IAEA's Nuclear Power Plant Simulators for Education and Training

Simulations for education and training…

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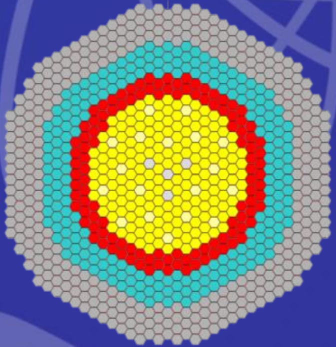

IAEA's Nuclear Power Plant Simulators for Education and Training

Speaker

- Technology development for advanced reactor lines
- Small Modular Reactors
- Trainer of NPP simulators for education and training
- Modelling and simulations
- Technical Project Management

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IAEA's Simulator Programme Overview

Programme objectives and types of simulators

IAEA Programme Objective

PC based basic education and training simulators

"Assist Member States in training nuclear researchers, engineers, and nuclear regulators"

IAEA arranges for:

- Management of existing suite of simulators and development of new
- Distribution of simulation software, and corresponding training materials
- Organization of training courses and workshops

The IAEA established Education & Training Courses based on **active learning (learning-by-doing)** with nuclear reactor simulation computer programs (**basic education and training simulators**) and other toolkits to assist Member States in educating & training their nuclear professionals on physics and technology of nuclear power reactor designs.

❖ Web site (or google "IAEA Simulators"):
https://www.iaea.org/topics/nuclear-power-reactors/nuclear-reactor-simulators-for-education-and-training

Simulators Classification

Nuclear Power Plant Simulators

Basic Education and Training

- Part Task
- Full Plant

IAEA Simulators

Professional Training

- Plant-specific Classroom
  - Part-task
  - Full Plant
- Maintenance /Component (part-task)
- Engineering
- Full-scope

Nuclear education understood to represent a knowledge building process enhancing indepth understanding of nuclear concepts, phenomena and related technologies. **Nuclear training** understood to represent a capacity building measure to enhance the skill sets, attitude and behaviour within the specific environment of an NPP.

Source: IAEA TECDOC 1887

Simulators Classification

Basic E&T Simulators

- Operate on personal computers (PC) and are provided for a broad audience of technical and non-technical personnel as an introductory educational & training set of tools
- Configuration suited to classroom & self-learning tool as complement to textbooks and manuals
- Provides subsystem training and overall plant training (startup, shutdown, malfunctions)

Full Scope Simulators

- Indispensable in the licensed training of the NPP control room operators on plant operation in a control room environment
- Procedure based, cognitive skill based, team work

Engineering Simulators

- Computer based simulation tools able to calculate and display in real time the physical parameters of NPP
- Maintenance and retraining

What are Basic E&T Simulators?

Basic E&T Simulators

ROLE

- Provide initial and fundamental educational training about NPP (including NPP personnel before NPP is built & full scope simulator in service)
- Provide knowledge of system interfaces, integration and interactions

SPECIFICS

- Relatively low cost and affordable
- Can use highly portable, standard PC platforms
- Mathematical models are easily configurable and provide flexibility of use
- Can use graphic icons, control pop-ups, time trends for user interfaces instead of hardwired panels

Application of Basic E&T Simulators

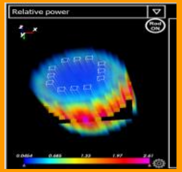
Basic E&T Simulators

The use of the IAEA PC based basic E&T simulators is aimed at **enhancing understanding of nuclear technologies through learning-by-doing:**

Hands-on experiential training is highly suitable for operators, maintenance technicians, suppliers, regulators and engineers.

Train the Trainer | Understand the technology

IAEA's Basic Education & Training Simulators:



Simulate nuclear power plants' operation and responses to various inputs (transients, accidents) with illustrative screens are a **state-of-the-art learning tool**

Designed to provide insight and understanding of the general design and operational characteristics of various power reactor systems

Focus on education and training in classrooms, and not licensing or reactor operator training, or benchmarks of other computer codes and methods

IAEA's Collection of Simulators

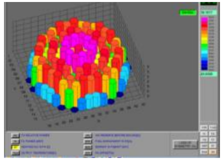
Basic Education & Training Simulators

PC Based Simulators

Innovative Energy Systems

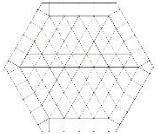
- SFR
- HTGR

Under Development



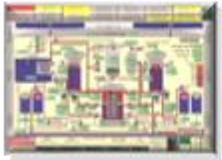
Part-task Simulators (1)

- Micro-Physics Simulator



Pressurized Water Reactors (4)

- Korean-type PWR
- Russian-type PWR
- Advanced Passive PWR
- Severe Accident Simulator Under Development
- iPWR



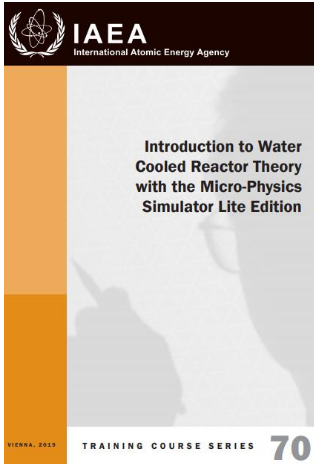
Boiling Water Reactors (2)

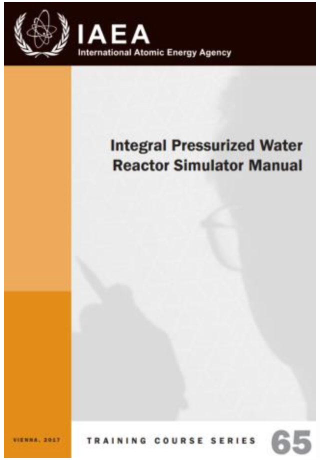
- Advanced BWR
- Advanced Passive BWR

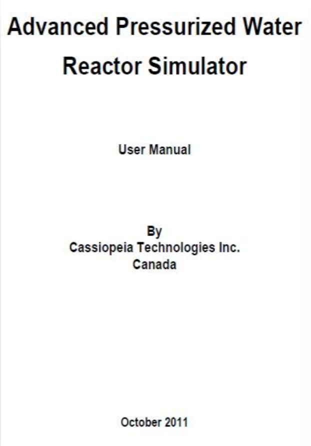
Pressurized Heavy water Reactors (2)

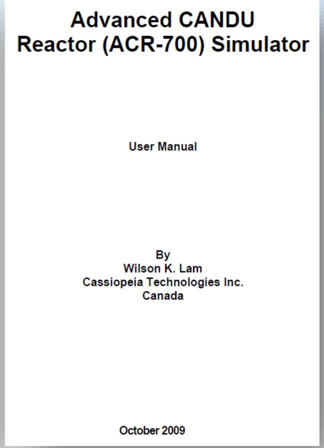
- Conventional PHWR
- Advanced PHWR

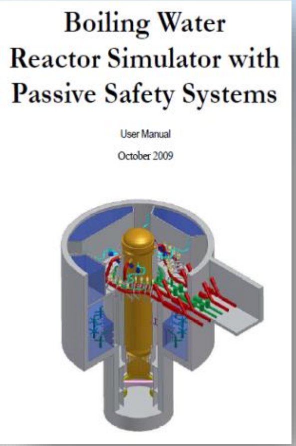
IAEA E&T Simulators: Publications





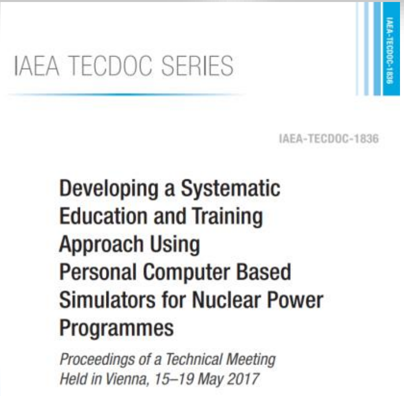


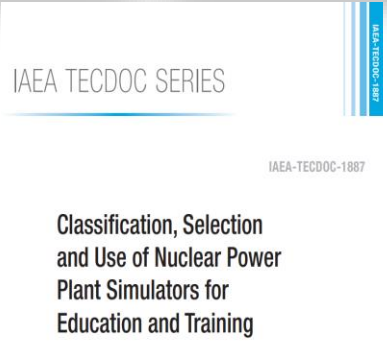


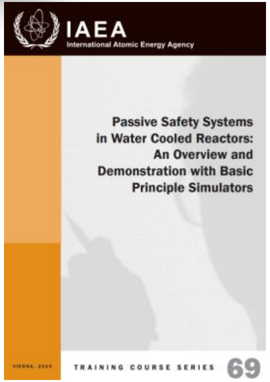


IAEA publications in support of education and training courses:

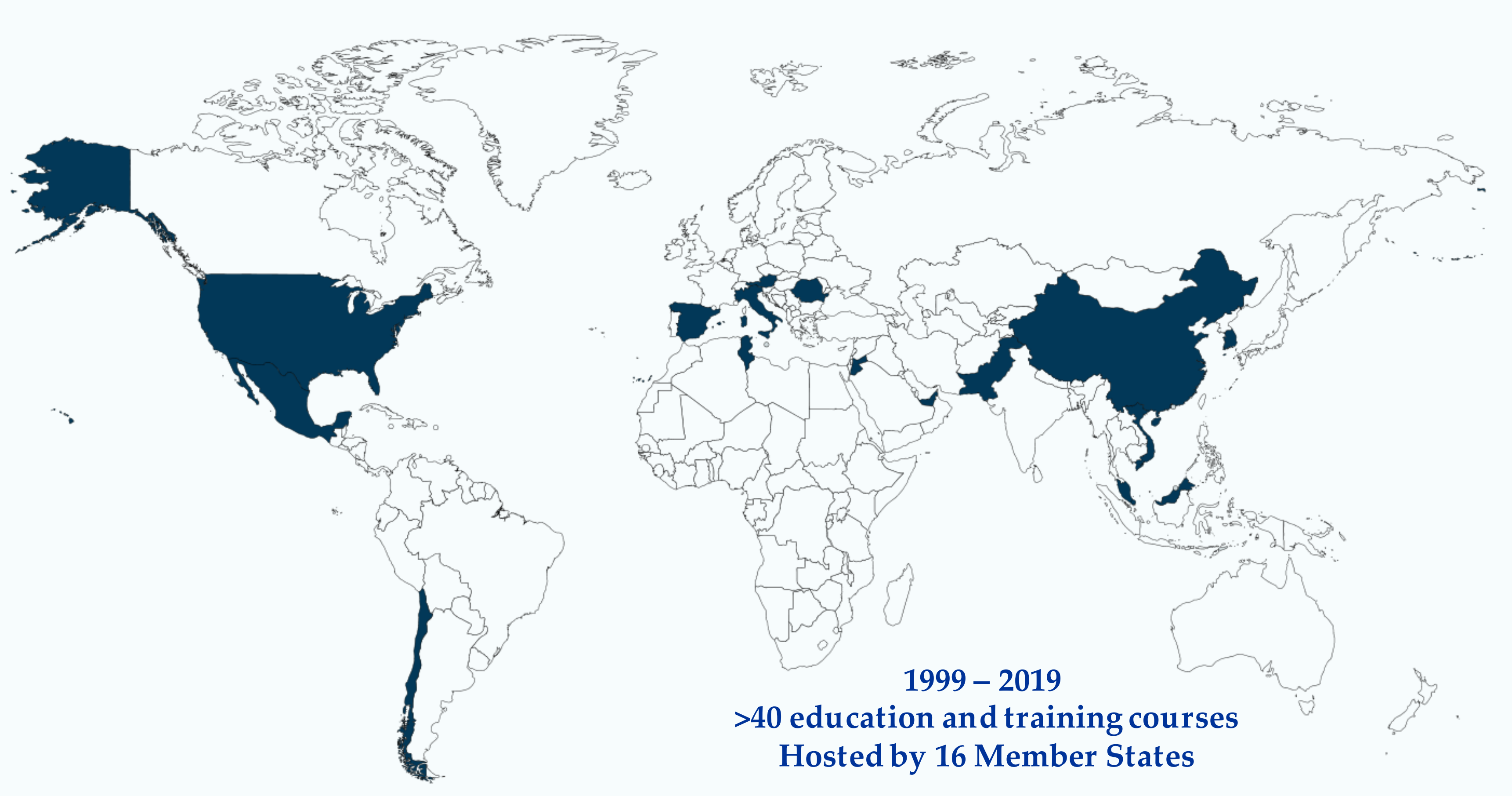
- Simulator manuals
- Simulator reports
- Training materials







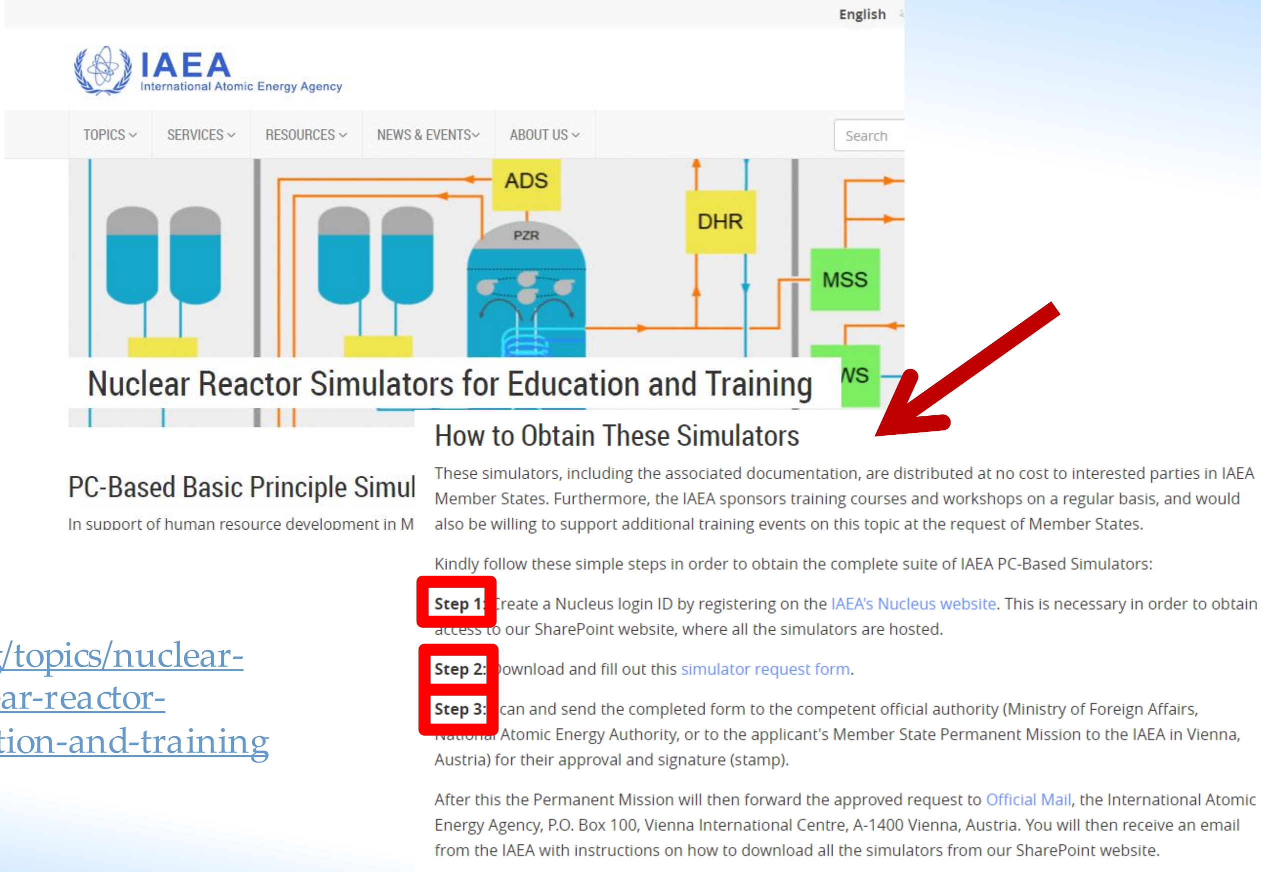
Training courses



How to Obtain the Simulators

3 step process to download IAEA's PC based basic principle simulators

How to Obtain these Simulators?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Nuclear Reactor Simulators for Education and Training

PC-Based Basic Principle Simul

In support of human resource development in M

How to Obtain These Simulators

These simulators, including the associated documentation, are distributed at no cost to interested parties in IAEA Member States. Furthermore, the IAEA sponsors training courses and workshops on a regular basis, and would also be willing to support additional training events on this topic at the request of Member States.

Kindly follow these simple steps in order to obtain the complete suite of IAEA PC-Based Simulators:

Step 1: Create a Nucleus login ID by registering on the IAEA's Nucleus website. This is necessary in order to obtain access to our SharePoint website, where all the simulators are hosted.

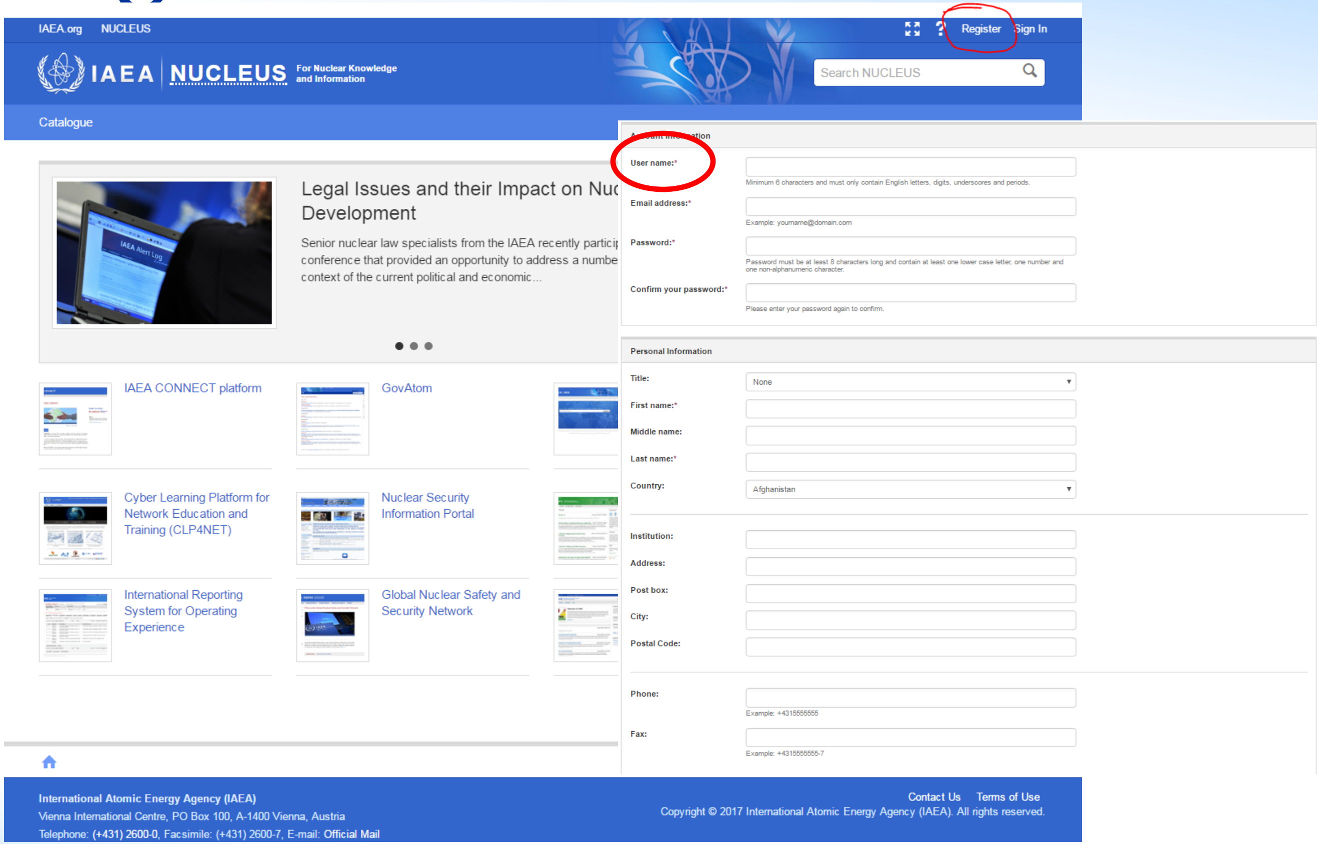
Step 2: Download and fill out this simulator request form.

Step 3: Scan and send the completed form to the competent official authority (Ministry of Foreign Affairs, National Atomic Energy Authority, or to the applicant's Member State Permanent Mission to the IAEA in Vienna, Austria) for their approval and signature (stamp).

After this the Permanent Mission will then forward the approved request to Official Mail, the International Atomic Energy Agency, P.O. Box 100, Vienna International Centre, A-1400 Vienna, Austria. You will then receive an email from the IAEA with instructions on how to download all the simulators from our SharePoint website.

https://www.iaea.org/topics/nuclear-power-reactors/nuclear-reactor-simulators-for-education-and-training

Step1: Register on Nucleus



https://nucleus.iaea.org

Step 2: Fill in the request Form

IAEA

Request Form

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Basic Principle Nuclear Power Plant Simulators for Education

The IAEA sponsors the development and distribution of basic principle nuclear power plant (NPP) simulators for education. These simulators provide insight into fundamental characteristics, system interactions, and plant responses to both transients and accidents. The purpose of this project is to provide quality educational tools for both university professors and trained engineers in all IAEA Member States involved in teaching topics related to nuclear reactor fundamentals and safety. These simulators, including associated documentation, are distributed at no cost to interested parties in IAEA Member States.

Additionally, the IAEA regularly arranges training courses in which experts provide detailed training in the use of these simulators while explaining the fundamentals of reactor physics and technology.

Access to our simulators and related training materials will be given through a SharePoint located within the ***IAEA's Nucleus website***. As a result, requestors will first need to register to obtain a Nucleus login ID at the following site: https://nucleus.iaea.org/Pages/default.aspx

By my signature below, I acknowledge that:

- I understand that the application of the simulators is limited to providing general response characteristics of selected reactor systems and that the IAEA simulators are not intended to be used for plant-specific purposes such as design, safety evaluation, or operator licensing training.
- The IAEA simulators are made available to me, and to colleagues in my organization, for teaching and self-learning in nuclear reactor technology. All other rights, including all ownership rights, remain with the owners of the simulation software.
- The IAEA simulators will NOT be shared outside my organization, nor distributed to individuals from other countries.
- The IAEA simulators cannot be sold or used to support training and educational events where a fee is charged for participation, unless specifically approved by the IAEA.

(Please type the following information) Date of request: ____________

Mr☐/Ms☐ First Name: ________________ Last Name: ____________________

Business / Home Address: ____
____

Organization: ____________________________________

Citizenship/Country: ________________________________

Intent of use: ____

Expected number of users within organization: ___________________________

Nucleus Username: _____________________ E-mail Address: _________________________

Telephone No.: __________________ Signature: ____________________________

Approval Authority: ______________________

Approval Authority Signature: ________________

Step 3: Send it for Approval

- After completely filling in the request form, it should be sent to the competent official authority* for their approval
- The national authority will then send it to us after the approval.
- Once the request comes to NPTDS, we provide access to the Simulator Portal, where all the simulators could be downloaded.

*Ministry of Foreign Affairs, National Atomic Energy Authority, or to the applicant's Member State Permanent Mission to the IAEA in Vienna, Austria

Distribution

IAEA.org NUCLEUS Browse Page

BATRA, Chirayu

IAEA | Basic Principle NPP Simulators for Education

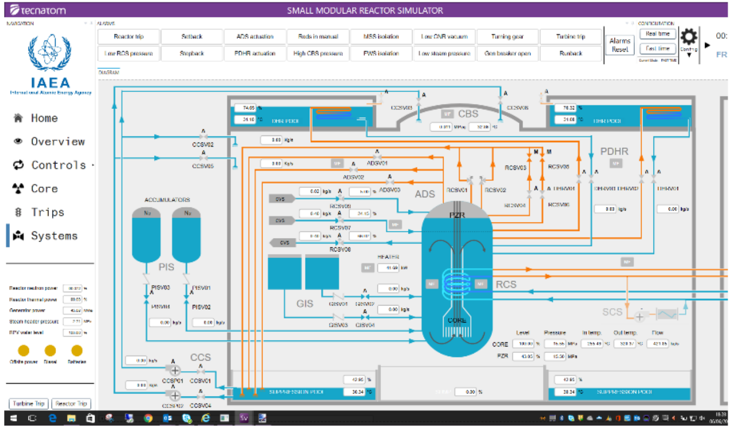
Search this site

PWR BWR PHWR Reactor Core Simulator iPWR

Home

iPWR Simulator

Integral Pressurized Water Reactor Simulator



Installation Files:

Windows 32-bit installer: SMR32.zip

Windows 64-bit installer: SMR64.zip

Works on Windows 7, 8 and 10

User Manuals:

Download User Manual: TCS-65_User_Manual.pdf

Download Exercise Handbook: TCS-65-Exercise_Handbook.pdf

Thank you

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Twitter:
@chirayubatra

"I have been driven by the conviction that much more than 1 percent of the energy contained in uranium must be utilized if nuclear power is to achieve its real long-term potential."

- Enrico Fermi

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G4SR Webinar: The IAEA Passive BWR Simulator Demonstration to Illustrate ESBWR Features, the Basis for BWRX300 Design

Evolution of BWRs – Conventional BWR, ABWR, ESBWR and BWRX – 300

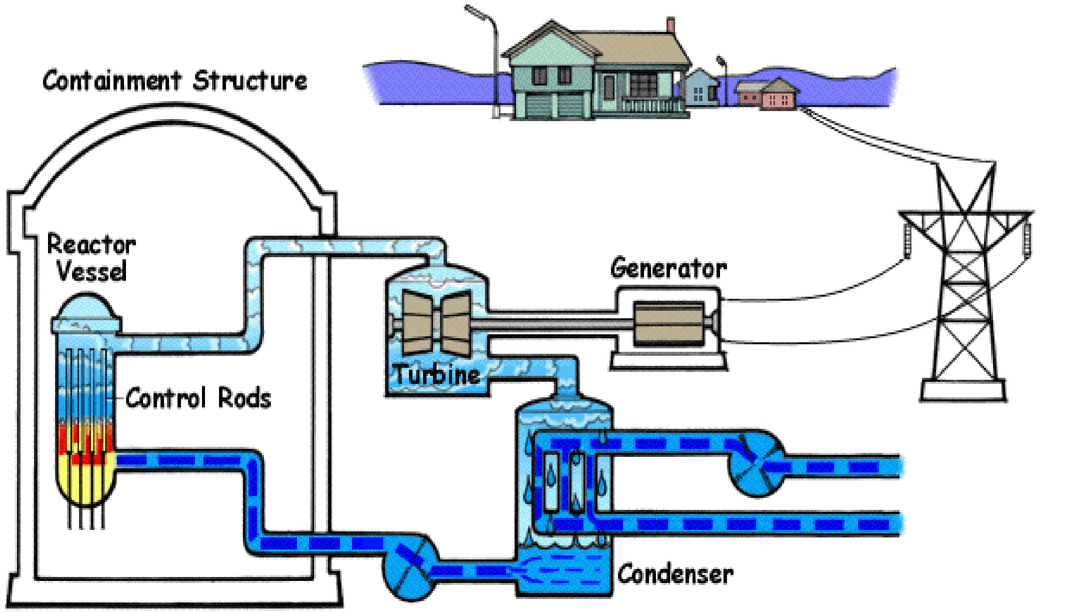
Dr Haseeb ur Rehman, Dr Tatjana Jevremovic

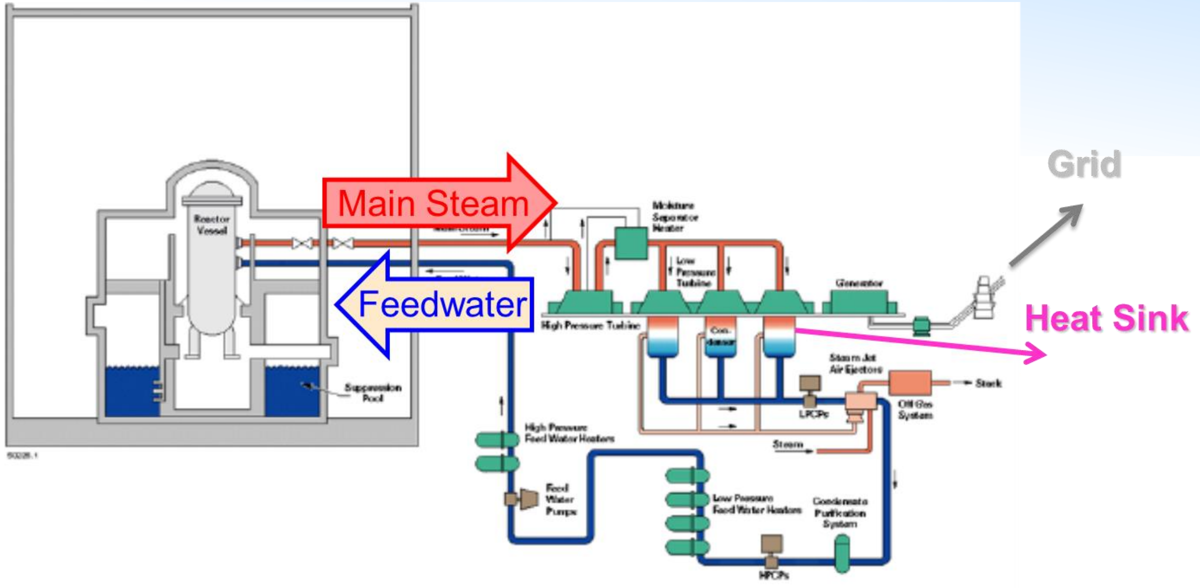
IAEA Department of Nuclear Energy
Nuclear Power Technology Development Section
Team – Water Cooled Reactor Technology Development

OUTLINE

1. Basic Concept
2. Nuclear Reactors’ Generations (Classification)
3. Early Boiling Water Reactor Designs
4. Advanced Boiling Water Reactors

Basic Concepts: Electricity Generation





- Second most common power reactor type
- Water: coolant & moderator
- Fission generates heat that causes cooling water to boil producing steam
- Steam drives the turbines directly: direct-cycle
- Cooling water is at low pressure, ~ 7.6 MPa (it boils in the core at ~ 285 C)

https://www.nrc.gov/reading-rm/basic-ref/students/animated-bwr.html

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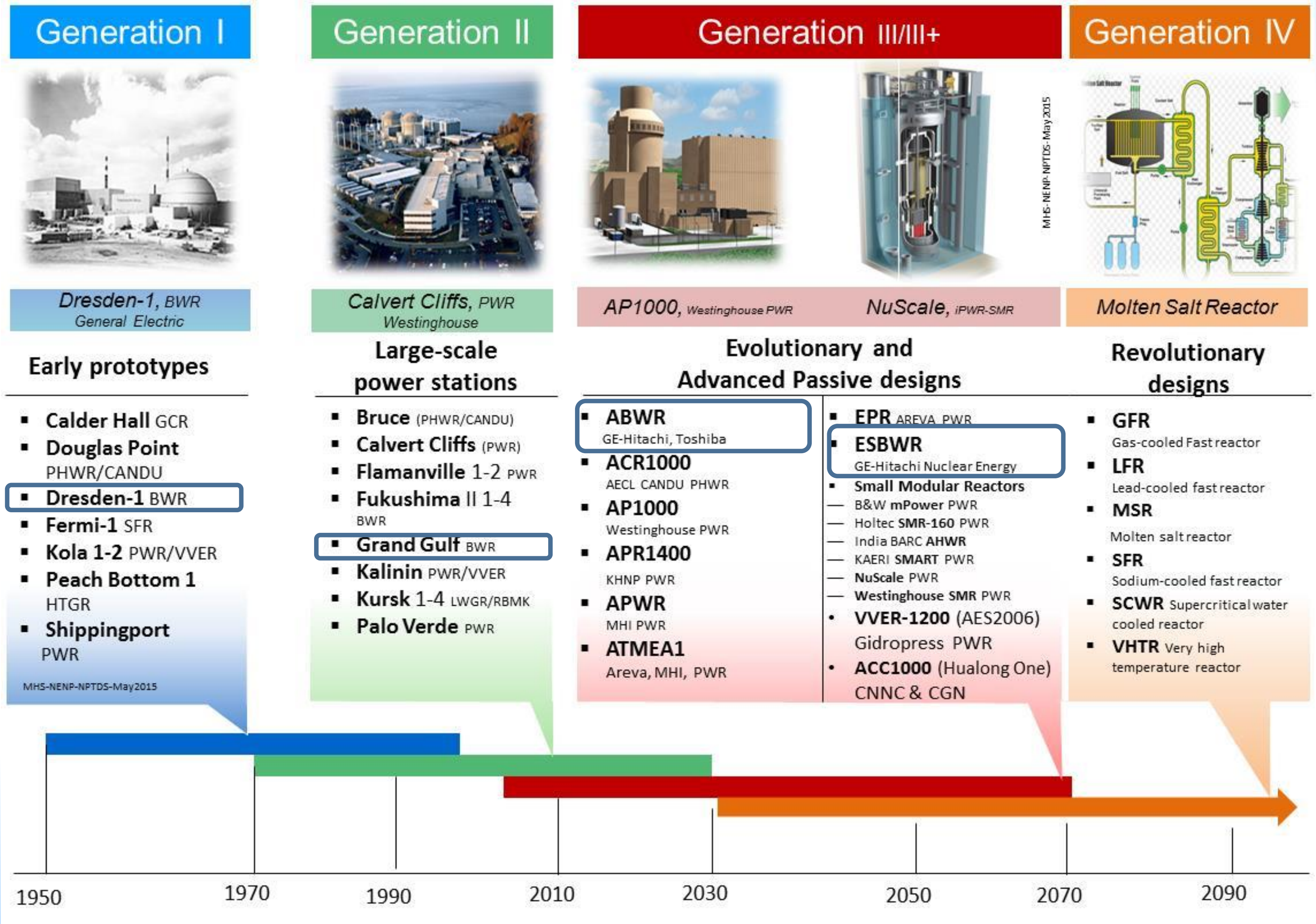
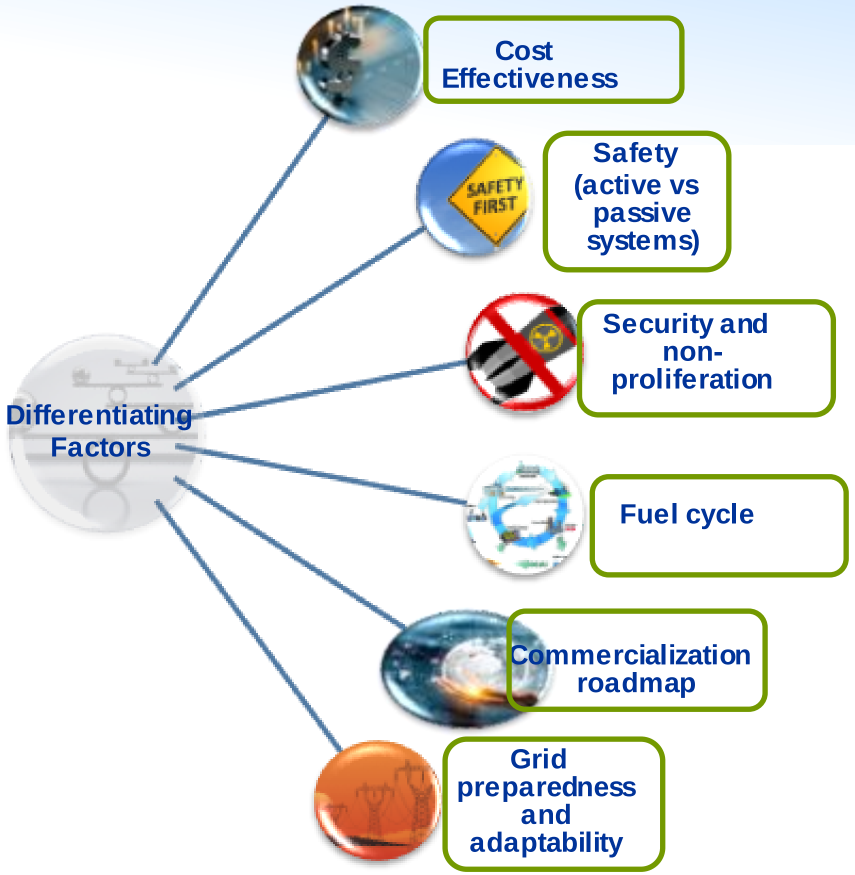
Nuclear Reactor Generations (Classification)

Early Boiling Water Reactor Designs

Advanced Boiling Water Reactors

Nuclear Reactor Classification





Advanced Nuclear Reactor Designs

*It is a design of **current interest** with substantial improvements over previous/existing designs.*

Advanced NPP Design

- Current Advanced NPP Designs
- Evolutionary NPP Designs
- Innovative NPP Designs

Current Advanced NPP Designs (Generation III)

- Advanced designs that have already been built, operated or are under construction.

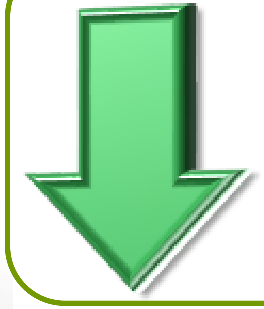
Evolutionary NPP Designs (Generation III+)

- Advanced designs that achieve improvements over existing designs thorough small modifications (maintaining proven designs)

Innovative NPP Designs (Generation IV)

- Advanced designs with conceptual changes in design approached or configuration comparing with existing practices

Common Goals



Economic Aspect:
Reduce Cost

Safety Aspect:
Improve Safety

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Nuclear Reactor Generations (Classification)

Early Boiling Water Reactor Designs

Advanced Boiling Water Reactors

History- All in One

Generation-I

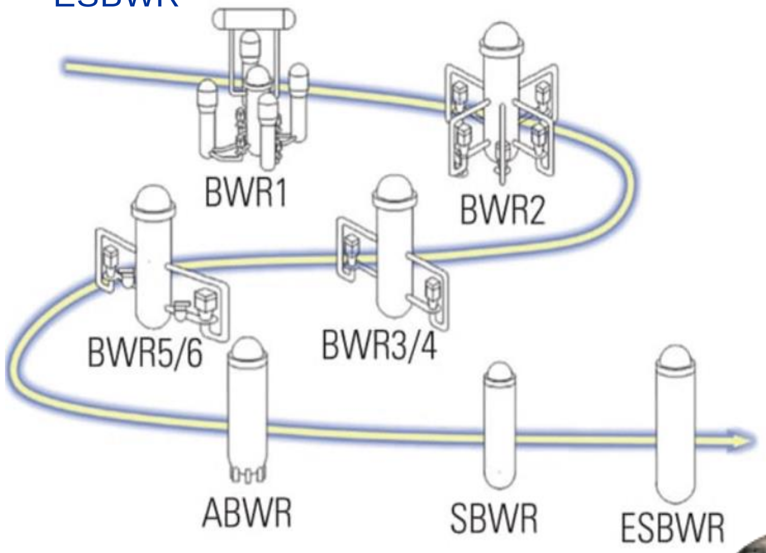
- VBWR
- BWR1 (Dresden)

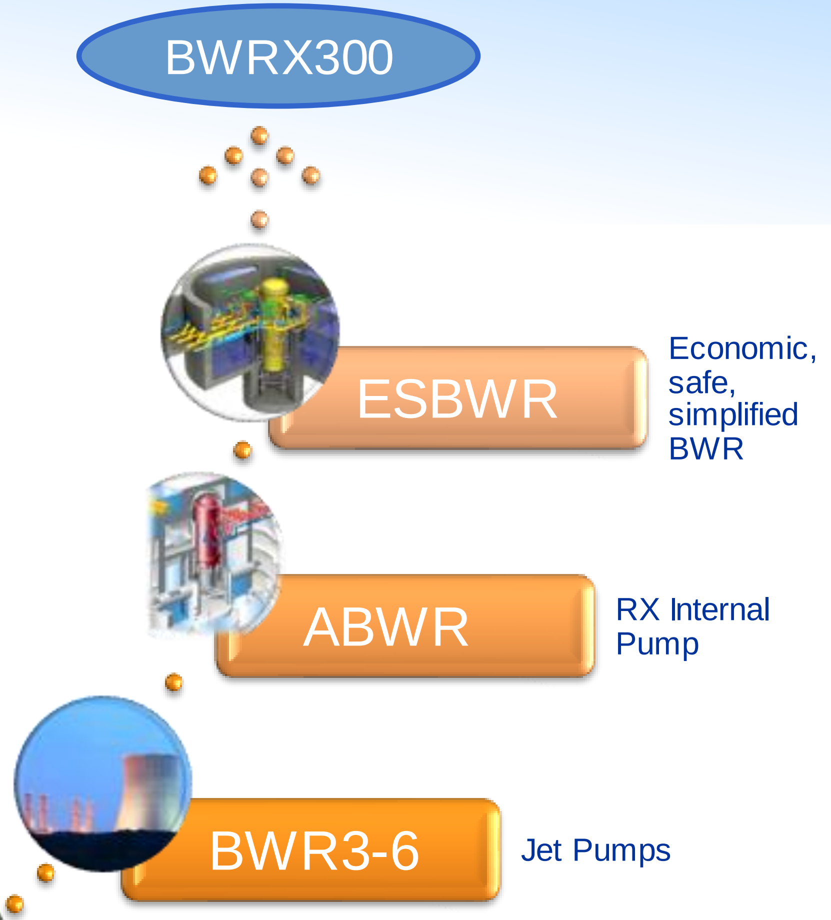
Generation-II

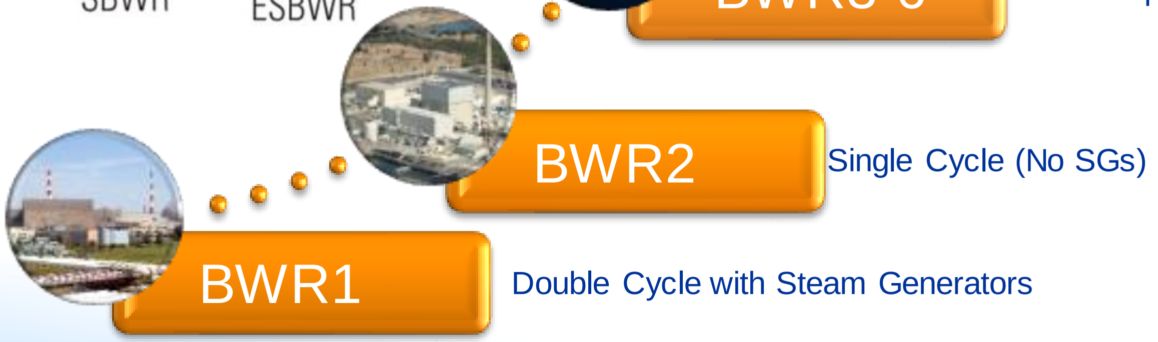
- BWR2 to BWR6

Generation-III/III+

- ABWR
- SBWR
- ESBWR





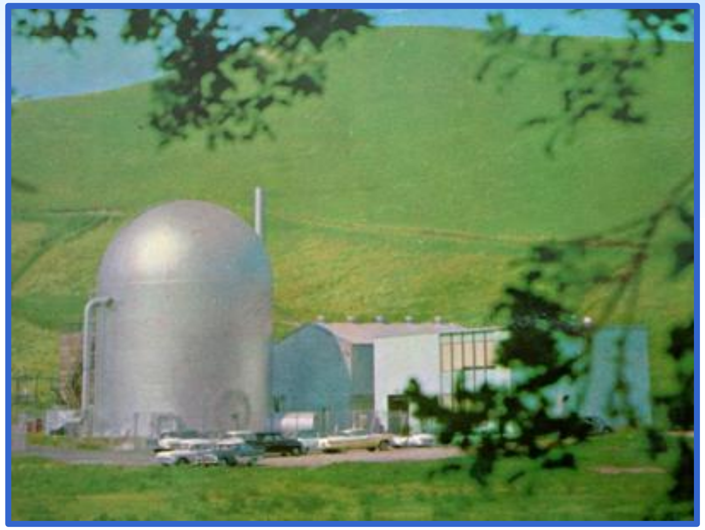


Early Prototype Reactors

<u>Vallecitos Boiling Water Reactor (VBWR)</u>

- Designer: General Electric
- Built in 1957 near San Jose, California
- 1st ever commercial BWR; 5 MWe supplied to Pacific Gas & electric grid
- Operating pressure: 7 MPa

Capable of producing safe electricity



With this success, GE pursued Dresden

<u>BWR1</u>

- Introduced in 1955
- 1st commercial plant in 1960 (Dresden 1)
- Use of dual steam cycle, steam drums, and steam generators
- Dry containment

US NRC
https://www.flickr.com/photos/nrcgov/17859259621

Commercial Power Reactors

BWR2

- Operation in 1969
- Internal steam separation
- 5 recirculation loops
- Lower power density core
- Flow control and load following
- Mark I containment

BWR3

Fukushima Daiichi Unit 1

- Operation in 1971
- Addition of jet pump
- 2 recirculation loops
- Lower power density core
- Improved ECCS with spray and flooding
- Mark I containment

BWR4

- Operation in 1972
- Internal steam separation
- Increase power density core
- Mark I or II Containment

Fukushima Daiichi Unit 2-5

Gradual Evolution

Commercial Power Reactors

BWR5 Fukushima Daiichi Unit 6

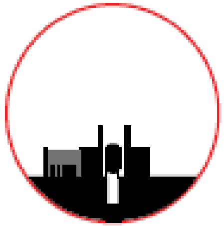
- Operation in 1977
- Improved safety systems
- Recirculation flow control valves
- Mark II containment

BWR6

- Operation in 1978
- Compact control room
- Solid state nuclear PPS
- Improved fuel safety margins
- Mark III containment

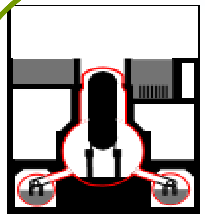
Gradual Evolution

Evolution of Containment Structure



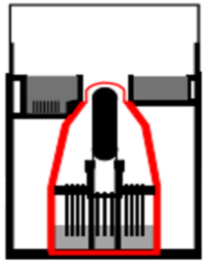
Dry

- No suppression pool
- Houses a steam separator and secondary steam generator for low pressure steam



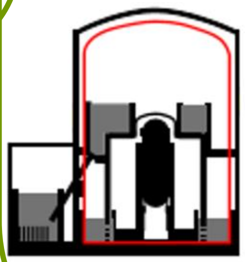
Mark I

- First pressure suppression design
- Inverted light bulb (dry well)
- Torus shaped (wet well)
- Steam vents between dry and wet wells



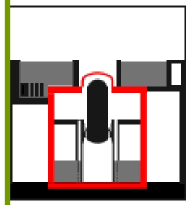
Mark II

- Truncated cone shaped dry well
- Cylindrical wet well is located directly below the dry well
- Increase size of dry well provides more room for steam and ECCS piping



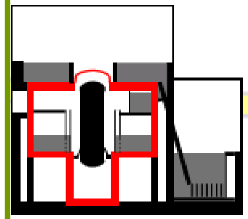
Mark III

- A major improvement in simplicity
- Cylindrical drywell design is used to confine steam during pipe rupture
- Released steam is channeled to wet well by horizontal vents



ABWR

- Much smaller than Mark III due to elimination of recirculation loops



ESBR

- Like ABWR slightly increased size

Early Prototype — *Commercial Power Reactors* — *Advanced Reactors*

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Nuclear Reactor Generations (Classification)

Early Boiling Water Reactor Designs

Advanced Boiling Water Reactors

Advanced Boiling Water Reactor (ABWR)

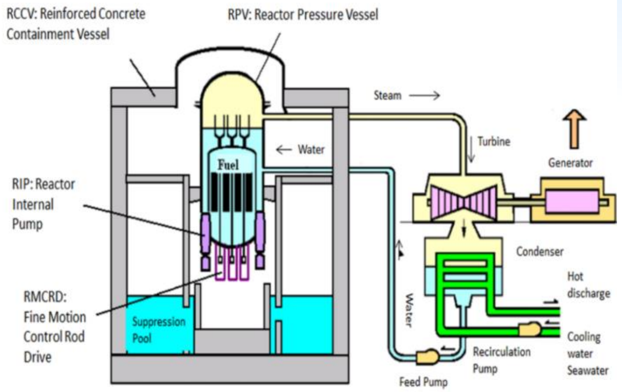
The design is based on integration of the most advanced and proven technologies to enhance construction, operation and maintenance.

ABWR

Designer	**GE-Hitachi**
Thermal power	3,926 MW(th)
Electrical power	1,420 MW(e)
Design status	In operation
Plant design life	60 years
Coolant	Light water
Moderator	Light water

Safety goal

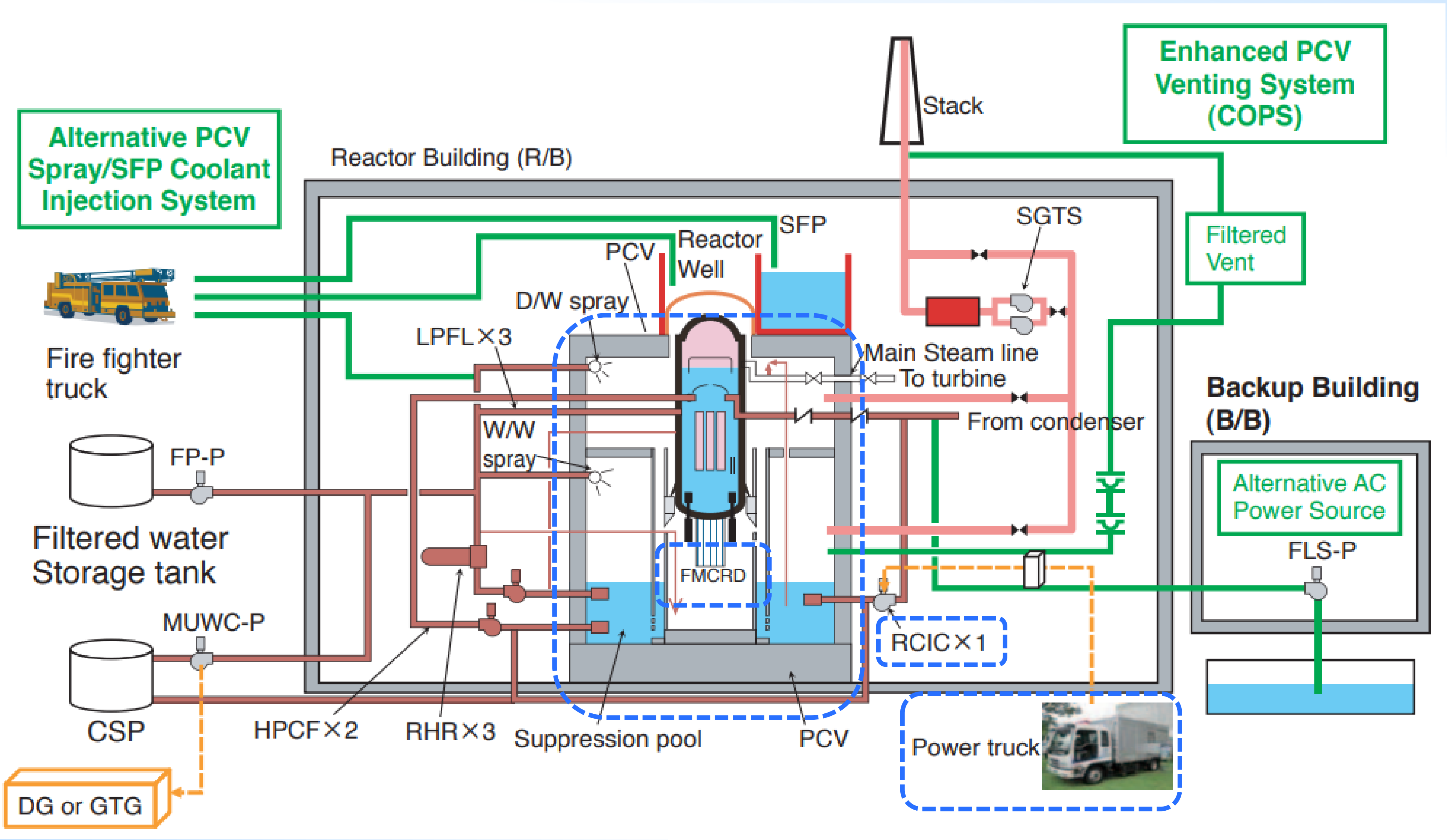
Core damage frequency <	10^{-5} /Reactor year
Large early release frequency <	10^{-6} /Reactor year
Occupational radiation exposure <	1.0 person-Sv /Reactor year



- **Emergency Core Cooling System (ECCS)** has improved in many areas.
  - ECCS is divided into 3 divisions with a high and low-pressure injection pump in each division whereas previous BWRs had 2 divisions.
  - Elimination of external recirculation pipes by using RIPs makes ECCS network smaller and decreases the possibility of LOCA.
- **Anticipated Transient Without Scram (ATWS)** countermeasures were diversified.

https://www.gov.uk/government/consultations/gda-of-hitachi-ge-nuclear-energy-ltds-uk-advanced-boiling-water-reactor/consultation-summary-document

ABWR – Main Features



Economic Simplified Boiling Water Reactor (ESBWR)

Simplified plant design enhances construction, operation, and maintenance.

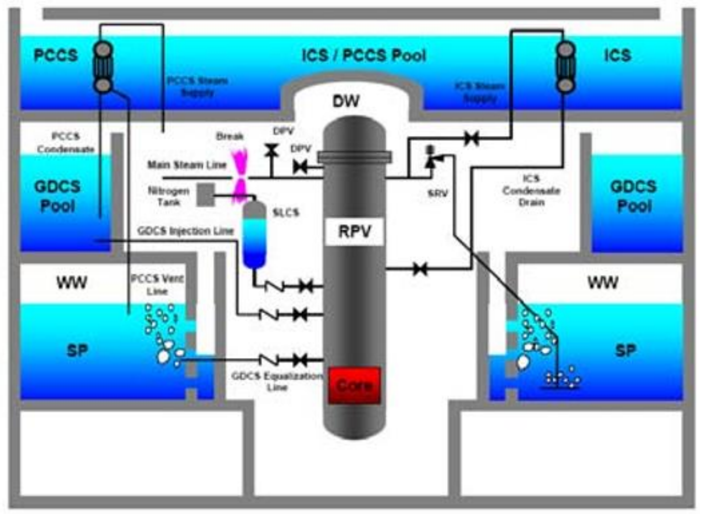
ESBWR

(Economic Simplified Boiling Water Reactor)

Designer:	GE-Hitachi
Thermal power:	4500 MW(th)
Electrical power:	1600 MW(e)
Design status*:	Licensed
Plant design life:	60 years
Coolant:	Light water
Moderator:	Light water

Safety goal

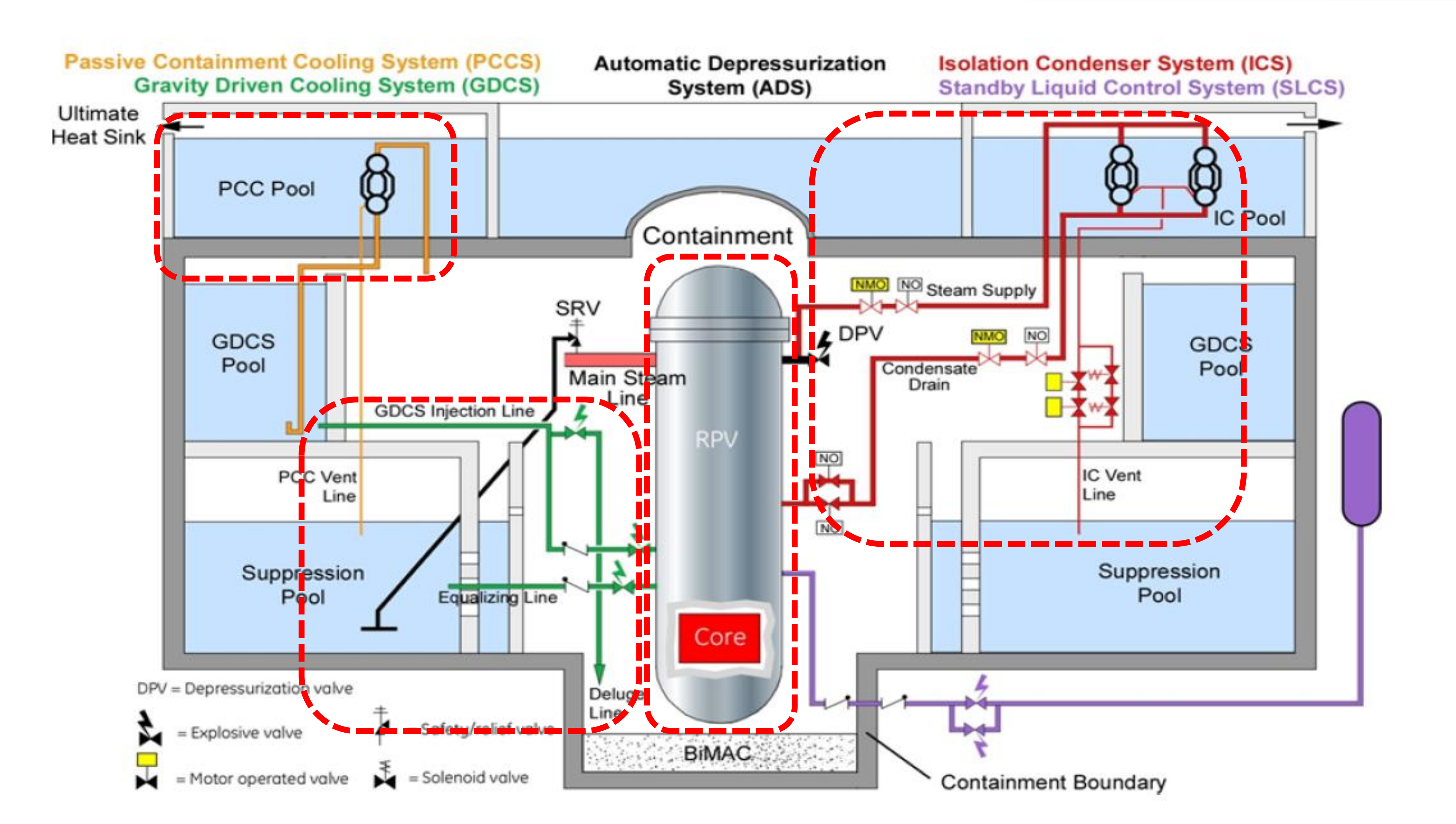
Core damage frequency <	10^{-6} /Reactor year
Large early release frequency <	5×10^{-8} /Reactor year
Occupational radiation exposure <	0.5 person-Sv /Reactor year



- ESBWR is designed to utilize inherent margins to remove system challenges
  1. **Normal operating system's ability** is **enhanced**
  2. **Passive safety-related system** are used
- Passive safety systems require no operator action for **72 hours** following a design basis accident

https://www.gov.uk/government/consultations/gda-of-hitachi-ge-nuclear-energy-ltds-uk-advanced-boiling-water-reactor/consultation-summary-document

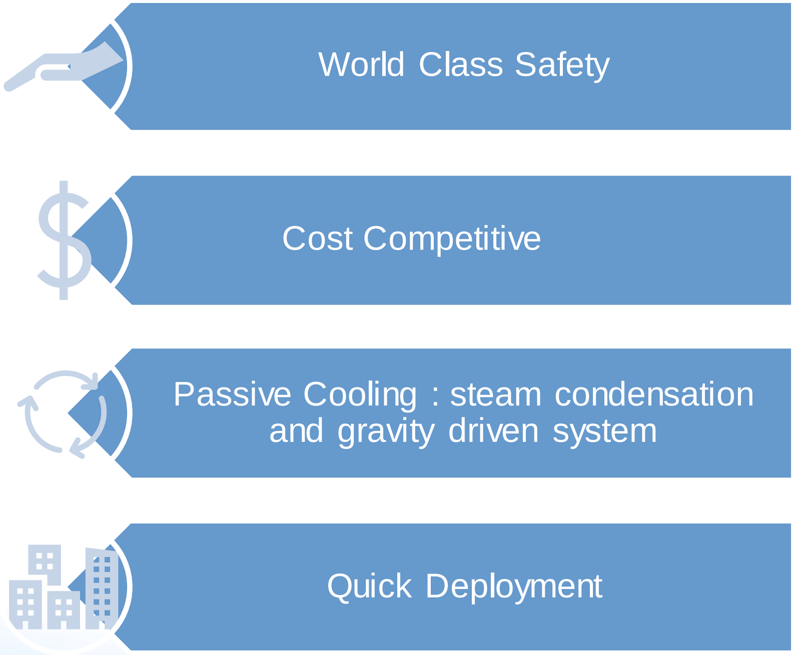
ESBWR – Main Features

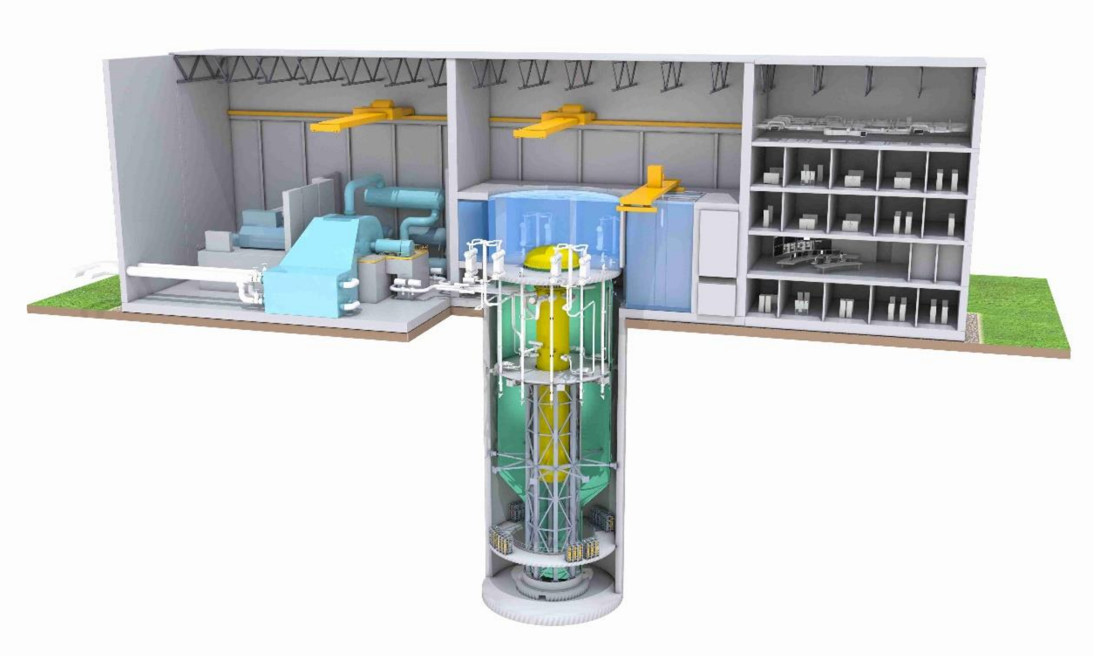


BWRX300 – Small Modular Reactor

- It is a 10th Generation SMR Boiling Water Reactor crated by General Electric – Hitachi.
- SMR evolution of the ESBWR

Characteristic Features





BW RX-300 (gepow er.com)

IAEA
International Atomic Energy Agency

Thank you!

G4SR Webinar - The IAEA Passive BWR Simulator Demonstration to illustrate ESBWR Features

Wilson Lam Wilson@cti-simulation.com

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CNS G4SR Technology Division Chair

CTI Simulation International Corp.

www.cti-simulation.com

In Cooperation with the IAEA

Disclaimer

- Chatham House Rules apply – materials presented here are solely from the speaker for educational purposes.
- The Passive BWR Educational Simulator is based on the ESBWR Design, derived from public domain information e.g. IAEA sources; the ESBWR Design Control Documents (DCD), and others etc.
- The Passive BWR simulator's fidelity has been checked by the IAEA BWR experts.
- While it mimics the realistic dynamic behavior of the ESBWR plant in normal operation, AOO and DBA, it is not identical to ESBWR's performance. Some plant dynamics will differ from the official ESBWR dynamic analysis due to simplification in modelling.
- More importantly, this Webinar is about ESBWR, and NOT about BWRX-300.

Context

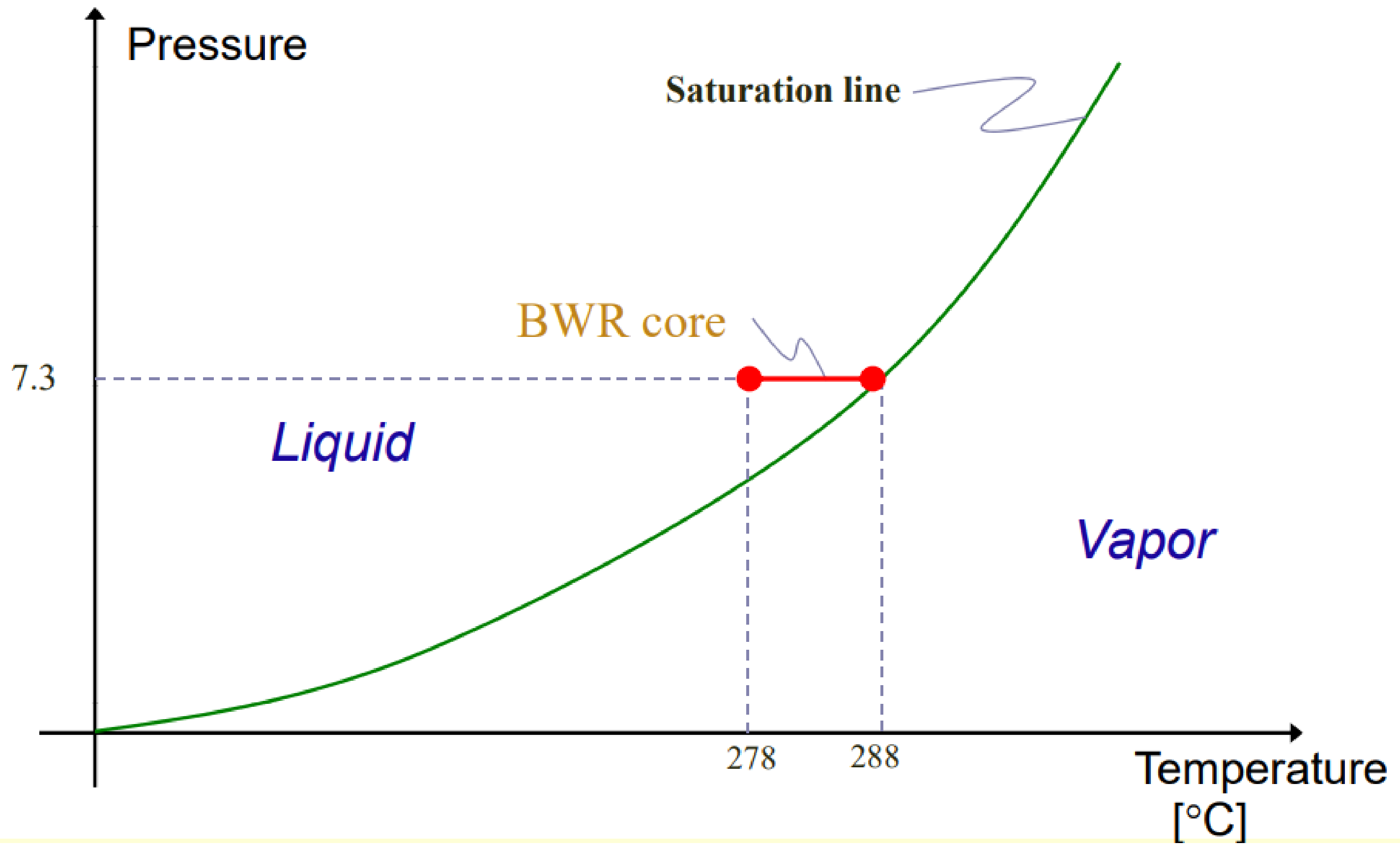
- This session presents the overview of Passive BWR design features: Direct cycle; Core Design; Fuel Assembly; Core Flux Shape properties; BWR Reactor Vessel, BWR Coolant Flow Circulation System; BOP; Safety System & Containment; Control Systems, etc.
- The intent is NOT to provide a complete overview of all the ESBWR design features, but rather, to present those key generic passive BWR design features that the Simulator models: neutronics, thermal hydraulics, safety systems, controls, and more importantly the passive safety features. The pertinent Simulator Display Screens are used to associate with the relevant BWR systems and controls.

Advanced Passive BWR Plant - generic features

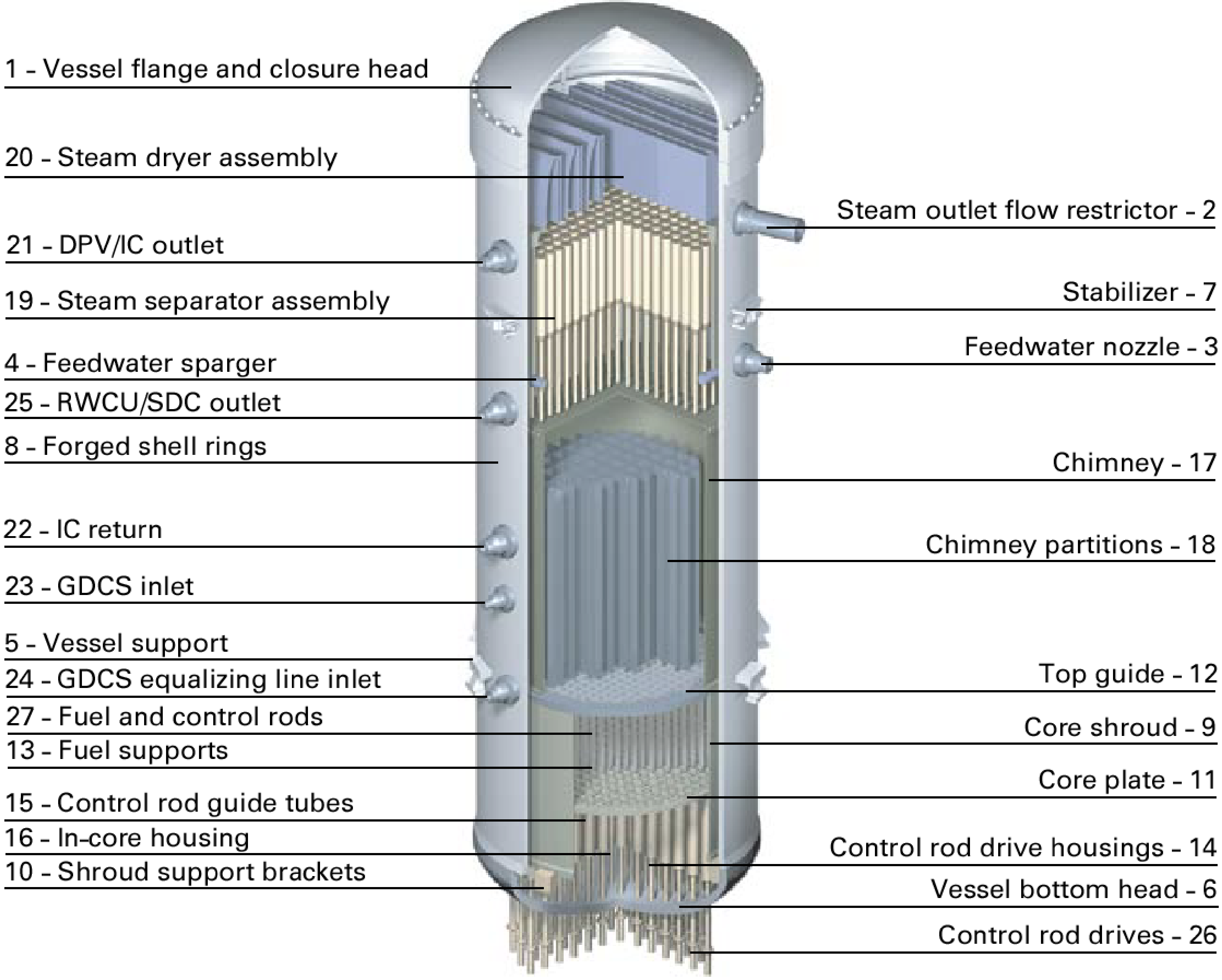
- Direct Cycle - heat generated in reactor core is directly utilized for steam generation inside the reactor vessel.
- Steam develops as small “bubbles” (void) entrained in core coolant, forming 2 phase saturated water and steam in core.
- The chimney located directly above the fuel assembly takes hot saturated water and steam, and channels it upward. The effect is to pull more water through the fuel bundles increasing natural circulation, thus no coolant recirculation pumps are needed.
- The saturated steam is separated in the coolant flow from “Steam Separators”, and dried in “Steam Dryer” arrangement - minimize water carry-over & minimize the radioactive products carried from the saturated steam. TG needs shielding.
- The quality of the coolant at the core exit is ~ 18 % (fraction (%) of coolant at core exit is saturated steam vapor.)

BWR Coolant Phase Diagram

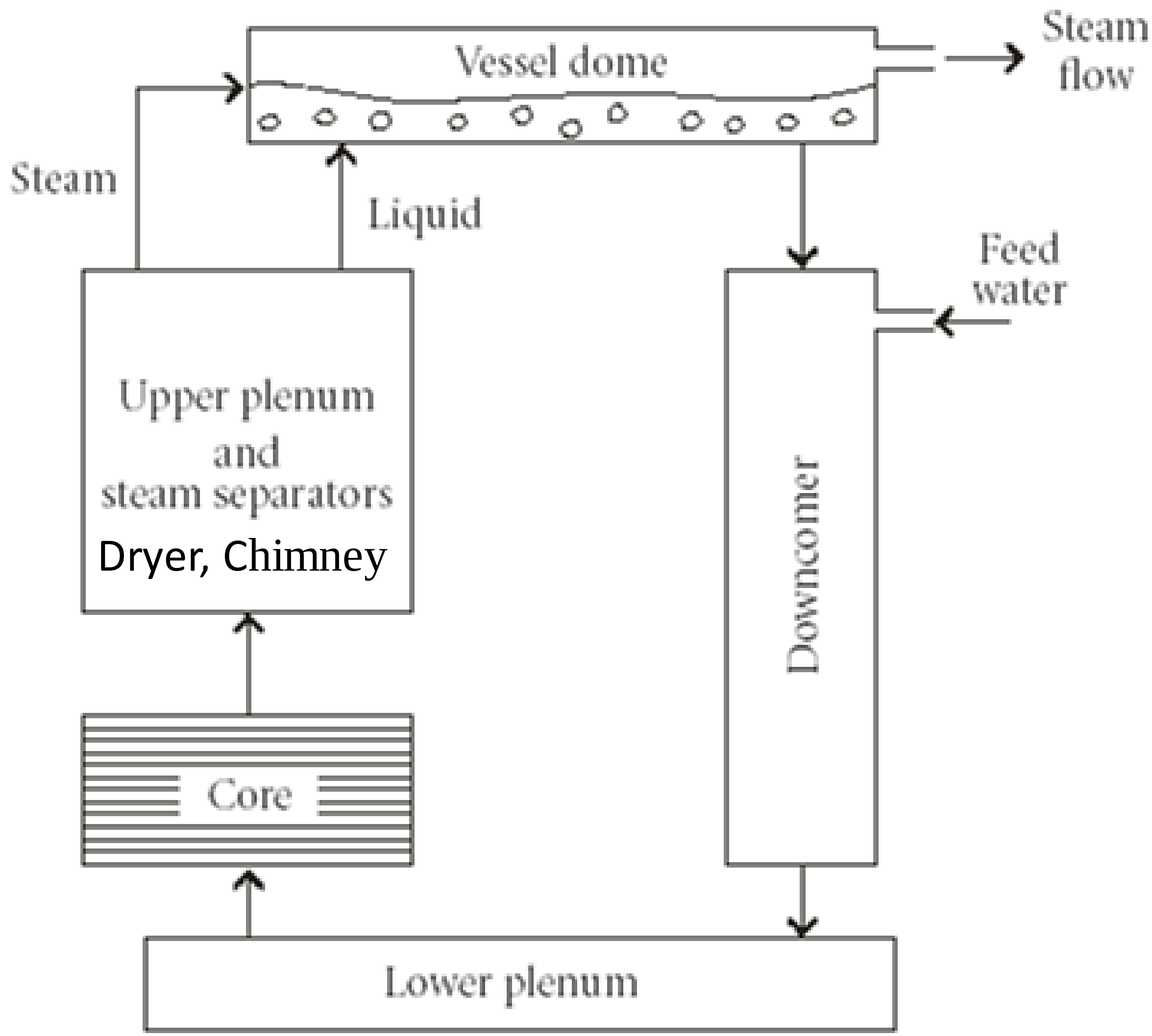
Phase Diagram of Water



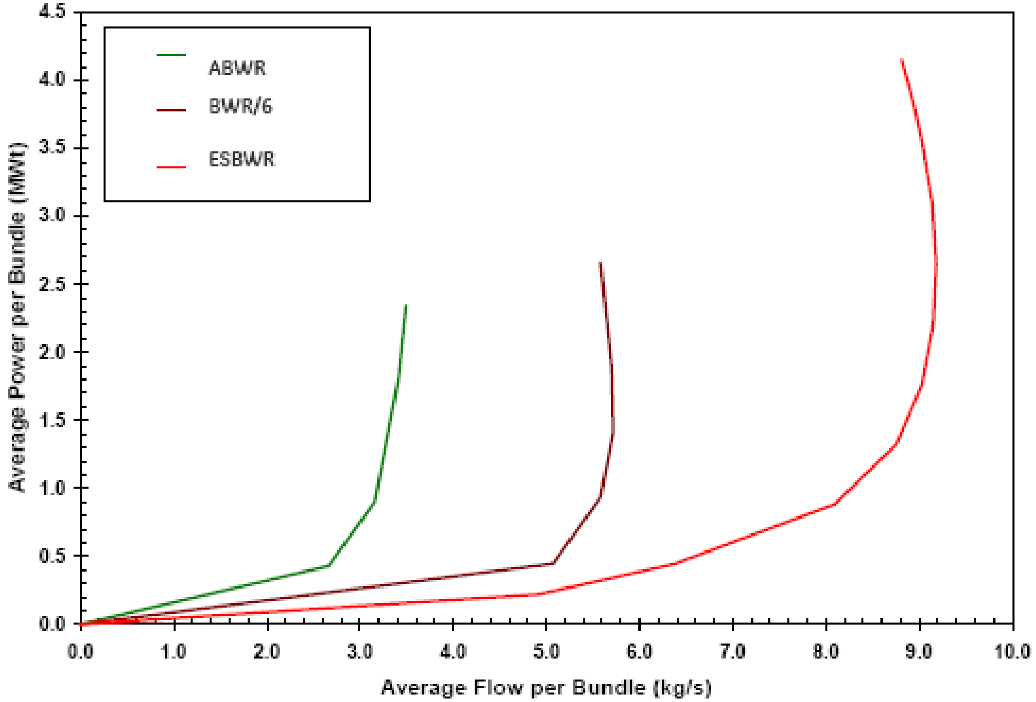
ESBWR

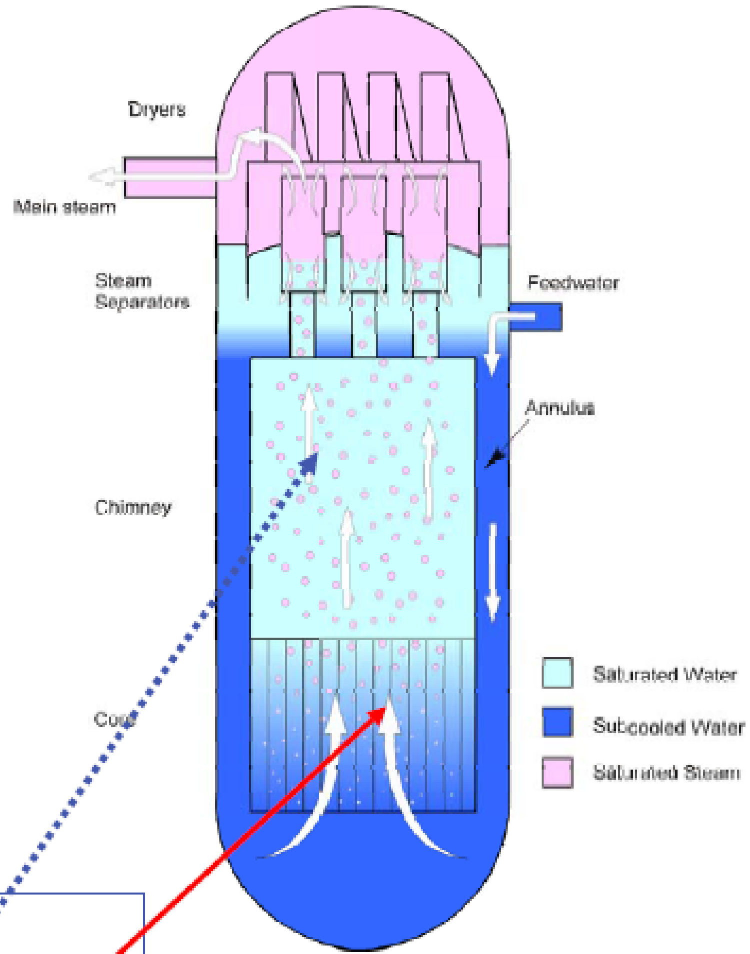

1 - Vessel flange and closure head
20 - Steam dryer assembly
21 - DPV/IC outlet
19 - Steam separator assembly
4 - Feedwater sparger
25 - RWCU/SDC outlet
8 - Forged shell rings
22 - IC return
23 - GDCS inlet
5 - Vessel support
24 - GDCS equalizing line inlet
27 - Fuel and control rods
13 - Fuel supports
15 - Control rod guide tubes
16 - In-core housing
10 - Shroud support brackets
Steam outlet flow restrictor - 2
Stabilizer - 7
Feedwater nozzle - 3
Chimney - 17
Chimney partitions - 18
Top guide - 12
Core shroud - 9
Core plate - 11
Control rod drive housings - 14
Vessel bottom head - 6
Control rod drives - 26

Divide the RPV into Control Volumes



Enhanced Natural Circulation





- Higher driving head
  - Chimney/taller vessel
- Reduced flow restrictions
  - Shorter core
  - Increase downcomer area

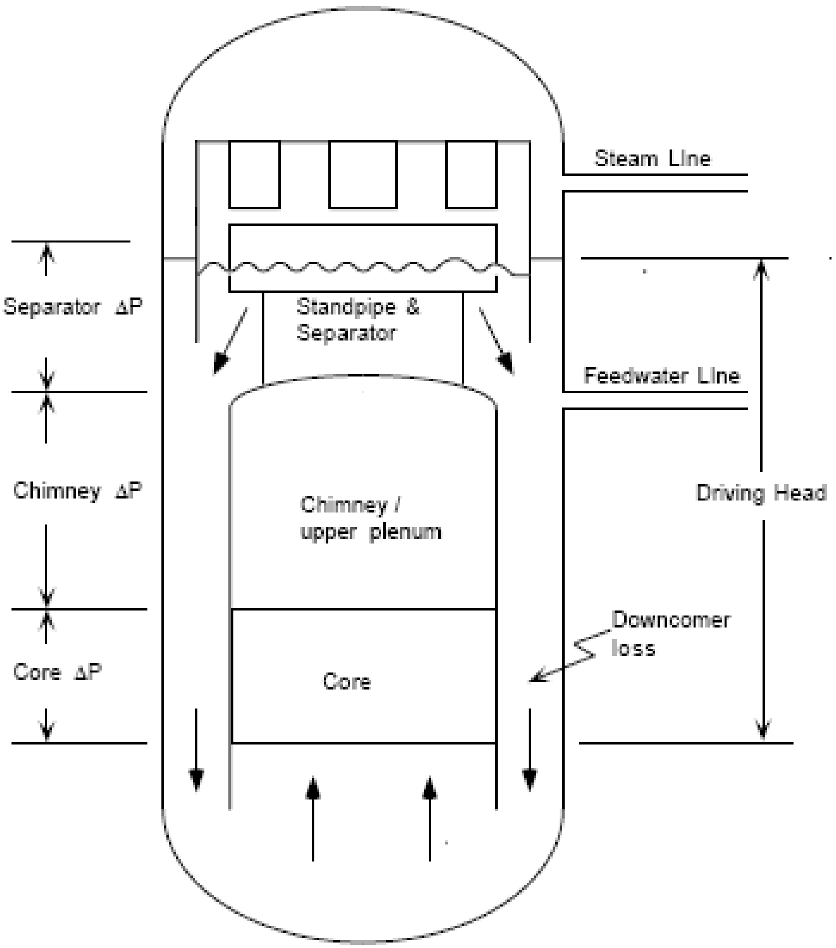
ESBWR vs BWRX-300

Parameters	ESBWR	BWRX-300	% of ESBWR
Power (MWth)	4000	870	21%
Power (MWe)	1380	270 - 290	21%
Vessel Height (m)	27.7	26	94%
Vessel Diameter (m)	7.1	4	14.4%
Fuel Bundles (number)	1132	208	18%
Active Core Height (m)	3	3.8	126%
Number of CRDs	269	57	21%
Coolant Inlet Temp (deg. C)	276	270	97%
Coolant Outlet Temp (deg.)	287	287	~100%
Reactor Operating Press (MPa)	7.17	7.2	~100 %
Coolant flow rate (Kg/s)	9570	1530	16 %

Source: [ARIS - Technical Data (iaea.org)]

Natural Circulation Flow

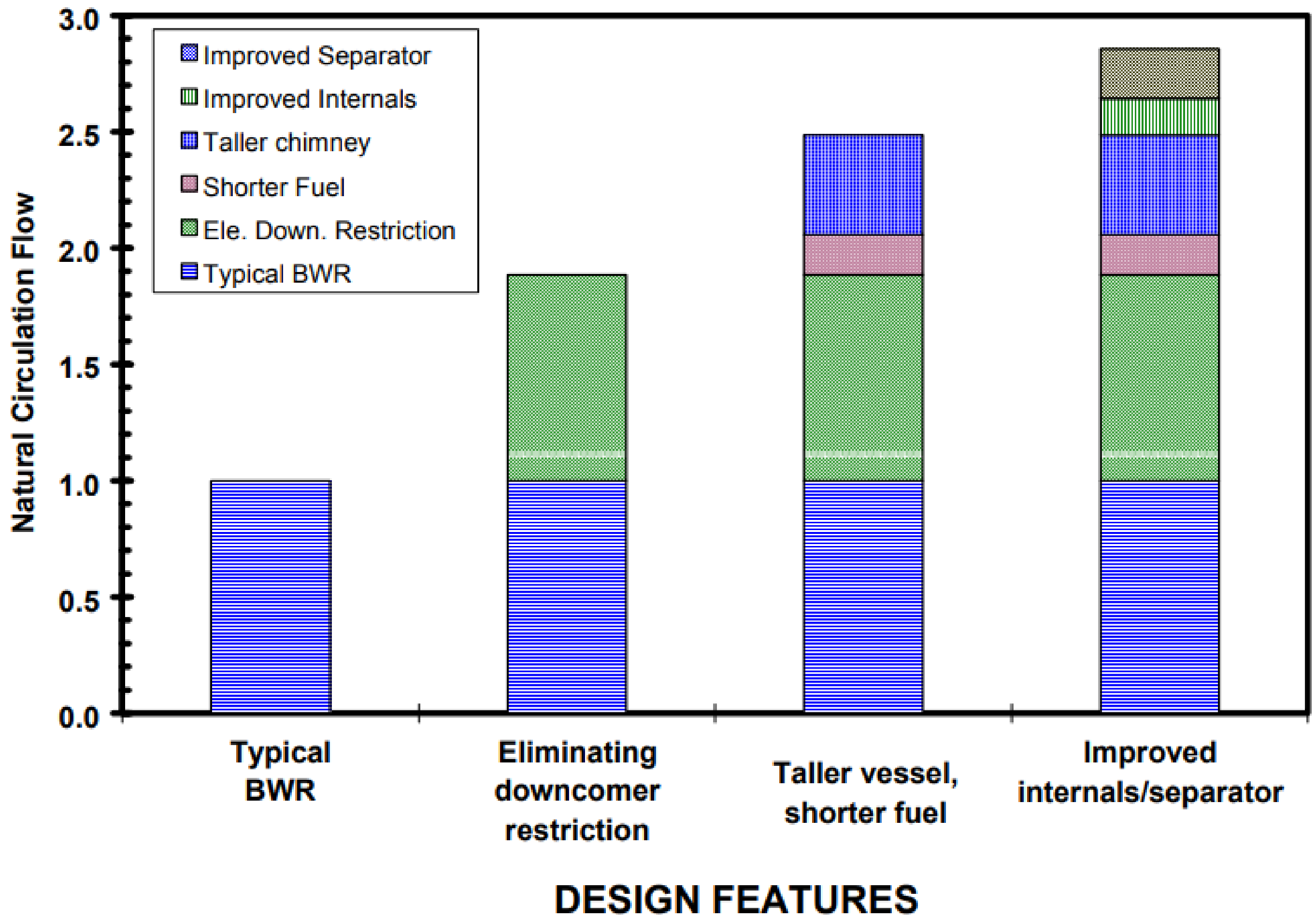
- Core flow depends on
  - > driving head
  - > losses through the loop
- Driving head
  - > proportional to core + chimney height
    - – **Void Fraction**
- Loop losses
  - > downcomer
    - – **Single-phase Δp, handbook loss coefficient**
  - > core (fuel bundle)
    - – **Two-phase Δp, test data/correlation**
  - > chimney ~ small
  - > separator
    - – **Two-phase Δp, test data/correlation**



Schematic of Flow and Pressure Drops in a Reactor

ESBWR design provides 2 to 3 times more natural circulation flow

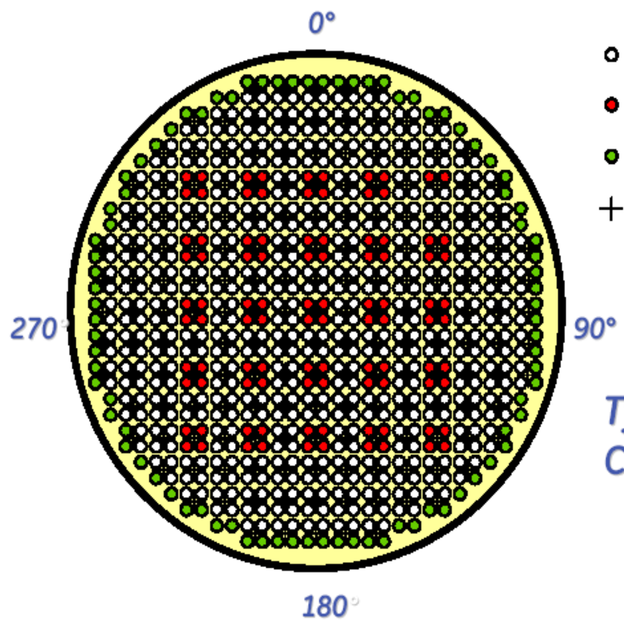
Effects of ESBWR Design Features on Natural Circulation Flow



Reactor Core & Fuel Design

- Passive BWR core consists of a number of fuel bundles (assemblies): ESBWR- 1132
- Each fuel bundle (assembly) consists of a number of UO_2 fuel rods with Zircaloy cladding arranged in 10 x 10 square lattice (slightly enriched uranium fuel pellets - typical enrichment 2 % to 5 % U-235 by weight). Average core power density ~ 60 % PWR.
- Some fuel rods have burnable poison Gd, to counteract the large positive reactivity contribution of fresh fuel in early core life.
- Number of control rods enter the core from the bottom, through guide tubes in the fuel assemblies :ESBWR - 269

BWR Core Design

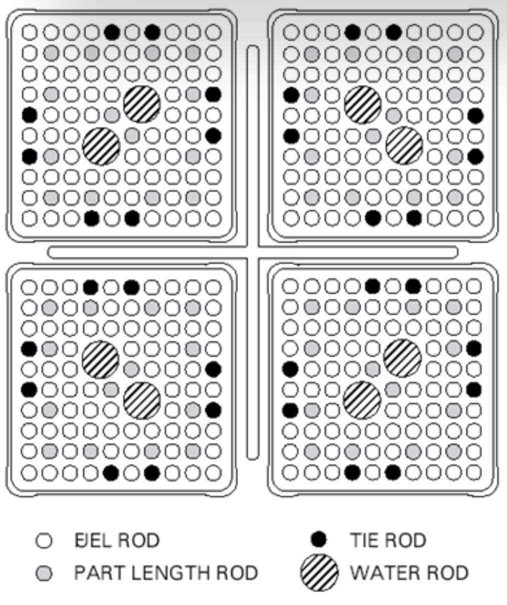


- Fuel Bundle
- Control Cell Bundle
- Peripheral Bundle
- Control Blade

Typical Control Cell Core (CCC) Layout

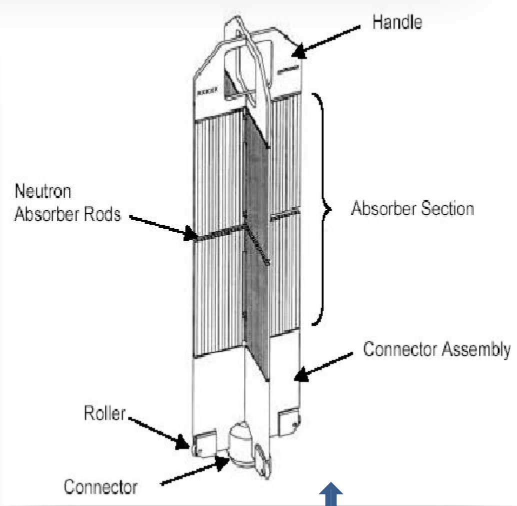
> Low reactivity bundles are placed in the control cells (CC)

> Control rod movement is done with CC rods during operation to compensate for burnup



Four Bundle Fuel Module (Cell)

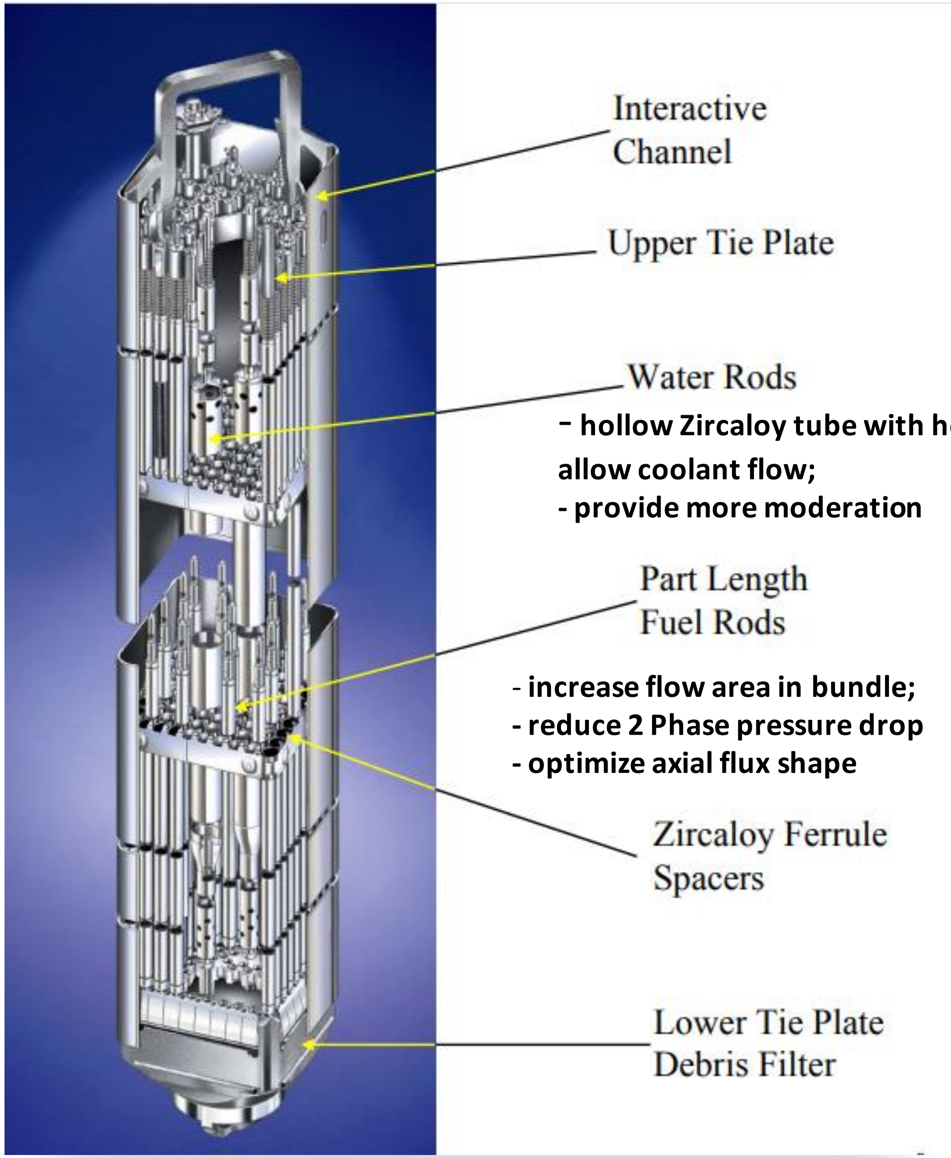
Control Rod

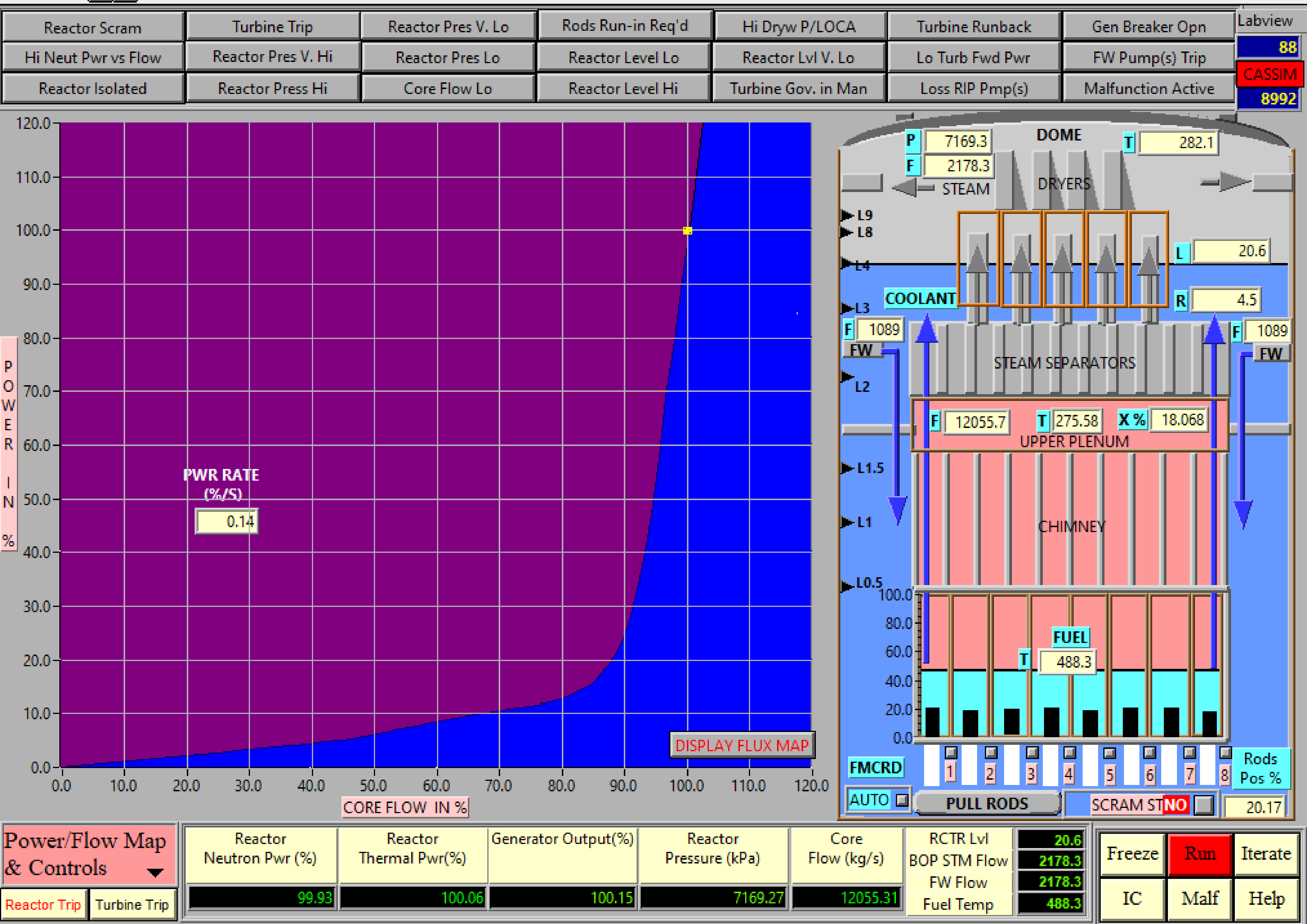


cruciform array of stainless steel tubes filled with the powdered form of boron carbide (B4C) poison.

ESBWR

- **Fuel Bundles – 1132**
- **Control Rods - 269**





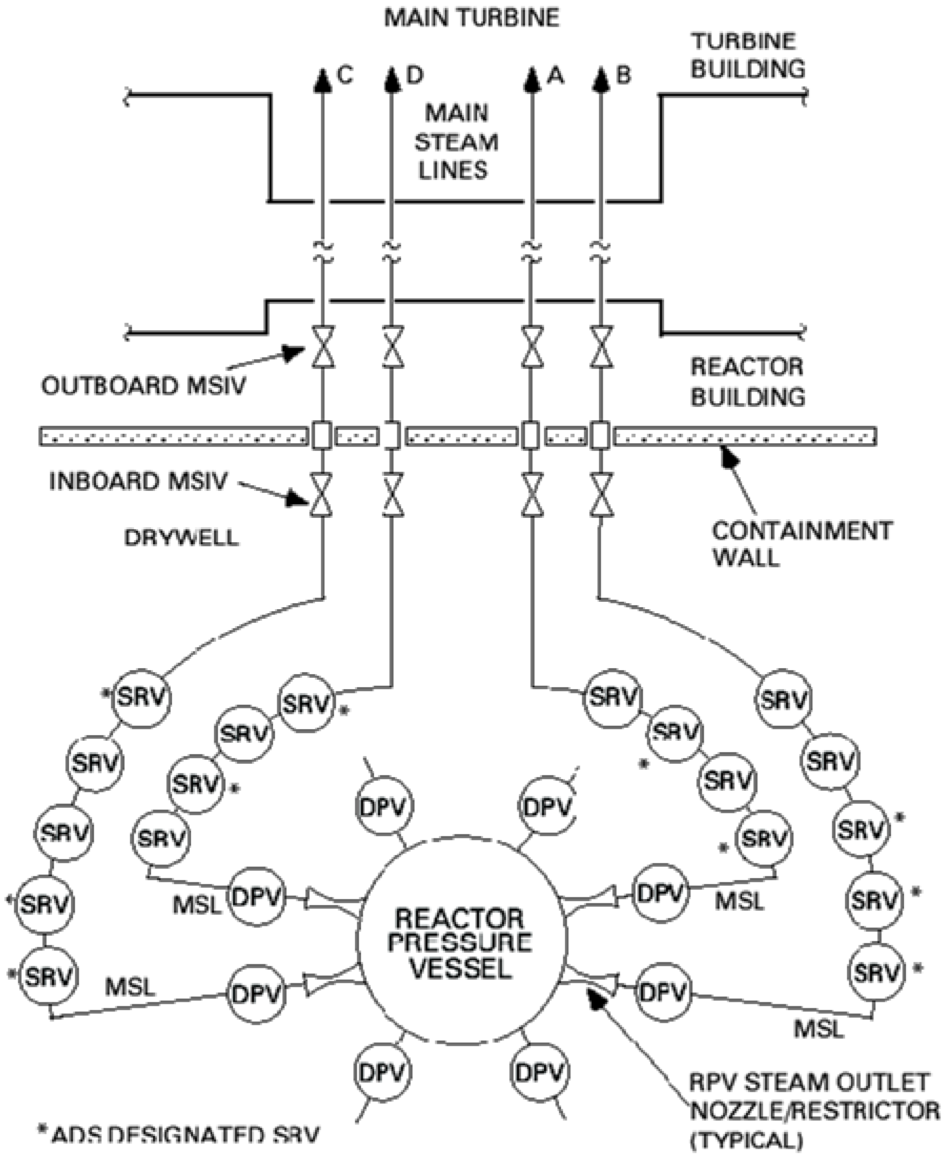
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
88
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
8992
POWER IN %
PWR RATE (%/S)
0.14
DISPLAY FLUX MAP
CORE FLOW IN %
0.0 10.0 20.0 30.0 40.0 50.0 60.0 70.0 80.0 90.0 100.0 110.0 120.0
DOME
P 7169.3
F 2178.3
T 282.1
STEAM
DRYERS
L9
L8
L4
L 20.6
COOLANT
R 4.5
L3
F 1089
FW
F 1089
FW
STEAM SEPARATORS
L2
F 12055.7
T 275.58
X % 18.068
UPPER PLENUM
L1.5
L1
CHIMNEY
L0.5
100.0
80.0
60.0
40.0
20.0
0.0
FUEL
T 488.3
FMCRD
1 2 3 4 5 6 7 8
Rods Pos %
AUTO
PULL RODS
SCRAM ST NO
20.17
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
99.93
Reactor Thermal Pwr(%)
100.06
Generator Output(%)
100.15
Reactor Pressure (kPa)
7169.27
Core Flow (kg/s)
12055.31
RCTR Lvl 20.6
BOP STM Flow 2178.3
FW Flow 2178.3
Fuel Temp 488.3
Freeze
Run
Iterate
IC
Malf
Help

Question #1

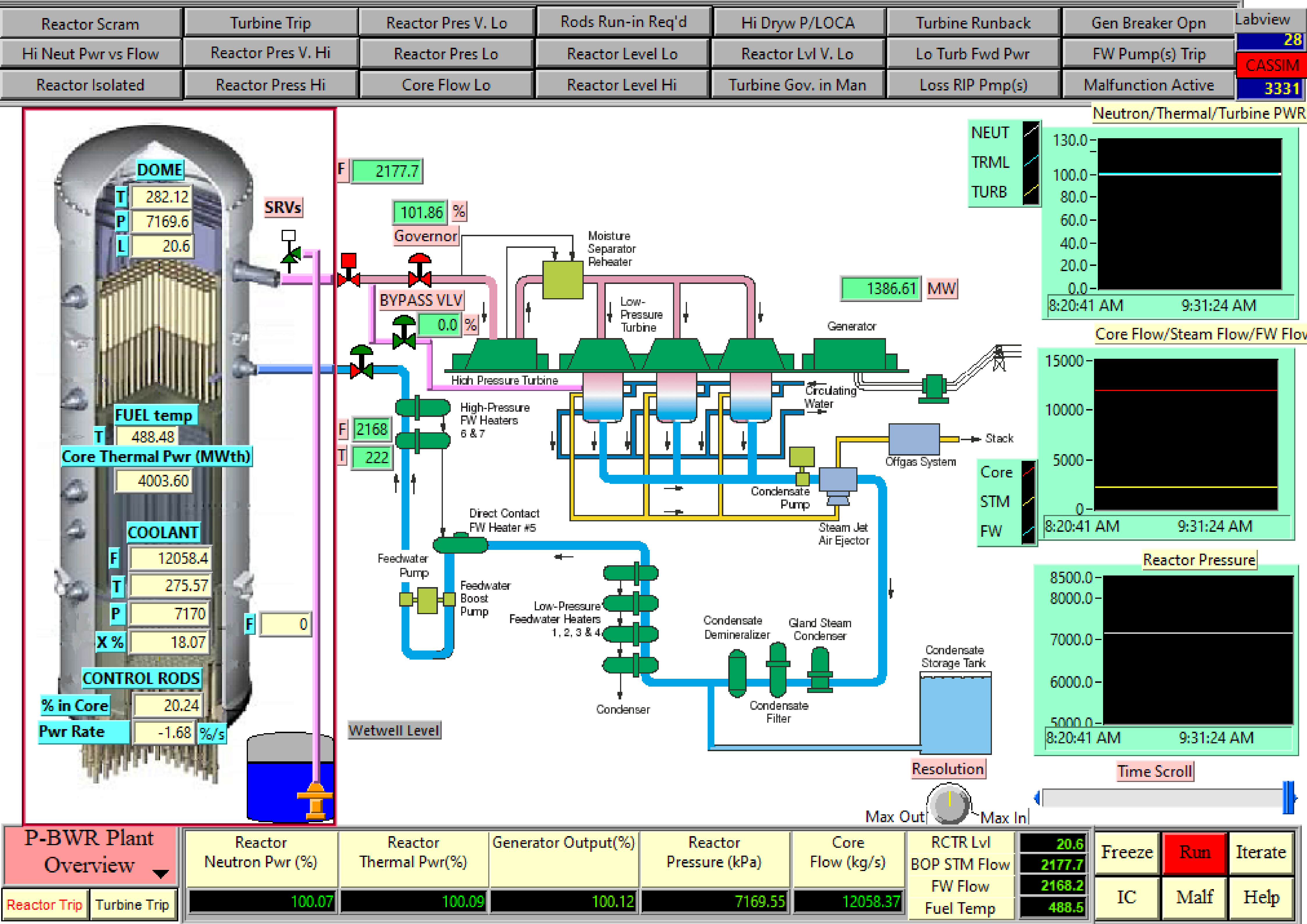
- Why Control Rods enter from the bottom of the core, as opposed to entering from the top of the core, like in PWR or PHWR ?

Main Steam System

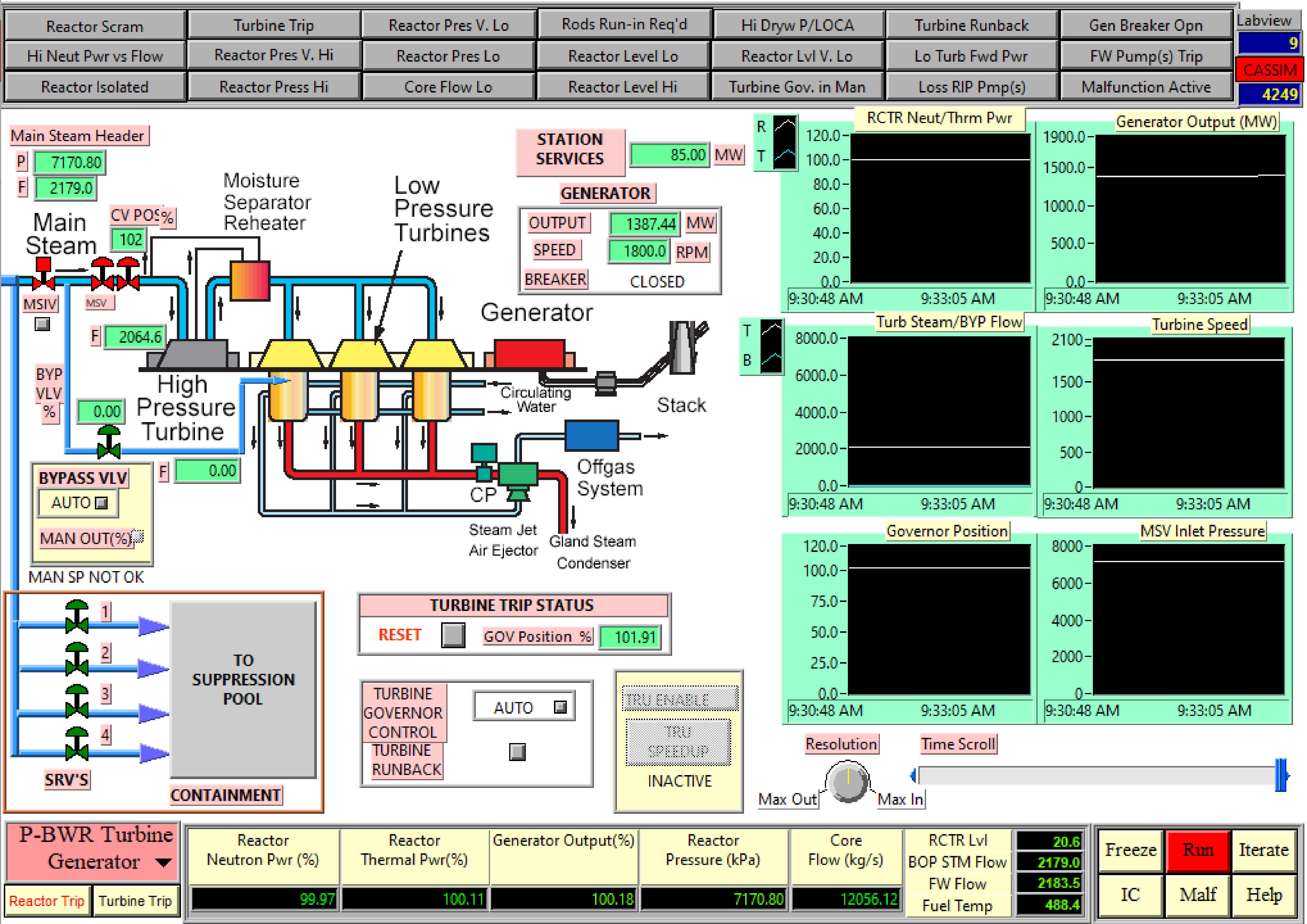
- "Dried steam" from Reactor Pressure Vessel (RPV) to the turbine plant through four steam lines connected to nozzles equipped with "flow limiters".
- Limit the coolant blowdown rate from the RPV = or < 200 % rated steam flow at 7.07 MPa upstream pressure in the event of steam line break occurs anywhere downstream the nozzle.
- Isolation valves inside and outside of containment wall.
- Safety Relief Valves (~16) connected to the four steam lines to prevent RPV overpressure, with blow down pipe to Suppression Pool.



MSIV, SRV and DPV Configuration



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
28
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
3331
DOME
T 282.12
P 7169.6
L 20.6
SRVs
F 2177.7
101.86 %
Governor
BYPASS VLV
0.0 %
Moisture Separator Reheater
Low-Pressure Turbine
High Pressure Turbine
1386.61 MW
Generator
Circulating Water
High-Pressure FW Heaters 6 & 7
F 2168
T 222
FUEL temp
T 488.48
Core Thermal Pwr (MWth)
4003.60
Offgas System
Stack
Condensate Pump
Steam Jet Air Ejector
Direct Contact FW Heater #5
Feedwater Pump
Feedwater Boost Pump
Low-Pressure Feedwater Heaters 1, 2, 3 & 4
Condensate Demineralizer
Gland Steam Condenser
Condensate Storage Tank
Condensate Filter
Condenser
COOLANT
F 12058.4
T 275.57
P 7170
X % 18.07
F 0
CONTROL RODS
% in Core 20.24
Pwr Rate -1.68 %/s
Wetwell Level
Resolution
Max Out
Max In
Neutron/Thermal/Turbine PWR
NEUT
TRML
TURB
130.0
100.0
80.0
60.0
40.0
20.0
0.0
8:20:41 AM 9:31:24 AM
Core Flow/Steam Flow/FW Flow
15000
10000
5000
0
Core
STM
FW
8:20:41 AM 9:31:24 AM
Reactor Pressure
8500.0
8000.0
7000.0
6000.0
5000.0
8:20:41 AM 9:31:24 AM
Time Scroll
P-BWR Plant Overview
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 100.07
Reactor Thermal Pwr(%) 100.09
Generator Output(%) 100.12
Reactor Pressure (kPa) 7169.55
Core Flow (kg/s) 12058.37
RCTR Lvl 20.6
BOP STM Flow 2177.7
FW Flow 2168.2
Fuel Temp 488.5
Freeze
Run
Iterate
IC
Malf
Help


Reactor Scram
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Hi Dryw P/LOCA
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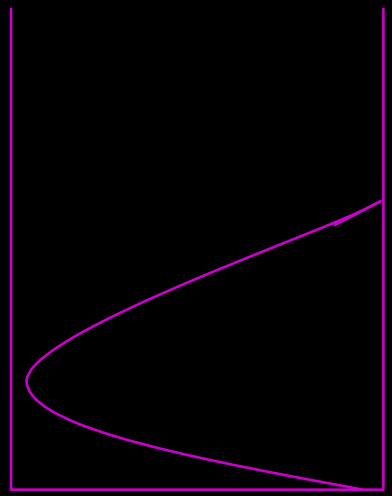
Gen Breaker Opn
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
9
CASSIM
4249
Main Steam Header
P
7170.80
F
2179.0
Main Steam
CV POS%
102
Moisture Separator Reheater
Low Pressure Turbines
MSIV
MSV
F
2064.6
BYP VLV %
0.00
High Pressure Turbine
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
F
0.00
STATION SERVICES
85.00
MW
GENERATOR
OUTPUT
1387.44
MW
SPEED
1800.0
RPM
BREAKER
CLOSED
Generator
Circulating Water
Stack
Offgas System
CP
Steam Jet Air Ejector
Gland Steam Condenser
1
2
3
4
SRV'S
TO SUPPRESSION POOL
CONTAINMENT
TURBINE TRIP STATUS
RESET
GOV Position %
101.91
TURBINE GOVERNOR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
R
T
RCTR Neut/Thrm Pwr
120.0
100.0
80.0
60.0
40.0
20.0
0.0
9:30:48 AM
9:33:05 AM
Generator Output (MW)
1900.0
1500.0
1000.0
500.0
0.0
9:30:48 AM
9:33:05 AM
T
B
Turb Steam/BYP Flow
8000.0
6000.0
4000.0
2000.0
0.0
9:30:48 AM
9:33:05 AM
Turbine Speed
2100
1500
1000
500
0
9:30:48 AM
9:33:05 AM
Governor Position
120.0
100.0
75.0
50.0
25.0
0.0
9:30:48 AM
9:33:05 AM
MSV Inlet Pressure
8000
6000
4000
2000
0
9:30:48 AM
9:33:05 AM
Resolution
Time Scroll
Max Out
Max In
P-BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
99.97
Reactor Thermal Pwr(%)
100.11
Generator Output(%)
100.18
Reactor Pressure (kPa)
7170.80
Core Flow (kg/s)
12056.12
RCTR Lvl
20.6
BOP STM Flow
2179.0
FW Flow
2183.5
Fuel Temp
488.4
Freeze
Run
Iterate
IC
Malf
Help

Feedback from the Audience on Q #1

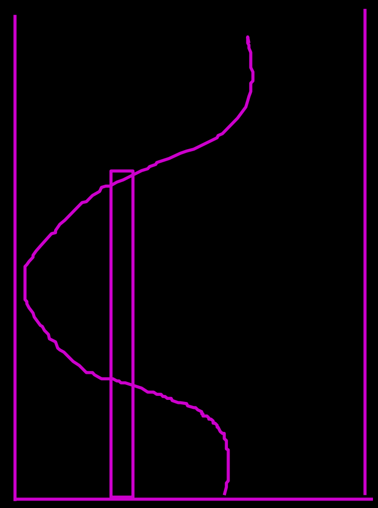
Answer #1

- Neutron flux distribution in BWR core is a function of void fractions in core.
- Since voids are abound in the upper part of the core, the moderating power is highest in the non- boiling section of the core (lower part).
- This causes the peak neutron flux (power density) for a boiling core to shift from the center position towards the bottom of the core. Control rods entering from the bottom can partially correct the skewed axial flux distribution

Answer #1



Before Control Rods
entry from bottom



After Control Rods
entry from bottom

BWR SPATIAL CORE PROPERTIES

(WITH CONTROL RODS PARTIALLY INSERTED)

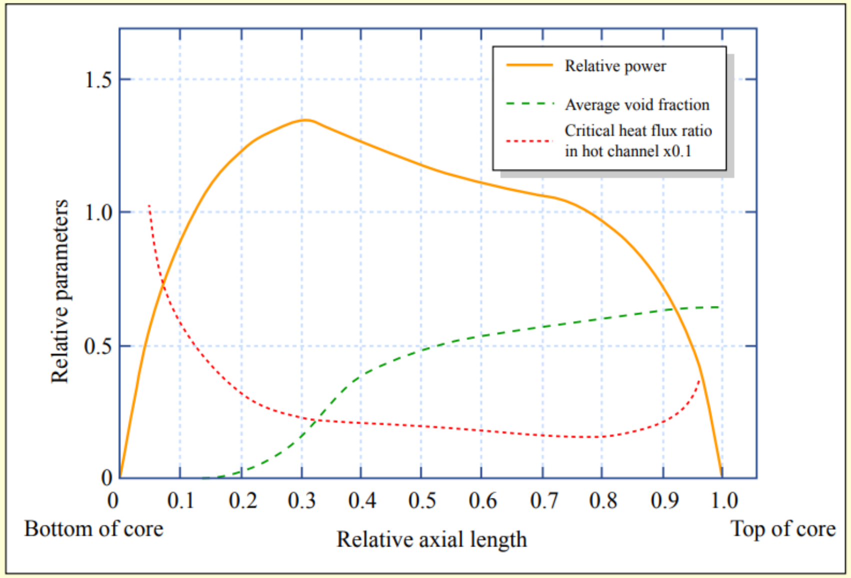


Image by MIT OpenCourseWare.

POWER IN FRESH FUEL ASSEMBLY AS ADJACENT CONTROL ROD IS WITHDRAWN TOWARD BOTTOM

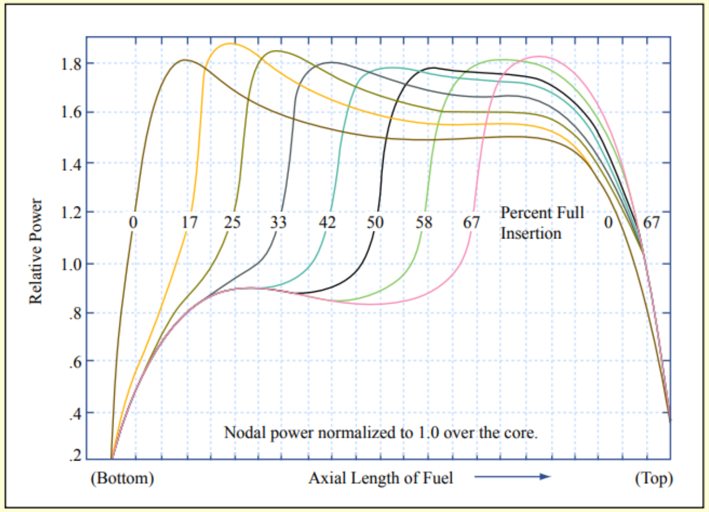
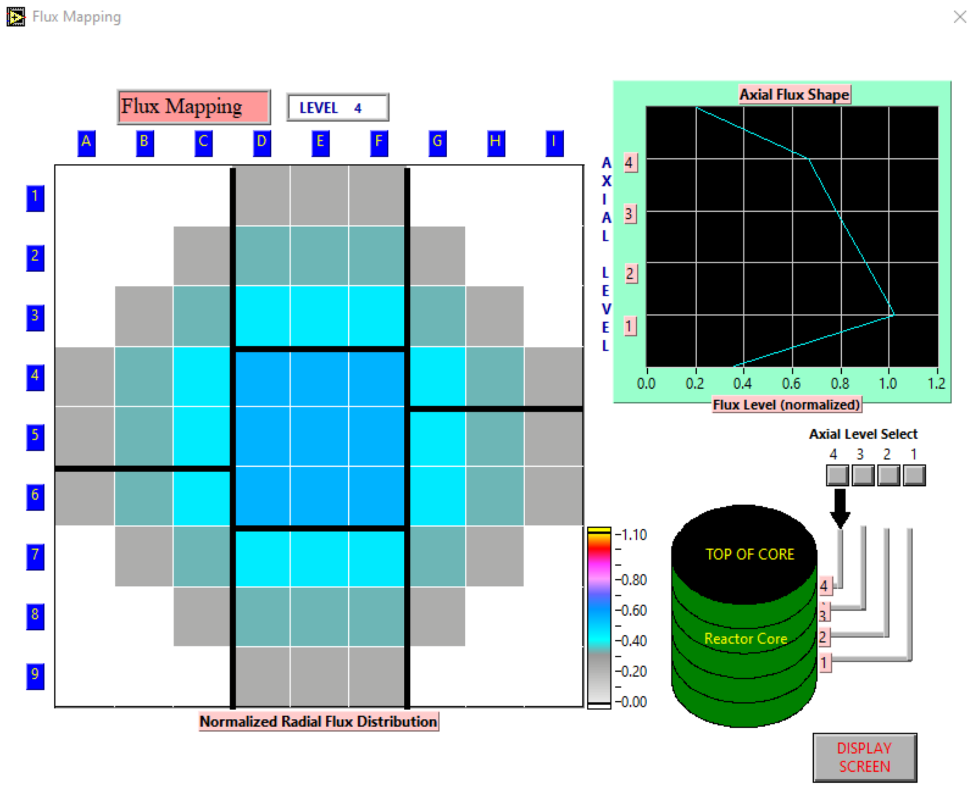
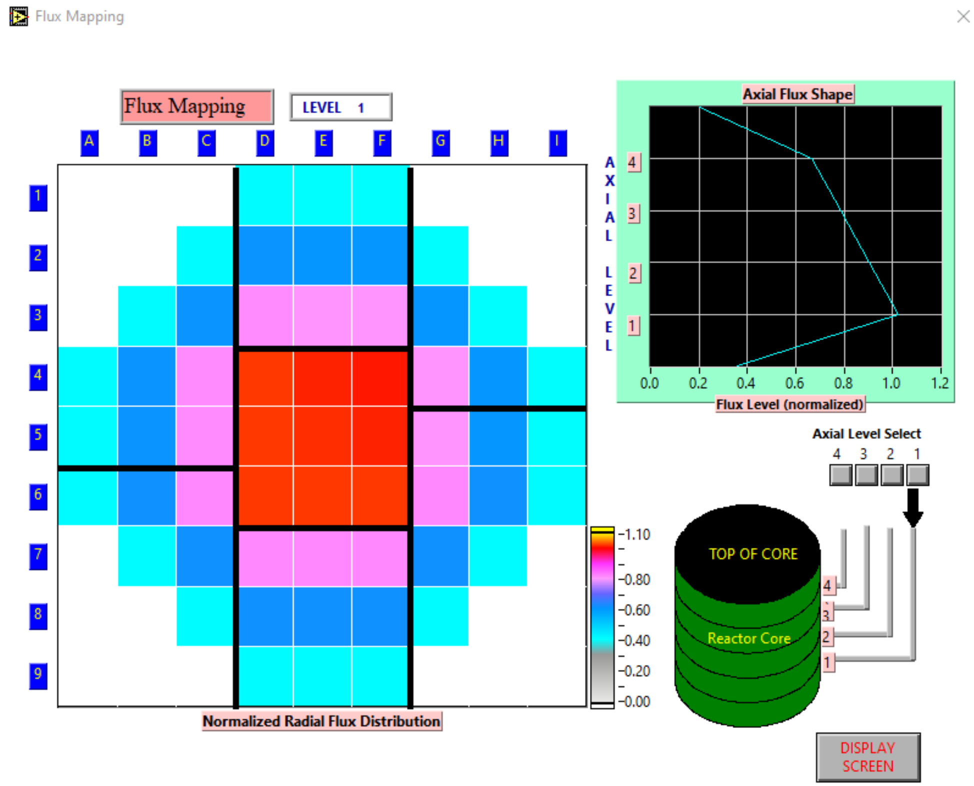


Image by MIT OpenCourseWare.

Passive BWR Simulator with Control Rods Partially Inserted



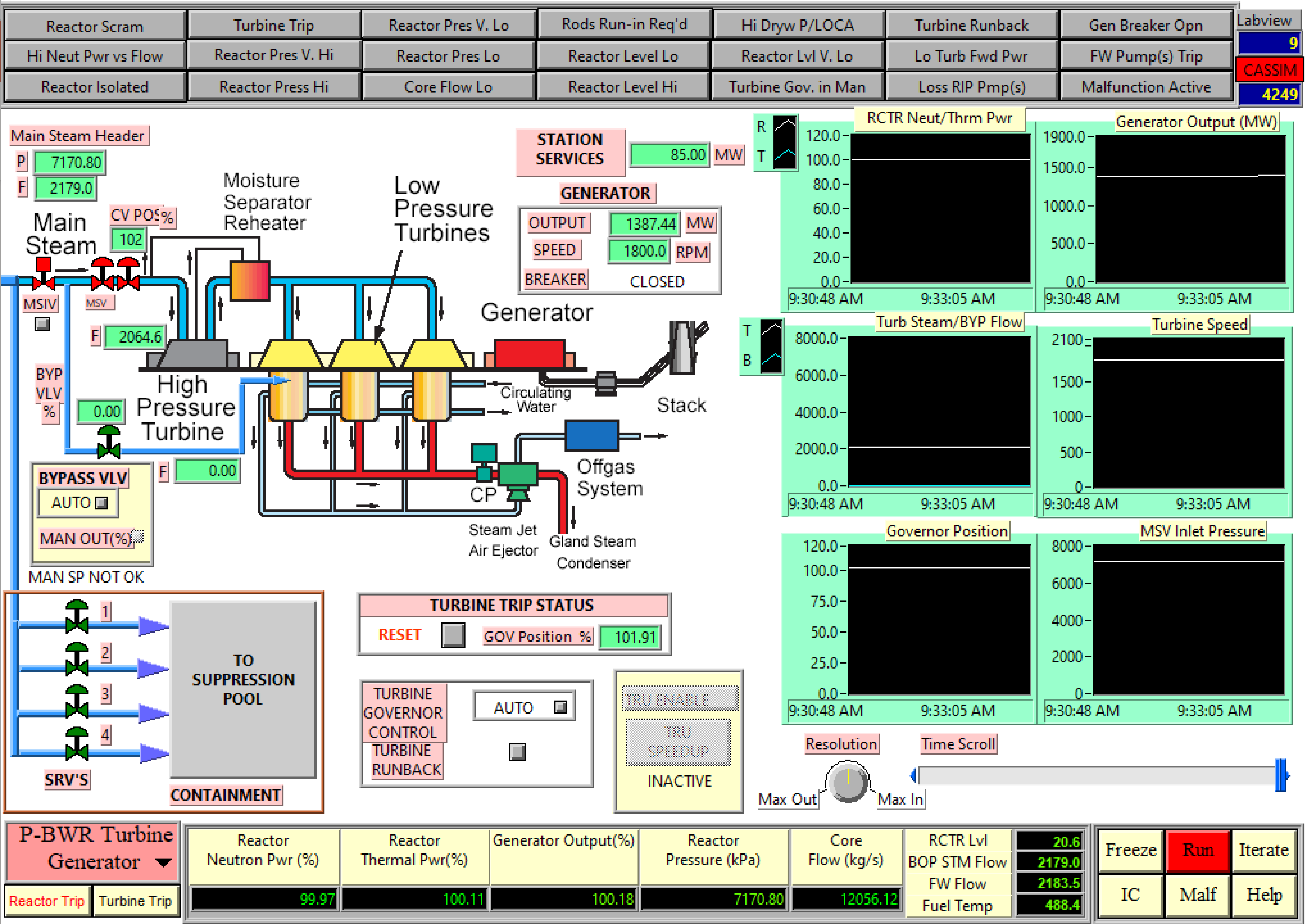


Question #2

- Why adjusting core flow rate in BWR will control reactor power ?

Turbine & Steam Bypass Systems

- Saturated steam from RPV main steam lines admitted to turbine HP cylinder via the governor valves. After HP section, steam passes through MSR to LP turbine cylinders.
- A special Steam Bypass line prior to the turbine governor valves, enables dumping the full nominal steam flow directly to condenser in the event of plant upset (e.g. turbine trip), in order to avoid severe pressure surges and corresponding power peaks in reactor.



Reactor Scram
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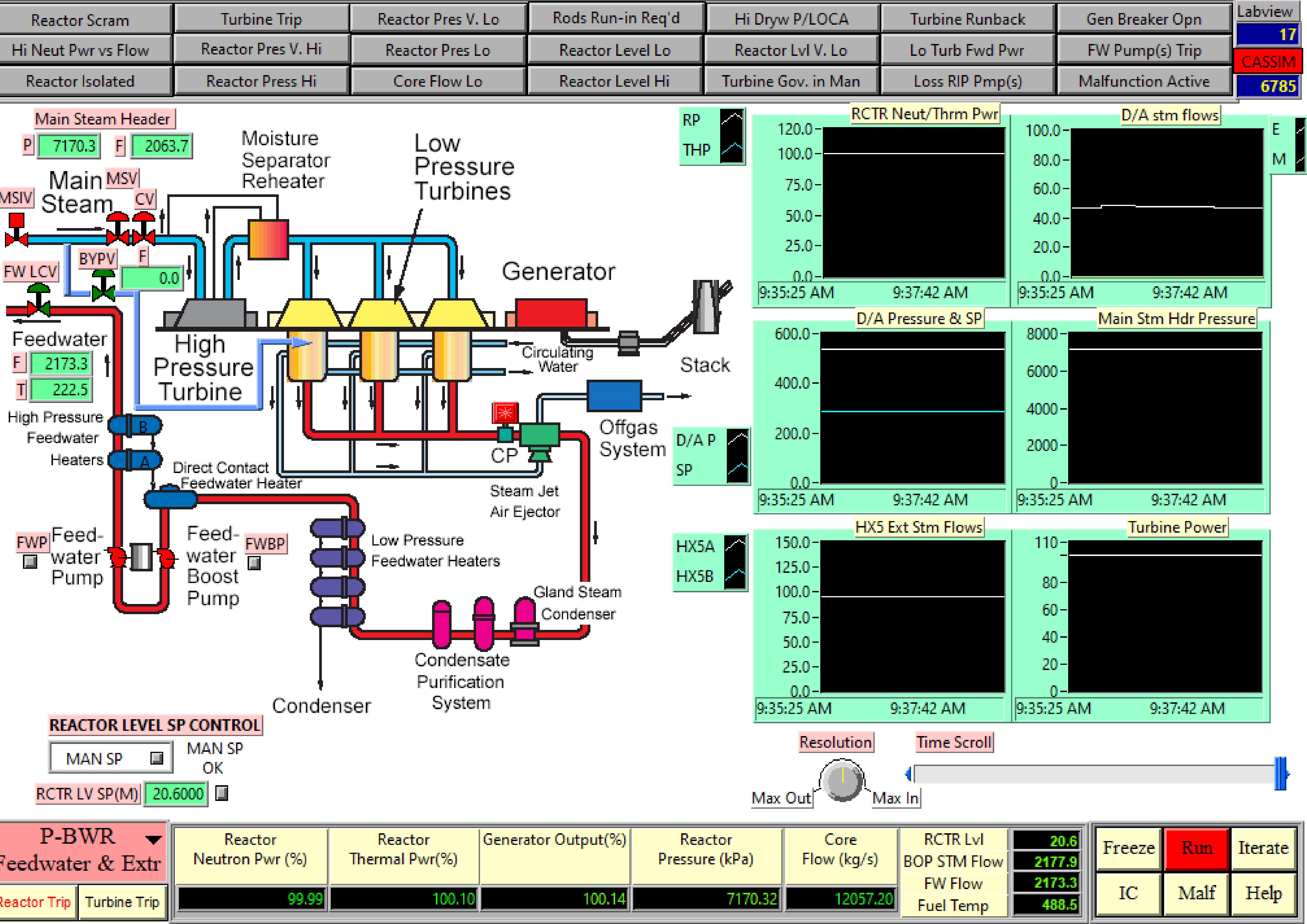
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
9
CASSIM
4249
Main Steam Header
P 7170.80
F 2179.0
CV POS%
102
Main Steam
MSIV
MSV
F 2064.6
BYP VLV %
0.00
Moisture Separator Reheater
Low Pressure Turbines
High Pressure Turbine
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
F 0.00
STATION SERVICES
85.00 MW
GENERATOR
OUTPUT 1387.44 MW
SPEED 1800.0 RPM
BREAKER CLOSED
Generator
Circulating Water
Stack
Offgas System
CP
Steam Jet Air Ejector
Gland Steam Condenser
1
2
3
4
SRV'S
TO SUPPRESSION POOL
CONTAINMENT
TURBINE TRIP STATUS
RESET
GOV Position % 101.91
TURBINE GOVERNOR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
RCTR Neut/Thrm Pwr
R
T
120.0 100.0 80.0 60.0 40.0 20.0 0.0
9:30:48 AM 9:33:05 AM
Generator Output (MW)
1900.0 1500.0 1000.0 500.0 0.0
9:30:48 AM 9:33:05 AM
Turb Steam/BYP Flow
T
B
8000.0 6000.0 4000.0 2000.0 0.0
9:30:48 AM 9:33:05 AM
Turbine Speed
2100 1500 1000 500 0
9:30:48 AM 9:33:05 AM
Governor Position
120.0 100.0 75.0 50.0 25.0 0.0
9:30:48 AM 9:33:05 AM
MSV Inlet Pressure
8000 6000 4000 2000 0
9:30:48 AM 9:33:05 AM
Resolution
Time Scroll
Max Out
Max In
P-BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.97
Reactor Thermal Pwr(%) 100.11
Generator Output(%) 100.18
Reactor Pressure (kPa) 7170.80
Core Flow (kg/s) 12056.12
RCTR Lvl 20.6
BOP STM Flow 2179.0
FW Flow 2183.5
Fuel Temp 488.4
Freeze
Run
Iterate
IC
Malf
Help

BOP & Feedwater System

- Typical BOP systems - condenser; condensate pumps; deaerator; feedwater heaters; Reactor Feed Pumps (RFP); Reactor Level Control Valves.



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
17
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
6785
Main Steam Header
P 7170.3
F 2063.7
Moisture Separator Reheater
Low Pressure Turbines
Main Steam
MSIV
MSV
CV
BYPV
F
0.0
FW LCV
Generator
Feedwater
F 2173.3
T 222.5
High Pressure Turbine
Circulating Water
Stack
High Pressure Feedwater Heaters
B
A
Direct Contact Feedwater Heater
CP
Offgas System
Steam Jet Air Ejector
FWP
Feed-water Pump
Feed-water Boost Pump
FWBP
Low Pressure Feedwater Heaters
Gland Steam Condenser
Condensate Purification System
Condenser
REACTOR LEVEL SP CONTROL
MAN SP
MAN SP OK
RCTR LV SP(M) 20.6000
RP
THP
RCTR Neut/Thrm Pwr
120.0 100.0 75.0 50.0 25.0 0.0
9:35:25 AM 9:37:42 AM
D/A stm flows
100.0 80.0 60.0 40.0 20.0 0.0
9:35:25 AM 9:37:42 AM
E
M
D/A P
SP
D/A Pressure & SP
600.0 400.0 200.0 0.0
9:35:25 AM 9:37:42 AM
Main Stm Hdr Pressure
8000 6000 4000 2000 0
9:35:25 AM 9:37:42 AM
HX5A
HX5B
HX5 Ext Stm Flows
150.0 125.0 100.0 75.0 50.0 25.0 0.0
9:35:25 AM 9:37:42 AM
Turbine Power
110 80 60 40 20 0
9:35:25 AM 9:37:42 AM
Resolution
Time Scroll
Max Out
Max In
P-BWR
Feedwater & Extr
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.99
Reactor Thermal Pwr(%) 100.10
Generator Output(%) 100.14
Reactor Pressure (kPa) 7170.32
Core Flow (kg/s) 12057.20
RCTR Lvl 20.6
BOP STM Flow 2177.9
FW Flow 2173.3
Fuel Temp 488.5
Freeze
Run
Iterate
IC
Malf
Help

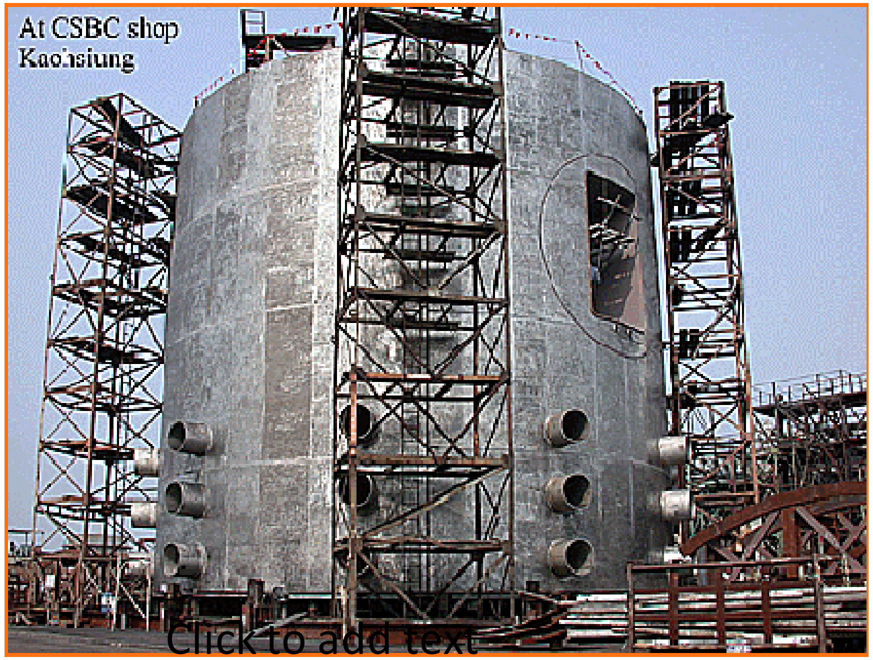
Feedback from the Audience on Q #2

Answer #2

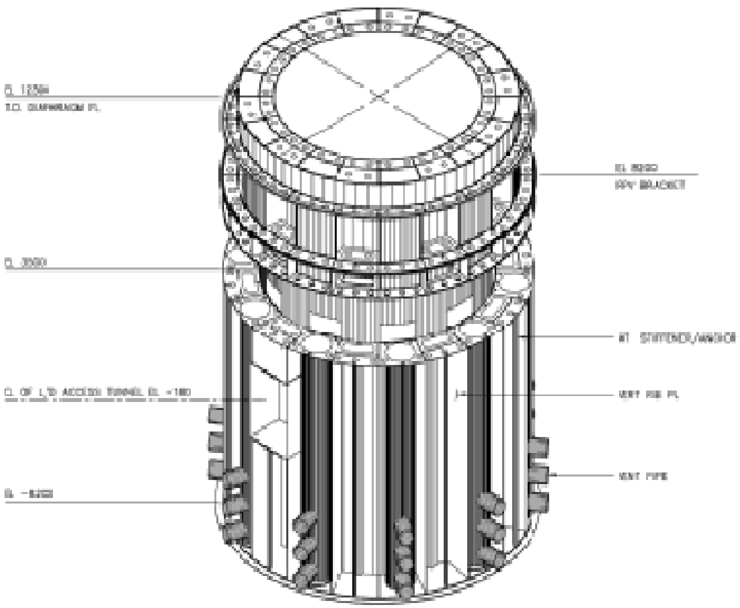
- Void in coolant has negative reactivity feedback - more steam bubbles, more void, more negative reactivity.
- Hence increasing the core flow rate will reduce void density, thus lesser negative reactivity feedback.
- If a critical reactor was at steady state ie. zero reactivity change situation before, any reduction of negative reactivity will inject a net positive reactivity change, causing the reactor neutronic power to increase.

Containment

- Containment - cylindrical prestressed concrete structure with embedded steel liner - encloses reactor RPV, passive reactor cooling systems, reactor coolant pressure boundary & important ancillary systems (RCCV).
- Includes pressure-suppression type with drywell and wetwell.
- Wetwell separated from drywell by partition floor. The wetwell's lower portion is filled with water - condensation pool. Upper portion serves a gas compression chamber.

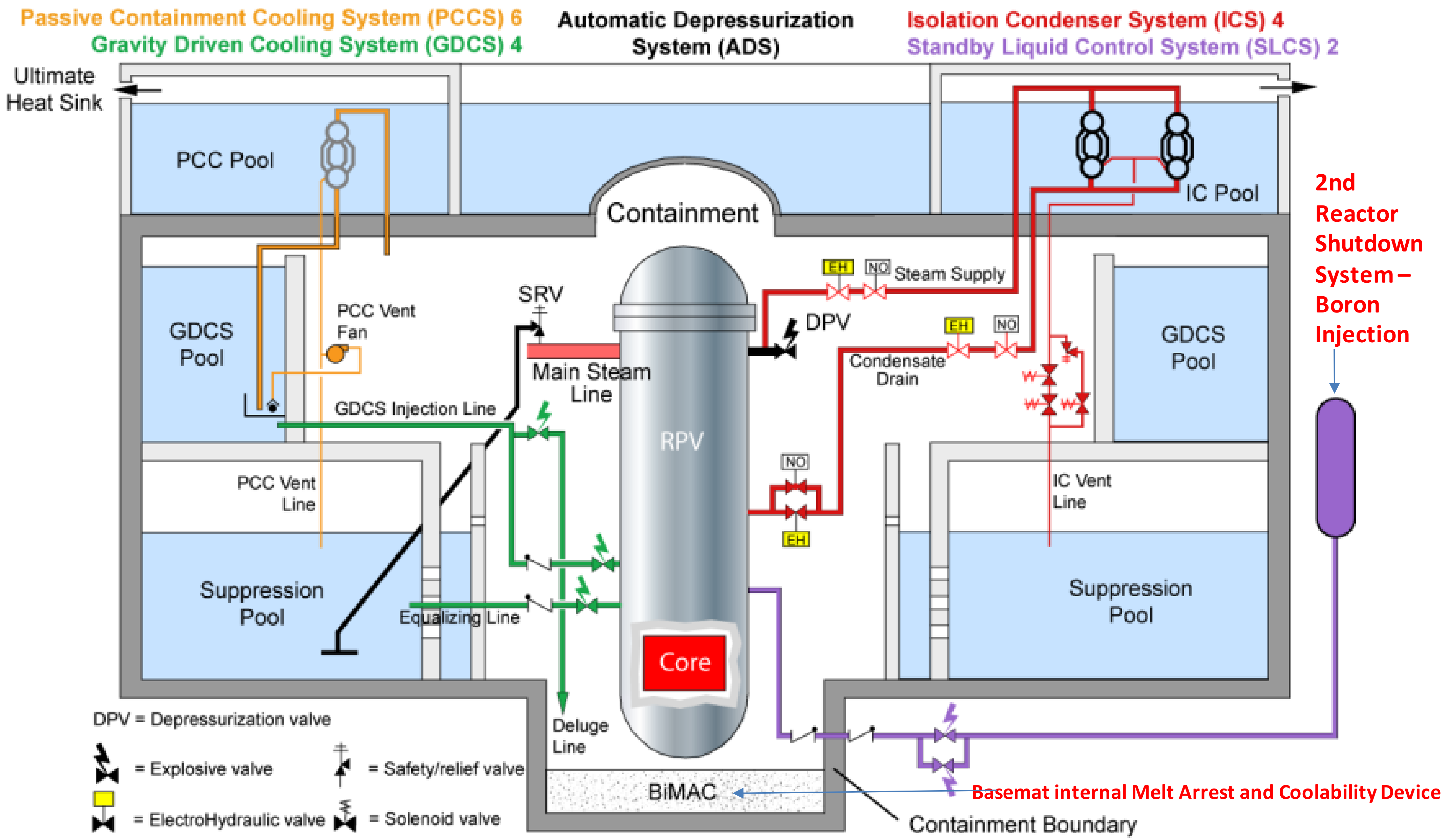


Click to add text



Concrete Containment Structure for ABWR

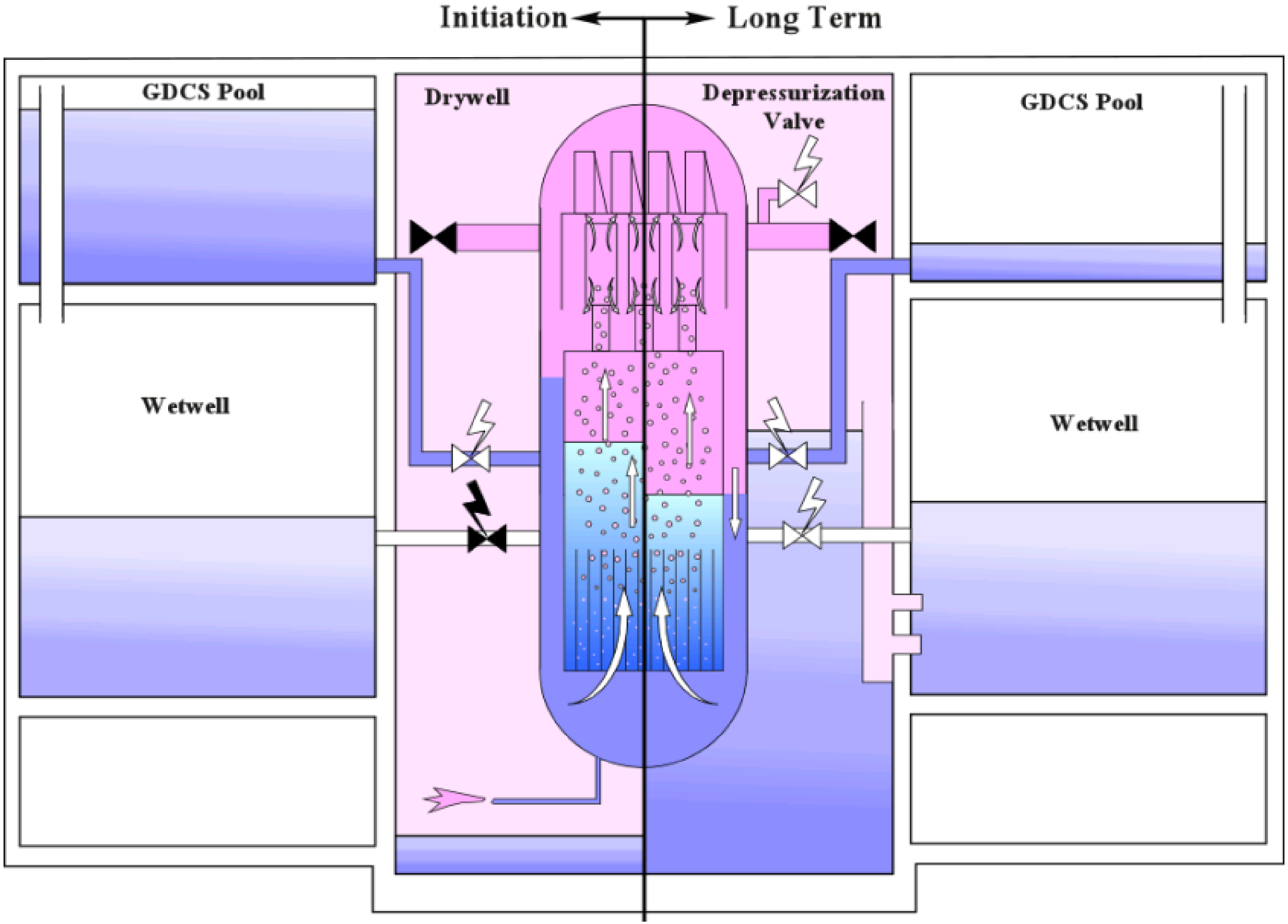
ESBWR Passive Safety

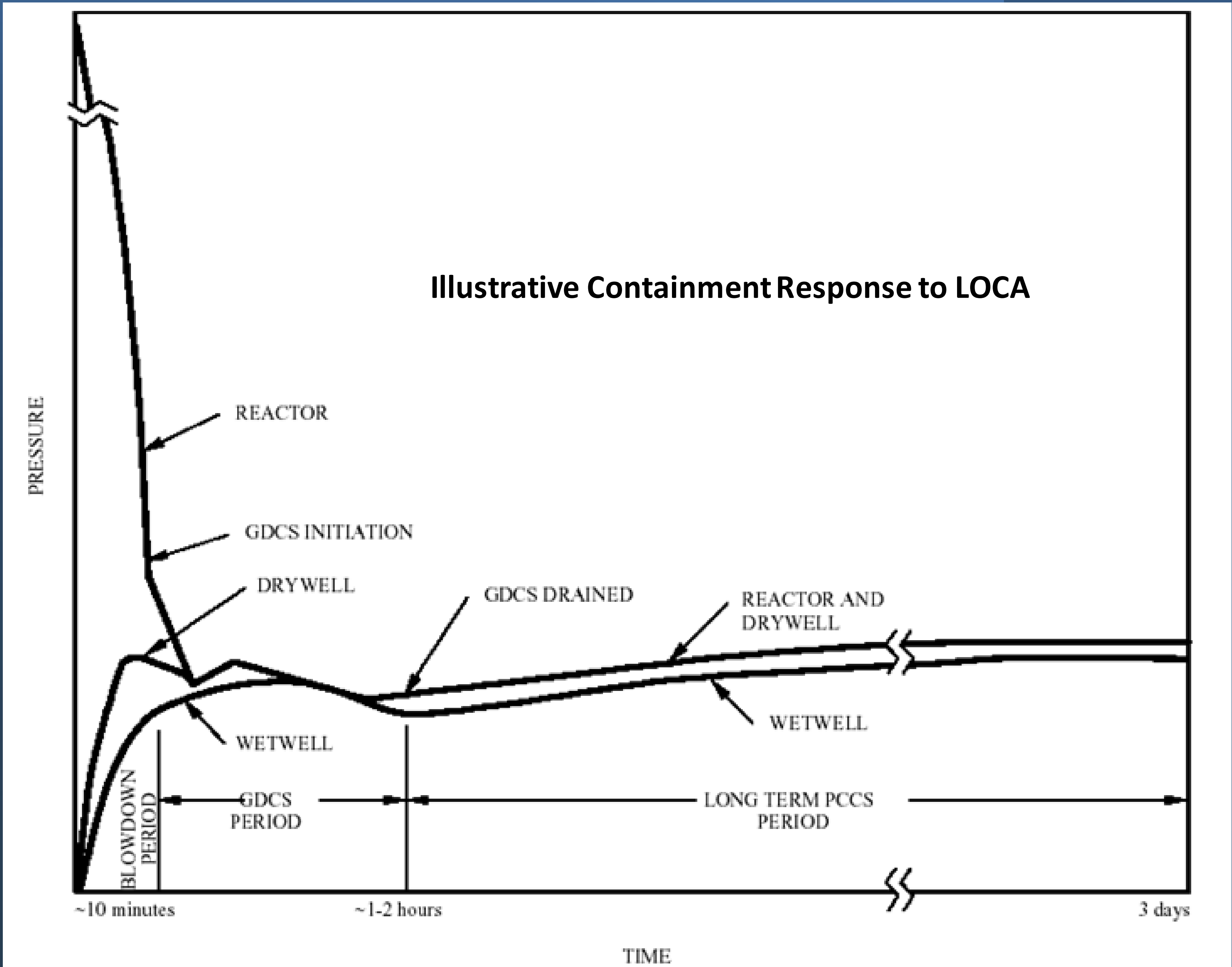


Containment - Emergency Core Cooling System

- Emergency Cooling for LOCA : Gravity-Driven Cooling System (GDCS) in conjunction with the Automatic Depressurization System (ADS) in case of a LOCA.
- Short-term gravity-driven water makeup from four separate water pools located within the upper drywell at an elevation above the active core region.
- Long-term post-LOCA makeup from the suppression pool via Equalizing Valve to meet long-term core decay heat boil-off requirements.
- Floods lower drywell via Deluge Line in the unlikely event of severe accident conditions – core melt.

Gravity Driven Cooling System (GDCS) - Small Pipe Break, Vessel Bottom

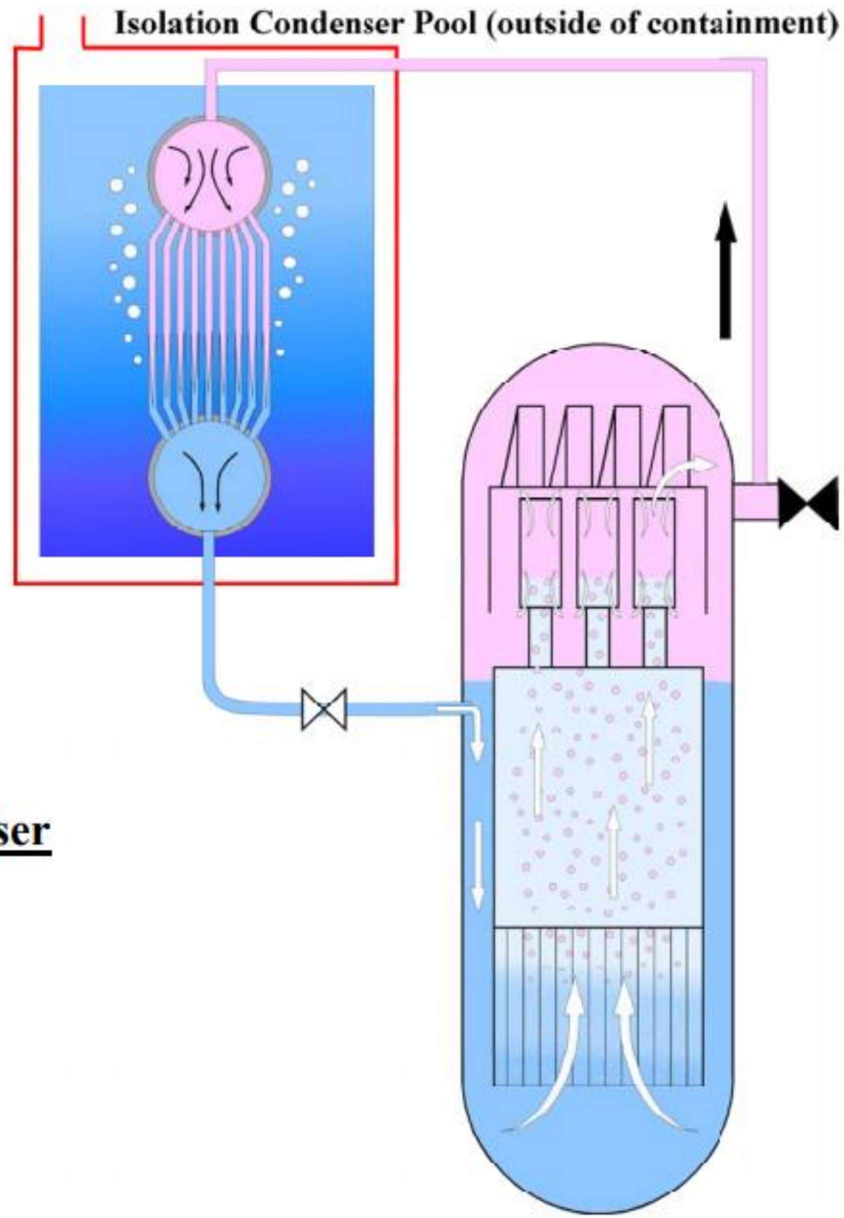



Illustrative Containment Response to LOCA
PRESSURE
REACTOR
GDCS INITIATION
DRYWELL
GDCS DRAINED
REACTOR AND DRYWELL
WETWELL
WETWELL
BLOWDOWN PERIOD
GDCS PERIOD
LONG TERM PCCS PERIOD
~10 minutes
~1-2 hours
3 days
TIME

Reactor Pressure Transient, Inventory Control & Passive Cooling

The Isolation Condenser System (ICS): four heat exchangers, each rated at 30 MWt

- Mitigates reactor pressure transients - limits reactor pressure surge and prevents Safety Relief Valve (SRV) operation following an isolation of the main steam lines.
- Controls coolant inventory to mitigates low reactor level triggering automatic reactor depressurization: the ICS, together with the water stored in the RPV, conserves sufficient reactor coolant volumes to avoid automatic depressurization caused by low reactor water level.
- Provides passive decay heat removal: the ICS removes excess sensible and core decay heat from the reactor via condensing heat exchangers, with minimal loss of coolant inventory.
- Initiation at MSIV closure, vessel high pressure, low level, manual

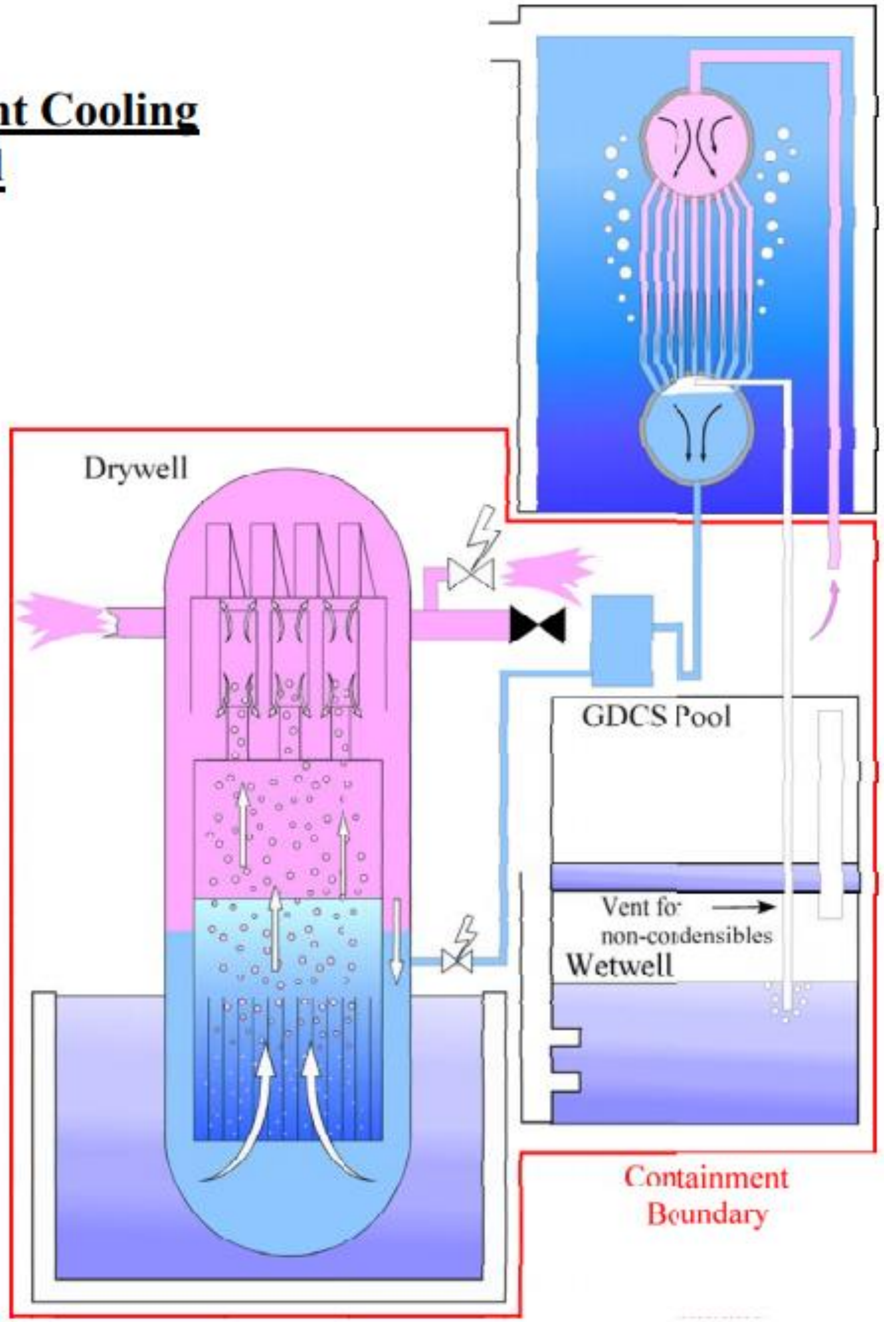


Isolation Condenser Simplified

Passive Containment Cooling System (PCCS)

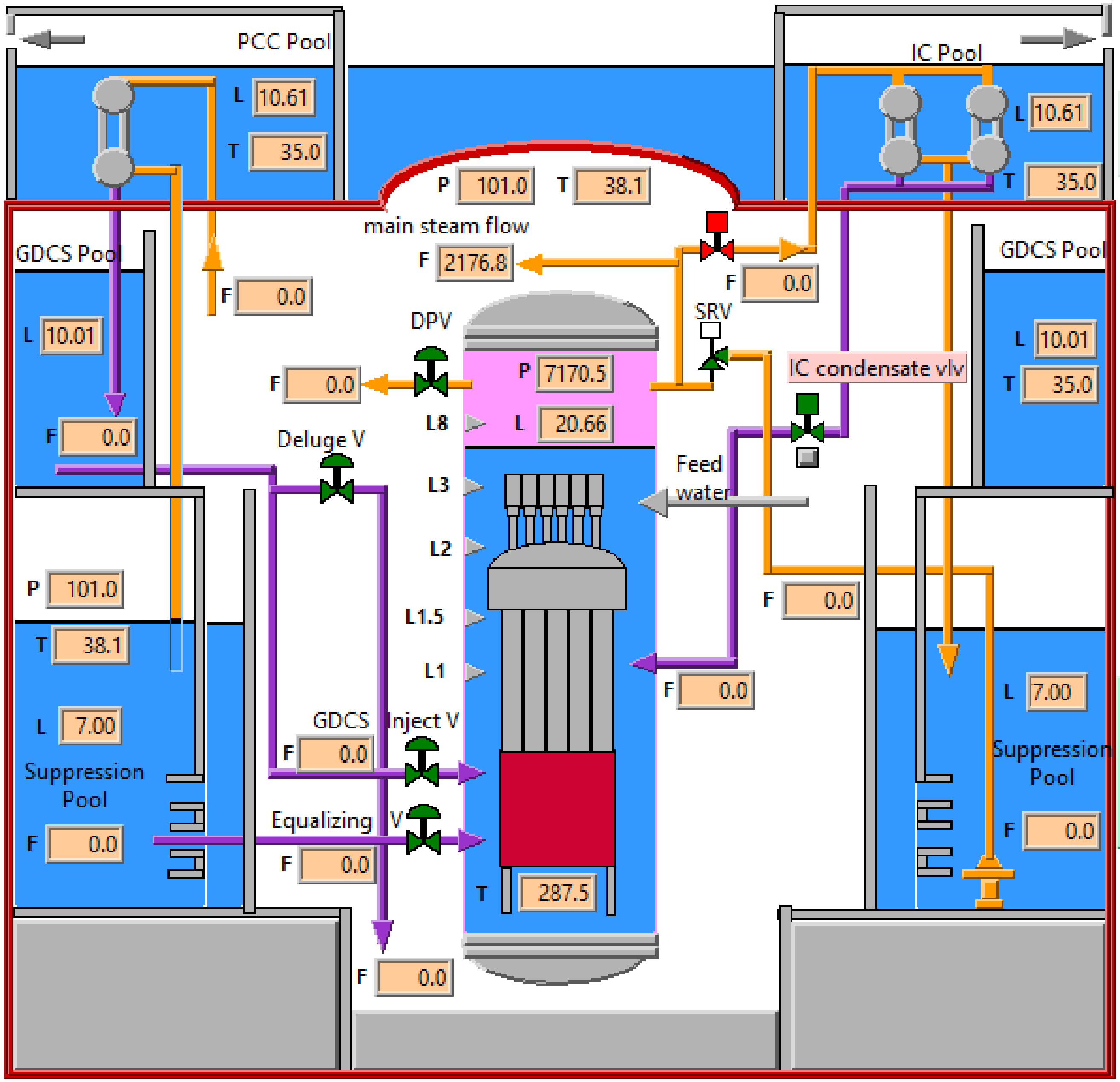
- The PCCS heat exchangers are located above and immediately outside of containment.
- There is sufficient water in the external pools to remove decay heat for at least 72 hours following a postulated design basis accident, and provisions exist for external makeup beyond that, if necessary

Passive Containment Cooling Simplified



Containment Pressure Suppression Control & non-condensables collection

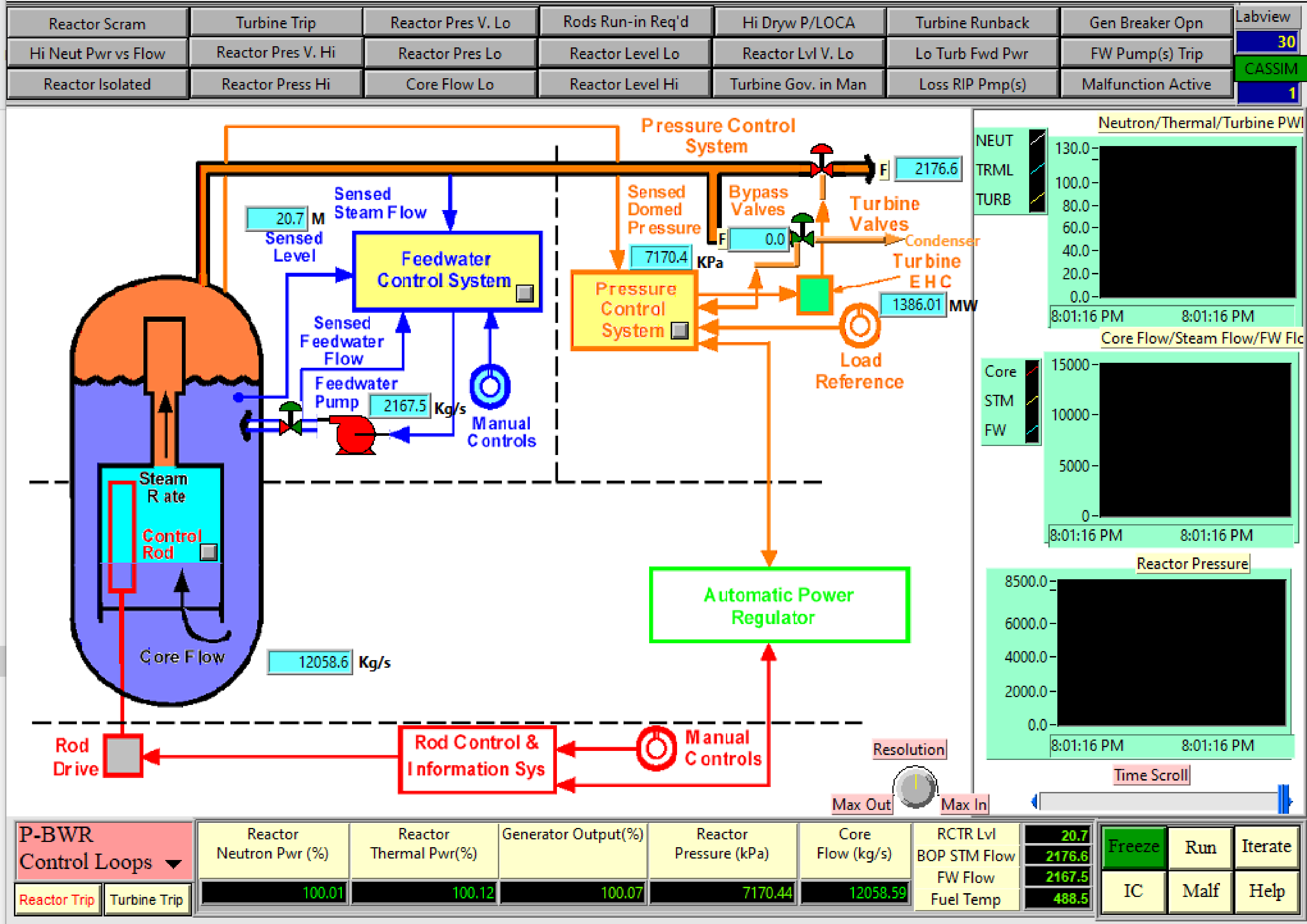
- Drywell pressurization (e.g steam line break, LOCA) - drywell atmosphere & steam pushed into the wetwell via a passage through the partition wall.
- Steam condensed in suppression pool. Non-condensables collected in the gas compression chamber
- Containment vessel can also be vented manually or via rupture disk, to the stack through filter system



- L 9 = 22.39 m – action: Reactor scram
- L 8 = 21.89 m – action: Turbine Trip.
- L 4 = 20.60 m – normal Reactor Level.
- L 3 = 19.00 m – action: Reactor Scram.
- L 2 = 16.50 m – action : initiate Isolation Condenser System; 29 sec delay, close MSIV
- L 1.5 = 13.00 m - action: start ADS blow down; initiate GDCS.
- L 1 = 11.00 m – action: start equalizing valve.

BWR Control Systems

- Reactor Power Control
- Reactor Pressure Control
- Reactor Water Level Control
- Turbine Control
- Turbine Steam Bypass Control



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
30
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
1
Pressure Control System
F 2176.6
Sensed Steam Flow
20.7 M
Sensed Level
Feedwater Control System
Sensed Domed Pressure
Bypass Valves
Turbine Valves
F 0.0
Condenser
7170.4 KPa
Turbine EHC
1386.01 MW
Pressure Control System
Sensed Feedwater Flow
Load Reference
Feedwater Pump
2167.5 Kg/s
Manual Controls
Steam Rate
Control Rod
Automatic Power Regulator
Core Flow
12058.6 Kg/s
Rod Drive
Rod Control & Information Sys
Manual Controls
Neutron/Thermal/Turbine PWI
NEUT
TRML
TURB
130.0
100.0
80.0
60.0
40.0
20.0
0.0
8:01:16 PM
8:01:16 PM
Core Flow/Steam Flow/FW Flc
Core
STM
FW
15000
10000
5000
0
8:01:16 PM
8:01:16 PM
Reactor Pressure
8500.0
6000.0
4000.0
2000.0
0.0
8:01:16 PM
8:01:16 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Control Loops
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
100.01
Reactor Thermal Pwr(%)
100.12
Generator Output(%)
100.07
Reactor Pressure (kPa)
7170.44
Core Flow (kg/s)
12058.59
RCTR Lvl
20.7
BOP STM Flow
2176.6
FW Flow
2167.5
Fuel Temp
488.5
Freeze
Run
Iterate
IC
Malf
Help

BWR Reactor Power Control

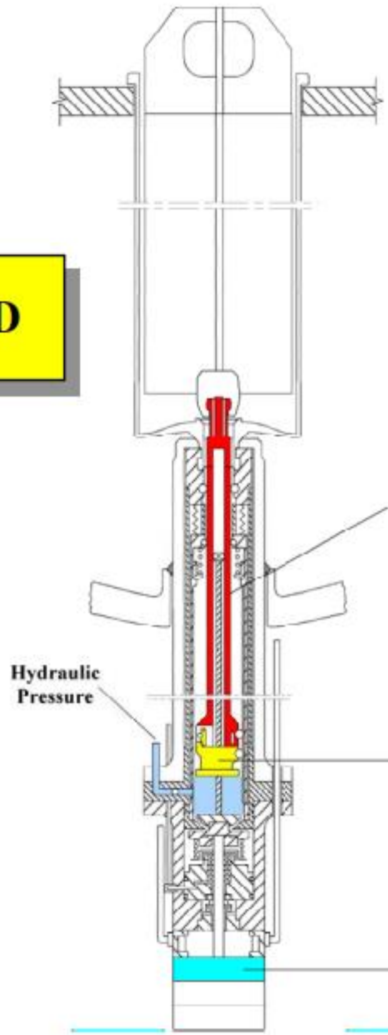
- Typical BWR Reactor power control consists of control rods and recirculation flow control
- Control Rods known as Fine Motion Control Rods (FMCRD) (neutron absorbing materials) maintain a constant desired power level by a adjusting their positions ~ 2 % per sec.
- In ABWR, recirculation flow control by coolant pumps also controls reactor power by altering the density of water used as moderator. The flow rate is adjusted by a variable speed pump. Power changes ~ 30 % per minute.
- In ESBWR & BWRX-300 (natural circulation), variable feedwater temperature control has been added to provide a second means of reactivity control in addition to the fine motion control rod drives.

Rod Control System

- The Control Rod Drive System is composed of three major elements using fine position digital motor drive & hydraulic drive:
- (1) the fine motion control rod drive, FMCRD mechanisms;
- (2) the hydraulic control unit (HCU) assemblies;
- (3) the control rod drive hydraulic subsystem (CRDH).

FMCRD

At SCRAM

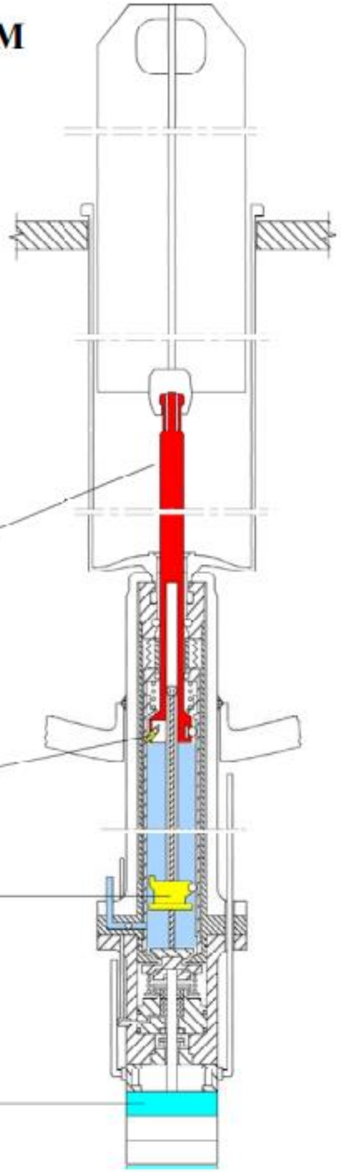


At SCRAM:
1. The hydraulic pressure pushes the hollow piston up and away from the ball nut, forcing the control rod into the core

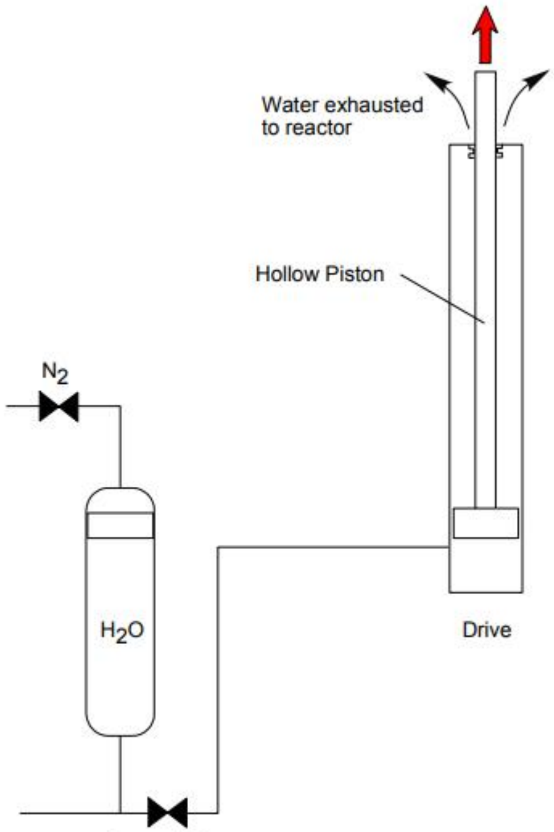
2. The motor starts to drive the ball nut up to mate with the hollow piston

3. The latches keep the hollow piston from dropping

After SCRAM



Fine Motion CRD



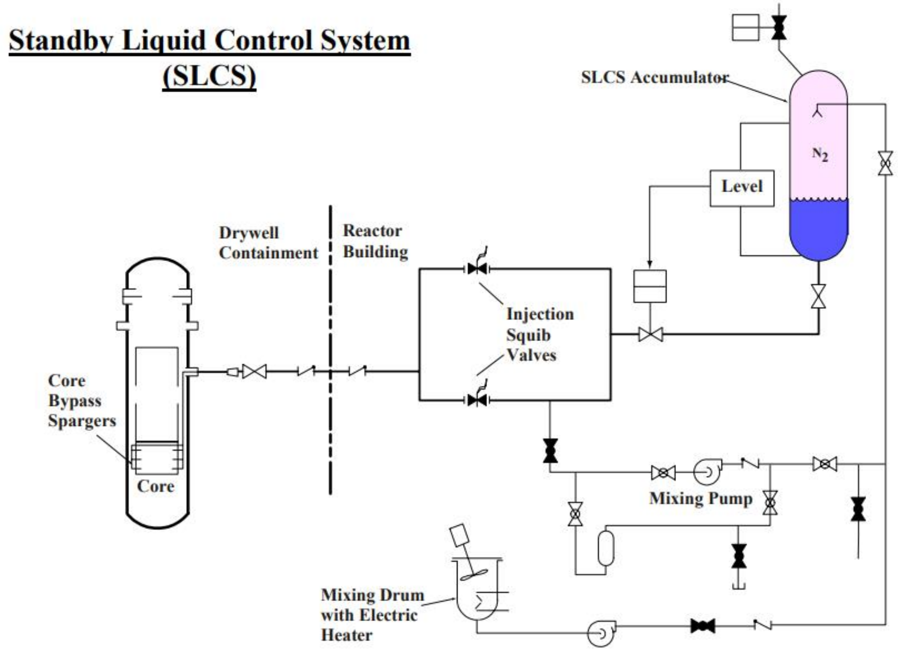
Rod Control System (cont'd)

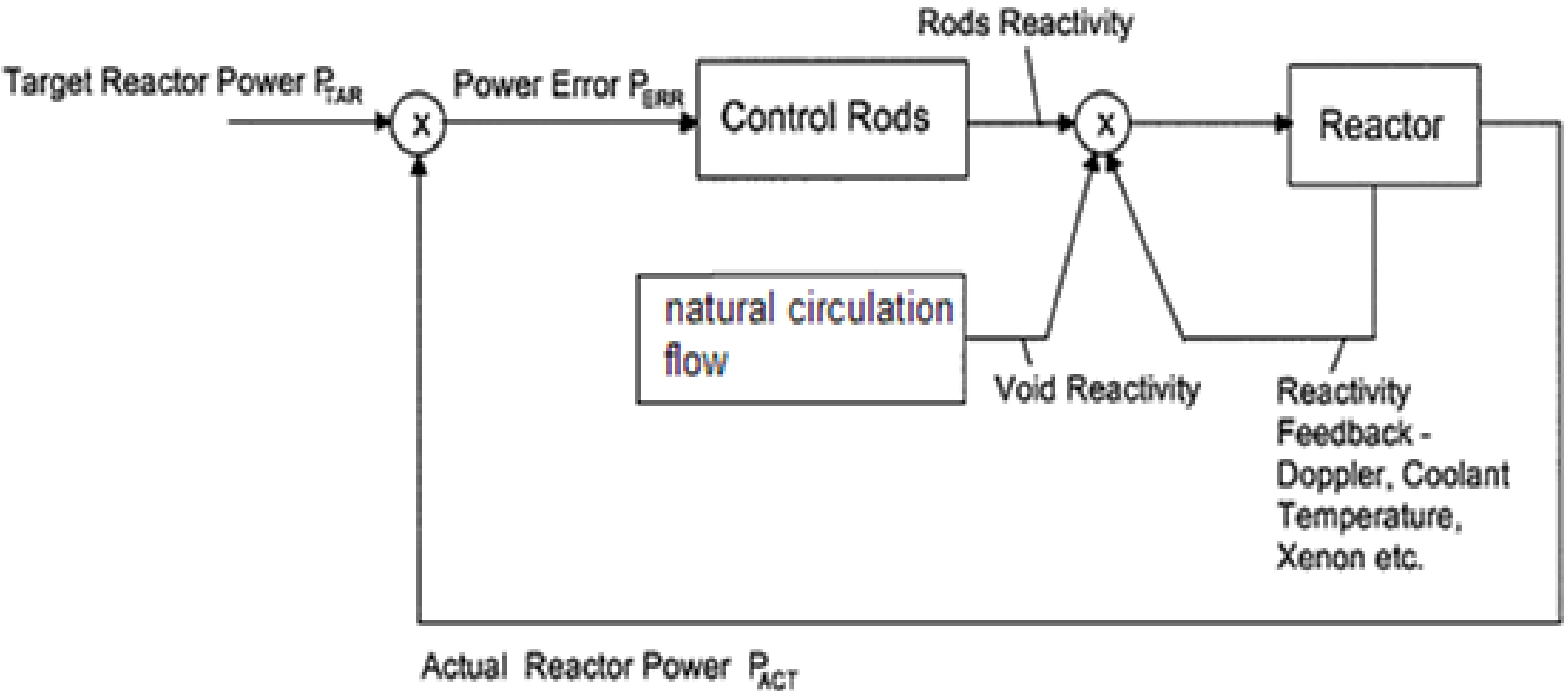
The FMCRDs together with the other components are designed to provide:

- electric-motor-driven positioning for normal insertion and withdrawal of the control rods;
- hydraulic-powered rapid control rod insertion (scram: 100 % insertion in 3 sec.) in response to manual or automatic signals from the Reactor Protection System (RPS);
- electric-motor-driven "Run-Ins" of some or all of the control rods as a path to rod insertion for reducing the reactor power by a sizable amount.

Reactivity Control - Backup

- **Standby Liquid Control System (SLCS)**
  - Meets alternate shutdown control requirements
    - » Also meets ATWS requirements (10CFR50.62)
  - Accumulator driven boron injection system
    - » Two divisions
  - Nitrogen pressurized accumulators

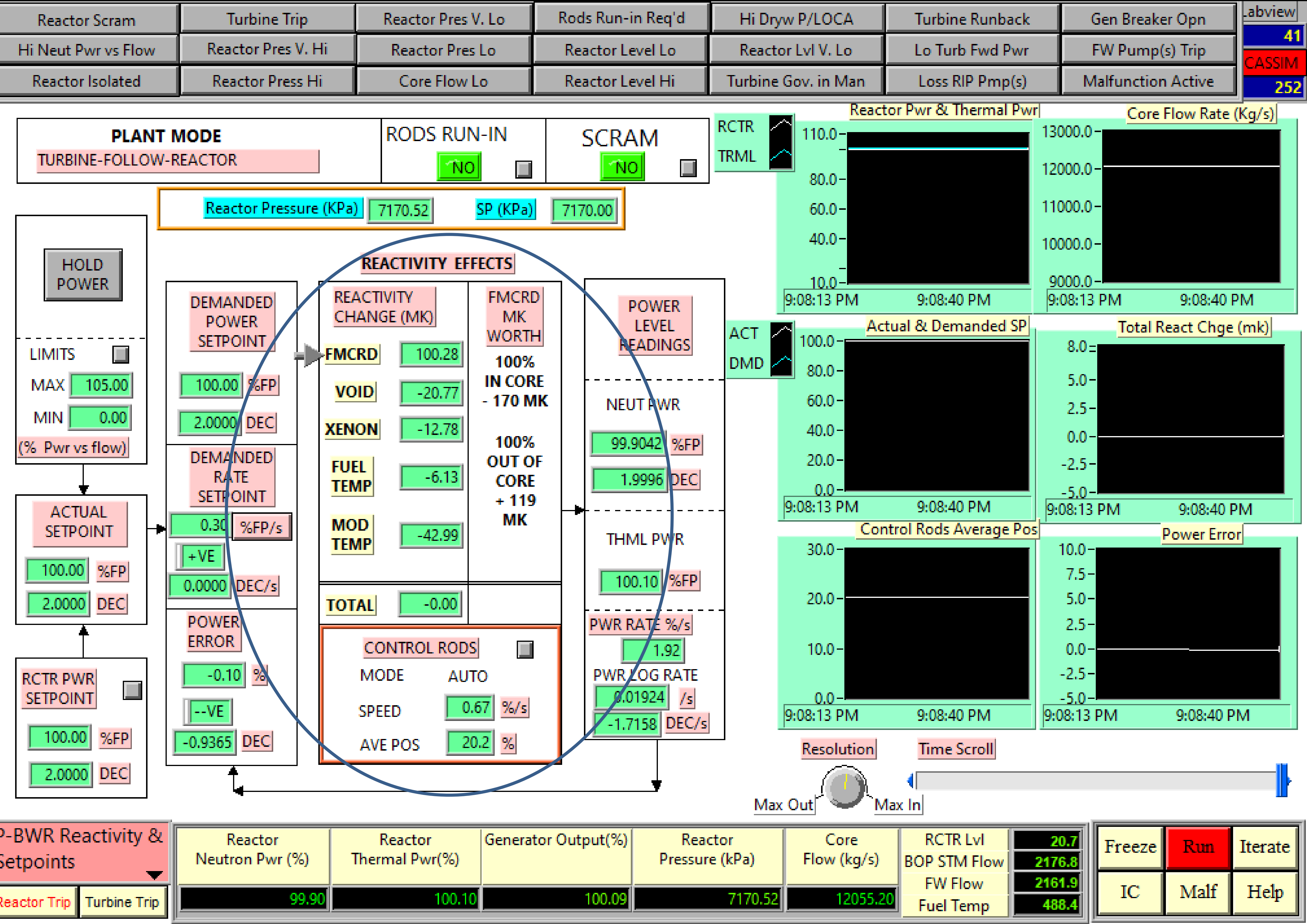




Target Reactor Power P_TAR
x
Power Error P_ERR
Control Rods
Rods Reactivity
x
Reactor
natural circulation flow
Void Reactivity
Reactivity Feedback - Doppler, Coolant Temperature, Xenon etc.
Actual Reactor Power P_ACT

Rod Control System & Reactivity Effects - Simulator

- There are 269 FMCRDs in total.
- For the Passive BWR Simulator, assumptions are taken as follows for reactivity worth, rods position at 100 % FP:
  - -170 mk when all of them are 100% IN-CORE;
  - +119 mk when all of them are 100 % Out-of-core
- At 100 % reactor FP, the control rods are ~ 20 % in core, providing +100 mk to counteract the negative reactivities feedback from:
  - Void : - 20 mk;
  - Xenon : - 13 mk;
  - Fuel Temperature: -6 mk
  - Moderator Temperature: - 43 mk;
  - Burnable Gd Poison in Fuel: - 18 mk
- To achieve the net reactivity change Delta K = 0

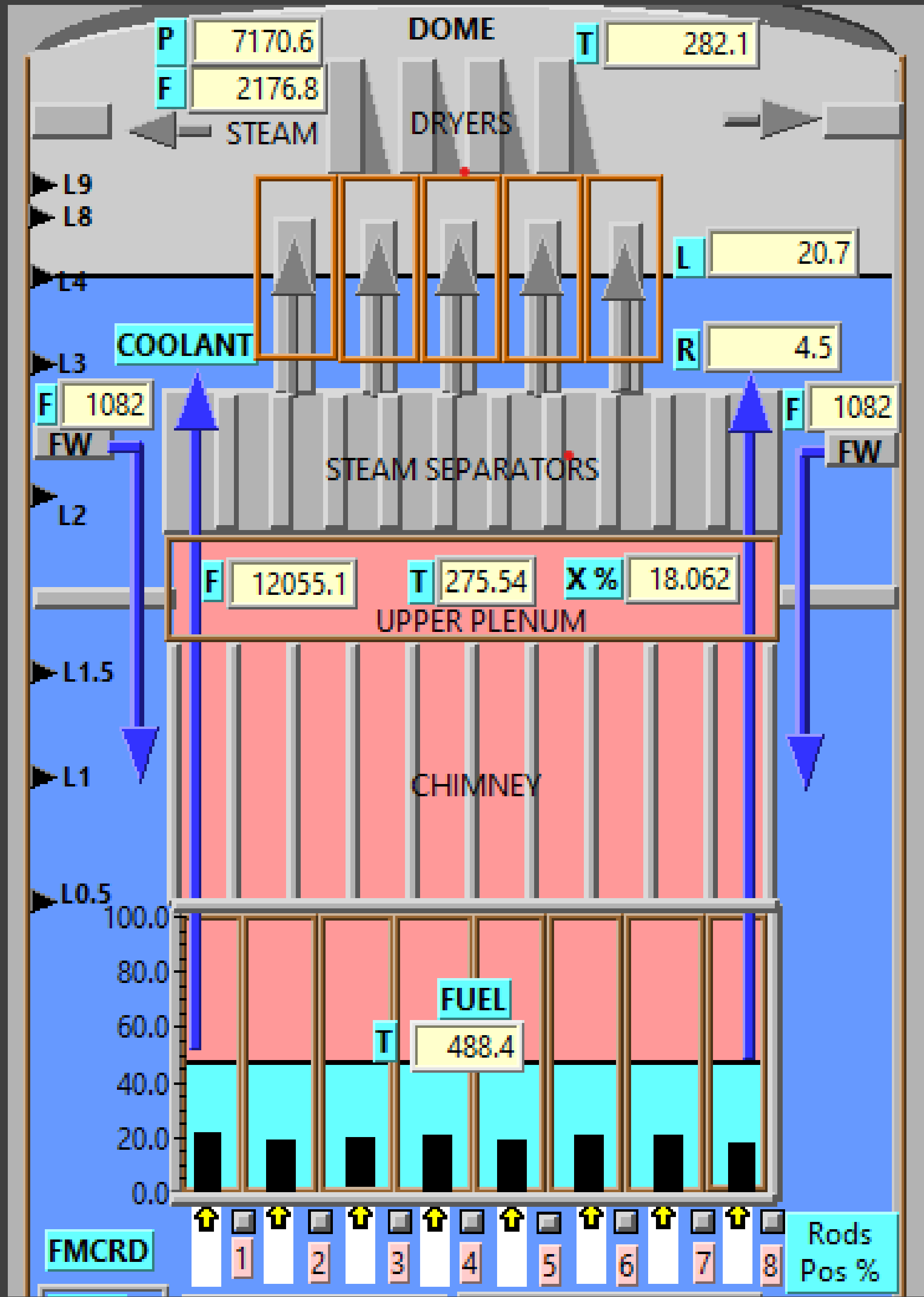

Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
41
CASSIM
252
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
NO
SCRAM
NO
Reactor Pressure (KPa)
7170.52
SP (KPa)
7170.00
HOLD POWER
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
ACTUAL SETPOINT
100.00 %FP
2.0000 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED POWER SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED RATE SETPOINT
0.30 %FP/s
+VE
0.0000 DEC/s
POWER ERROR
-0.10 %
--VE
-0.9365 DEC
REACTIVITY EFFECTS
REACTIVITY CHANGE (MK)
FMCRD 100.28
VOID -20.77
XENON -12.78
FUEL TEMP -6.13
MOD TEMP -42.99
TOTAL -0.00
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 20.2 %
POWER LEVEL READINGS
NEUT PWR
99.9042 %FP
1.9996 DEC
THML PWR
100.10 %FP
PWR RATE %/s
1.92
PWR LOG RATE
0.01924 /s
-1.7158 DEC/s
RCTR
TRML
Reactor Pwr & Thermal Pwr
110.0
80.0
60.0
40.0
10.0
9:08:13 PM
9:08:40 PM
Core Flow Rate (Kg/s)
13000.0
12000.0
11000.0
10000.0
9000.0
9:08:13 PM
9:08:40 PM
ACT
DMD
Actual & Demanded SP
100.0
80.0
60.0
40.0
20.0
0.0
9:08:13 PM
9:08:40 PM
Total React Chge (mk)
8.0
5.0
2.5
0.0
-2.5
-5.0
9:08:13 PM
9:08:40 PM
Control Rods Average Pos
30.0
20.0
10.0
0.0
9:08:13 PM
9:08:40 PM
Power Error
10.0
7.5
5.0
2.5
0.0
-2.5
-5.0
9:08:13 PM
9:08:40 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
99.90
Reactor Thermal Pwr(%)
100.10
Generator Output(%)
100.09
Reactor Pressure (kPa)
7170.52
Core Flow (kg/s)
12055.20
RCTR Lvl 20.7
BOP STM Flow 2176.8
FW Flow 2161.9
Fuel Temp 488.4
Freeze
Run
Iterate
IC
Malf
Help

The Passive BWR Reactor Neutronic data used for the Simulator

Passive BWR Reactor Neutronics Data – Only used for Simulator	
Reactivity Coefficients (nominal values; actual value vary with temperature). Note: Core fuel burnup reactivity change is not simulated.	Source: US NRC BWR Training Manual (public domain)
Moderator Void Coefficient (dK/K/% voids)	- 1 x 10-3; ~ - 1 mk/% voids or –100 PCM/% voids
Moderator Temperature Coefficient (dK/K/deg C)	- 1.8 x 10-4; ~ - 0.18 mk/deg C or – 18 PCM/deg C
Fuel Temperature Coefficient (dK/K/deg C)	- 1.8 x 10-5 ~ - 0.018 mk/deg C or – 1.8 PCM/deg C

Simulator Assumptions:

- The 269 FMCRDs are divided into 8 Banks of control rods; each bank reactivity worth ~ 36 mk
- The FMCRDS can be in Auto or Manual Control.
- In Auto, the rods banks move together and are controlled by the reactor power regulation system.
- In actual design, the Rod Sequence Control System (RSCS) provides control rod selection and movement interlocks within banks to enforce a mandatory sequence of movements when withdrawing or inserting control rods.
- A mandatory sequence is required to maintain relative positions between adjacent control rods within predetermined limits in order to limit the reactivity worth of any given amount of control rod motion.
- RSCS is not modelled in the Simulator, as individual FMCRD is not simulated.
- In Manual, the individual banks of rods are controlled manually: "IN"; "OUT"; "STOP"



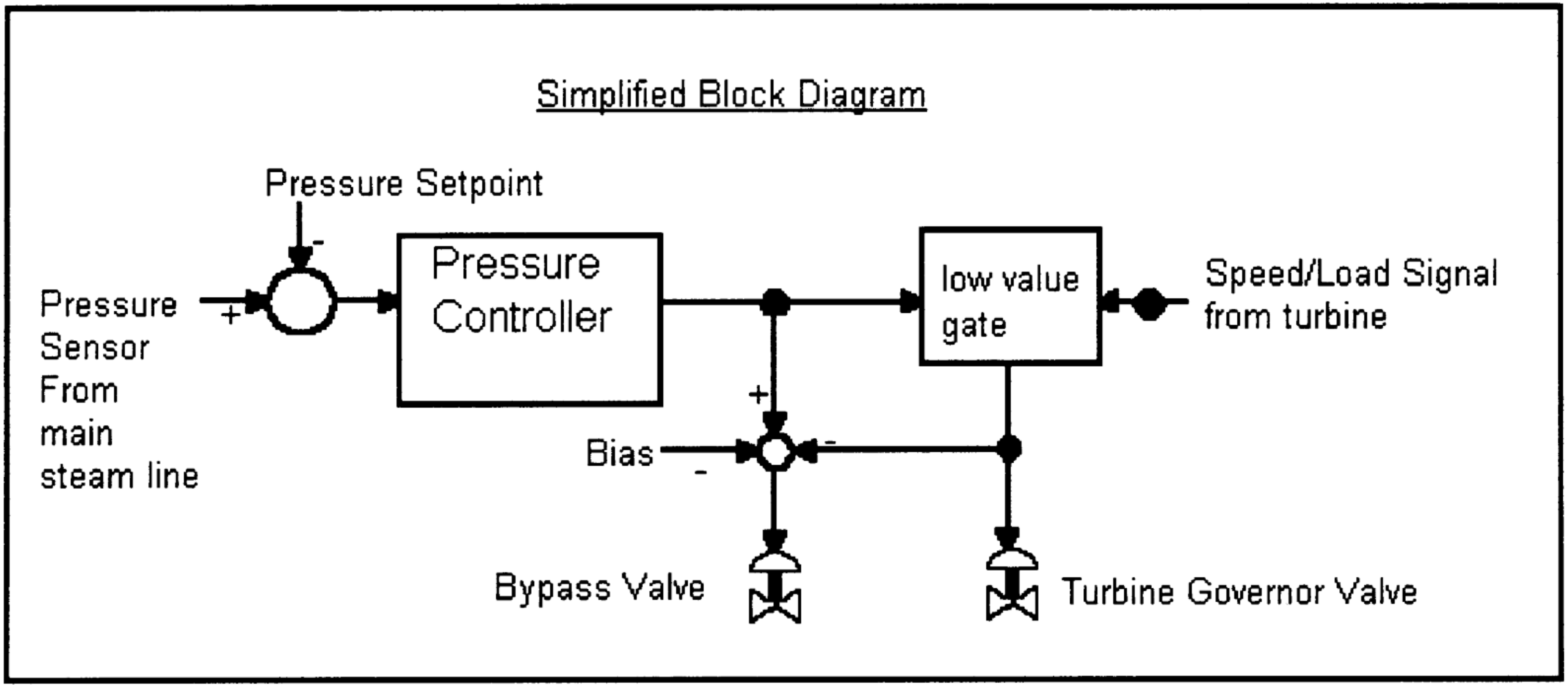
Rod Control System – Simulator Assumptions

- The full speed travel time for the rod movement during power maneuvering is typically 60 sec., or considering for the total FMCRDs in Auto mode, where all the rods move together, the reactivity change rate is ~ 4.8 mk per sec.

- Considering moving the banks of rods individually under Manual Mode, then the reactivity change rate for each bank under manual mode control is ~ 0.6 mk per sec.

Reactor Pressure Control

- In normal operation, the reactor pressure is automatically controlled to be constant.
- A pressure controller to regulate the turbine inlet steam pressure by opening and closing the turbine governor control valve and the turbine bypass valve. Currently, the reactor pressure setpoint is set at plant design pressure of 7170 KPa.



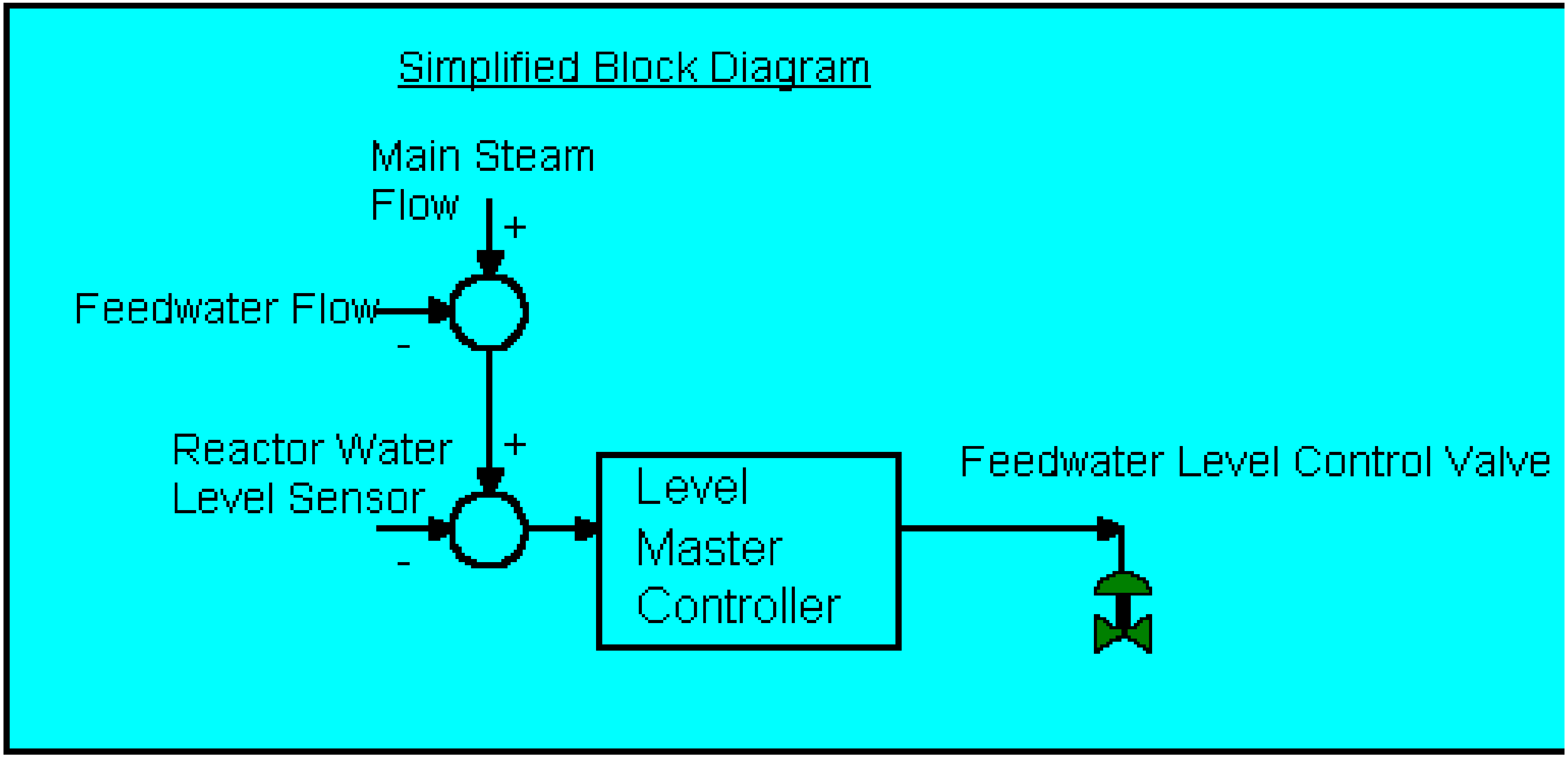
Simplified Block Diagram

Turbine Control

- The turbine control employs an electrohydraulic control system (EHC) to control the turbine valves.
- Under normal operation, the Reactor Pressure Control (RPC) unit keeps the inlet pressure of the turbine constant, by adjusting the opening of the turbine "speeder gear" which controls the opening of the turbine governor valve opening.
- Should the generator speed increase due to sudden load rejection of the generator, the speed control unit of the EHC has a priority to close the turbine governor valve over the Reactor Pressure Control (RPC) unit.

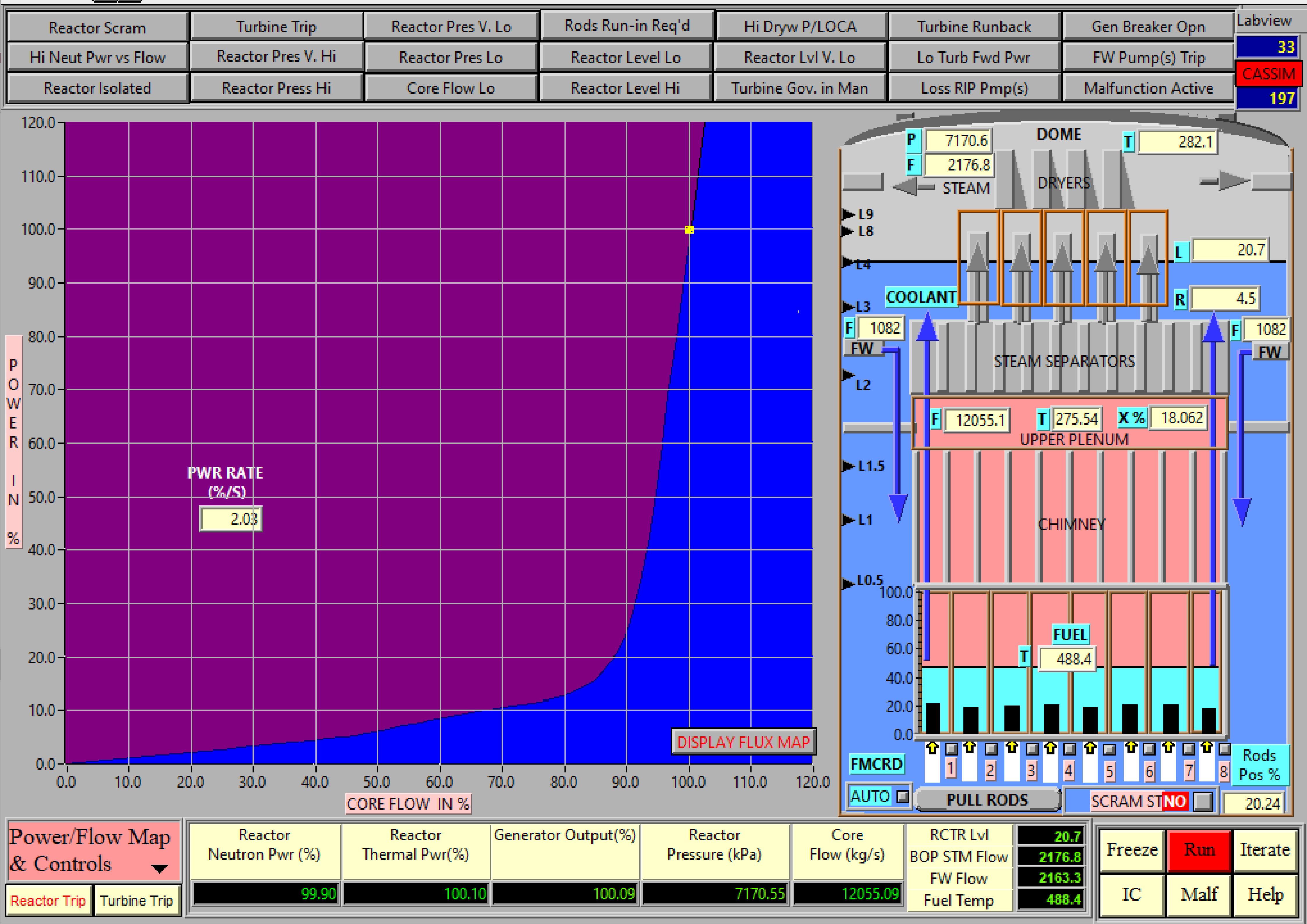
Reactor Water Level Control

- The flow of feedwater is automatically controlled to maintain the specified water level by a "three elements" control scheme: steam flow, feedwater flow, water level.
- The valve opening of the feedwater control valve provided at the outlet of the feedwater pumps is regulated by the control signal as result of this "three-element" control scheme.


Simplified Block Diagram
Main Steam Flow
+
Feedwater Flow
-
Reactor Water Level Sensor
+
-
Level Master Controller
Feedwater Level Control Valve

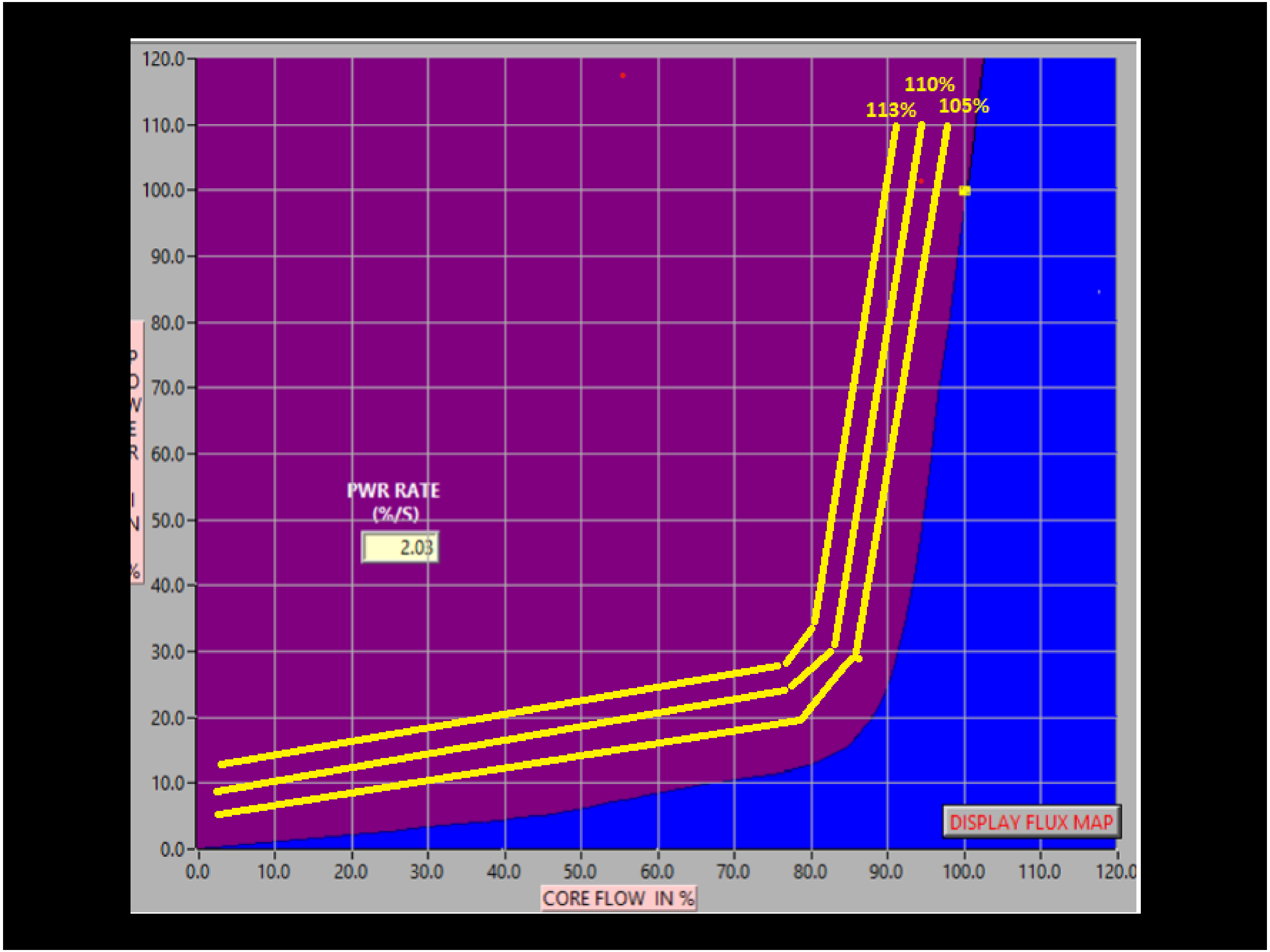
Power/Flow Map

- The Power Flow Map is a representation of reactor power vs. Core flow. The horizontal axis is the core flow in % of full power flow. The vertical axis is reactor neutron power in % full power.
- Any operation path that changes the power and the flow from one condition to another condition through control rod maneuver and/or recirculation flow change can be traced on this map.

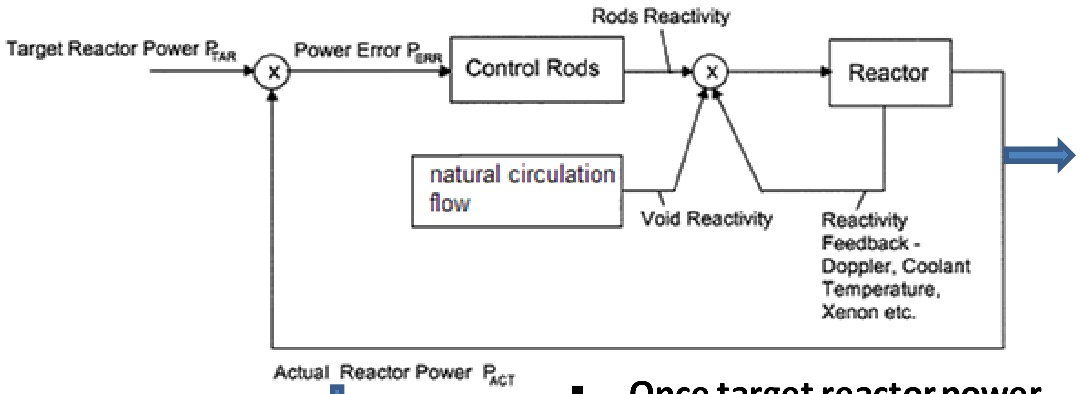

Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
33
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
197
POWER IN %
CORE FLOW IN %
PWR RATE (%/S)
2.03
DISPLAY FLUX MAP
DOME
P 7170.6
F 2176.8
T 282.1
STEAM
DRYERS
L9
L8
L4
L 20.7
R 4.5
COOLANT
L3
F 1082
FW
F 1082
FW
STEAM SEPARATORS
L2
F 12055.1
T 275.54
X % 18.062
UPPER PLENUM
L1.5
L1
CHIMNEY
L0.5
FUEL
T 488.4
FMCRD
1 2 3 4 5 6 7 8
Rods Pos %
AUTO
PULL RODS
SCRAM ST NO
20.24
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.90
Reactor Thermal Pwr(%) 100.10
Generator Output(%) 100.09
Reactor Pressure (kPa) 7170.55
Core Flow (kg/s) 12055.09
RCTR Lvl 20.7
BOP STM Flow 2176.8
FW Flow 2163.3
Fuel Temp 488.4
Freeze
Run
Iterate
IC
Malf
Help

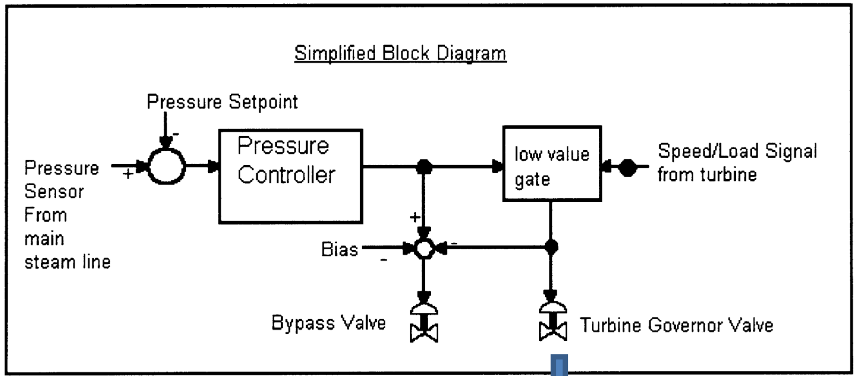
Power /Flow Map

- Limits are imposed to prevent operation in certain areas of the Power - Flow Map to maintain core thermal limits and to avoid operation above licensed power level - there are three measures to prevent that:
- Control Rods Withdrawal "Blocked" (if> 105%) ; Control Rods "Run-in" (if > 110%); Scram (if > 113%).
- In particular, limits are imposed to avoid "low flow - high power" region – reactor scram on high neutron flux and low core flow



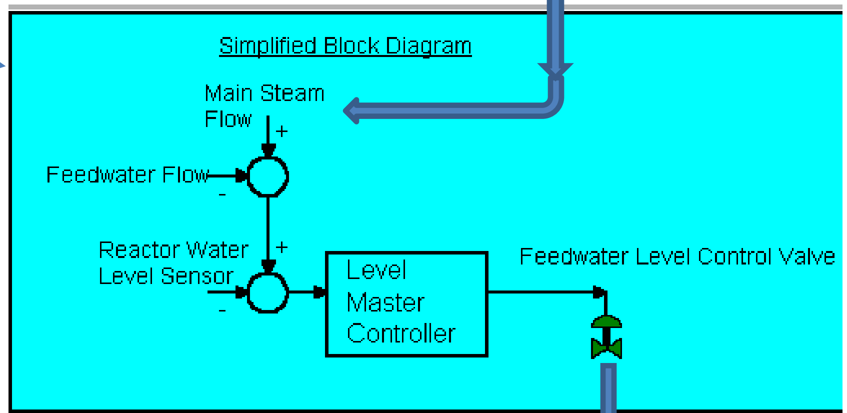
113%
110%
105%
PWR RATE (%/S)
2.03
DISPLAY FLUX MAP
POWER IN %
CORE FLOW IN %
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0





Power – Flow Map

- Once target reactor power reached, rod movement is stopped temporarily.
- Check core flow, if delta core flow error exists in Power Flow Map, add a bias to flow controller, to increase or decrease core flow until delta flow error = 0
- If in the process of changing core flow, reactor power deviates from setpoint, rod movement resumes to reach setpoint again



Click to add text

Automatic Power Regulator

Reactor Power-Flow Controls Schematic

Protection Systems -

- Reactor Power Protection
  - Reactor Scram – FMCRD;
  - Backup SLCS for ATWS

REACTOR SCRAM PARAMETERS

FIRST OUT	SCRAM CAUSES
○	High Neutron Flux / Low Core Flow
○	High Drywell Pressure /LOCA detected
○	Reactor Water Level Low
○	Reactor Pressure High
○	Reactor Water Level Abnormally High
○	Main Steam Isolation Valve Closed/ Reactor Isolated
○	Main Steam Line Radioactivity High
○	Turbine Power/Load Unbalance - Loss of Line
○	Earthquake Acceleration Large
○	Manual Scram

- Control Rods withdrawal blocked > 103 power; Control Rods Run-in > 110 % power; Feedwater temp deviation > limit

- Reactor Pressure
  - Turbine Bypass
  - ICS
  - SRV
- Reactor Coolant Inventory
  - ICS
  - GDCS
  - Wetwell
  - External sources
- Reactor Decay Heat Removal
  - ICS
  - PCCS
- Containment Pressure
  - Wetwell
  - Emergency vent

Layers of Defense against DBA

The safety systems for passive BWR involve the following layers of defense against design basis events accidents:

(1) Reactor protection — control rods blocked; control rods run-in, and reactor scram.

(2) SLC will be activated in the event of ATWS.

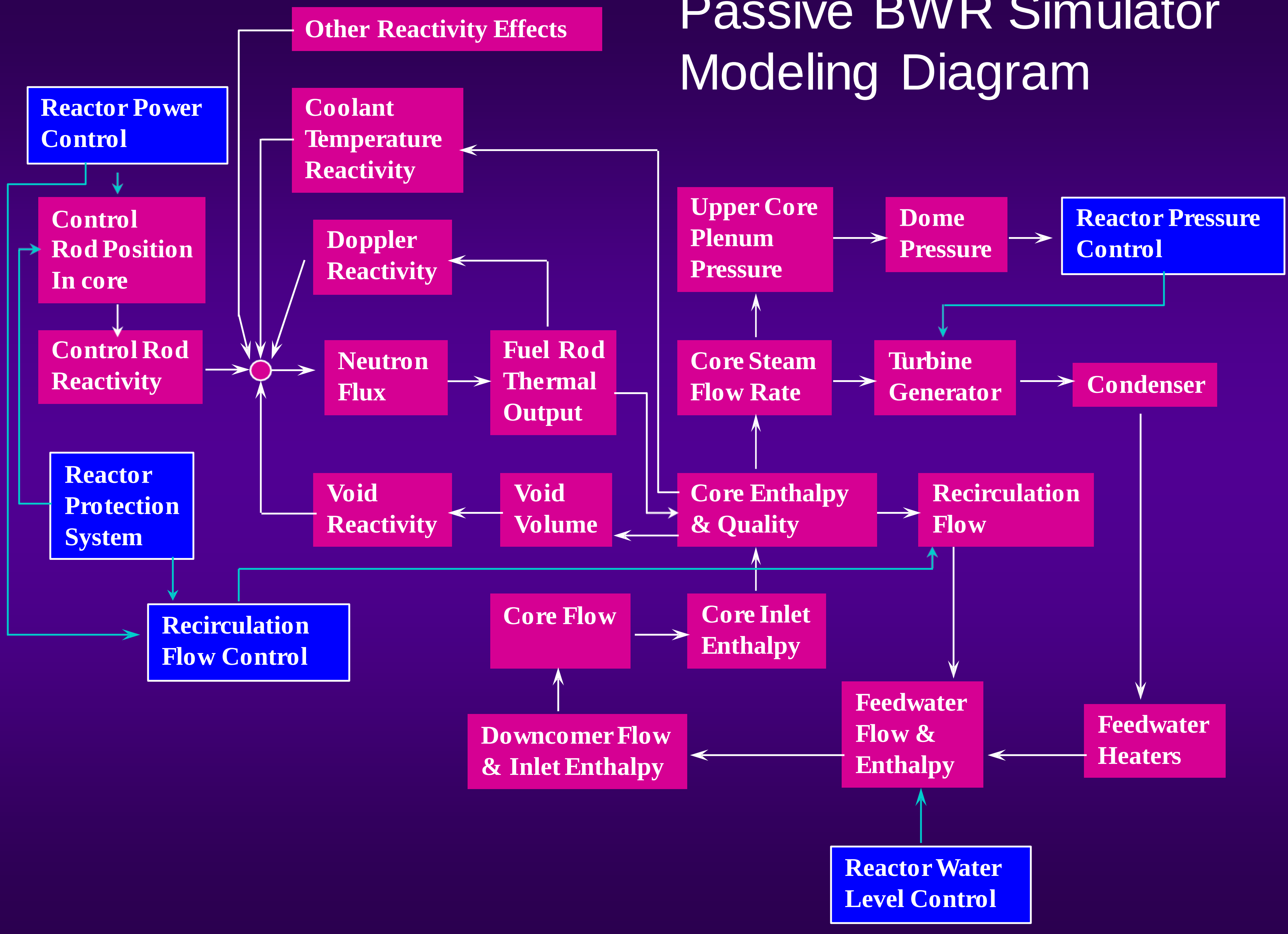
(3) Containment isolation — in the event of steam line break, feedwater line break, LOCA.

(4) ECCS actuation — ICS for decay heat removal; PCC to remove steam from drywell; Drywell Vents open to Suppression Pool to allow quick pressure relief for containment, and to discharge non-condensable (e.g. hydrogen) to Wetwell.

(5) ADS actuation to depressurize reactor pressure to a level to bring in GDCS flows by gravity.

(6) Suppression pool cooling - to provide long term cooling of the isolated containment in case of LOCA.

Passive BWR Simulator Modeling Diagram



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TCS 23 (iaea.org)

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