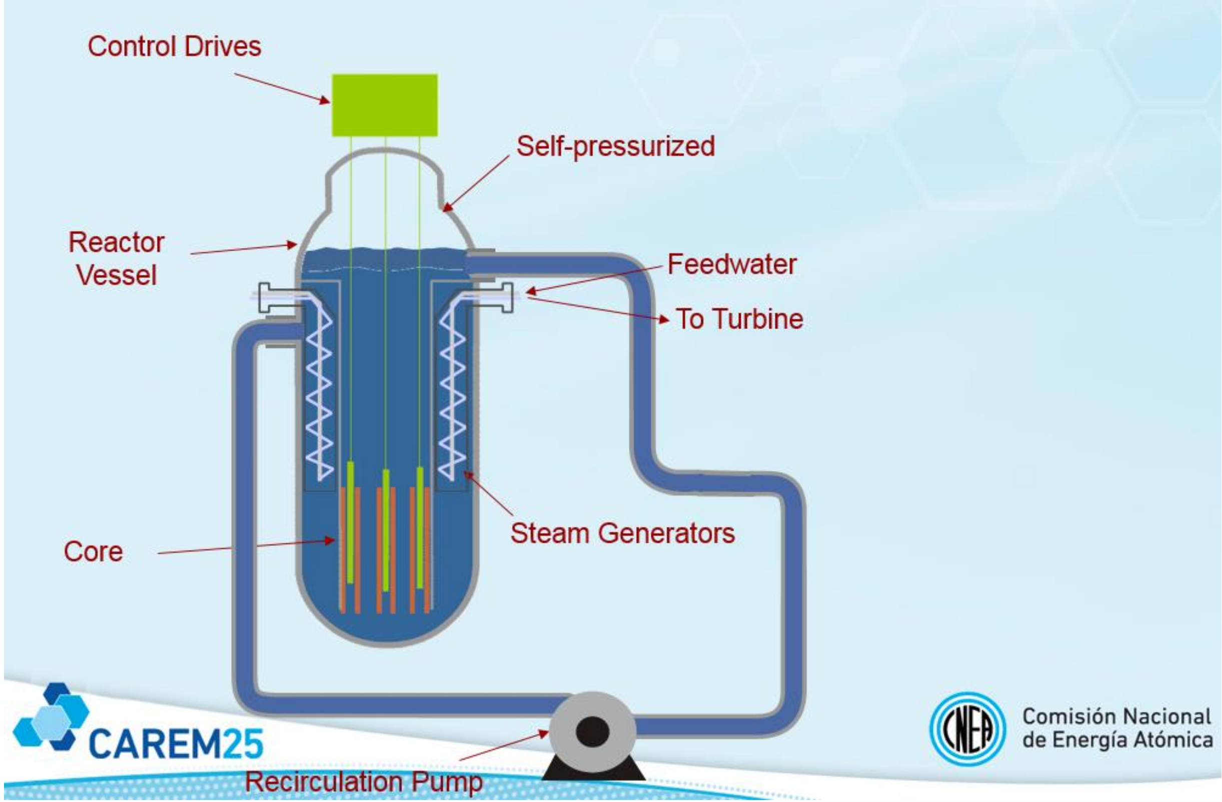
TRANSFORMATION: PRESSURIZER ELIMINATION



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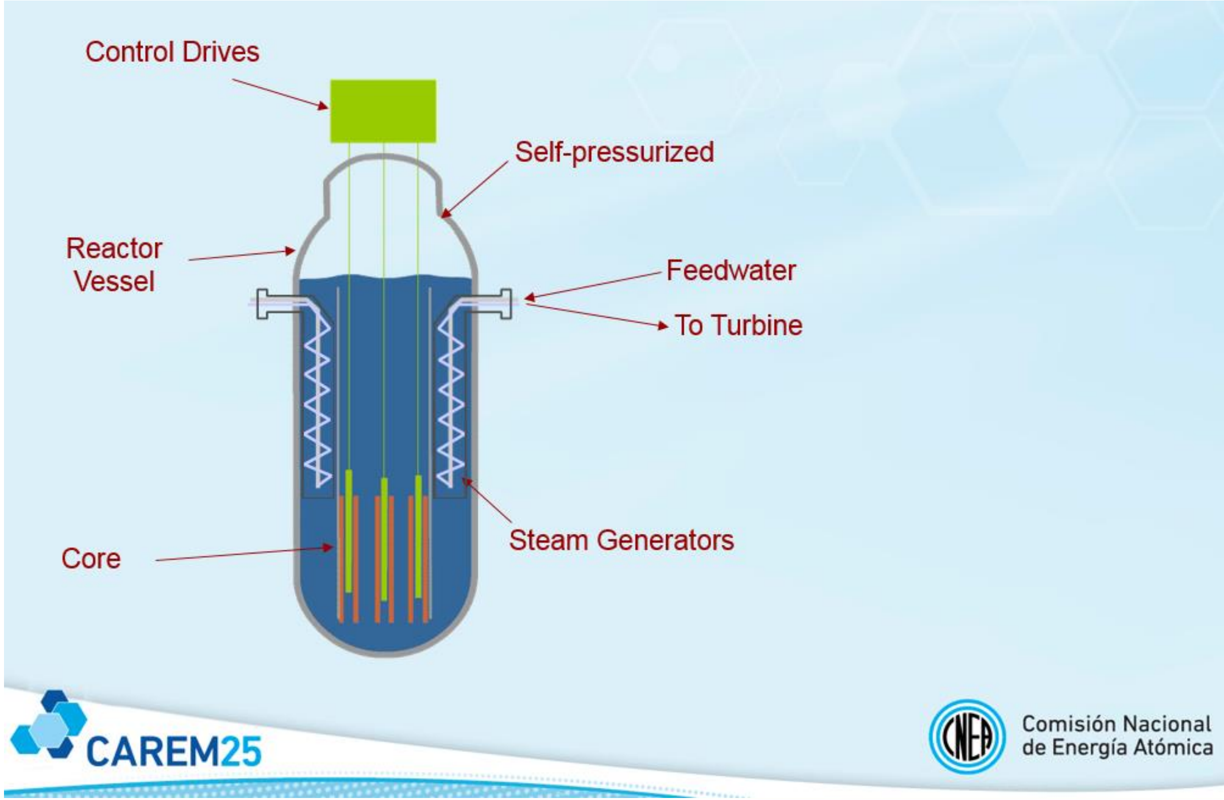
TRANSFORMATION: PUMPS ELIMINATION



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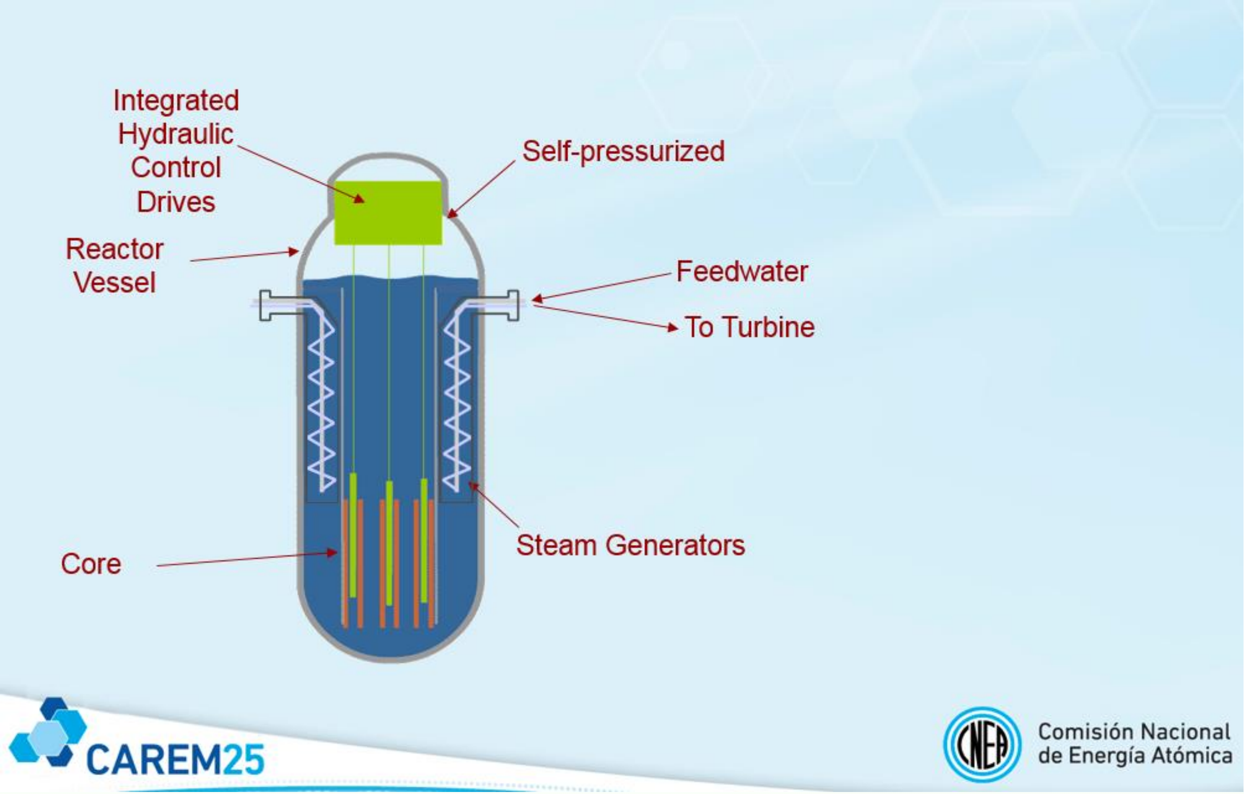
TRANSFORMATION: CONTROL DRIVES INTEGRATION

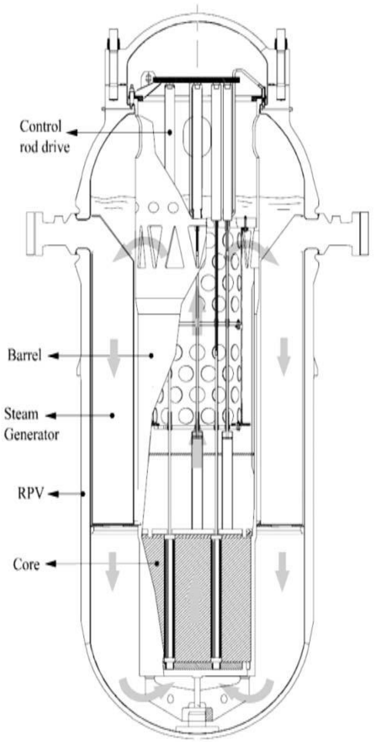


This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

CAREM-25 main features

- Integrated primary cooling system
- Primary cooling by natural circulation
- Self-pressurized
- Safety systems relying on passive features,
- and active and self powered systems to cool the reactor and the containment after the grace period.

Long grace period



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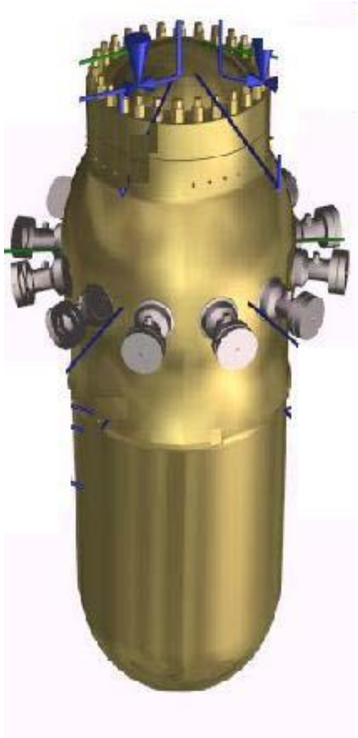
Integrated Primary System

In order to simplify the design the whole high-energy primary system, core, steam generators, pressurizer and pumps, are contained inside a single pressure vessel.

This considerably reduces the number of pressure vessels and simplifies the layout.



This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Integrated Primary System

Due to the absence of large diameter piping associated to the primary system, no large LOCA has to be handled by the safety systems. The elimination of large LOCA considerably reduce the needs in ECCS components, AC supply systems, etc.

Large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or accidents.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

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Eliminating primary pumps and pressuriser results in added safety (LOFA elimination), and advantages for maintenance and availability.

Self-pressurization of the primary system in the steam dome is the result of the liquid-vapor equilibrium. The large volume of the integral pressurizer also contributes to the damping of eventual pressure perturbations. Heaters and sprinkles typical of conventional PWR’s are thus eliminated.

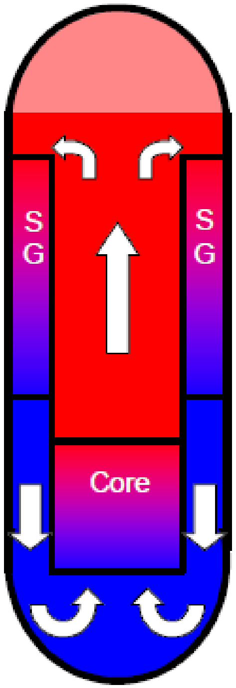
This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Self-pressurization and Natural Circulation

The flow rate in the reactor primary systems is achieved by natural circulation.

The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing the adequate flow rate in the core in order to have the sufficient thermal margin to critical phenomena.

Reactor coolant natural convection is produced by the location of the steam generators above the core.



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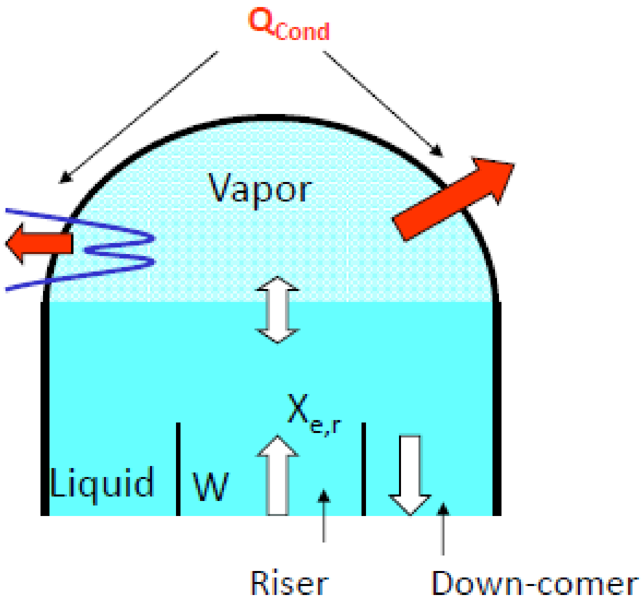
Self-pressurization

Dome



Thermal Balance

$$Q_{Cond} = W \,.\, X_{e,r} \,.\, h_{fg}$$

$$X_{e,r} = Q_{Cond} \,/\, (W \,.\, h_{fg})$$

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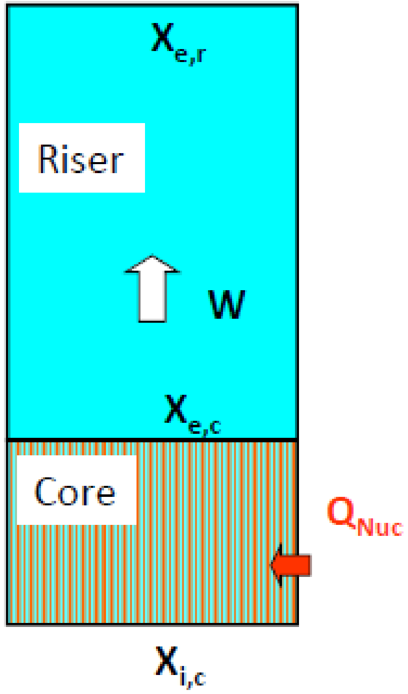
Self-pressurization



Thermal Balance

$$Q_{Nuc} = W \cdot (h_{e,c} - h_{i,c})$$

$$Q_{Nuc} = W \cdot h_{fg} \cdot (X_{e,c} - X_{i,c})$$

Assuming an adiabatic riser and an uniform pressure

$$X_{i,c} = (Q_{Cond} - Q_{Nuc}) / (W \cdot h_{fg})$$

For constants Q_{Cond} and Q_{Nuc} :

if	Core Flow Rate decrease	⇩
then	Core Quality decrease	⇩

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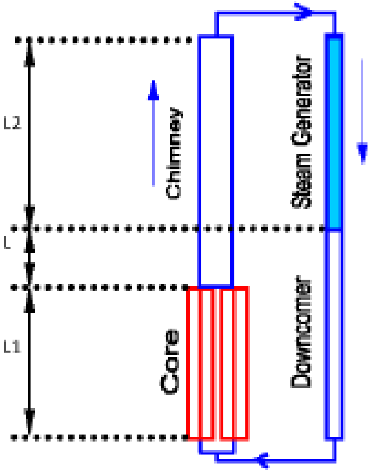
Natural Circulation

In CAREM-25 reactor the steam quality is very low and therefore the largest contribution driving term in the momentum balance is due to single-phase buoyancy forces. Considering:

- Only single-phase natural circulation,
- the Boussinesq approximation.
- And that the heat flux is uniform regarding the axial direction.



$$\dot{m} = \sqrt[3]{\frac{2A^2 Q_{Nuc}\, g \rho_l^2 \beta}{K c_p}\left[L + \frac{L1 + L2}{2}\right]}$$

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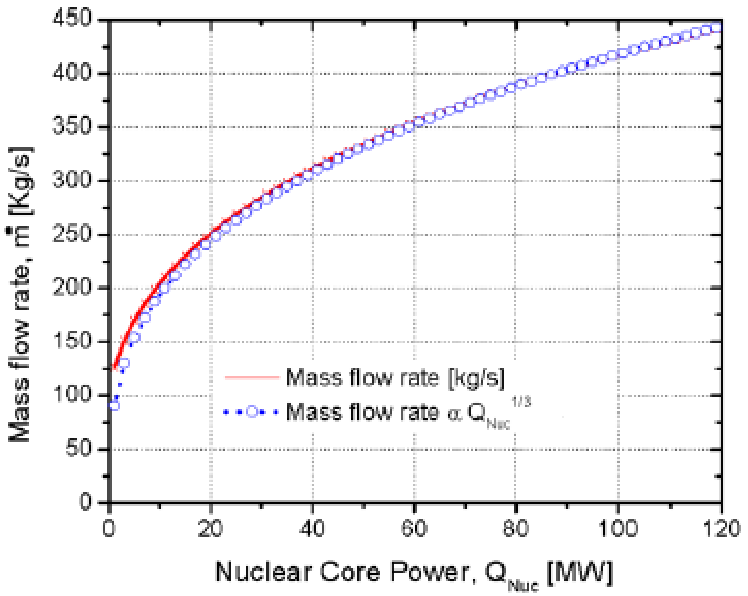
Natural Circulation



C.P. Marcel, H.F. Furci, D.F. Delmastro, V.P. Masson.: "Phenomenology involved in self-pressurized, natural circulation, low thermo-dynamic quality, nuclear reactors: The thermal–hydraulics of theCAREM-25 reactor", Nuclear Engineering and Design 254 (2013) 218– 227.

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Natural Circulation

$$Q_{Nuc} = Q_{SG} + Q_{Cond}$$

$$Q_{Cond} = \dot{m}(h_{Nuc,e} - h_{sat})$$

$$h_{Nuc,i} = h_{sat} - (Q_{Nuc} - Q_{Cond})\sqrt[3]{\frac{Kc_p}{2A^2 Q_{Nuc}\, g\rho_l^{\,2}\,\beta}\left[L + \frac{L1+L2}{2}\right]^{-1}}$$

From this result it can be observed that in a self-pressurized, natural circulation such as CAREM-25, the core inlet enthalpy cannot be controlled directly but it is a result of the combination of the produced and condensed power in the system.

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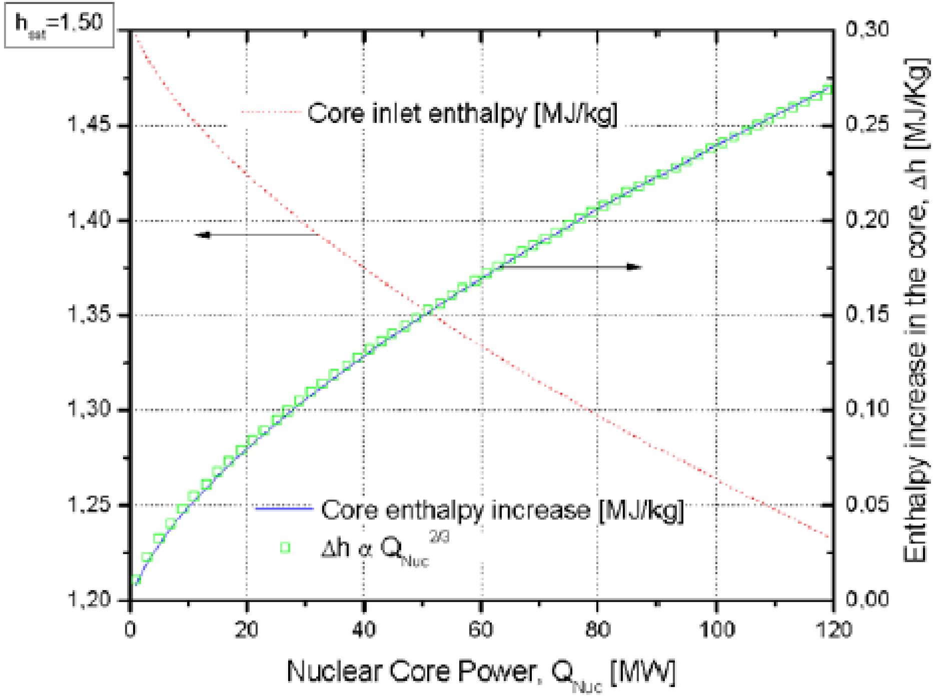
Natural Circulation



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Thank you!

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Questions

1) In a primary system with natural circulation, an initiating event can be eliminated?

- Yes, L-LOCA
- Yes, LOFA
- Yes, RIA
- No

2) In CAREM at a steady state, the primary mass flow rate does not depends on the core power?

- Yes
- No, the primary flow rate is lower at a higher core power
- No, the primary flow rate is higher at a higher core power
- No, the primary flow rate increases in a linear way with the core power

3) In CAREM at a steady state at a higher core power the core inlet enthalpy is higher than in one at a lower core power?

- Yes, in a linear way with the core power
- Yes, in a cuadratic way with the core power
- No, the core inlet enthalpy does not depends on the core power
- No, the core inlet enthalpy is lower if the core power is higher

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

CAREM-25 main core features

Course on “SMR LWR technologies”

Edmundo Lopasso
Comisión Nacional de Energía Atómica (CNEA) - Centro Atómico Bariloche – Argentina

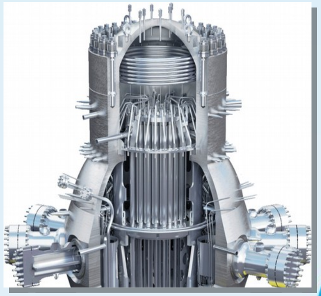
27/01/2021

This project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945063.

Presentation content

Overview of CAREM reactor core design

- ✓Introduction to CAREM25 Demonstration NPP
- ✓Core behaviour.
- ✓Fuel design.
- ✓Control and shutdown.
- ✓Monitoring.



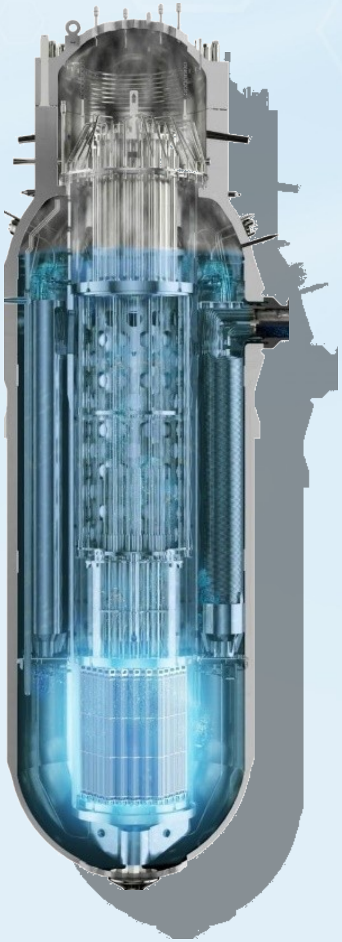
CAREM25

Comisión Nacional de Energía Atómica

CAREM25 project

CAREM25 is one of the main projects of the National Atomic Energy Commission (CNEA) of Argentina. It aims to cover all the stages: **design**, **construction** and **operation** of a **demonstration plant** as a first stage towards a commercial version

CAREM25 site
Paraná river, Atucha



Reactor Pressure vessel

CAREM25 main features

✓**Integrated type PWR** with “mini-helical” Steam Generators.

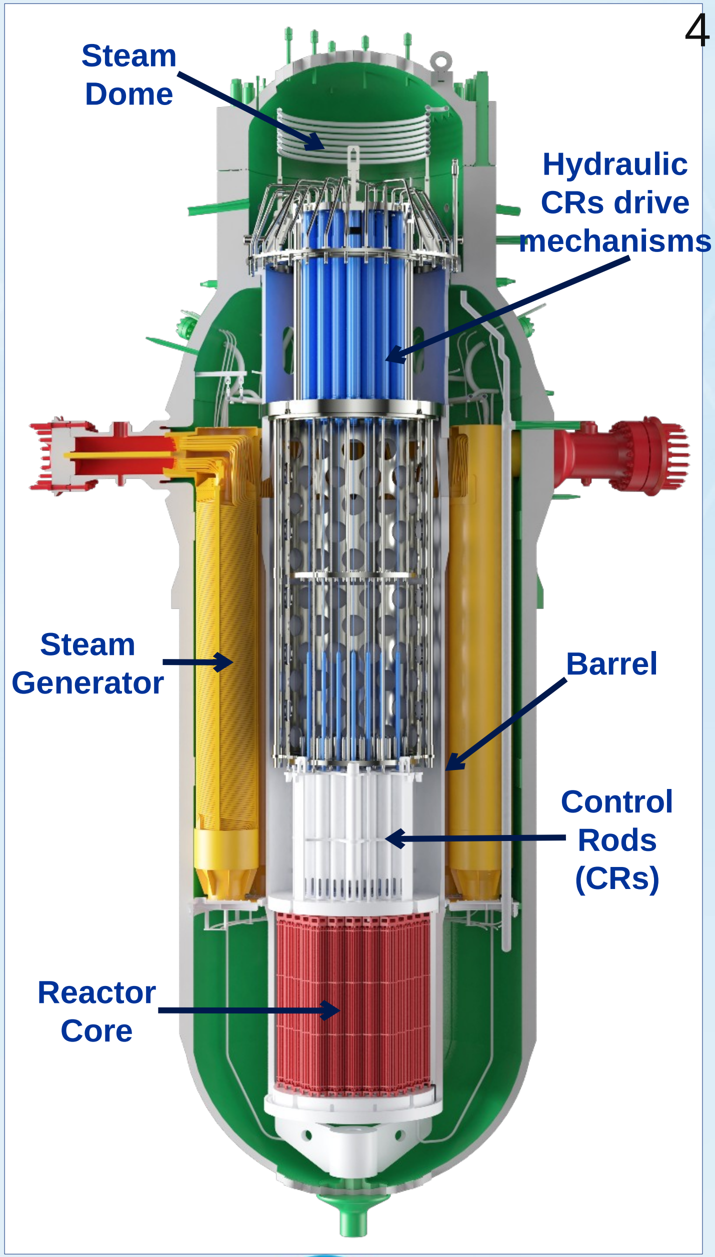
✓Power: 100 MWth / 27 Mwe.

✓**Self-pressurized (12,25 MPa):** core outlet, riser and dome temperature ~ saturation = 326ºC (no heaters, spray, relief valves) and **Natural Circulation**.

✓**In-vessel Control Rods Drive Mechanisms**.

✓**No boron** for reactivity control.

✓Passive safety systems, grace period.



12 Helical Steam Generators

CAREM25

Comisión Nacional de Energía Atómica

CAREM25 main features

✓**Core design:** 61 fuel elements. Hexagonal arrangement similar to a VVER.

✓**Fuel elements:** Zyrcaloy cladding and sintered UO_2 pellets, some of them with Gd_2O_3 burnable poison to lower the number of control rods.

✓**Control rods:** Ag-In-Cd alloy, with stainless steel cladding. They are driven through zircaloy guide tubes inside the fuel.

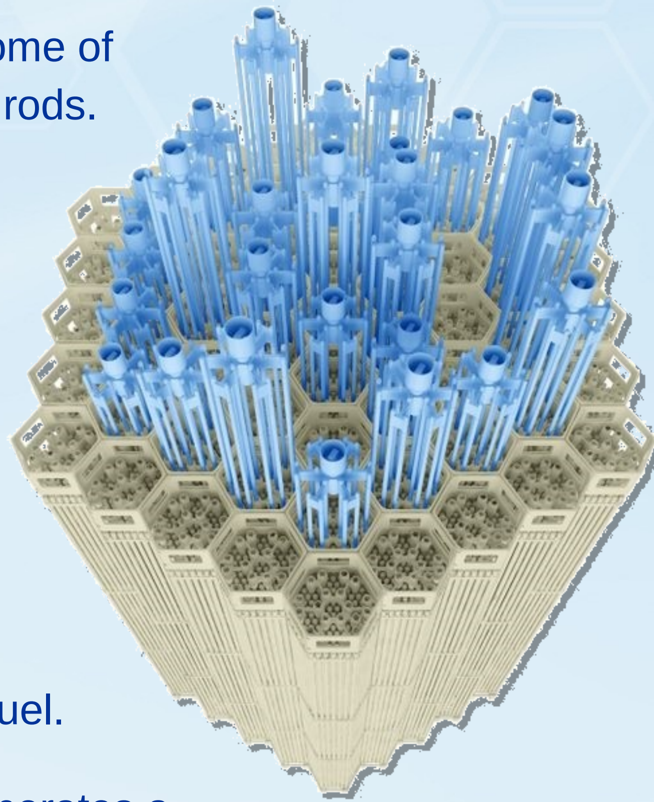
✓**Stainless steel radial neutron reflector**.

✓**Core exit temperature is fixed** by the dome pressure (saturation conditions).

✓**Sub-cooled void fraction** (up to 15%).

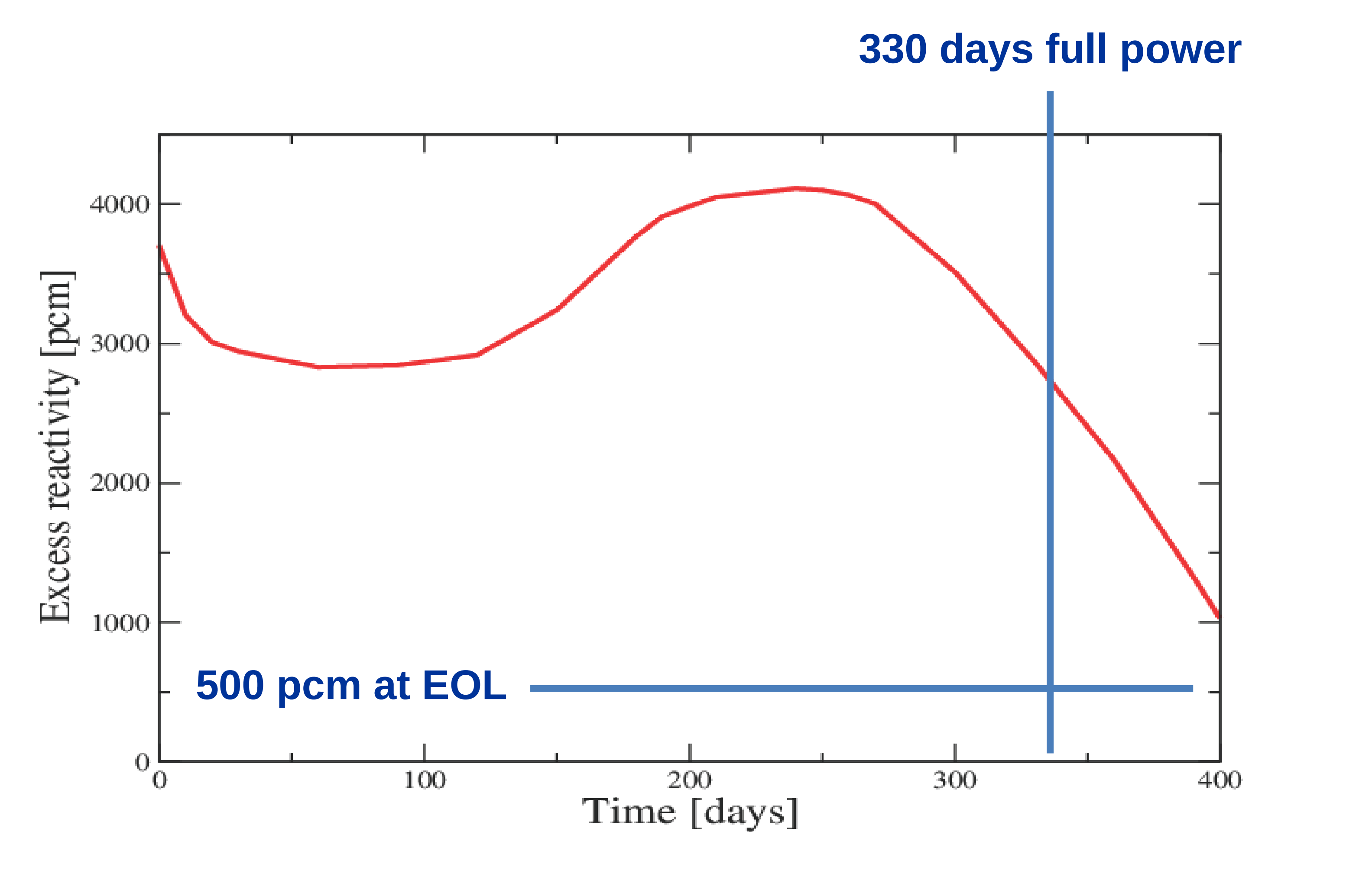
✓**Core management**: Two fuel replacement zones and central.fuel.

✓**Cross flow**: There are no channels. The power distribution generates a three dimensional mass flow rate distribution.

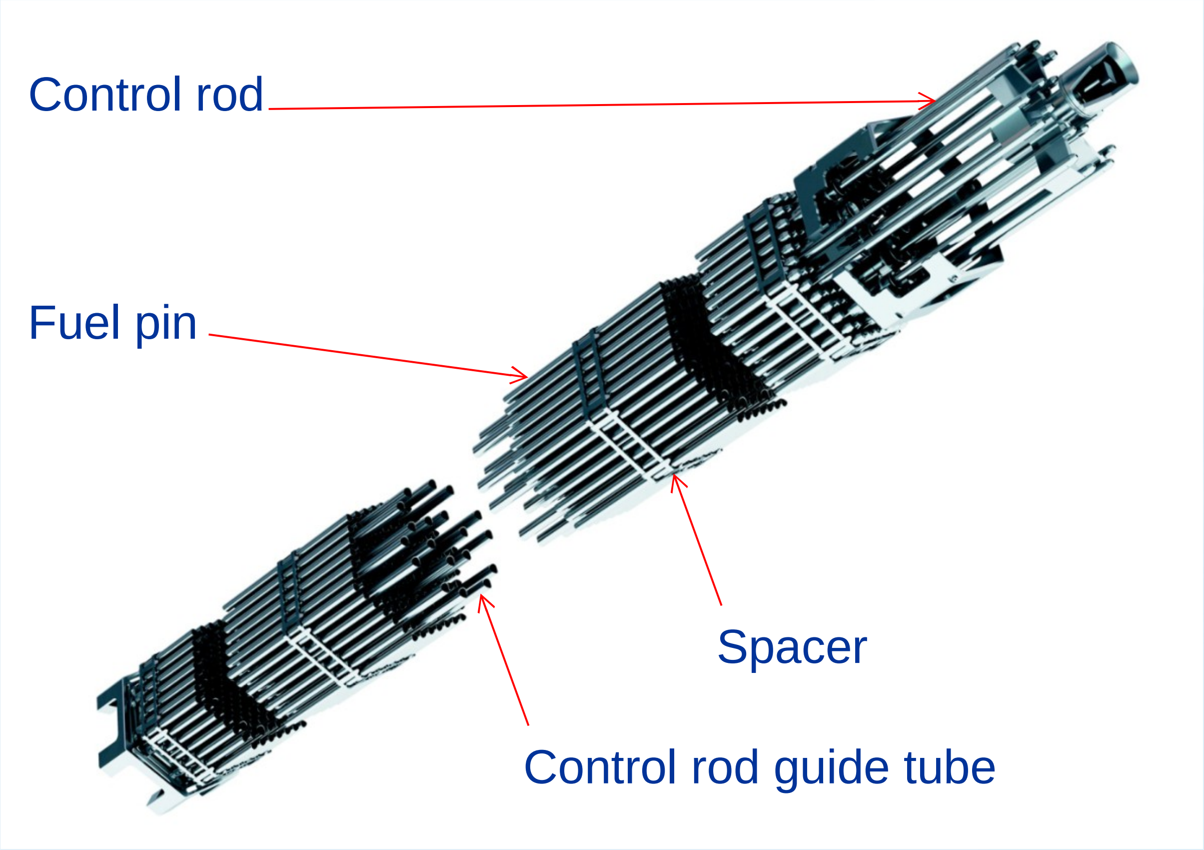


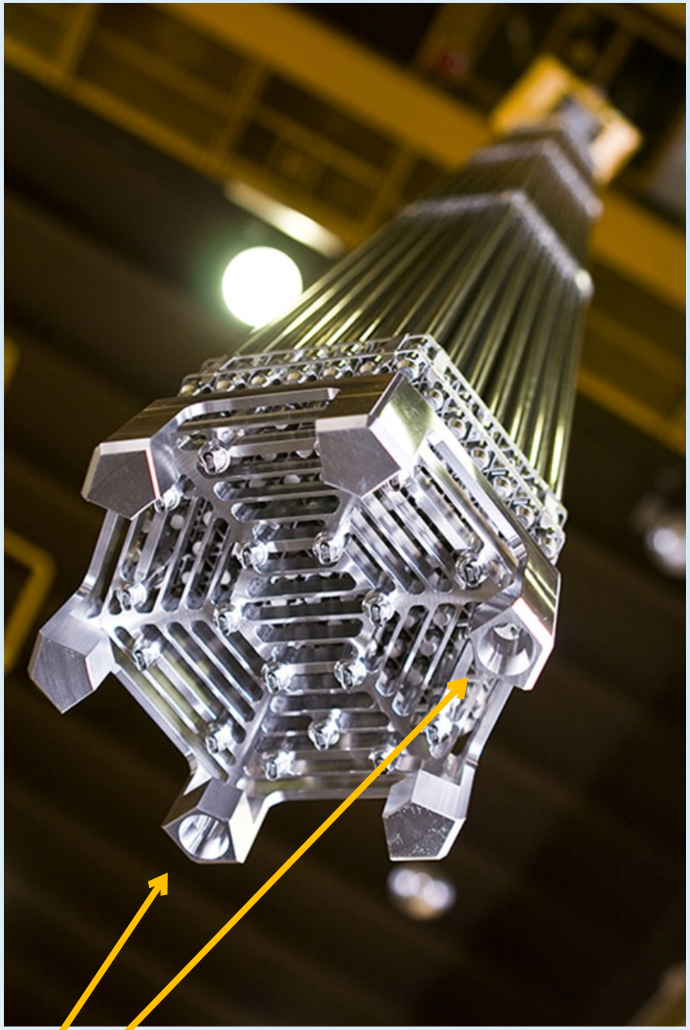
CAREM25 Reactivity and cycle length desing criteria

Shutdown margin > 3000 pcm



CAREM25 Fuel element

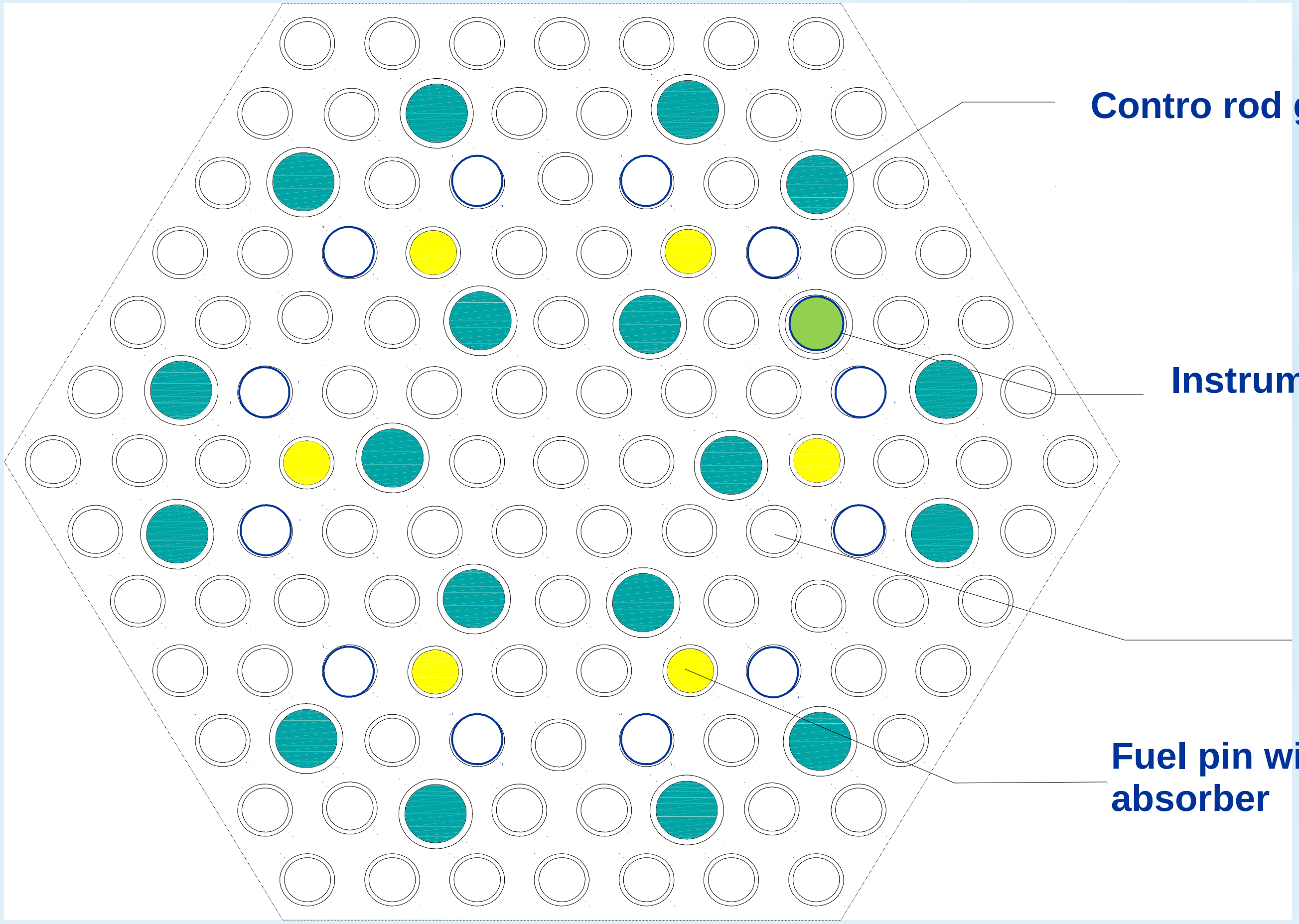




Fuel elements stand on the bottom grid, correctly set on position

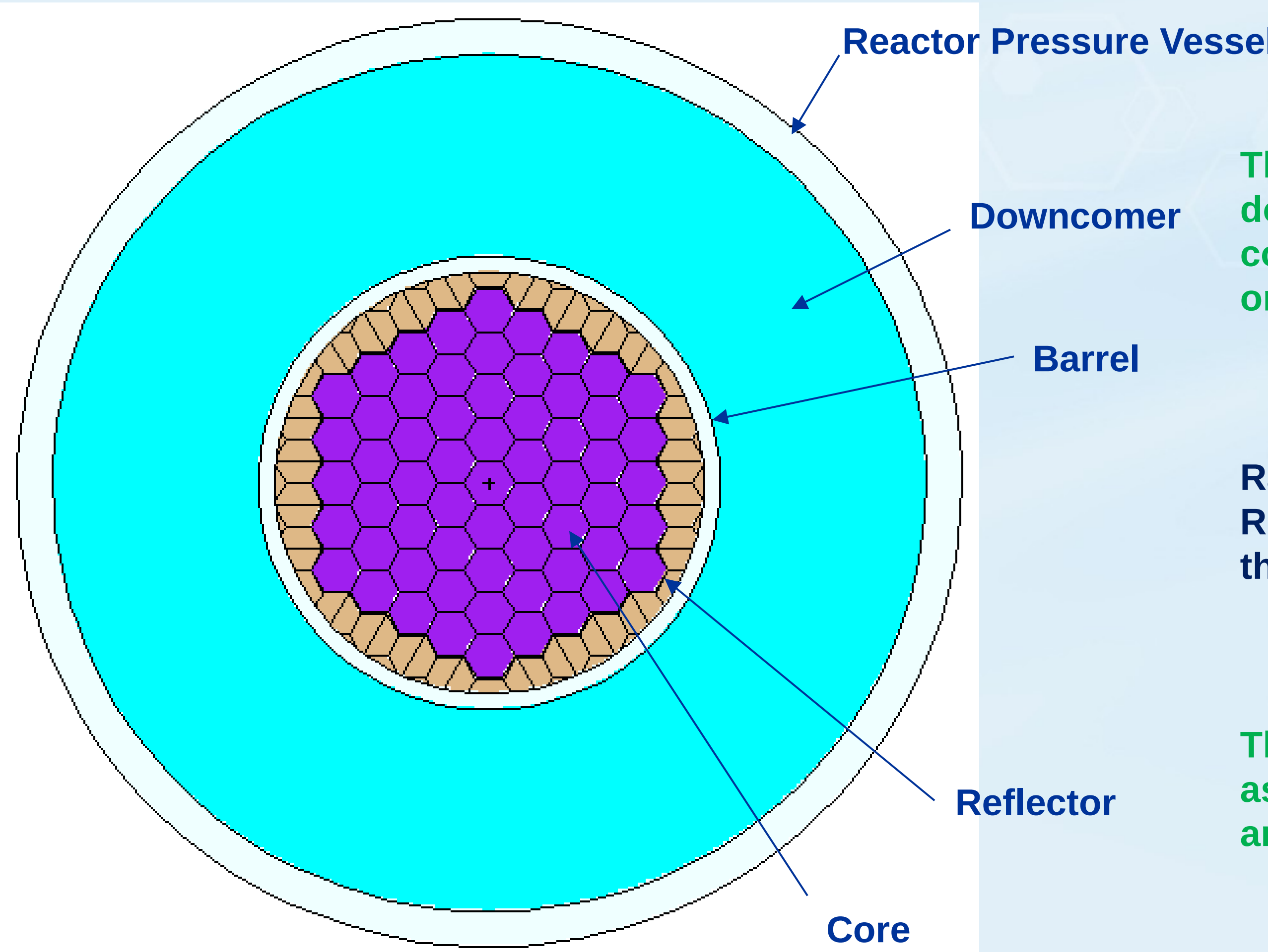
CAREM25 Fuel element

Fuel structure is fixed by guide tubes, and the top and bottom components



Fuel pins are the same diameter and length

CAREM25 Core description

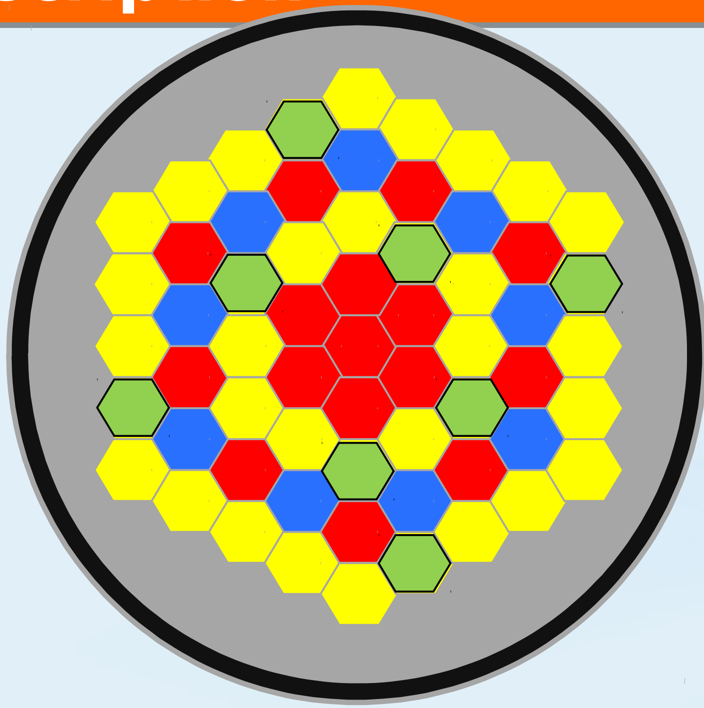


The large water inventory in the downcomer reduces the fast component of the neutron flux on the Reactor Pressure Vessel

Radiation damage on the Reactor Pressure Vessel is lower than in typical PWRs

The stainless steel reflector acts as a shielding between the core and the Reactor Pressure Vessel

CAREM25 Core description



Neutron detectors

Reactivity control system

Fast shutdown system

Fuel elements have the same geometry.
They can be located in any position:
with/without control rods or instrumentation

In core Self Powered Neutron Detectors:
- Vanadium (neutron response)
- Platinum (gamma response)

Total of seven per the active lenght at each location

- Fast shutdown system: Withdrawn in operation
- Reactivity control system: Only seven are needed in operation
- Both conform the First shutdown system

CAREM25 Core behaviour

✓**Cross flow**: Due to the absence of coolant channels and the evolution of the power profiles on operation, the coolant flow is three dimensional and the pattern changes between Beginning and End of Life.

✓**Coupled calculations**: The power distribution and fuel burnup are evaluated in a continuous feedback scheme between neutronic and thermal hydraulics codes.

✓**Stable neutronic-thermalhydraulic feedback**: Any power increase leads to a higher sub-cooled vapor generation which decreases the local moderator density.

✓**Low coolant speed**: Inside the core it is lower than 1 m/s.

✓**Large water inventory**: The downcomer volume reduces the fast neutron flux component (> 1 MeV) on the Reactor Pressure Vessel. However, it affects the response of ex vessel neutron detectors in power transients.

CAREM25 Core behaviour

✓**Fuel design limits**: In spite of the fact that the Power Peaking Factor is below the design limit of 3.0, attention is given to the evaluation of the Departure from Nuclear Boiling Ratio, which is larger than the design limit of 1.7.

✓**Dome pressure and core mass flow**: Depend on the power balance between the core power and steam generators.

✓**Core outlet temperature**: Fixed by the dome pressure (saturation conditions).

✓**Core inlet temperature**: Determined by the core power.

✓**Neutronic-thermalhydraulic feedback**: Due to the coupling of pressure, coolant flow, temperatures and power balance.

CAREM25 Fuel design

✓**Radiation damage**: The fast neutron fluence in fuel components is low enough to guarantee the structural stability.

✓**Power density**: Lower than in typical PWRs.

✓**Cladding**: Designed to withstand volatile fission products, non collapsable

✓**Experimental evaluation**: Irradiation test performed at the Research Reactor in Halden, Norway, on standard and poisoned pellets.

CAREM25 Control and Shutdown

✓**Reactivity control system**: With 16 control rods, not all necessarily inserted during operation.

✓**First Shutdown System**: With 25 control rods. The 16 control rods of the reactivity control system plus 9 fast drop rods.

✓**Secondary Shutdown System**: Boron solution injection, designed to operate only in case of failure of the first shutdown system.

✓**Boron in normal operation**: Not used as routine reactivity control, and not needed in regular shutdown or refuelling.

CAREM25 Monitoring

✓**In core neutron flux detectors**: Self Powered Neutron Detectors Vanadium (delayed response to neutrons in transients) and Platinum (fast gamma response) for monitoring and protection.

✓**Ex vessel neutron flux detectors**: Fission chambers, and BF3 in startup (withdrawn at medium and large core power).

✓**Range of flux detectors**: The proper switching allows to cover all power levels.

✓**Coolant temperature**: Measured at downcomer and dome.

✓**Core mass flow**: Evaluated from the power balance in secondary system and core inlet temperature measurements.

✓**Reactor pressure vessel parameters**: Dome pressure and liquid level.

CAREM25 Summary

✓CAREM25 is the 27 MWe prototype for the commercial unit of 120 MWe.

✓It is a PWR, **boron free** during normal operation and cold shutdown. Boron injection only as a second shutdown system.

✓Its primary circuit is integrated, self-pressurized and cooled by natural convection.

✓It has very low RPV radiation damage

✓Deep N-TH coupled behavior, due to self-pressurization, natural convection and negative feedback.

✓Some traditional design basis accidents excluded by design (like L-LOCA).

✓Long grace period

✓Simple core: 1 type of FA needed for refueling operations.

Thank you!

Questions

1) How many refueling zones are implemented by design?
- 1 (cassette type)
- 2 + center fuel
- 3 + center fuel

2) Which are the available shutdown systems?
- Control rods and boron acid injection
- Control rods and burnable absorbers
- Control rods
- Passive shutdown based on negative feedback

3) Does the fuel include burnable absorbers?
- Yes, boron compound
- Yes, Gadolinium Oxide mixed with fuel
- Yes, Gadolinium Oxide
- No, design is boron free

4) Is it possible a L-LOCA accident?
- Yes, but low probatility
- Yes, but the core has negative void reactivity
- Yes, but at low pressure
- No

5) Is boron or a boron compund, used as a routine reactivity control mechanism?
- Yes, to compensate for burnup
- Yes, as a burnable absorber
- Yes, to lower power peaking factors
- No