The Abdus Salam International Centre for Theoretical Physics

2374-8

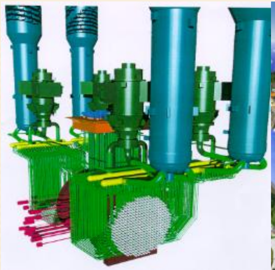
Joint ICTP-IAEA School of Nuclear Energy Management

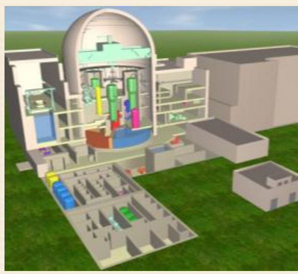
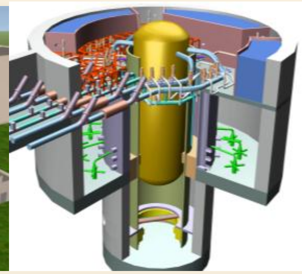
5 - 23 November 2012

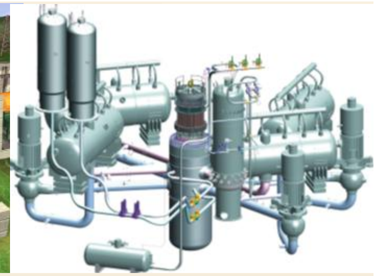
Advanced Nuclear Reactor Designs and Technologies for Near Term Deployment

SUBKI Mochammad Hadid

International Atomic Energy Agency Division of Nuclear Power
Nuclear Power Engineering Section
Wagramerstrasse 5, P.O. Box 100
A-1400 Vienna
AUSTRIA





The ICTP-IAEA School on *"Nuclear Energy Management"*, 5 – 23 November 2012, in Trieste, ITALY:

Advanced Nuclear Reactor Designs and Technologies for Near Term Deployment

Dr. Mochammad Hadid Subki
Technical Lead, SMR Technology Development
Nuclear Power Technology Development Section
Division of Nuclear Power, Department of Nuclear Energy

IAEA
International Atomic Energy Agency

Outline

- Near Term Deployable Reactor Designs
- Current Newcomer Countries Plans
- Near Term Deployment Issues
- Global Trends in Advanced Reactor Designs
- OPERATING FUNDAMENTALS (BWR, PWR and PHWR)
  - 11 pages basic presentations on BWR, PWR and PHWR principles
- MAIN TECHNICAL FEATURES of ADVANCED REACTORS for NEAR TERM DEPLOYMENT
  - 63 pages featured technical presentations of 9 advanced reactor designs
- Identified Issues from the TEPCO Fukushima Daiichi Nuclear Accident
- Safety and Technical Improvements in Operating Reactors
- What's New in Global SMR Development Activities?
- IAEA Responses to Global Trends
- Summary and Conclusions

Near Term Deployable Reactor Designs

Country –Designer	Reactor Identification	Type	Power, MW(e)	Status
CANADA				
Candu Energy Inc.	EC6 & ACR-1000	PHWR	740 & 1000	Detailed design
CHINA				
CNNC	ACP-1000	PWR	1000	Detailed design
CGNPC	ACPR-1000	PWR	1150	Detailed design
CNNC	CNP-300	PWR	375	In operation & under construction
FRANCE				
AREVA	EPR	PWR	1600	Under construction
FRANCE-JAPAN				
ATMEA	ATMEA1	PWR	1100	Detailed design
INDIA				
NPCIL	PHWR-220, PHWR-540 & -700	PHWR	220, 540, and 700	In operation & under construction
IGCAR	PFBR-500	SFR	500	**In commissioning - Prototype**

Near Term Deployable Reactor Designs

Country –Designer	Reactor Identification	Type	Power, MW(e)	Status
JAPAN-USA				
Hitachi-GE, Toshiba-Westinghouse	ABWR	BWR	1350 – 1550	In operation & under construction
Mitsubishi Heavy Industries, Ltd.	APWR	PWR	1400 – 1700	Detailed design
REPUBLIC of KOREA				
KHNP	OPR-1000	PWR	1000	In operation & under construction
KHNP	APR-1400	PWR	1400	Under construction
KAERI	SMART	Integral PWR	100	Certified design
RUSSIAN FEDERATION				
OKB Gidropress	VVER-1000	PWR	1000	In operation
OKB Gidropress	VVER-1200	PWR	1200	Under construction
OKBM Afrikantov	KLT-40s	Floating PWR	35	Under construction

Near Term Deployable Reactor Designs

Country –Designer	Reactor Identification	Type	Power, MW(e)	Status
RUSSIAN FEDERATION				
AKME Engineering	SVBR-100	LBFR	100	Detailed design
UNITED STATES of AMERICA				
GE Hitachi	ESBWR	BWR	1550	Detailed design
Toshiba – Westinghouse	AP1000	PWR	1000	Under construction
ARGENTINA				
CNEA	CAREM-25	Integral PWR	27	Prototype under construction

IAEA

Current Newcomer Countries Plan

Country	Grid Capacity in GWe	Current Deployment Plan
Bangladesh	5.8	2 x 1000 MWe PWRs in Rooppur in 2018
Vietnam	15.19	4 x 1000 MWe PWRs in Ninh Thuan #1 by 2020 4 x 1000 MWe PWRs in Ninh Thuan #2 by 2025
Jordan	2.6	2 x 1000 - 1100 MWe PWR in + possible interest in SMR
UAE	23.25	4 x 1400 MWe PWR in Braka by 2018
Belarus	8.03	2 x 1200 MWe PWR in Ostrovets by 2018
Turkey	44.76	4 x 1200 MWe PWR in Akkuyu by 2022
Malaysia	25.54	2 x 1000-1200 MWe LWRs, 1st unit by 2021

IAEA

Near Term Deployment Issues

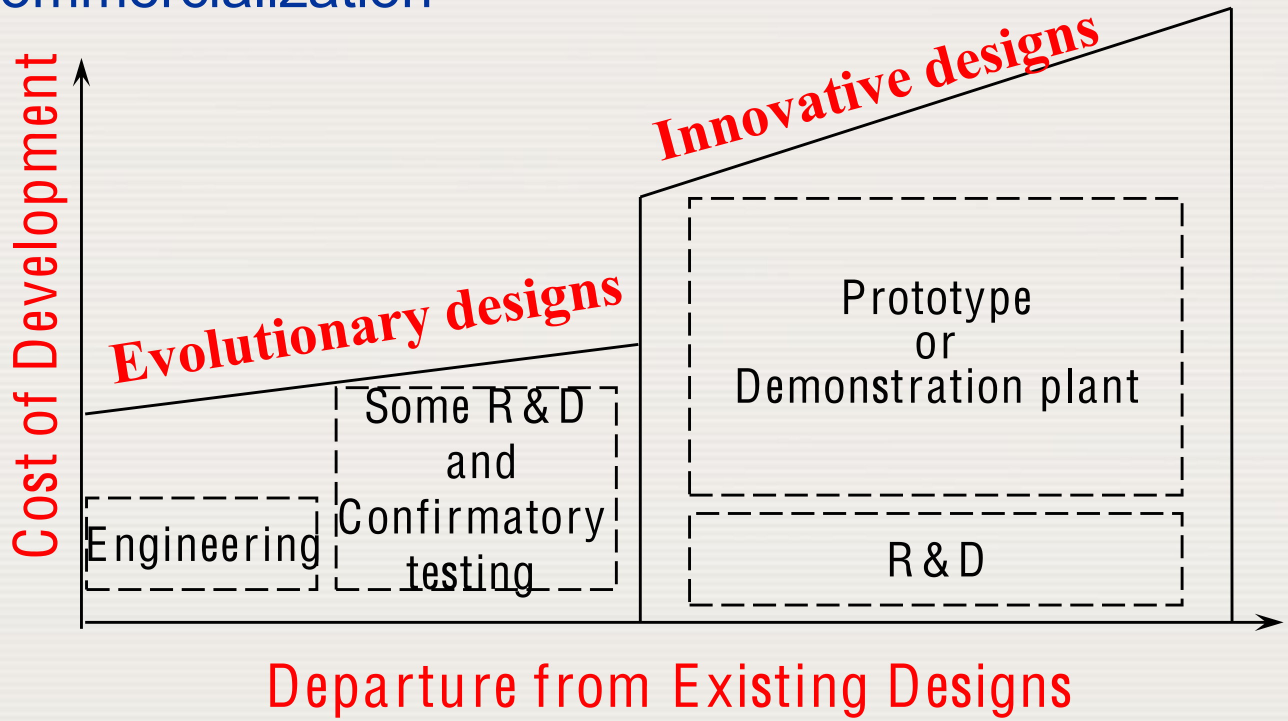
- **Advanced reactors face several challenges:**
  - Regain public acceptance after the Fukushima Daiichi nuclear accident
  - Perceived as safe by the global public
  - Be competitive in deregulated market
  - Capacity building in newcomer countries
  - Final radioactive-waste disposal/repository site
- **4 categories of reactors by the IAEA:**
  - Current Technology Plants (e.g. Tomari-3 PWR, Sizewell-B PWR)
  - Evolutionary (e.g. ABWR, APR-1400, VVER-1000,…)
  - Advanced (e.g. AP1000, EPR, ESBWR, APWR, ATMEA1,…)
  - Innovative (e.g. Fast reactors, SMRs, advanced gas cooled reactors, Gen-IV designs)

IAEA

Advanced Reactor Designs

(As defined in IAEA-TECDOC-936)

- **Evolutionary Designs** - achieve improvements over existing designs through small to moderate modifications
- **Innovative Designs** - incorporate radical conceptual changes and may require a prototype or demonstration plant before commercialization



IAEA

Global Trends in Advanced Reactor Design

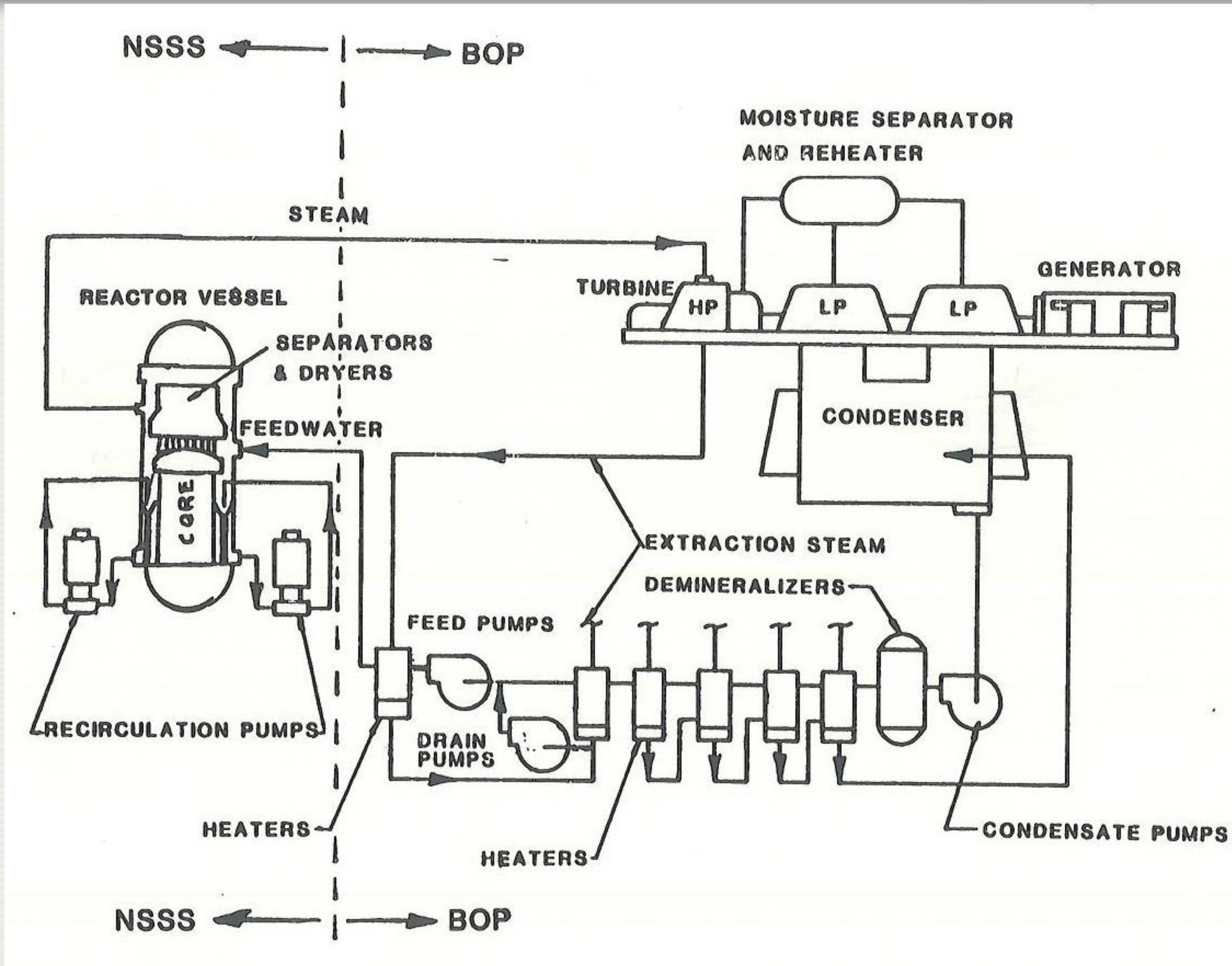
- **Cost Reduction**
  - Standardization and series construction
  - Improving construction methods to shorten schedule
  - Modularization and factory fabrication
  - Design features for longer lifetime
  - Fuel cycle optimization
  - Economy of scale → larger reactors
  - Affordability → SMRs
- **Performance Improvement**
  - Establishment of user design requirements
  - Development of highly reliable components and systems, including "smart" components
  - Improving the technology base for reducing over-design
  - Further development of PSA methods and databases
  - Development of passive safety systems
  - Improved corrosion resistant materials
  - Development of Digital Instrumentation and Control
  - Development of computer based techniques
  - Development of systems with higher thermal efficiency and expanded applications (Non-electrical applications)

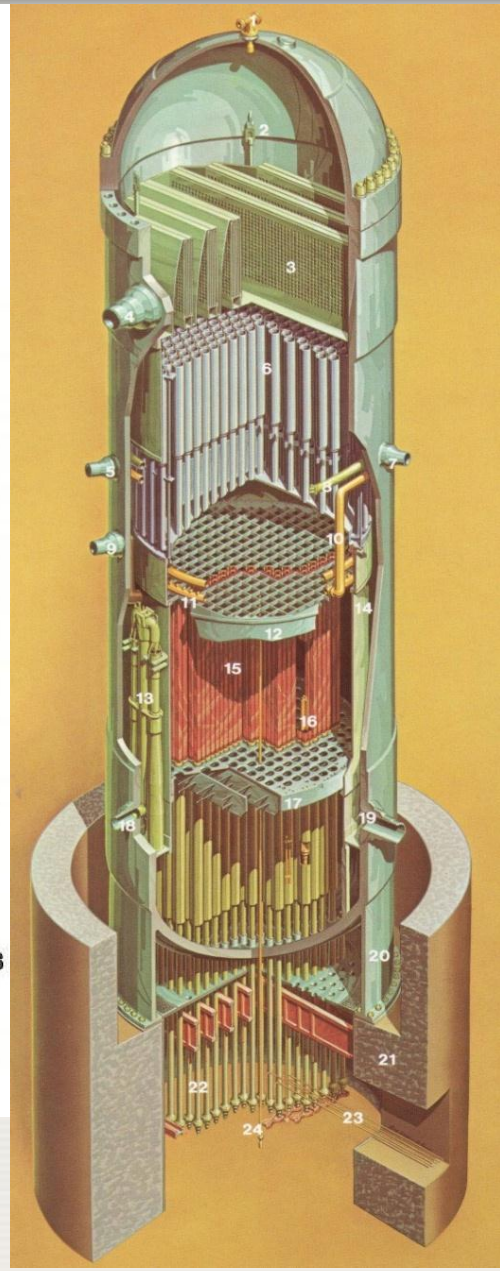
OPERATING FUNDAMENTALS

!!! Plant Layouts are similar even for different designs …

BWR Operating Fundamental (1)

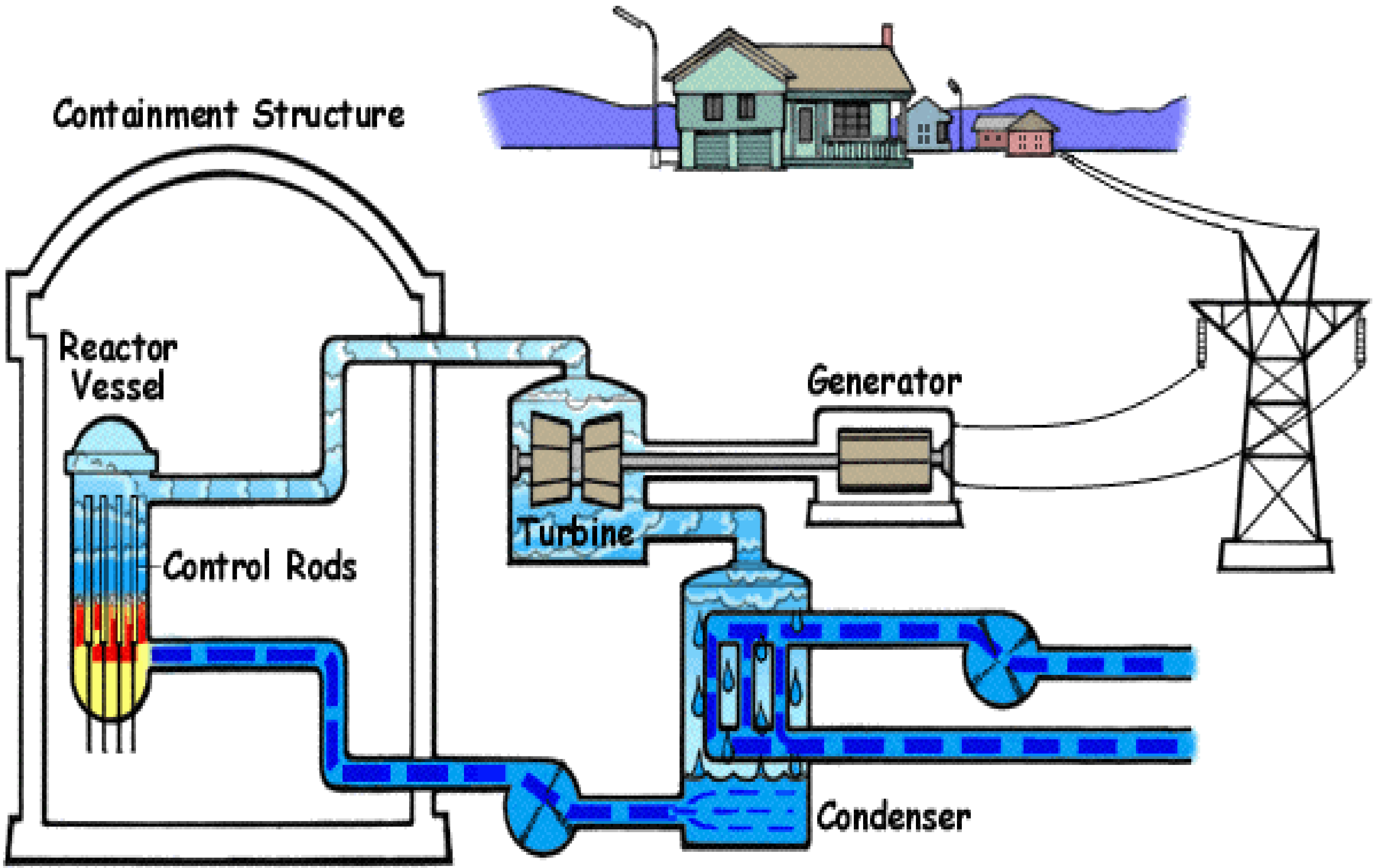
Courtesy of GE Nuclear Energy, USA.





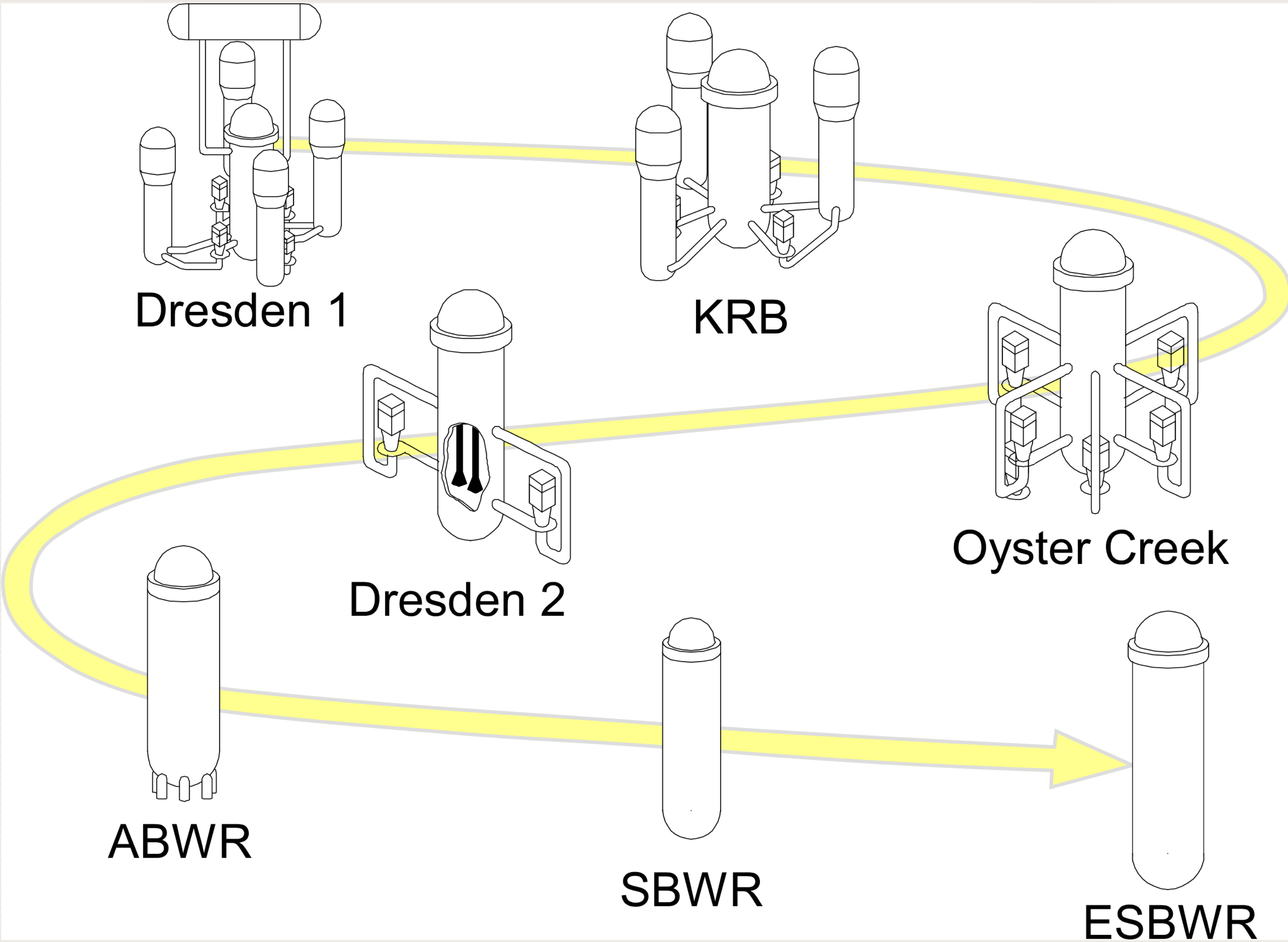
BWR Operating Fundamental (2)

Courtesy of NRC, USA.



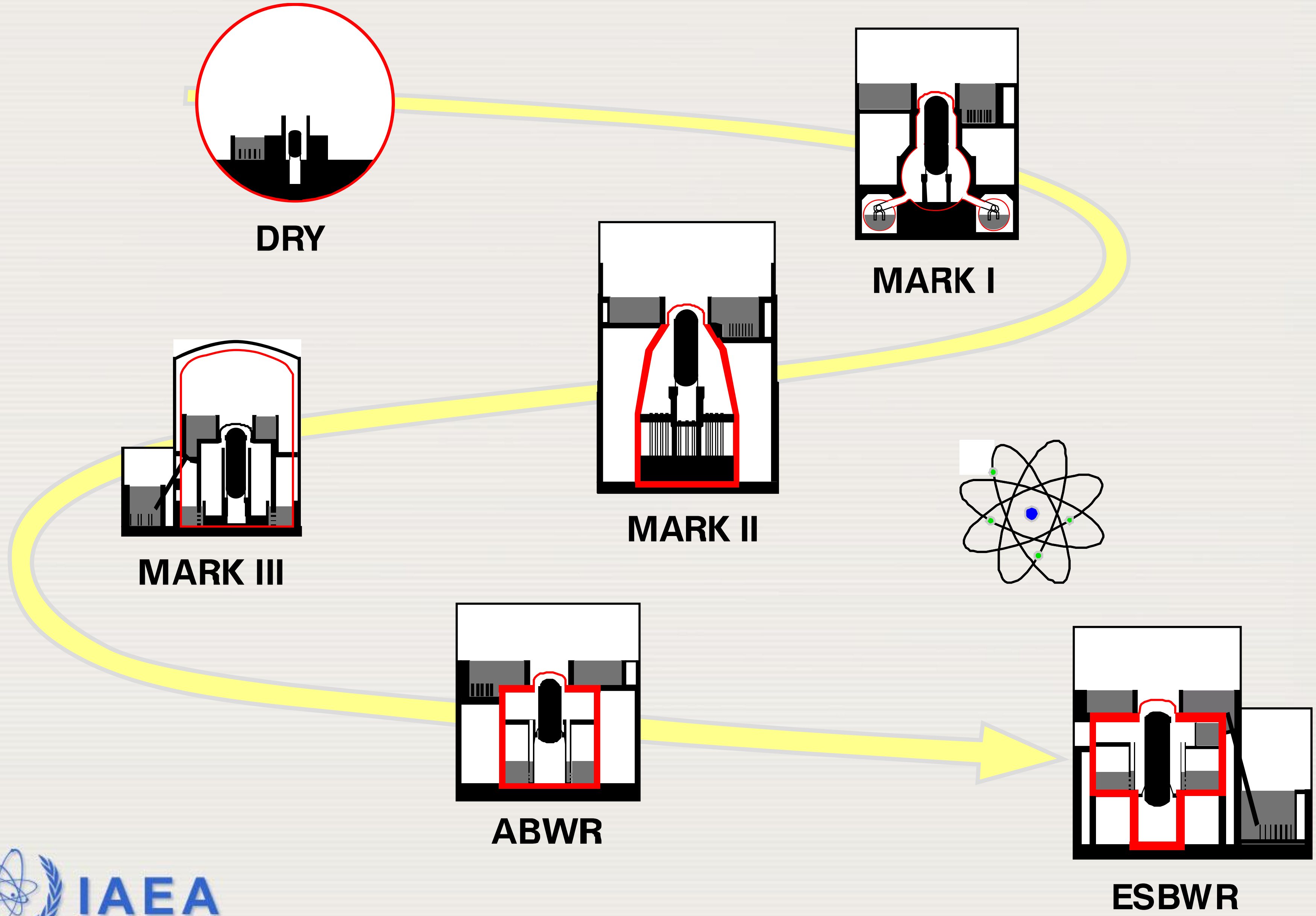
Evolution of BWR steam supply system

Courtesy of GE Nuclear Energy, USA.

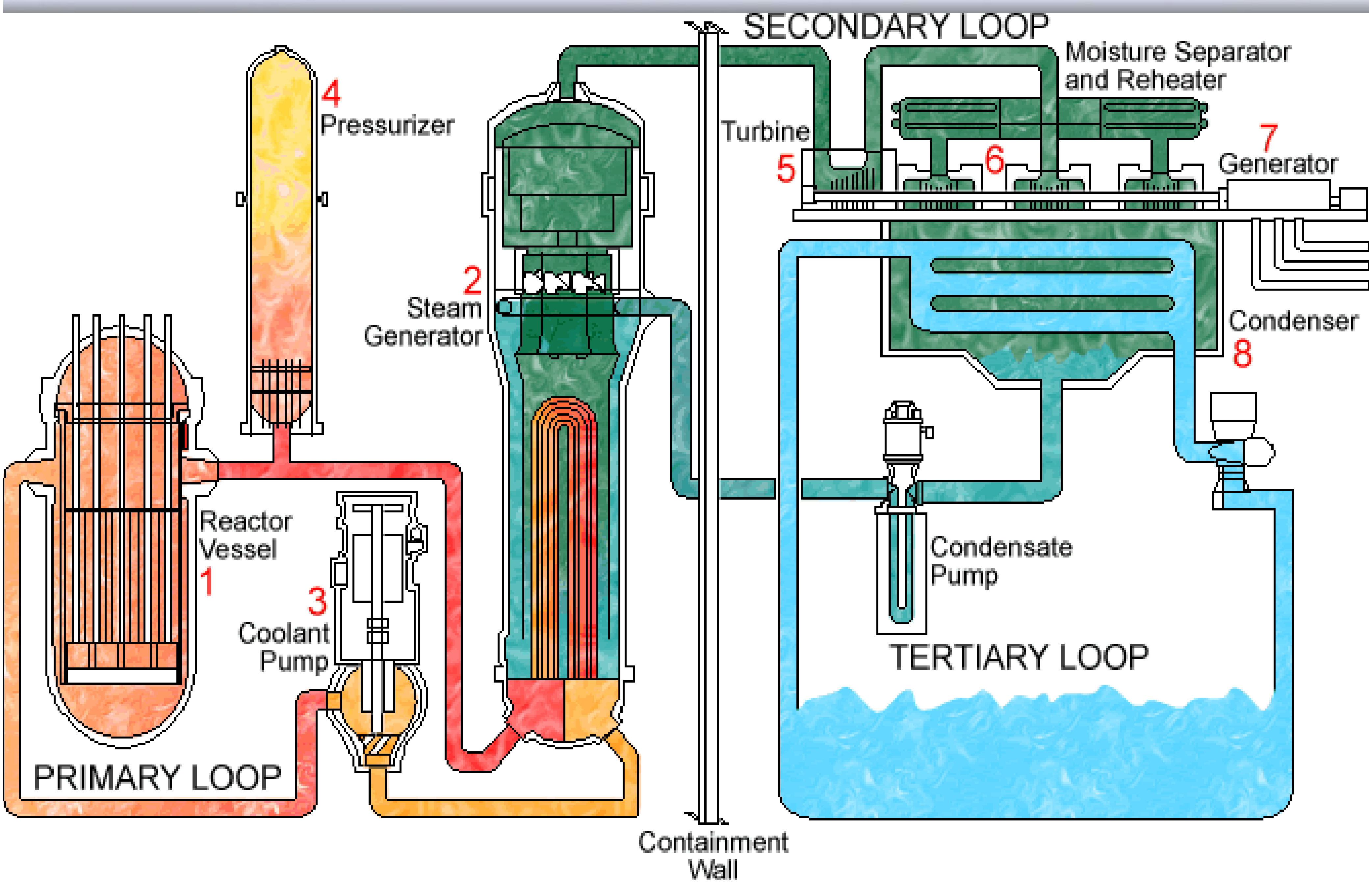


Evolution of BWR Containment

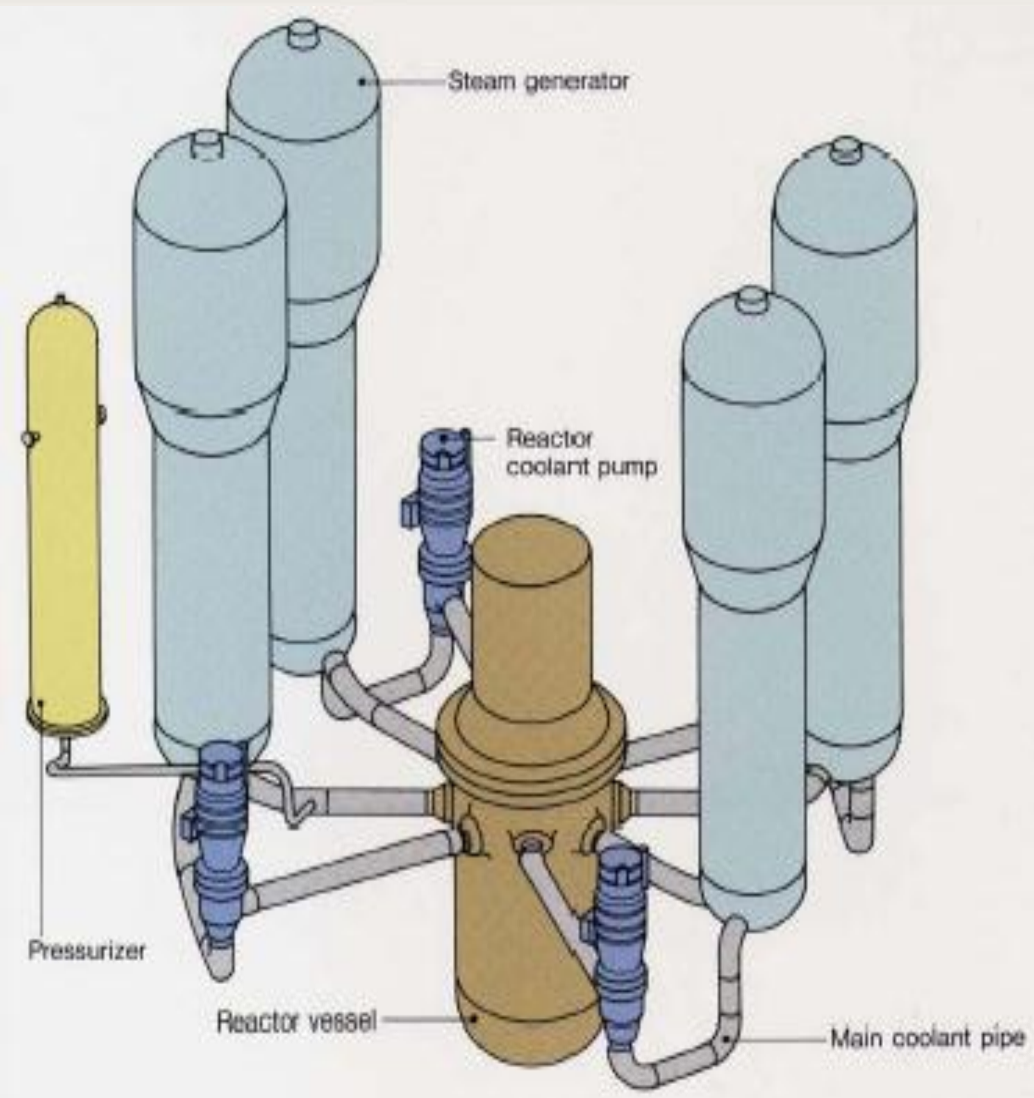
Courtesy of GE Nuclear Energy, USA.

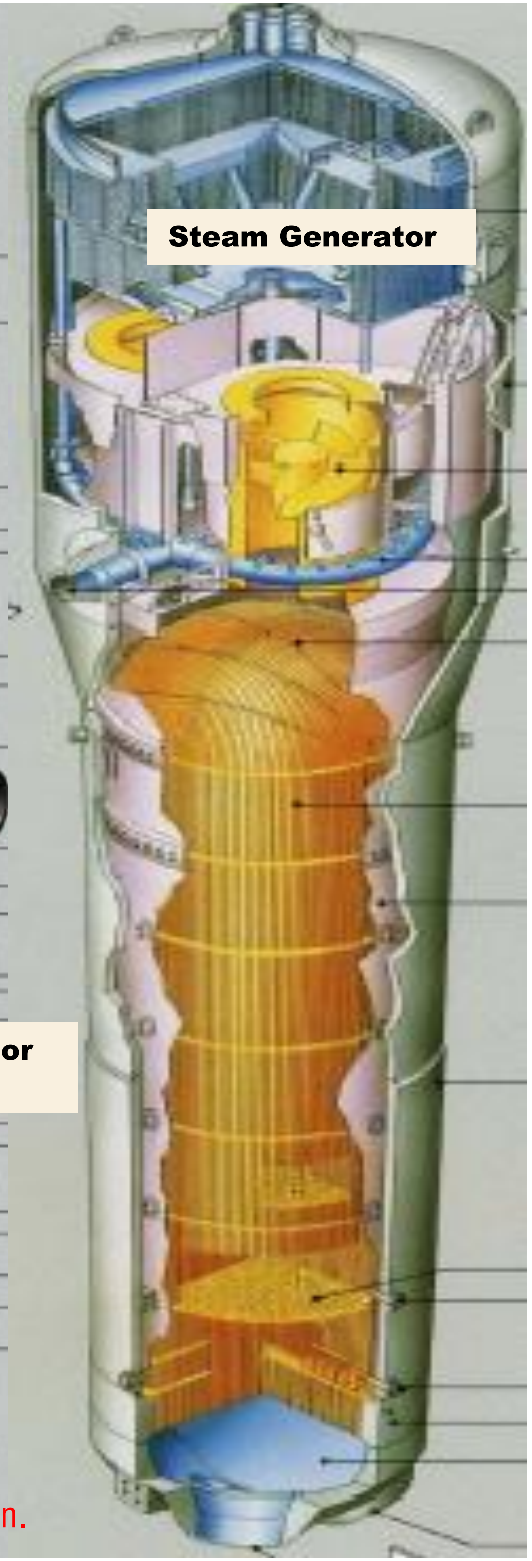


PWR Operating Fundamentals (1)



Reactor Coolant System of PWR (1)

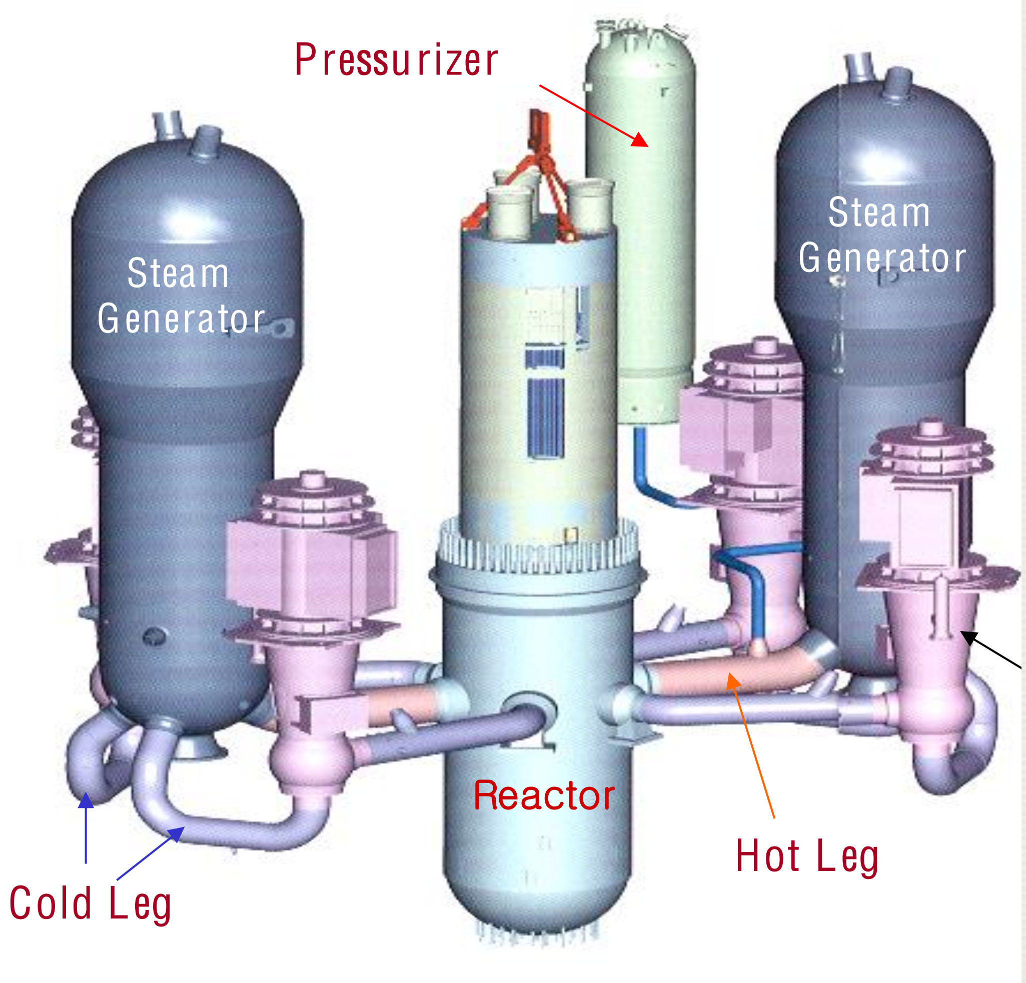


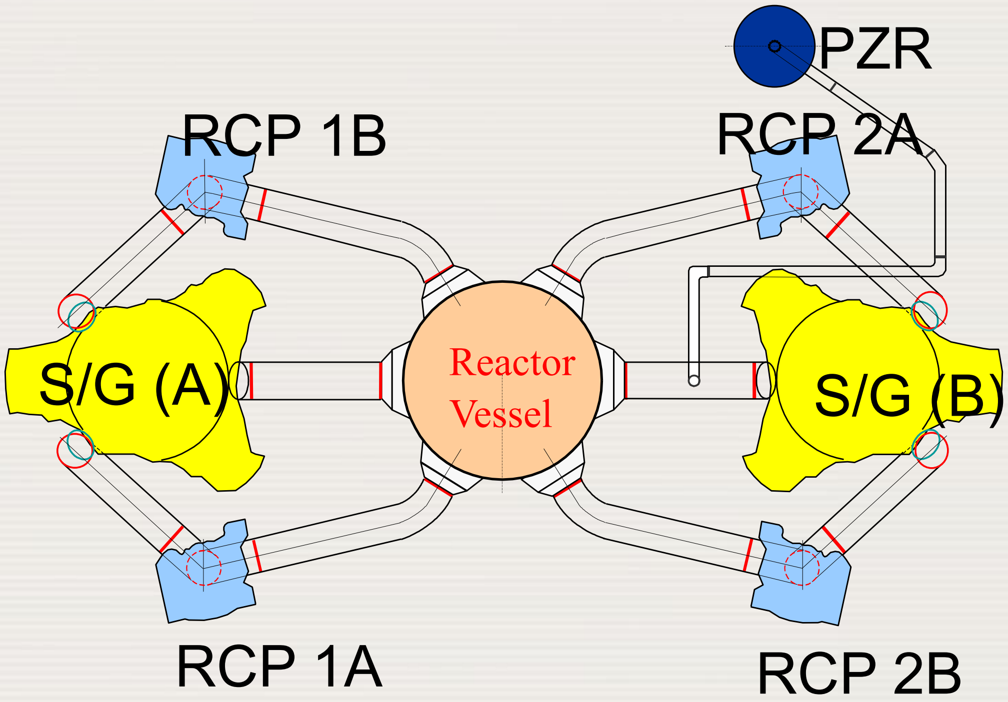


Courtesy of Mitsubishi Heavy Industries, Japan.

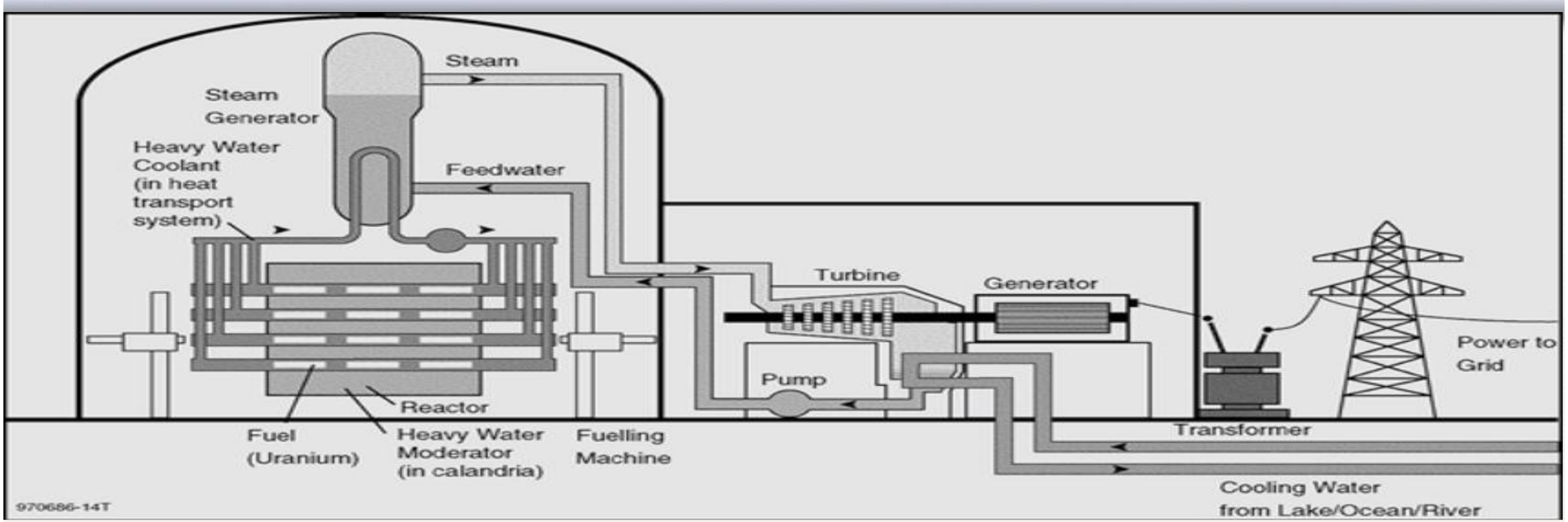
Reactor Coolant System of PWR (2)

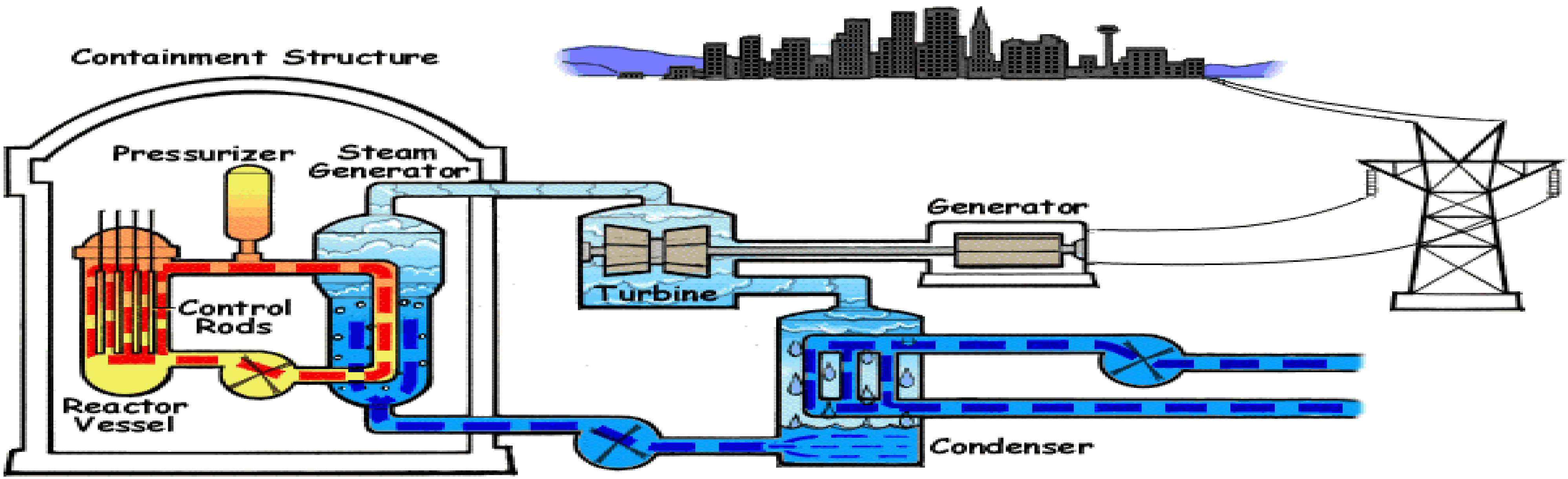
Courtesy of KHNP, Korea.





CANDU Operating Fundamentals

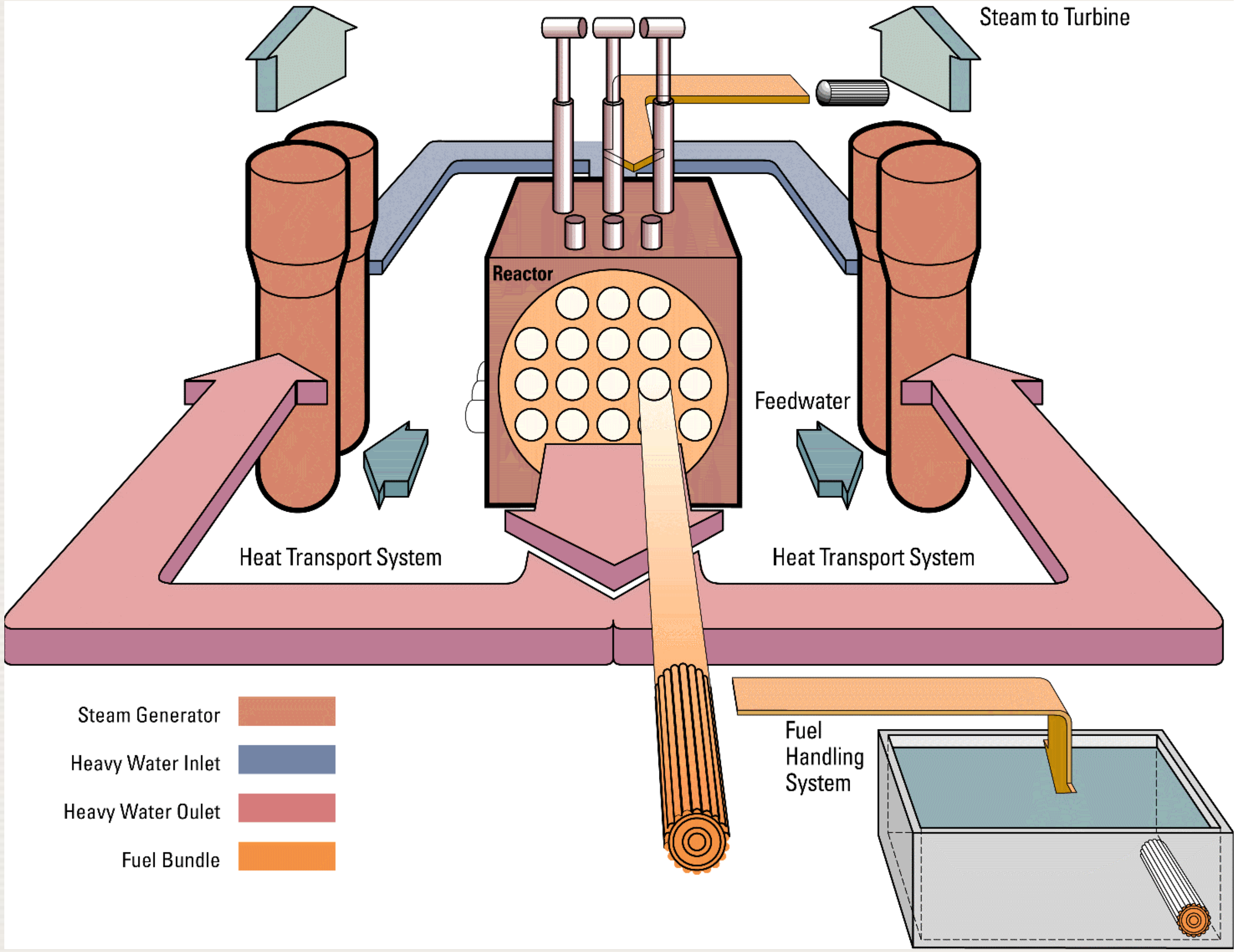




CANDU Primary Components

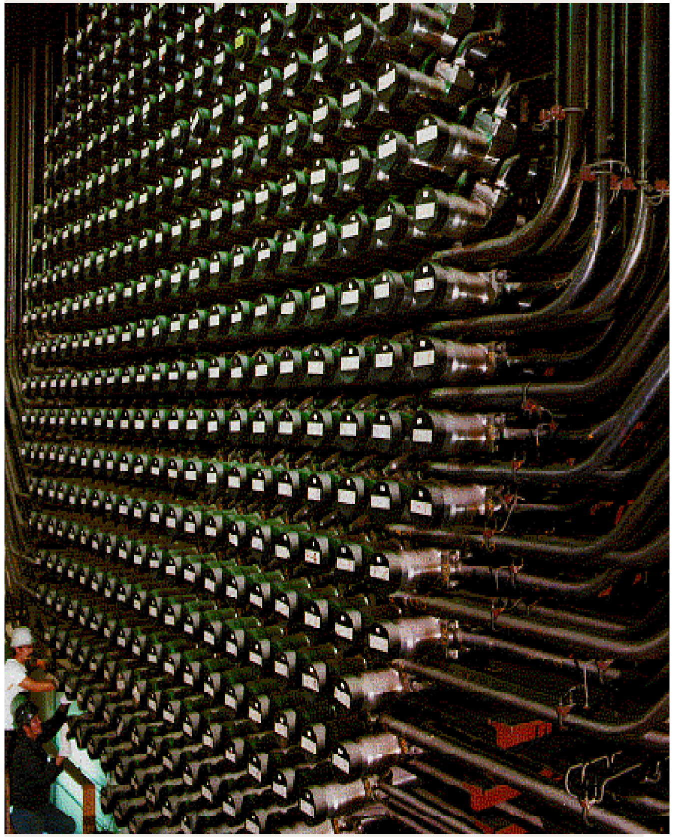
Courtesy of AECL, Canada.

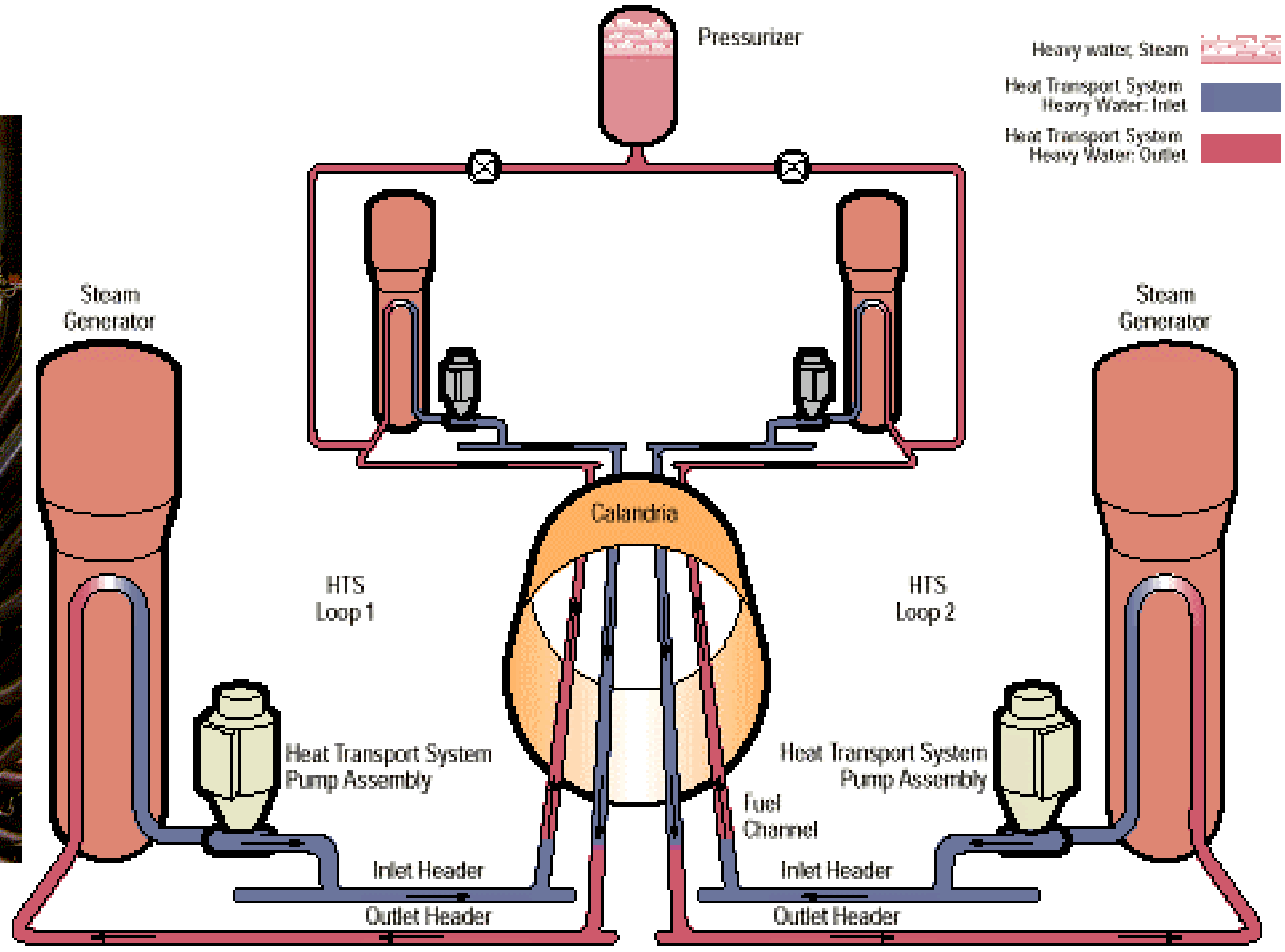
- Fuel bundles
- Fuel channels
- CALANDRIA
- Control system
- Heat Transport System
- Moderator System
- Fuel Handling System
- Auxiliary System
- Safety System
- Support System



CANDU Heat Transport System

Courtesy of AECL, Canada.





Currently Competing Advanced Reactors

1. VVER-1000, VVER-1200	Russia
2. ABWR	Japan-USA
3. APR-1400, OPR-1000	Korea
4. EPR	France
5. AP1000	USA-Japan
6. ESBWR	USA-Japan
7. APWR	Japan
8. ATMEA1	France-Japan
9. EC-6, ACR-1000	Canada

IAEA

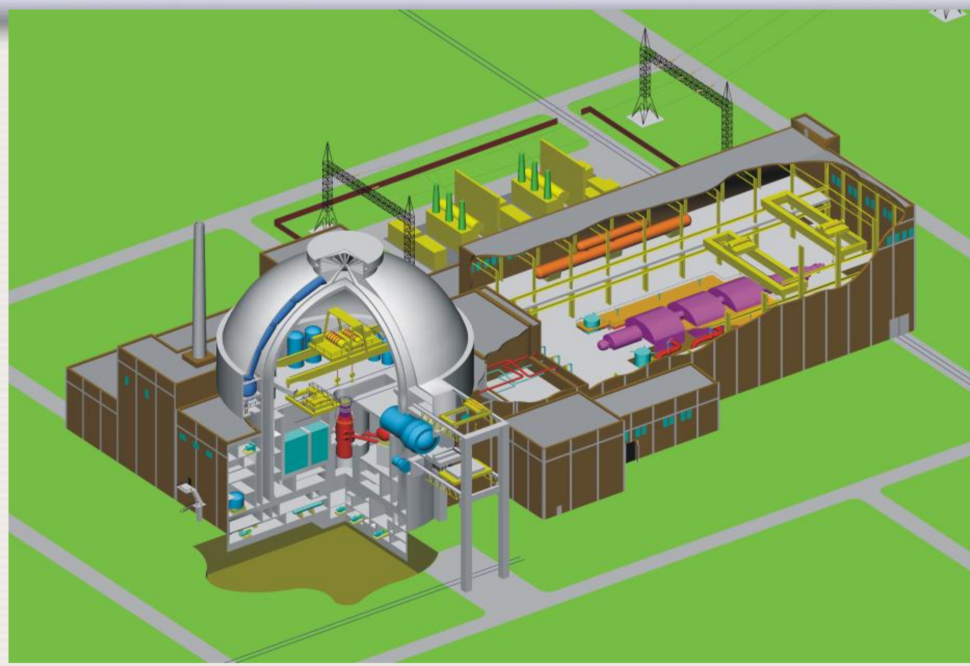
What does the above order imply?

Russian Federation

VVERs & AES 2006

OKB Gidropress

WWER-1000 / 1200 (AEP)

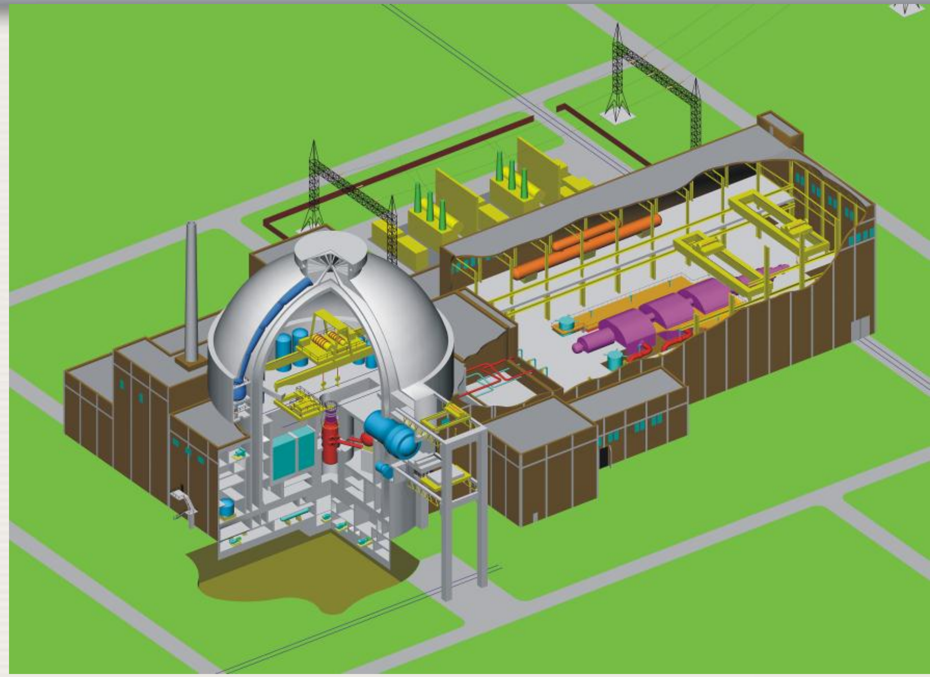


- The state-owned AtomEnergoProm (AEP), and its affiliates (including AtomStroyExport (ASE) et.al) is responsible for nuclear industry activities, including NPP construction
- Advanced designs based on experience of 23 operating WWER-440s & 27 operating WWER-1000 units
- Present WWER-1000 construction projects
  - Kudankulam, India (2 units)
  - Belene, Bulgaria (2 units)
  - Bushehr, Iran (1 unit) - completed
- WWER-1200 design for future bids of large size reactors

- Tianwan
  - first NPP with corium catcher
  - Commercial operation: Unit-1: 5.2007; Unit-2: 8.2007
- Kudankulam-1 & 2
  - Commercial operation expected in 2010
  - Core catcher and passive SG secondary side heat removal to atmosphere

IAEA

WWER-1200



Commissioning of 17 new WWER-1200s in Russia expected by 2020

- Novovoronezh – 2 units
- Leningrad – 4 units
- Volgodon – 2 units
- Kursk – 4 units
- Smolensk – 4 units
- Kola – 1 unit

- Uses combination of active and passive safety systems
- One design option includes core catcher; passive containment heat removal & passive SG secondary side heat removal
- 24 month core refuelling cycle
- 60 yr lifetime
- 92% load factor

IAEA

VVER, NPP 2006

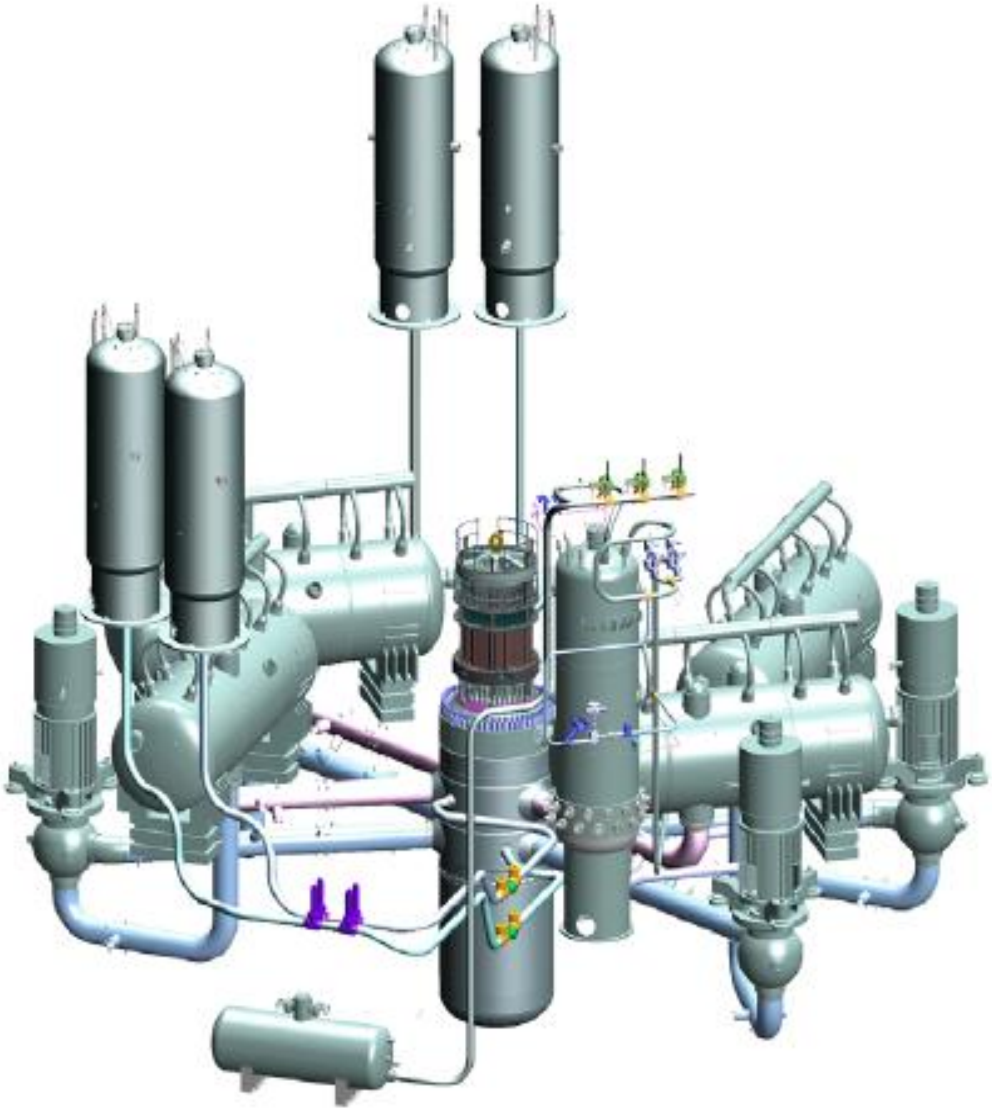
Courtesy of OKB Gidropress, Russia.

What is VVER

ВВЭР: водо-водяной энергетический р

WWER: water-cooled water-moderated e

- Pressurized light water reactor.
- Loop-type reactor plant.
- Horizontal steam generators.
- Hexagonal fuel assemblies.
- High level of inherent safety.
- About 1500 reactor-years of operating time



IAEA

NPP 2006 Basis

Now **VVER-1200** (also known as **NPP-2006**) is the main design for near term nuclear power program in Russia and for bidding at international market.

NPP-2006 is the evolution of VVER-1000s by improving plant performance and increasing plant safety.

Plant performance	Plant safety
Rated power – 1170 (1000) MWe	Passive reactor scram system
Primary pressure – 16.2 (15.7) MPa	Passive core flooding system
Secondary pressure – 7.0 (6.3) MPa	Passive decay heat removal system
Coolant temperature – 329 (320) C	Passive containment cooling system
Refueling period – 24 (12) months	Passive hydrogen removal system
Burn-up in FA – 70 (50) MWD/kgU	Passive corium catcher

NPP 2006 Major Parameters

Parameter	Value
NSSS equipment lifetime, years	60
Load factor, %	92
NSSS availability factor	99
Efficiency factor, net %	35.7
Length of fuel cycle, years	4 - 5
Frequency of refueling, months	12 (18-24)
FA maximum burn-up, MW day/kgU	70
Inter-repair period length, years	4 - 8
Refueling length, days	$\leq$ 16
Number of unplanned reactor shutdowns per year	$\leq$ 1

Japan

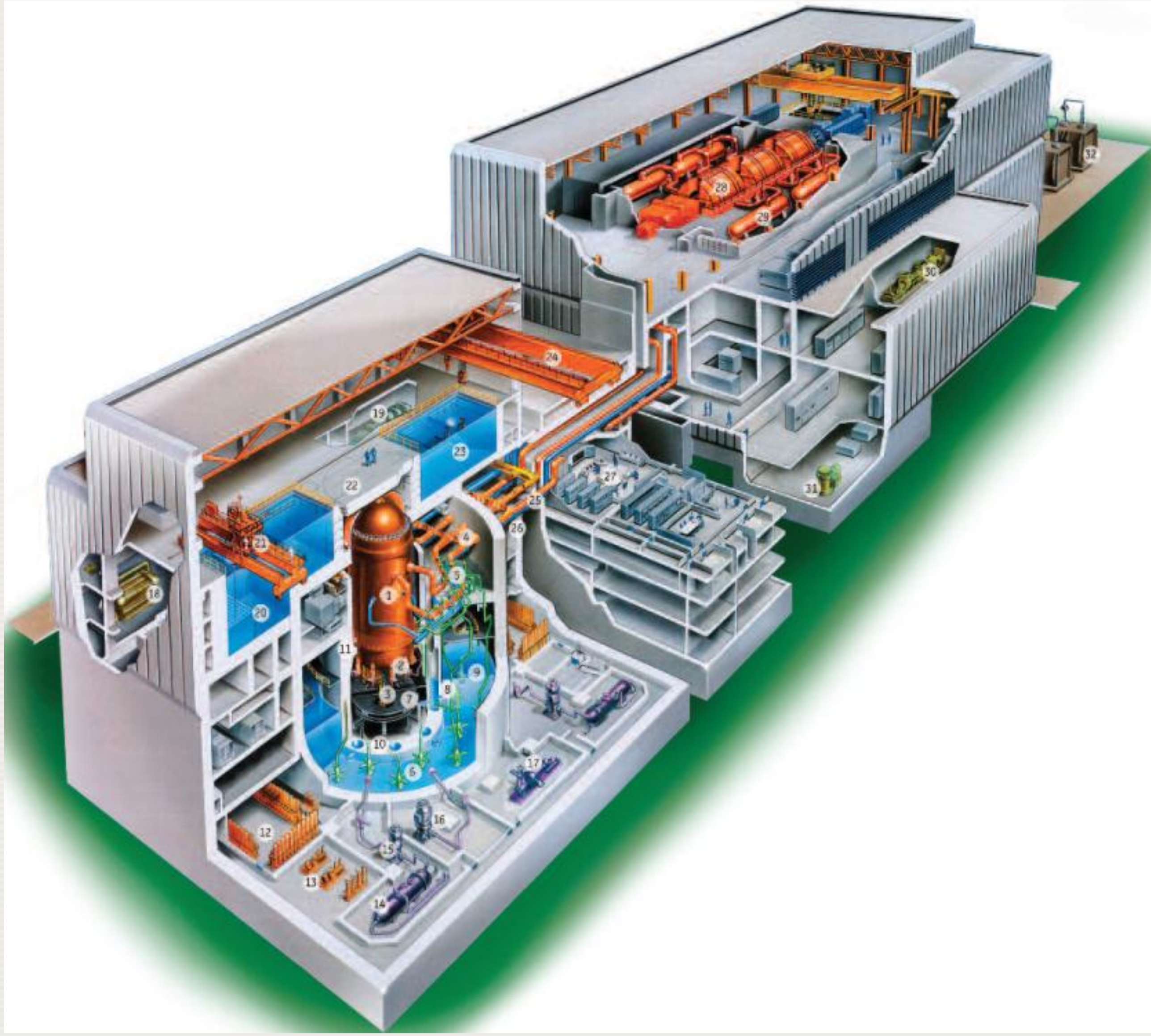
US-ABWR and **EU-ABWR**

Toshiba Corporation &
Hitachi GE Nuclear Energy

Advanced Boiling Water Reactor (ABWR)

Courtesy of GE Hitachi Nuclear Energy

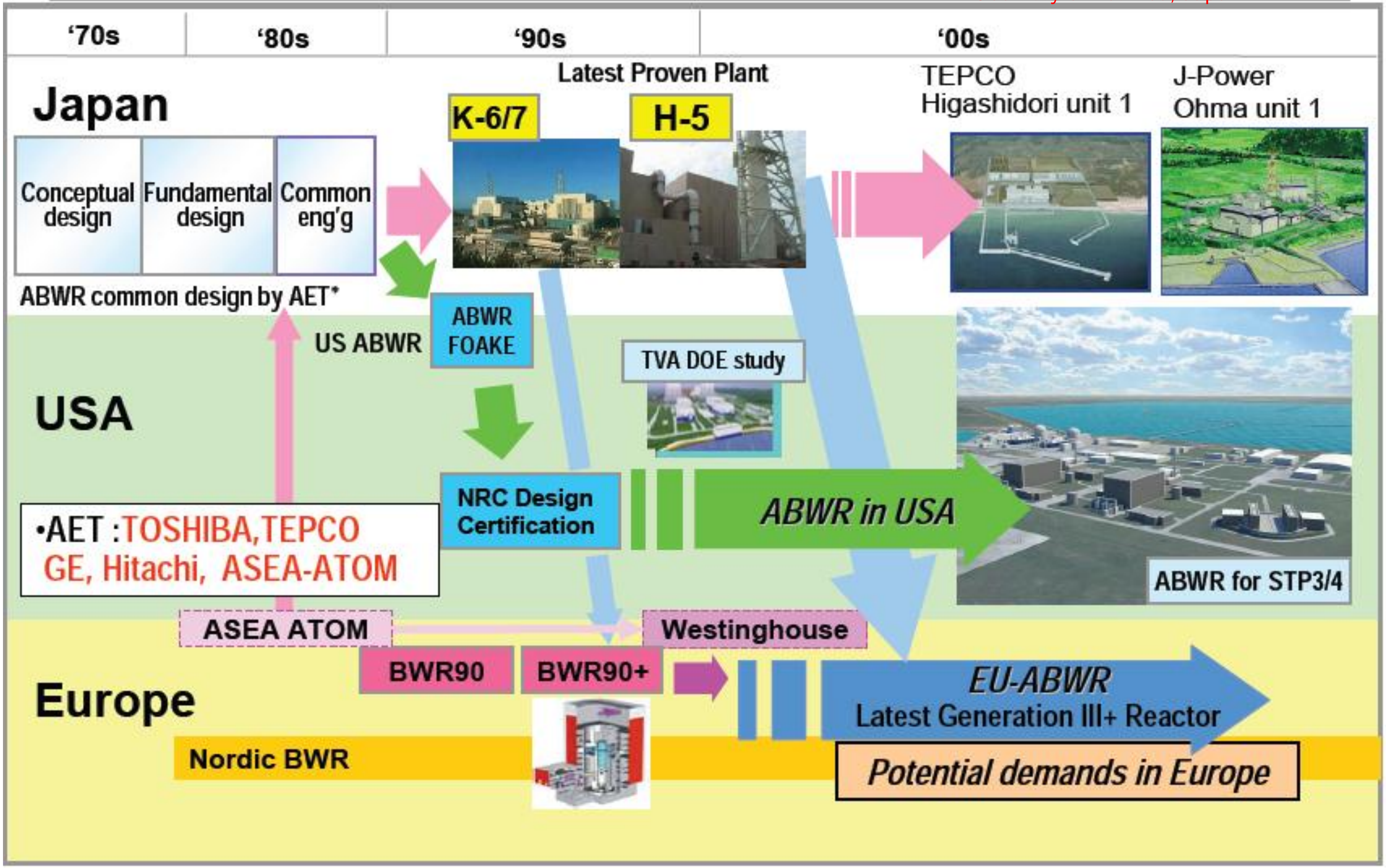
- Originally by GE, then Hitachi & Toshiba
- Developed in response to URD
- First Gen III reactor to operate commercially
- Licensed in USA, Japan & Taiwan, China
- 1380 MWe - 1500 MWe
- Shorter construction time
- Standardized series
  - 4 in operation (Kashiwazaki-Kariwa -6 & 7, Hamaoka-5 and Shika-2)
  - 7 planned in Japan
  - 2 under construction in Taiwan, China
  - Proposed for South Texas Project (USA)



IAEA

ABWR Development History

Courtesy of Toshiba, Japan.



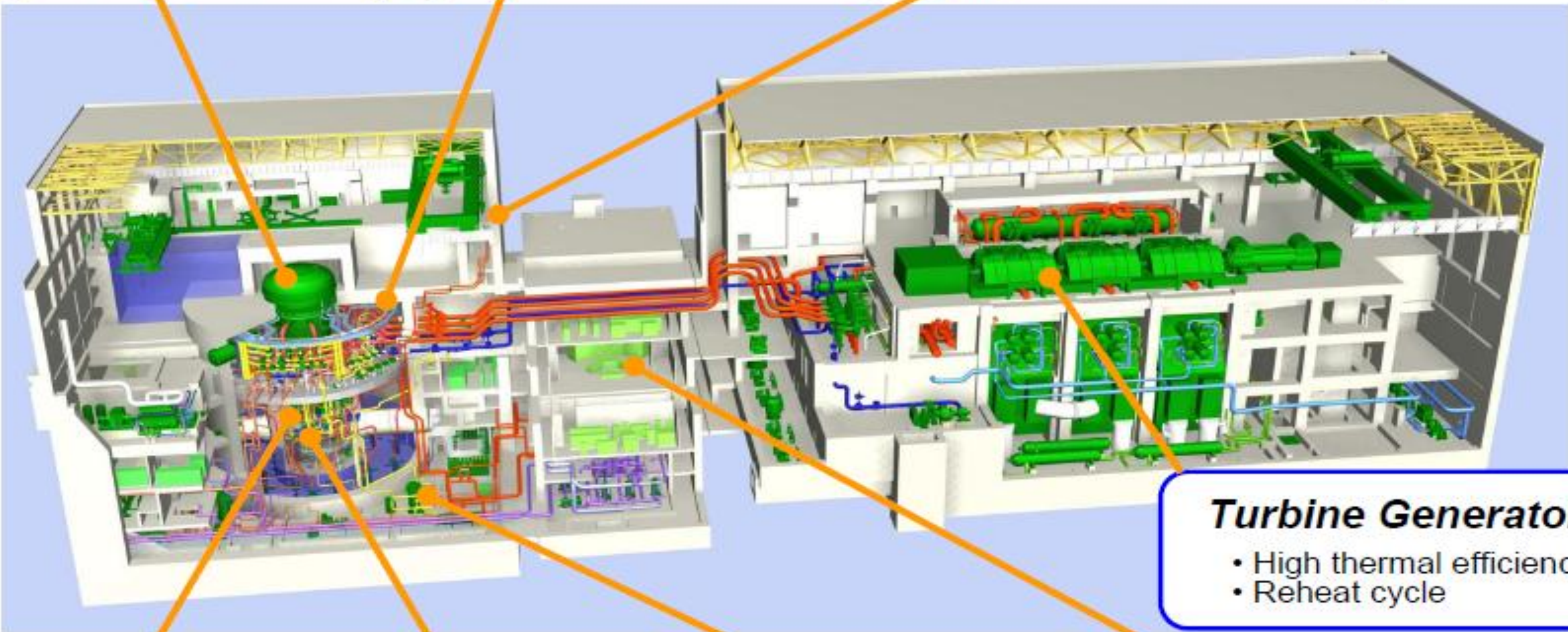
US-ABWR Overview

Courtesy of Toshiba, Japan.

Reactor Pressure Vessel, Core
- Improved core
- Improved internals

Reinforced Concrete Containment Vessel
- Short construction period
- Low construction cost

Reactor Building
- Compact building
- Strong structure to withstand high vibration load



Turbine Generator
- High thermal efficiency
- Reheat cycle

Reactor Internal Pump
- High safety
- Simplicity

Fine Motion Control Rod Drive
- High reliability
- High operability

Emergency Core Cooling System
- Reduced capacity
- High safety

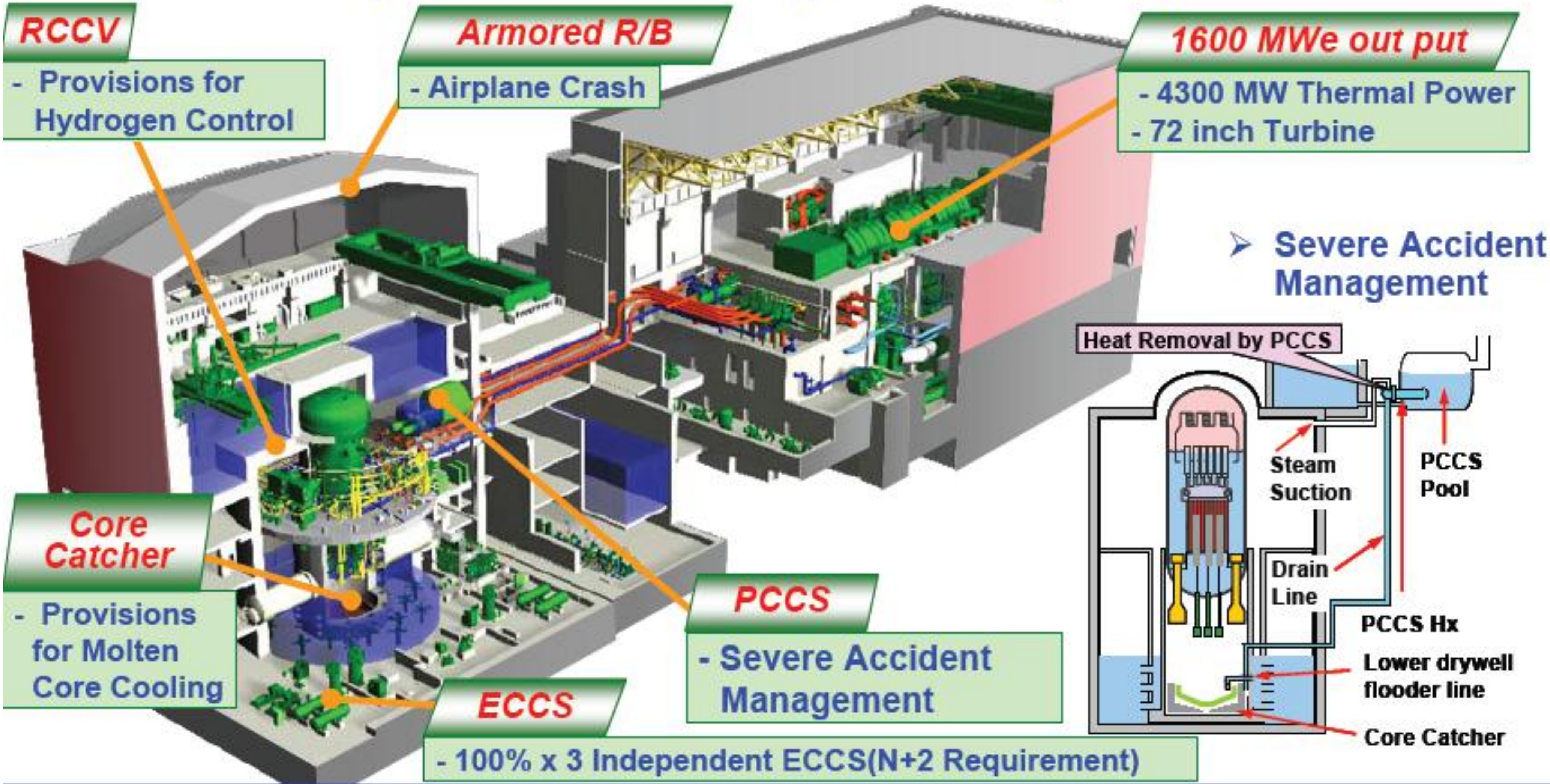
Main Control Room
- Automatic operation to reduce startup time
- Fully digital system

IAEA

EU-ABWR Overview

Courtesy of Toshiba, Japan.

- European BWR Technologies are applied in EU-ABWR.
- ABWR is adapted to meet European safety requirements.

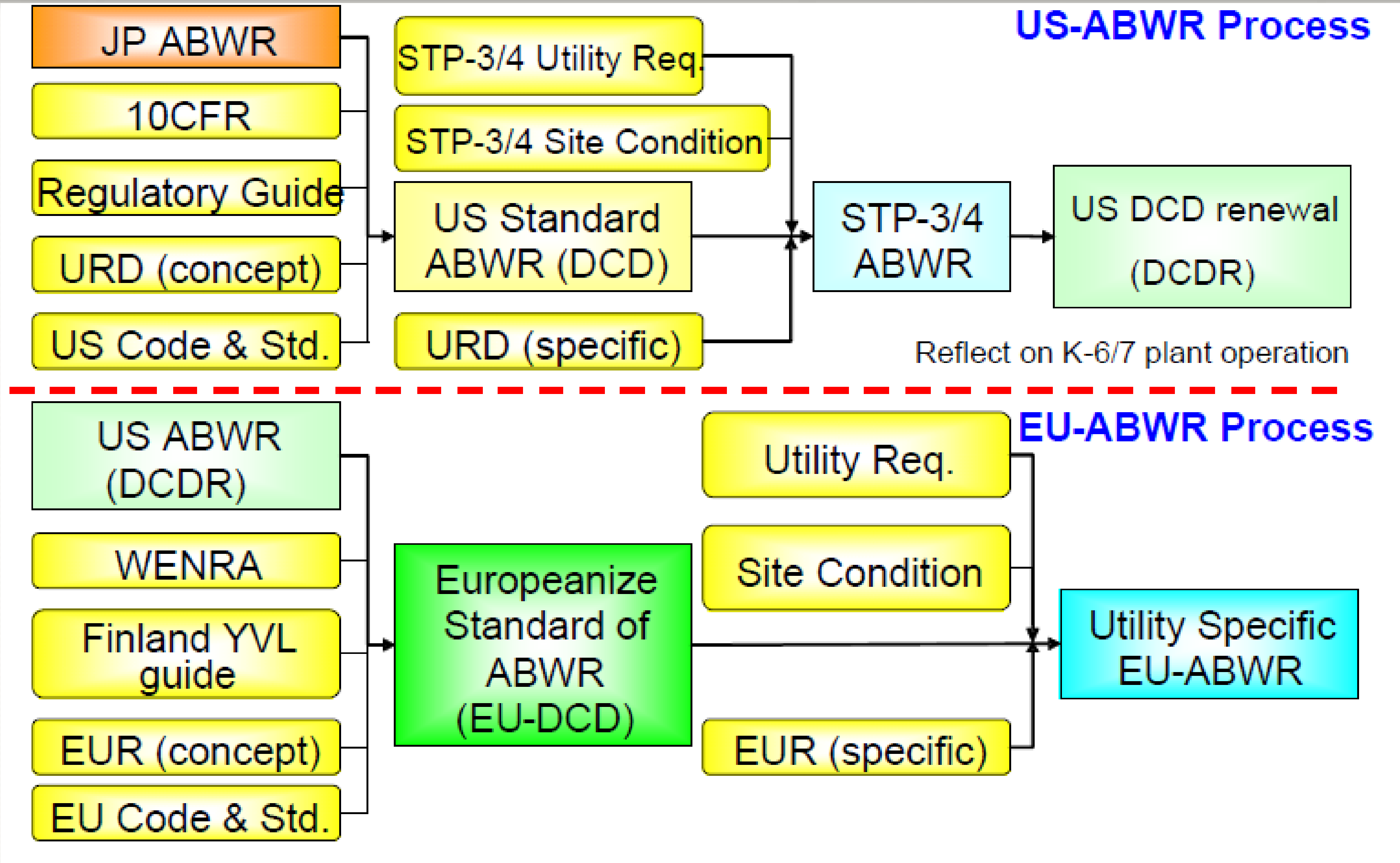


Key Features of US, EU ABWR

- **Principle**
  - ABWR design based on the integration of the proven technologies of BWR fleet
  - Plentiful construction and operation experience of Japanese BWRs and ABWRs
  - Latest Westinghouse BWR design mixture
- **Approach**
  - Extension of the JP- ABWR design for wide application accounting for US,European safety requirements

	US-ABWR	EU-ABWR
Thermal Power	3,926 MWt	4,300 MWt
Electrical Output	1,400 MWe	1,600 MWe
Design Life time	60 years	60 years
Plant Availability	>90 %	>90 %

Approach to US, EU-nized ABWR



Republic of Korea

APR1400 and OPR1000

Korea Hydro and Nuclear Power (KHNP)

IAEA

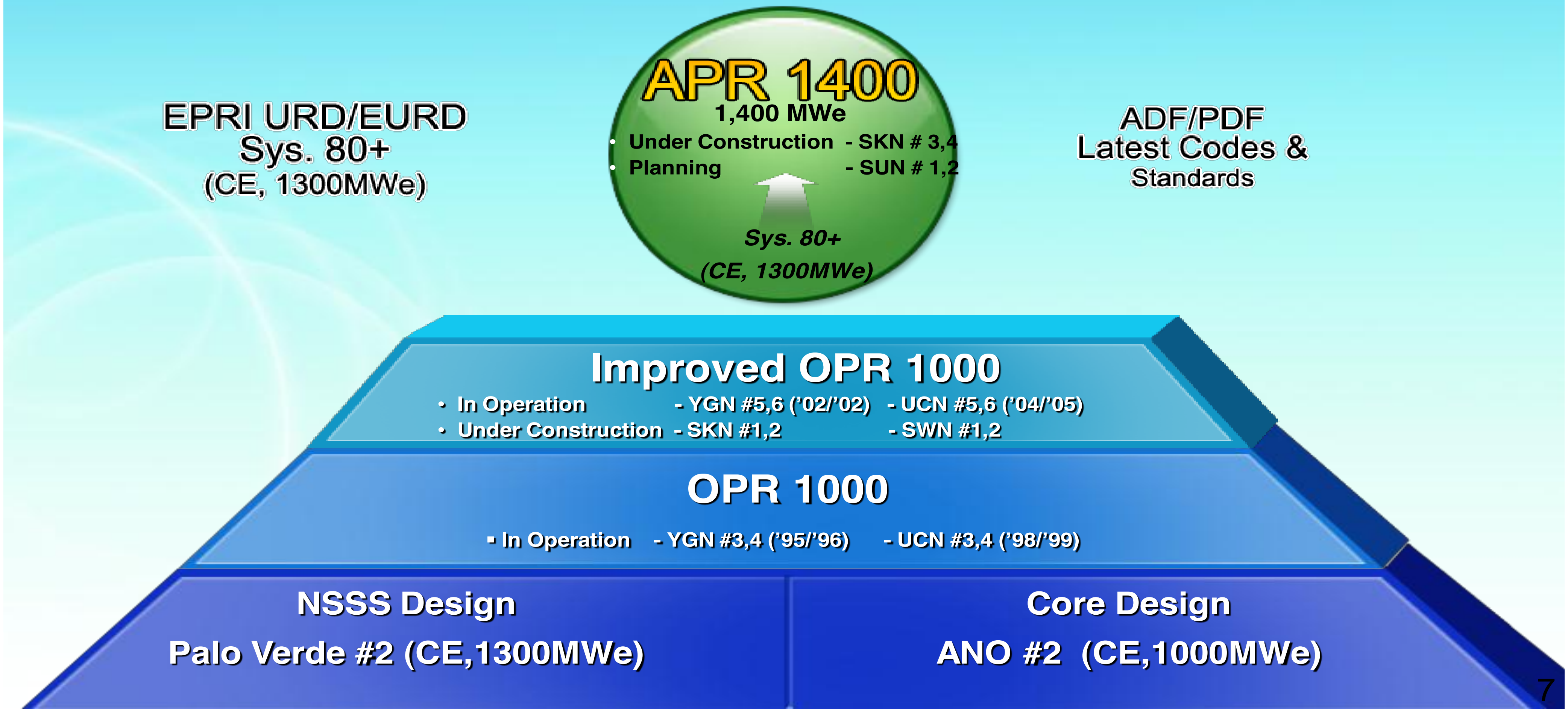
APR-1400

- Developed in Rep. of Korea (KHNP and Korean Industry)
- 1992 - development started
- Based on CE's System 80+ design (NRC certified)
- 1400 MWe - for economies of scale
- Incorporates experience from the 1000 MWe Korean Standard Plants
- Relies primarily on well proven active safety systems
- First units will be Shin-Kori 3,4
  - completion 2013-14
- Design Certified by Korean Regulatory Agency in 2002
- 4 units to be built in UAE

IAEA

Phase IV – Technology advancedment

- Development of Advanced Power Reactor 1400 (1992~2001)
- Licensing agreement with ABB-CE
  - Perfect technology self-reliance & technology ownership



Quick Comparison of OP1000 & APR1400

OPR1000

APR1400

Parameters	OPR1000	APR1400
Power capacity (MWe)	1000	1400
Design life time (yr)	40	60
Seismic design criteria	0.2g	0.3g
Core damage frequency	6.8×10^{-6} /RY	2.4×10^{-6} /RY
Emergency core cooling	2 Train	4 Train
Main control type	Analog + Digital	Digital

- **OPR1000: Optimized Power Reactor 1000MW**
- **APR1400: Advanced Power Reactor 1400MW**

IAEA

KHNP

Design Principles

Adoption of ADF based on proven technology

- Direct Vessel Injection of Safety Injection System
- Passive Flow Regulator or Fluidic Device in Safety Injection Tank
- In-containment Refueling Water Storage Tank & Sparger
- Fully Digitalized I&C and Operator-Friendly Man-Machine Interface

Enhanced Plant Safety and Cost Effectiveness

- Improved Severe Accident Mitigation System
- Reinforced Seismic Design Basis (0.3 g)
- Extended plant design lifetime (60 years)
- R educed construction time (48 months for Nth unit)

Improved O & M Convenience

- Extended operator response time
- Reduced occupational exposure
- Easier In-Service Inspection and maintenance for components

APR 1400

Design Goals

Safety

- Core Damage Frequency < 1.0E-5/RY (2.25×10-6/RY)
- Containment Failure Frequency < 1.0E-6/RY (2.84×10-7/RY)
- Seismic Design Basis : 0.3 g
- Occupational radiation exposure < 1 man·Sv/RY

Performance

- Thermal Margin (is greate than) > 10 %
- Plant Availability > 90 %
- Unplanned Trip (less than) < 0.8/RY

Economy

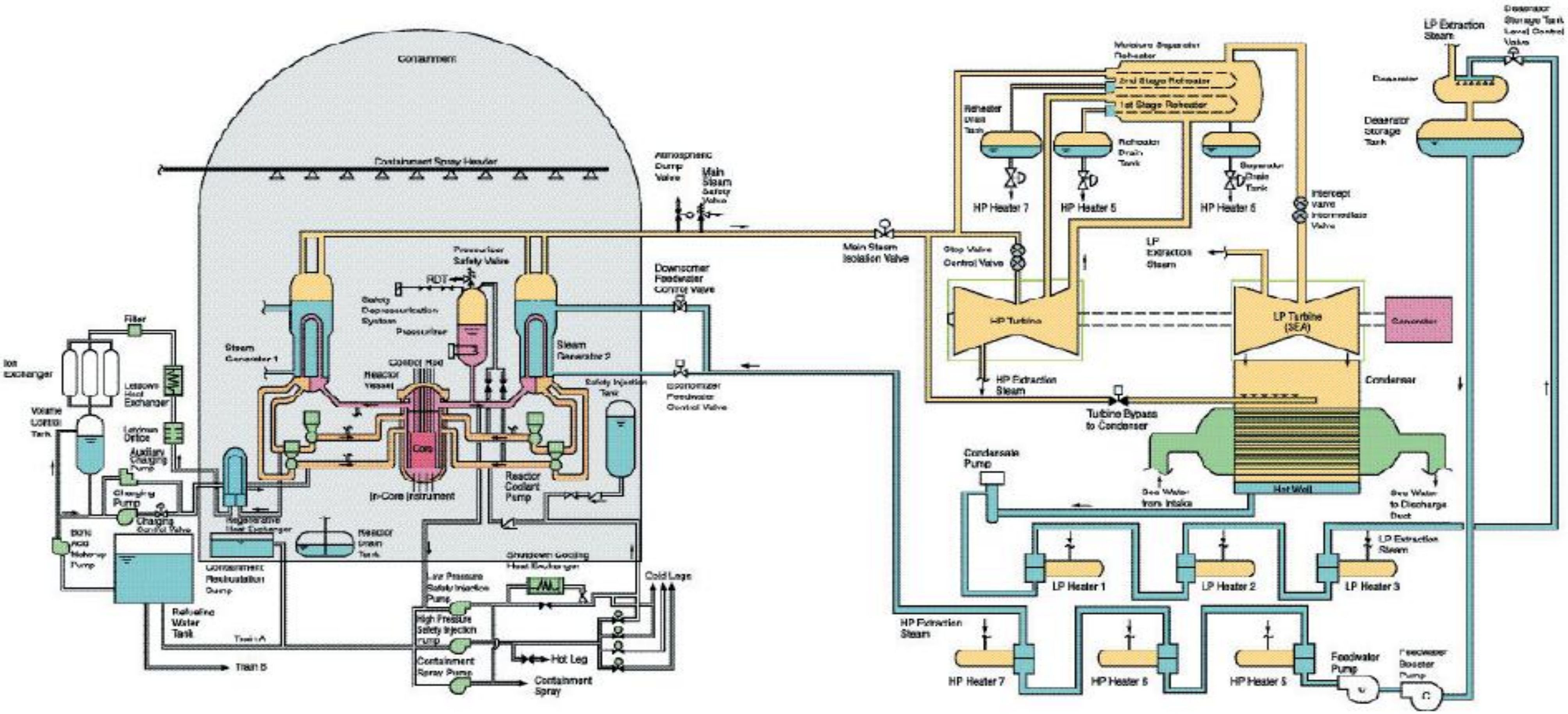
- Plant Capacity (Gross) : 1,455 Mwe
- Plant Lifetime : 60 years
- Refueling Cycle ≥ 18 months
- Construction Period : 48 months (Nth Unit)

KHNP

Top-Tier of APR1400 vs. OPR1000

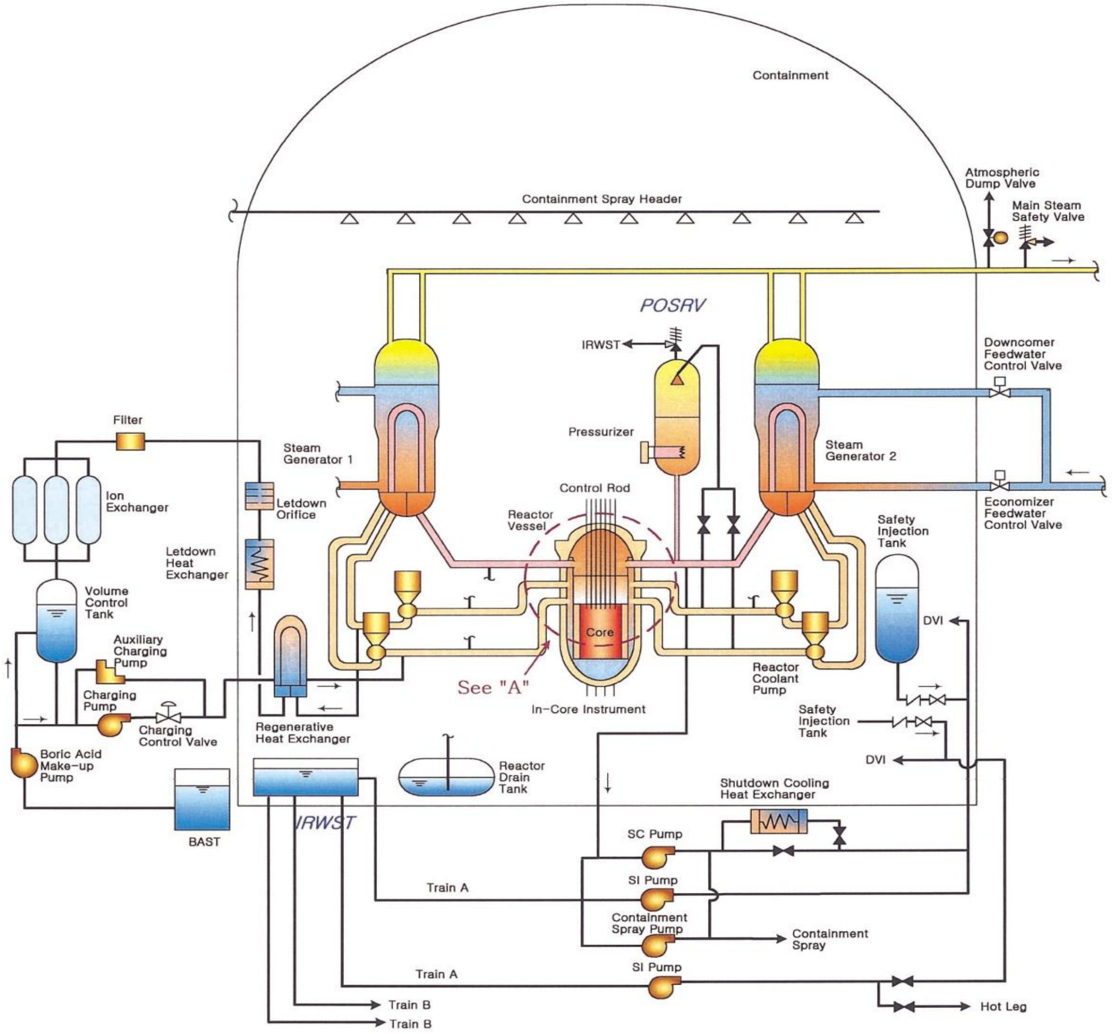
Items	APR1400	OPR1000
[Economics/Performance]		
- Design life time	60 yrs	40 yrs
- Capacity	1400 MWe	1000MWe
- Construction period	58 Months(Unit 1,2)	62 Months
- Daily load following	Automatic	Manual
- Refueling interval	18 Months	18 Months
- I&C	Full digital	Partial digital
[Safety]		
- CDF	<1.0E-5/RY	<1.0E-4/RY
- CFF	<1.0E-6/RY	<1.0E-5/RY
- Seimic design	0.3g	0.2g
- Operator action time	30 mins	10 mins
- SBO scoping time	8 hours	4 hours
- ECCS	4 Trains DVI	2 Train CLI
- SG plugging margin	10% (Inconel 690)	8% (Inconel 690)

Overall Configuration of OPR1000



Overall Configuration of APR1400

- **Basically same configuration with OPR1000 except safety system (DVI, IRWST and so on)**



12

Construction Schedule

France

EPR and US-EPR

AREVA

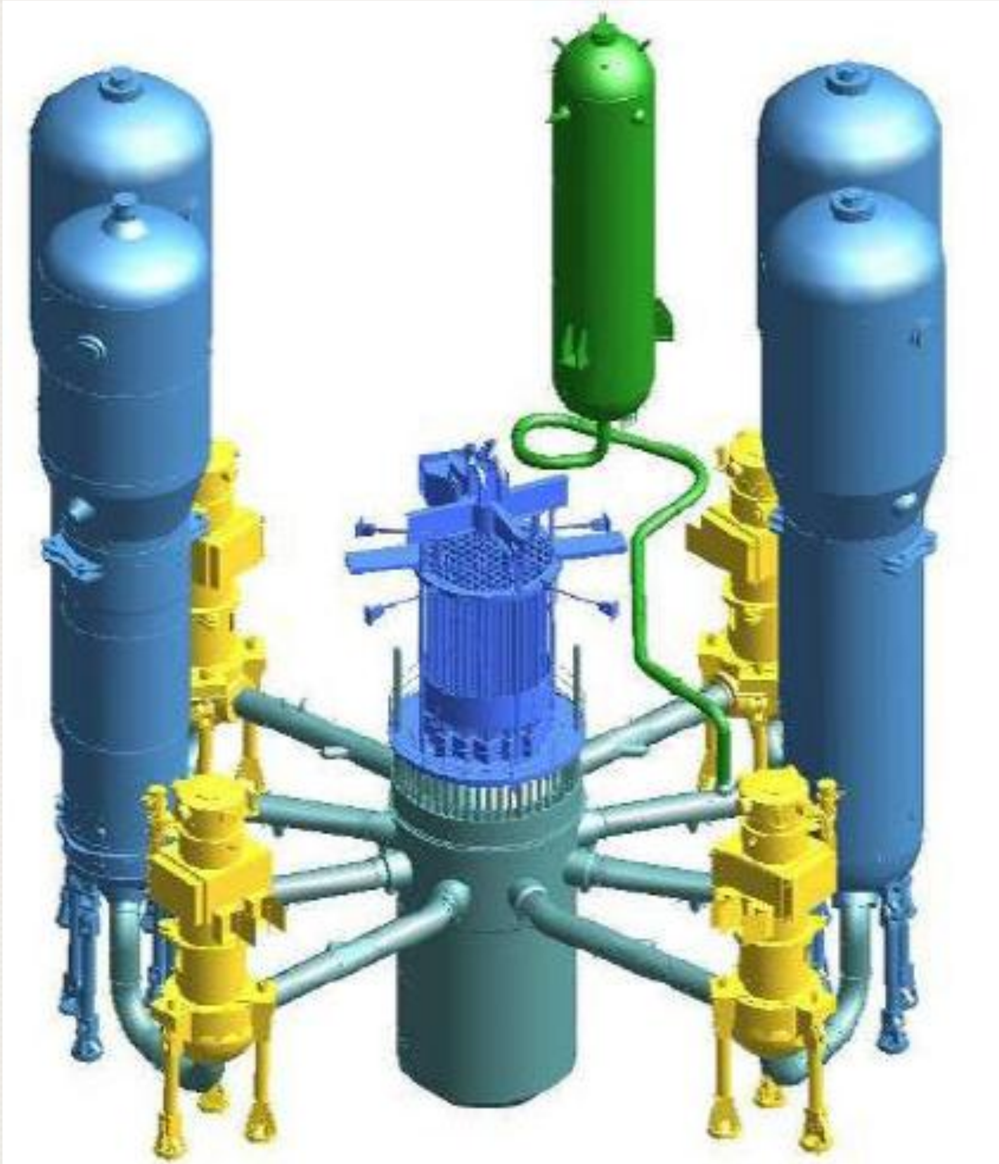
A Plant Layout of EPR

Courtesy of AREVA, France.

EPR Primary System Features

Courtesy of AREVA, France.

- 4 loop configuration, similar to those operating designs
- Enlarged capacity to enhance grace period in transients and accidents
- Extensive use of forgings with integral nozzles
- Adopt corrosion- and cracking-resistance materials

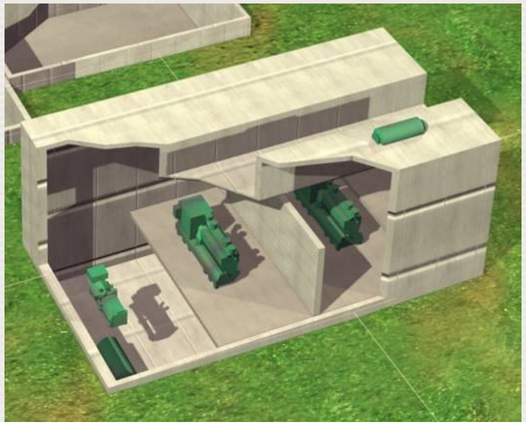


IAEA

EPR Safety Features

Courtesy of AREVA, France.

- Redundancy to overcome single failure
- Four 100% safety trains and physical separation
- Each enough to extract decay heat and cool the core
- Additional safety valves on the pressurizer allow faster reactor cool-down
- Diversity to avoid common cause failure
- 2 SBO DG + 4 main EDG →



IAEA

United States of America

AP1000

Westinghouse Electric Company LLC

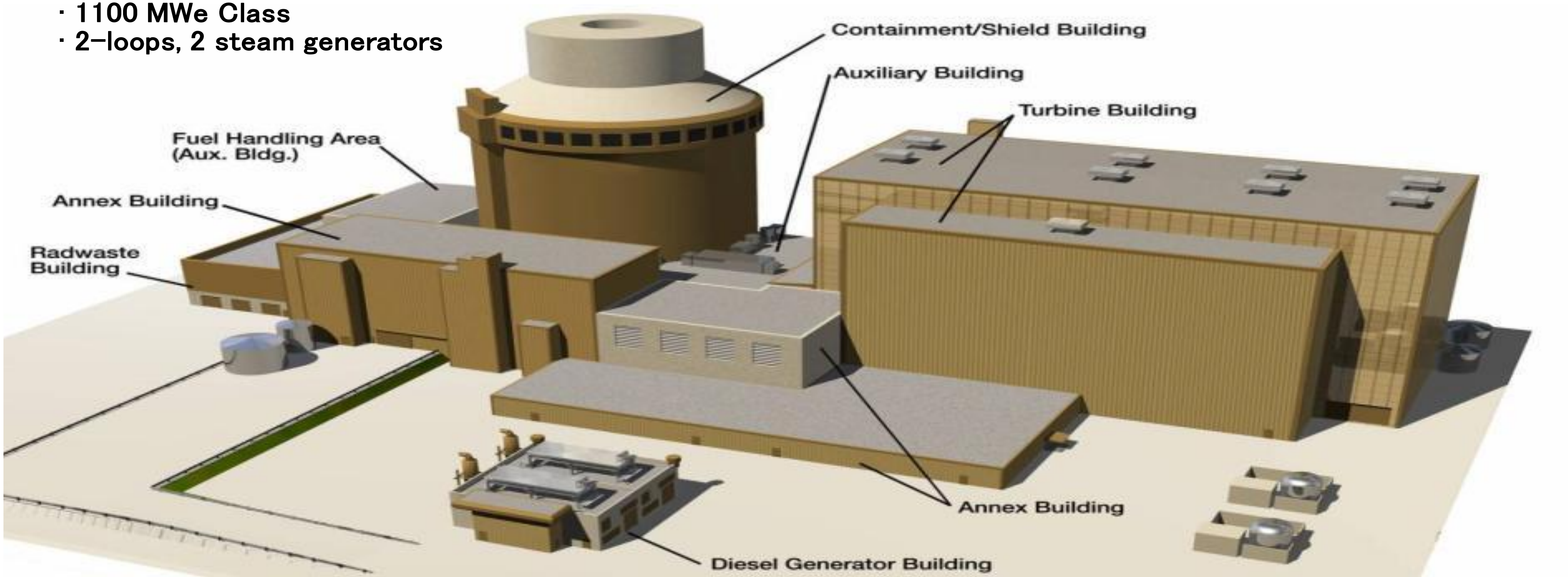
IAEA


The Westinghouse AP600

Westinghouse AP1000

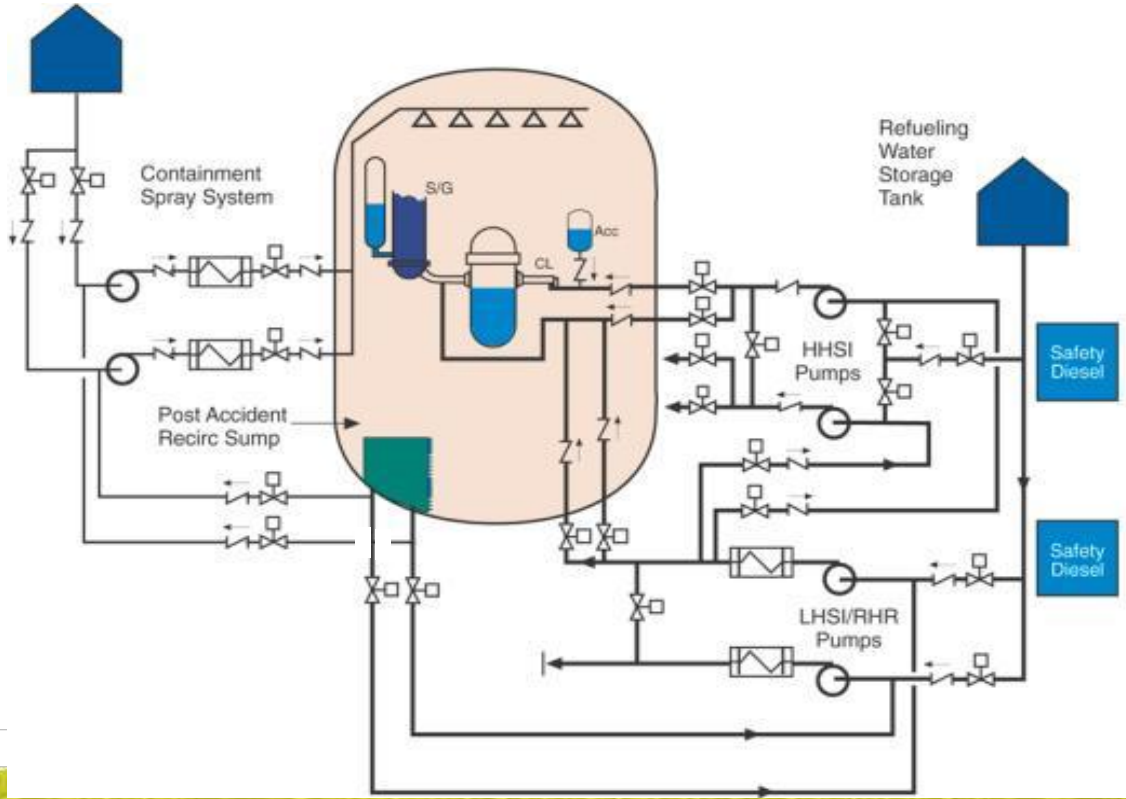
A compact station

- 3415 MWt. Primary system
- 1100 MWe Class
- 2-loops, 2 steam generators

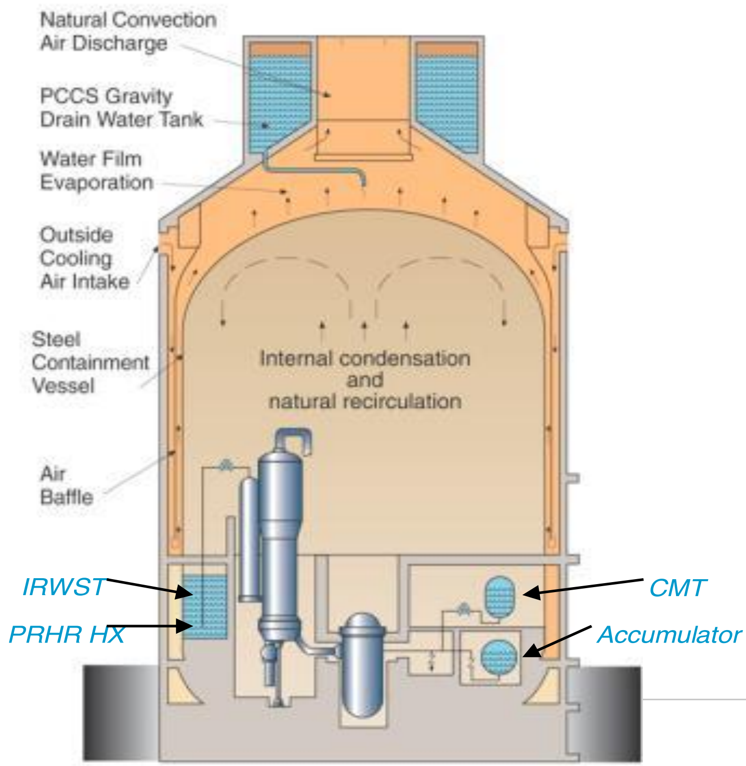


Simplification of Safety Systems Dramatically Reduces Building Volumes

Standard PWR



AP1000

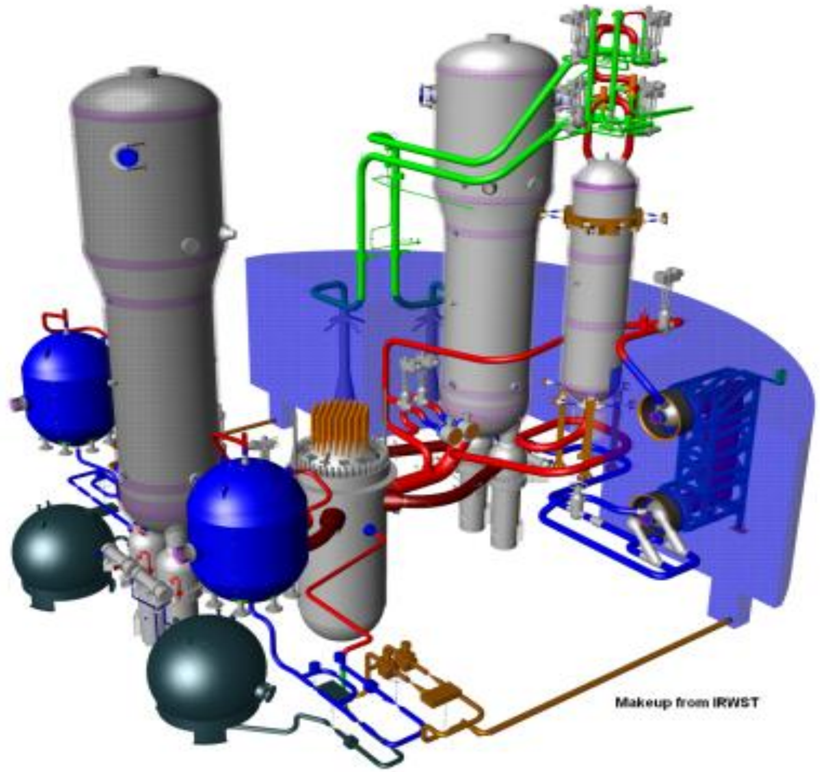


Westinghouse

IAEA

AP1000 Passive Core Cooling System Eliminate the need for AC Power

- **Passive Residual Heat Removal** (PRHR HX)
  - Natural circ. heat removal replaces auxiliary feedwater pumps
- **Passive Safety Injection**
  - Core Makeup Tanks (CMT)
    - Full RCS pres, natural circ. inject (replaces high head injection pumps)
  - Accumulators (ACC)
    - Similar to current plants
  - In-containment Refueling Water Storage Tank (IRWST) Injection
    - Low pres (replaces low head injection pumps)
  - Containment Recirculation
    - Gravity recirc. (replaces pumped recirc)
  - Automatic RCS Depressurization
    - Staged, controlled depressurization



United States of America

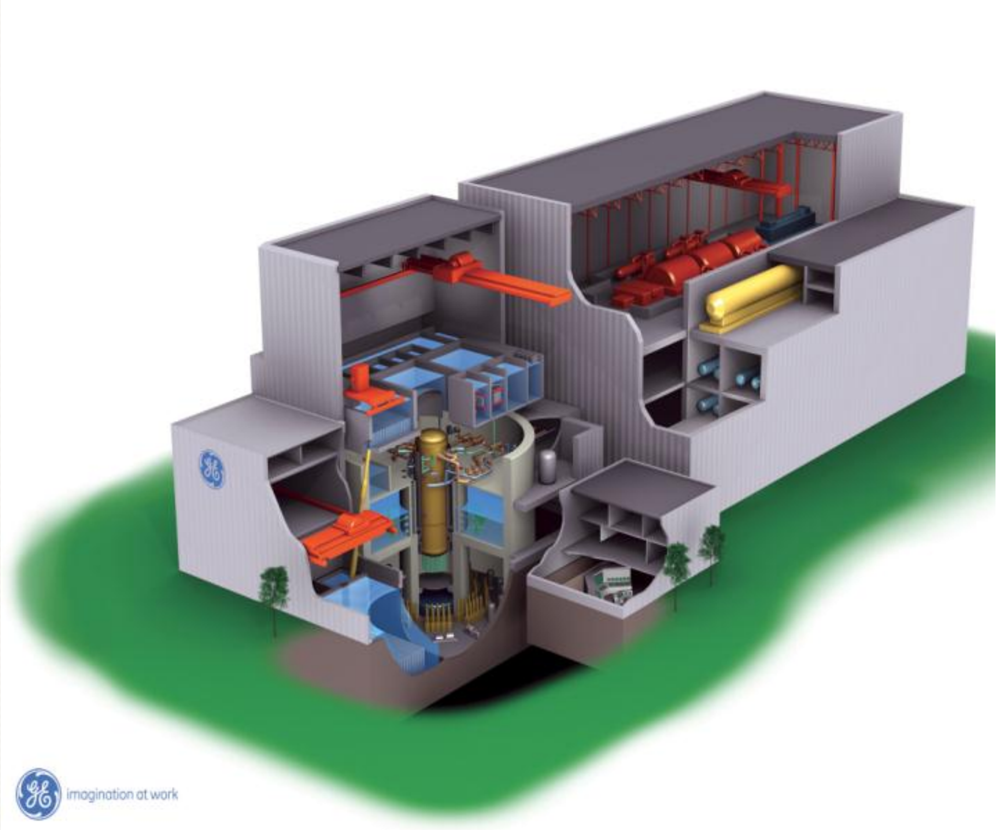
ESBWR

GE Hitachi Nuclear Energy

ESBWR

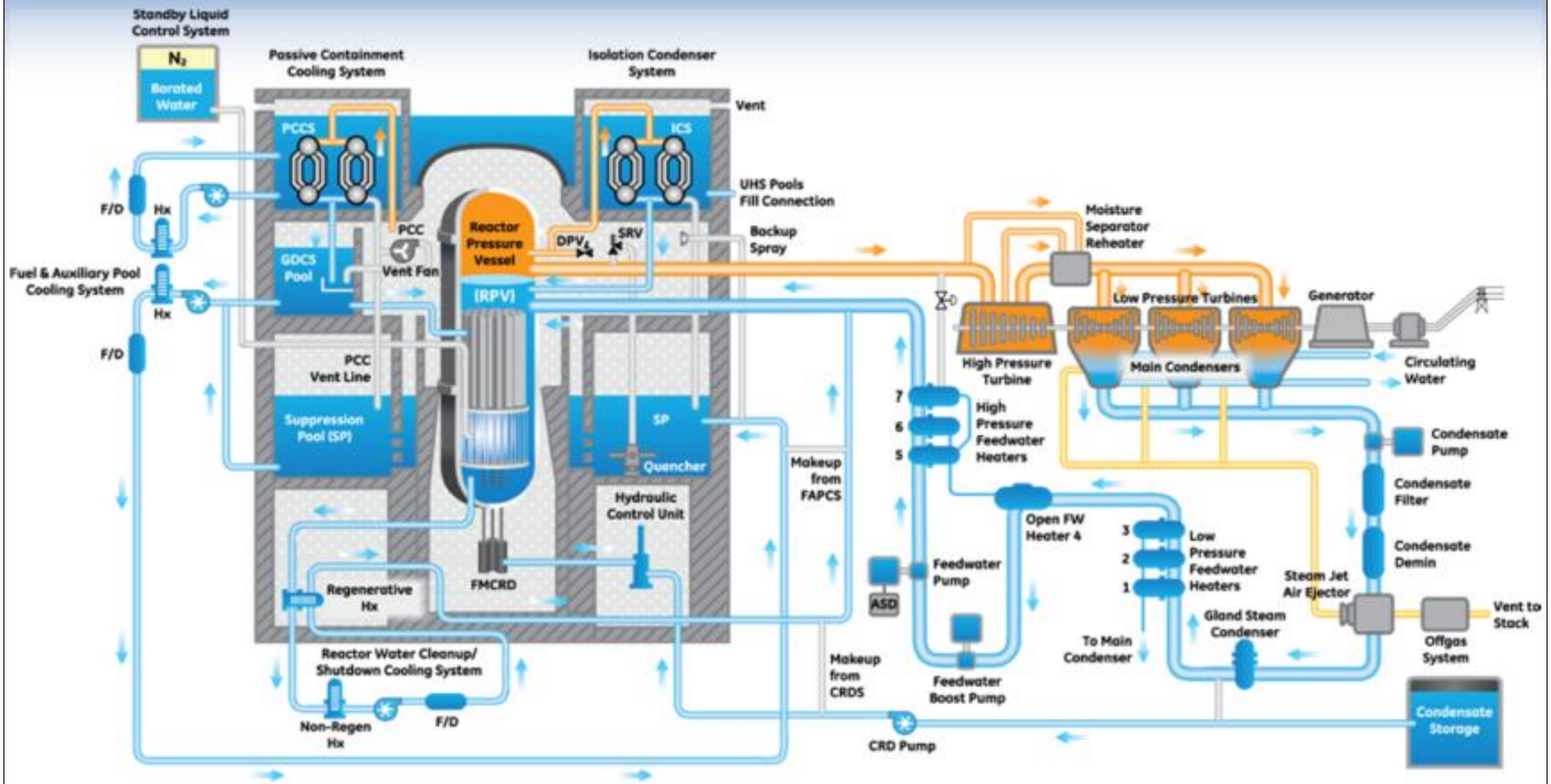
Courtesy of GE Nuclear Energy, USA

- Developed by GE
- Development began in 1993 to improve economics of SBWR
- 4500 MWt (~ 1550 MWe)
- In Design Certification review by the U.S.NRC – approved 10/2010
- Meets safety goals 100 times more stringent than current
- 72 hours passive capability
- Key Developments
  - NC for normal operation
  - Passive safety systems
    - Isolation condenser for decay heat removal
    - Gravity driven cooling with automatic depressurization for emergency core cooling
    - Passive containment cooling to limit containment pressure in LOCA
  - New systems verified by tests



IAEA

ESBWR Overall Flowchart

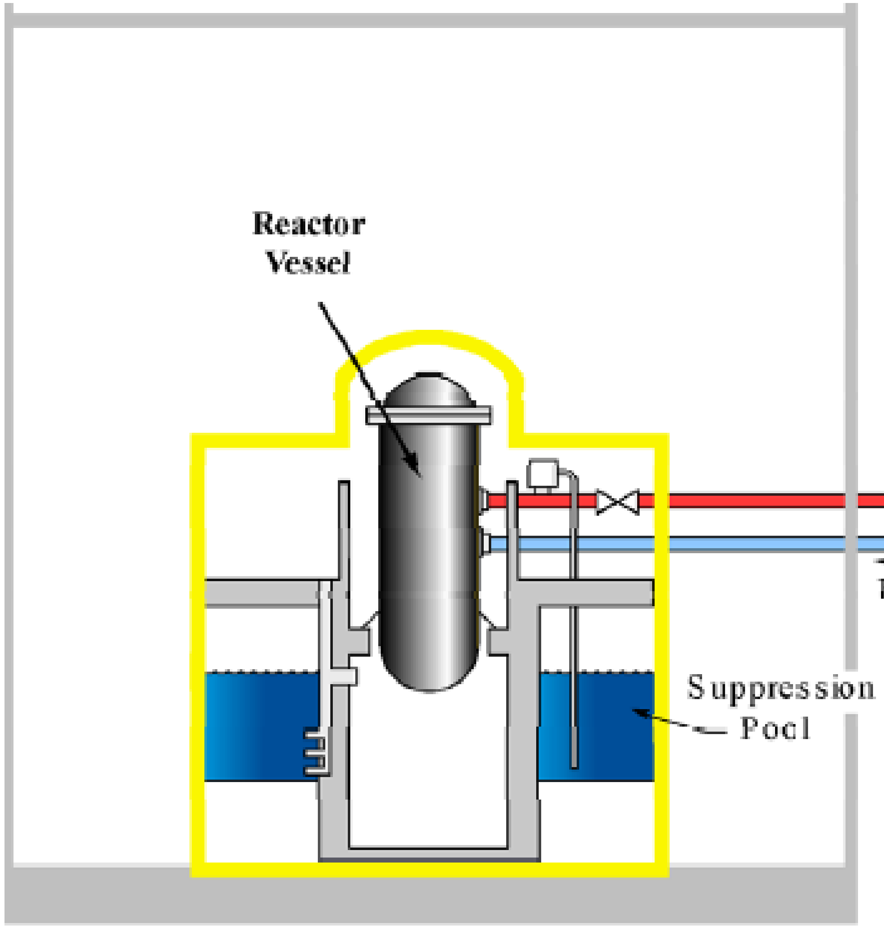


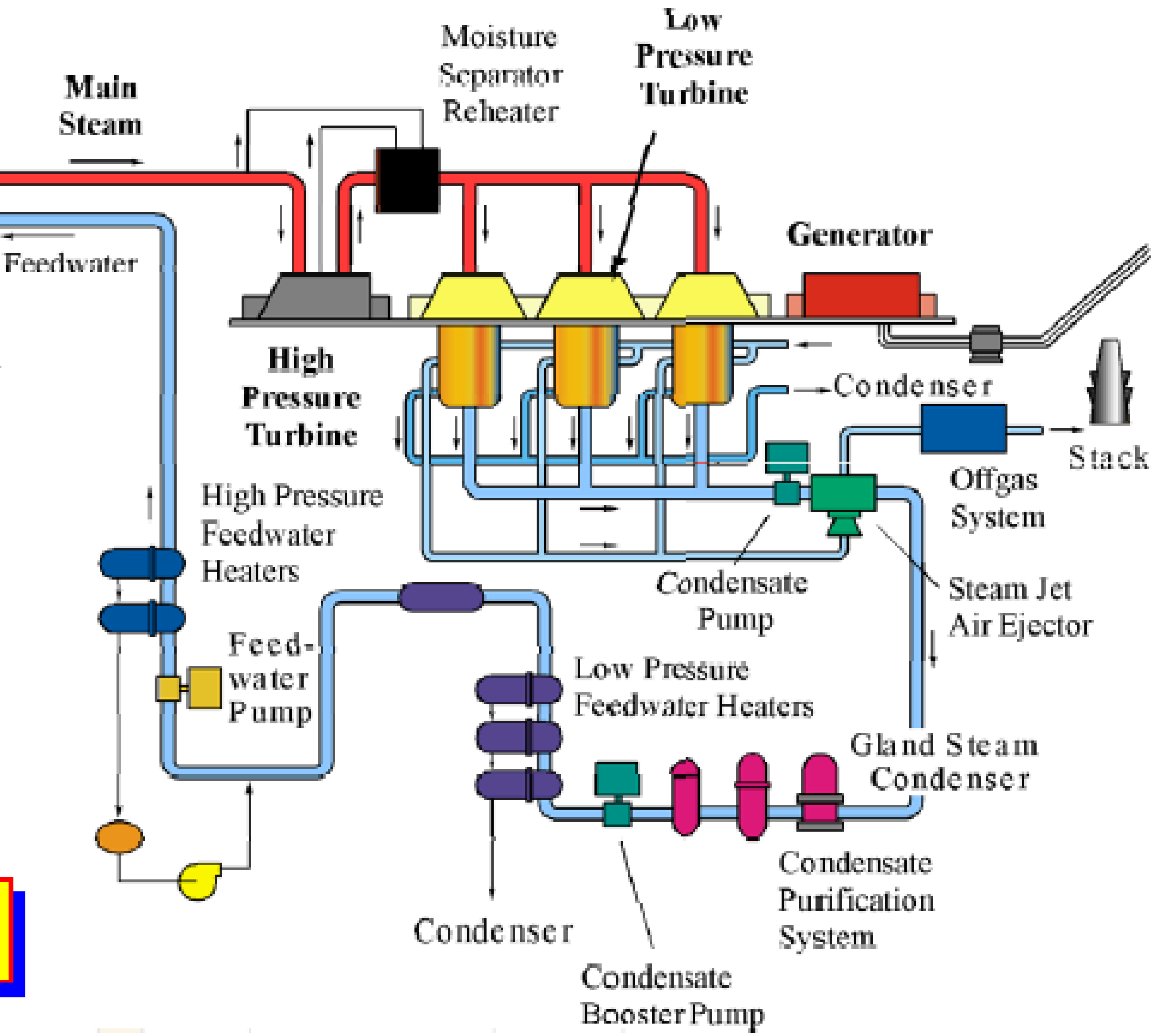
HITACHI

IAEA

Courtesy of GE Hitachi Nuclear Energy

ESBWR Plant Schematic



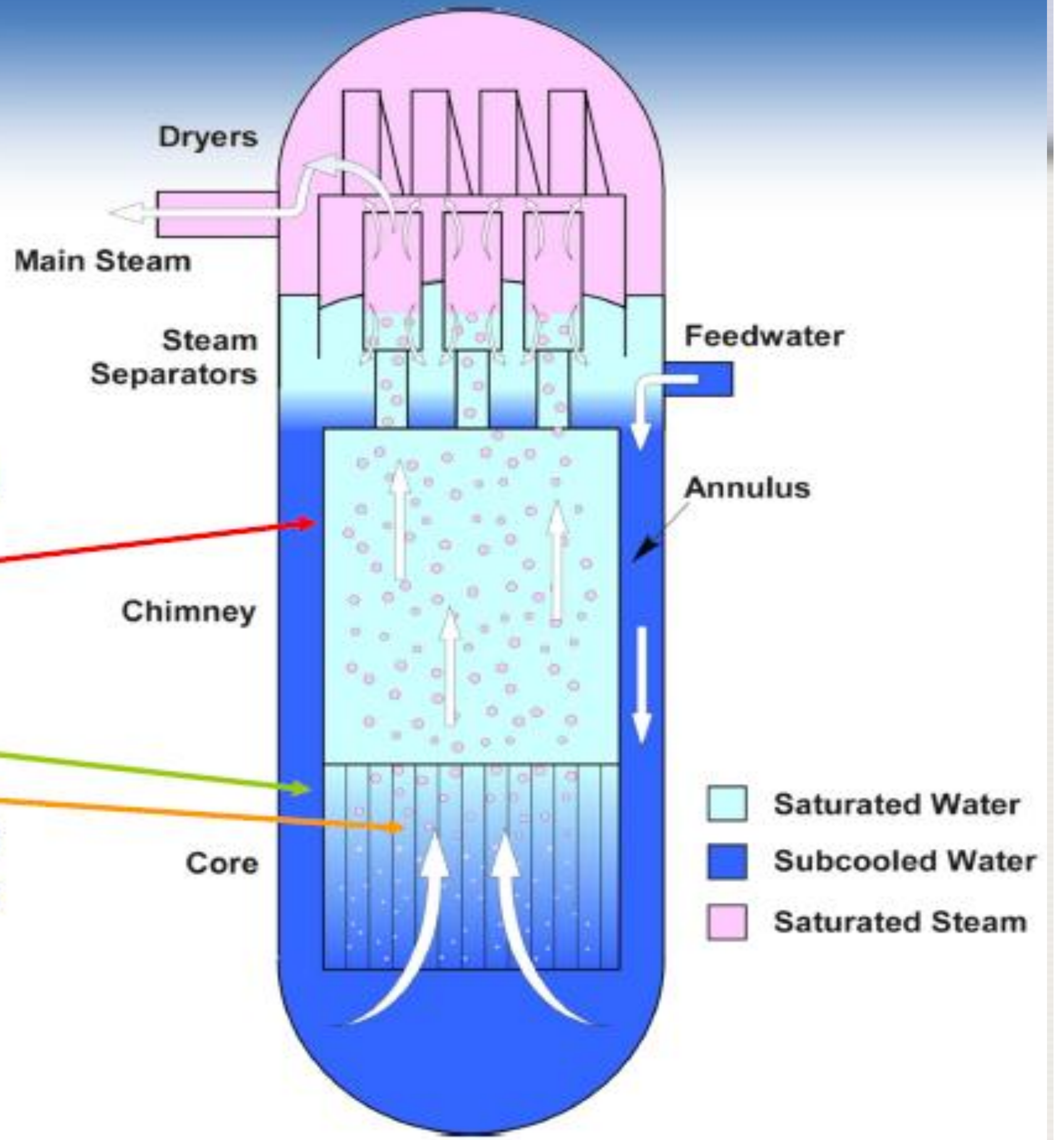


A typical direct cycle plant

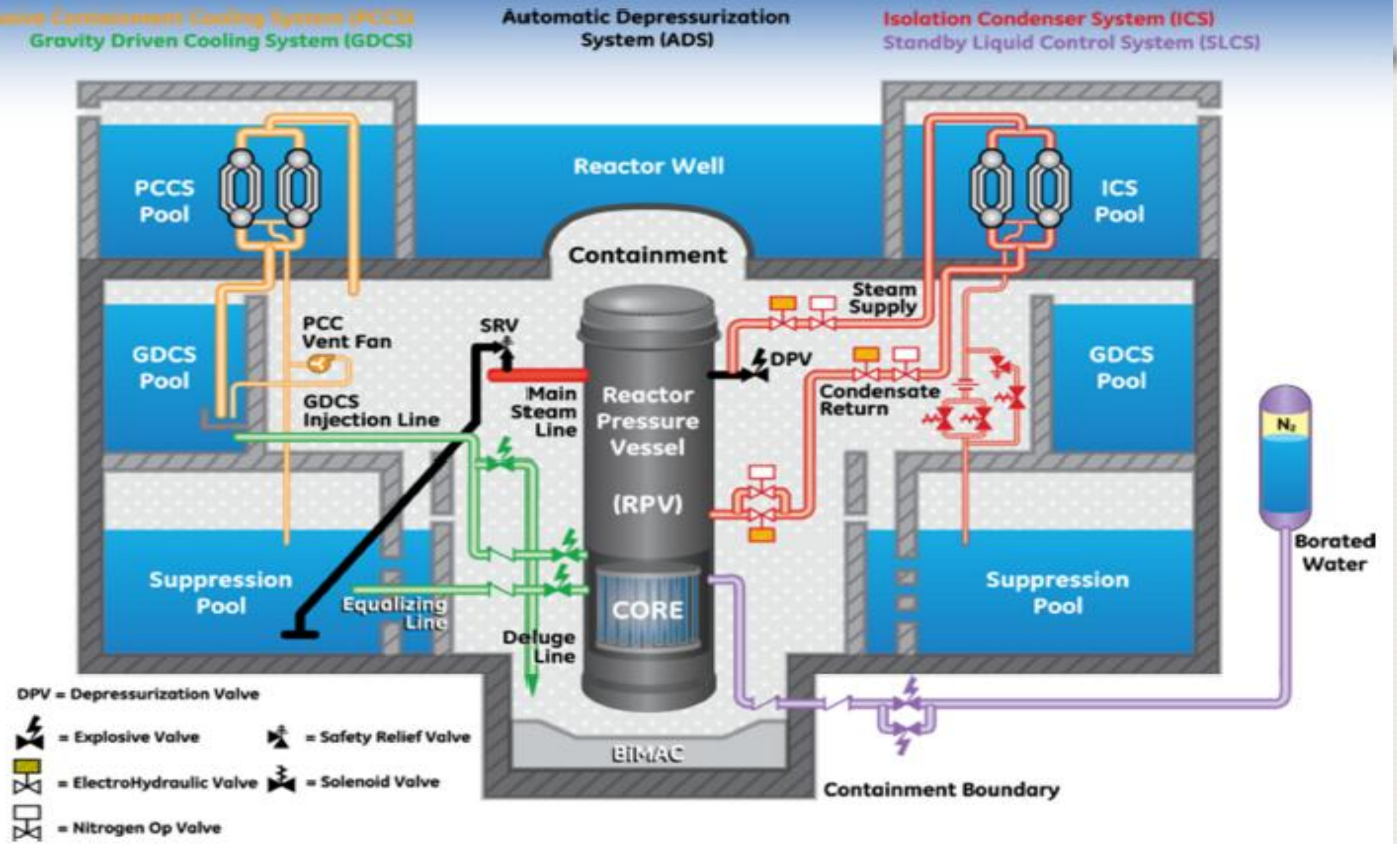
Natural Circulation

Simplification without performance loss …

- Passive safety/natural circulation
  - Increase the volume of water in the vessel
  - Increase driving head
    - Chimney, taller vessel
  - Reduce flow restrictions
    - Open downcomer
    - Shorter core
- Significant reduction in components
  - Pumps, motors, controls, Heat Exchangers
- **Power Changes with Feedwater Temperature and Control Rod Drives**
  - Minimal impact on maintenance

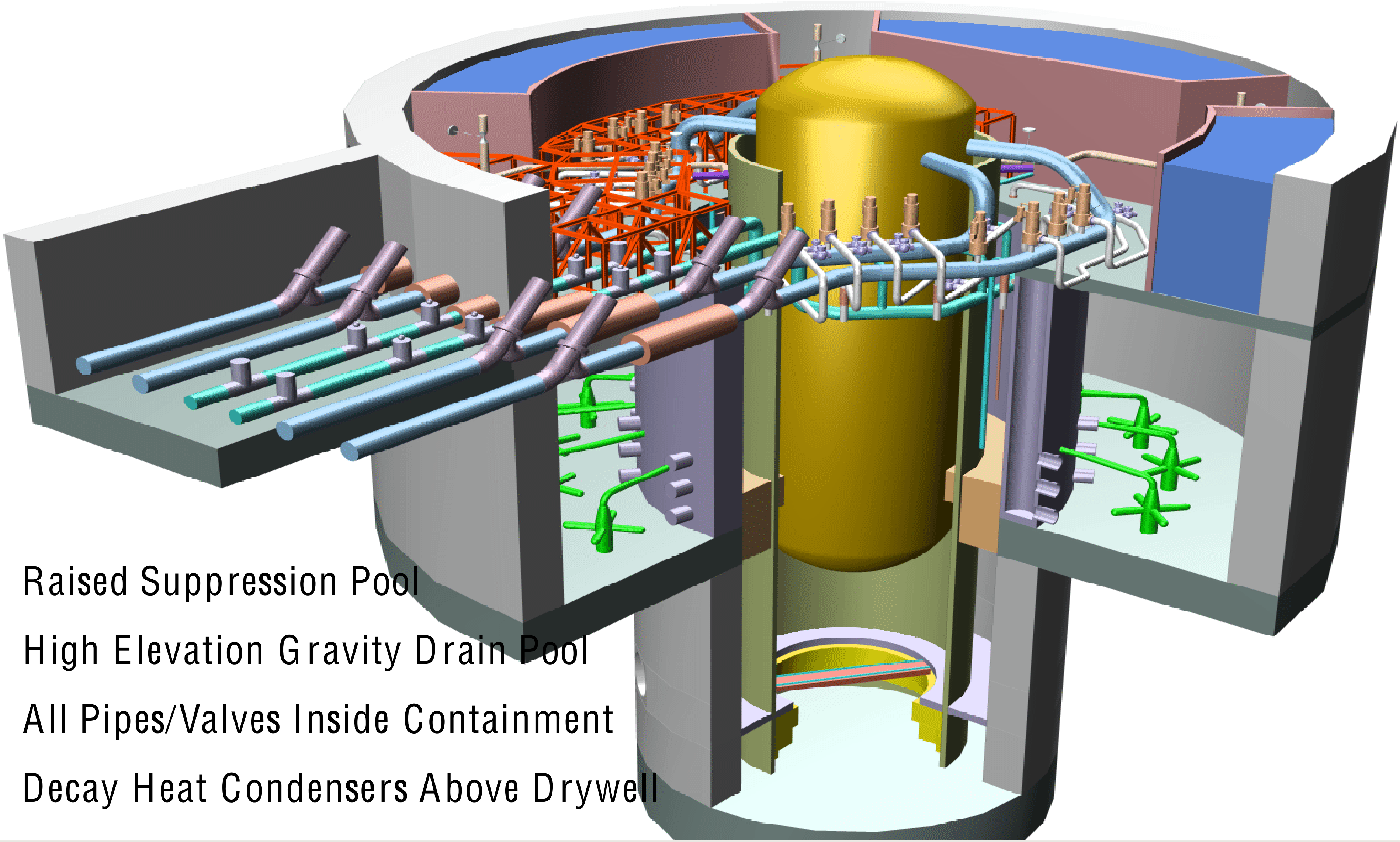


ESBWR passive safety systems



ESBWR Containment System

Courtesy of GE Hitachi Nuclear Energy



- Raised Suppression Pool
- High Elevation Gravity Drain Pool
- All Pipes/Valves Inside Containment
- Decay Heat Condensers Above Drywell

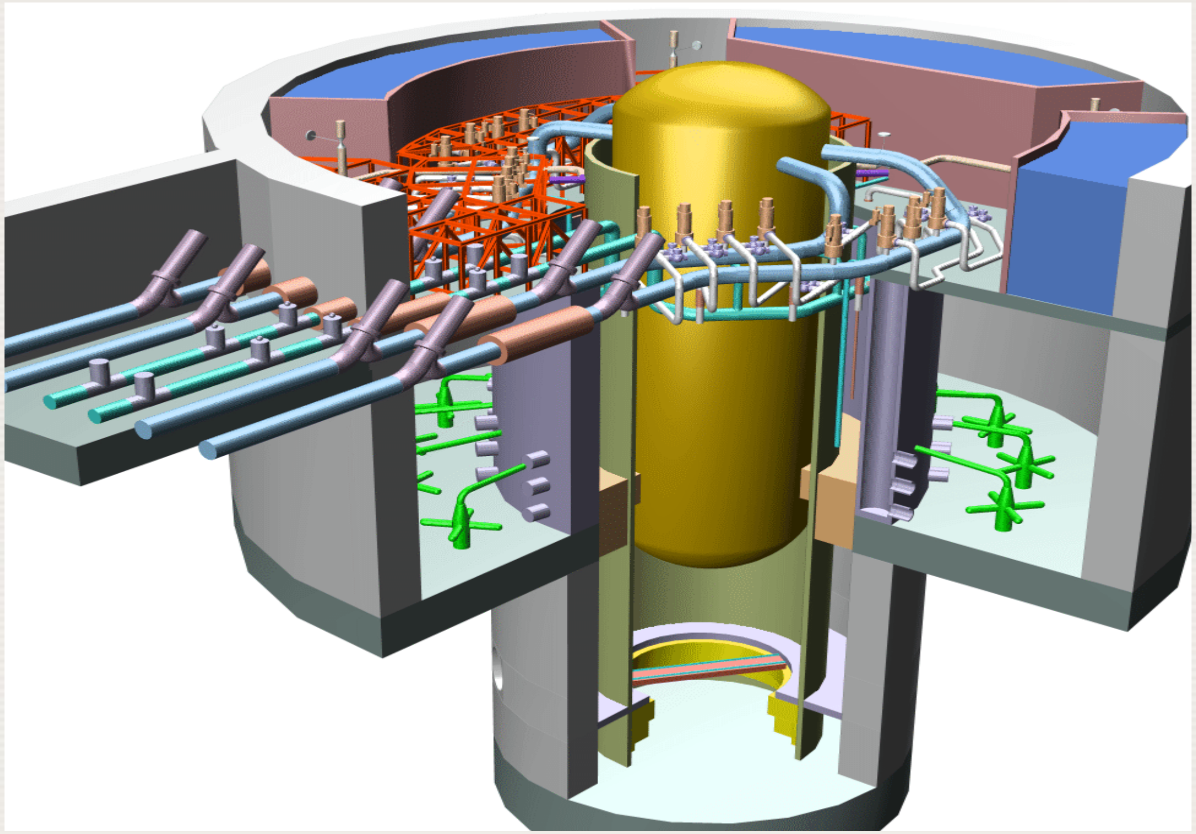
Comparison of BWR System

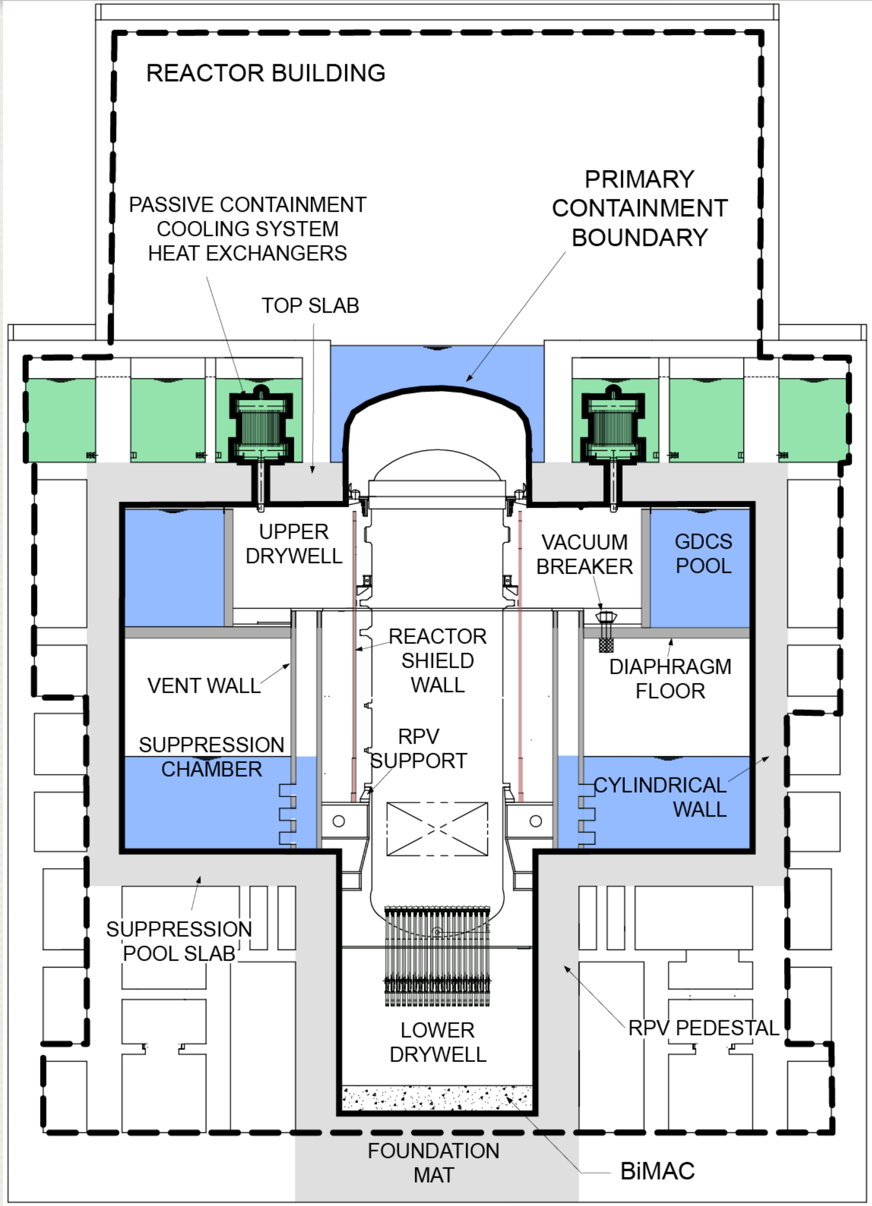
Feature	BWR/6	ABWR	ESBWR
Recirculation System inside RPV	Two external loop Recirc system with jet pumps	Vessel-mounted reactor internal pumps	Natural circulation
Control Rod Drives	Locking piston CRDs	Fine-motion CRDs	Fine-motion CRDs
ECCS	2-division ECCS plus HPCS	3-division ECCS	4-division, passive, gravity-driven
Reactor Vessel	Welded plate	Extensive use of forged rings	Extensive use of forged rings
Primary Containment	Mark III - large, low pressure, not inerted	Compact, inerted	Compact, inerted
Isolation Makeup Water	RCIC	Fluidic Controlled RCIC	Isolation condensers, passive
Shutdown Heat Removal	2-division RHR	3-division RHR	Non-safety system combined with RWCU
Containment Heat Removal	2-division RHR	3-division RHR	Passive
Emergency Power	3 safety-related D/G	3 safety-related D/G	Safety related batteries
Alternate shutdown	2 SLC pumps	2 SLC pumps	2 SLC accumulators
Control & Instrumentation	Analog, hardwired, single channel	Digital, multiplexed, fiber optics, multiple channel	Digital, multiplexed, fiber optics, multiple channel
In-core Monitor Calibration	TIP system	A-TIP system	Gamma thermometers
Control Room	System-based	Operator task-based	Operator task-based
Severe Accident Mitigation	Not specifically addressed	Inerting, drywell flooding, containment venting	Inerting, drywell flooding, core catcher

IAEA

ESBWR Containment

Courtesy of GE Nuclear Energy, USA

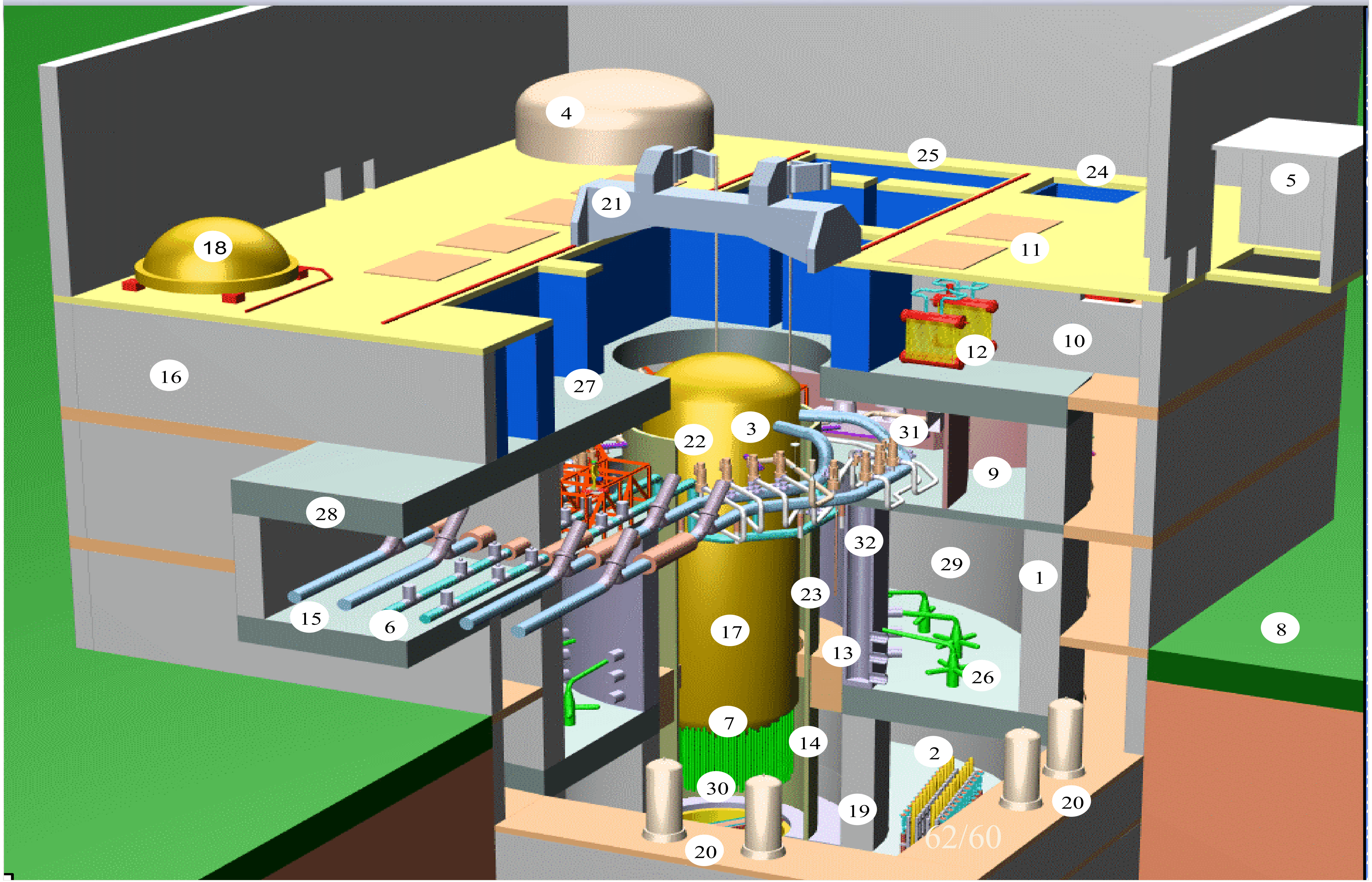




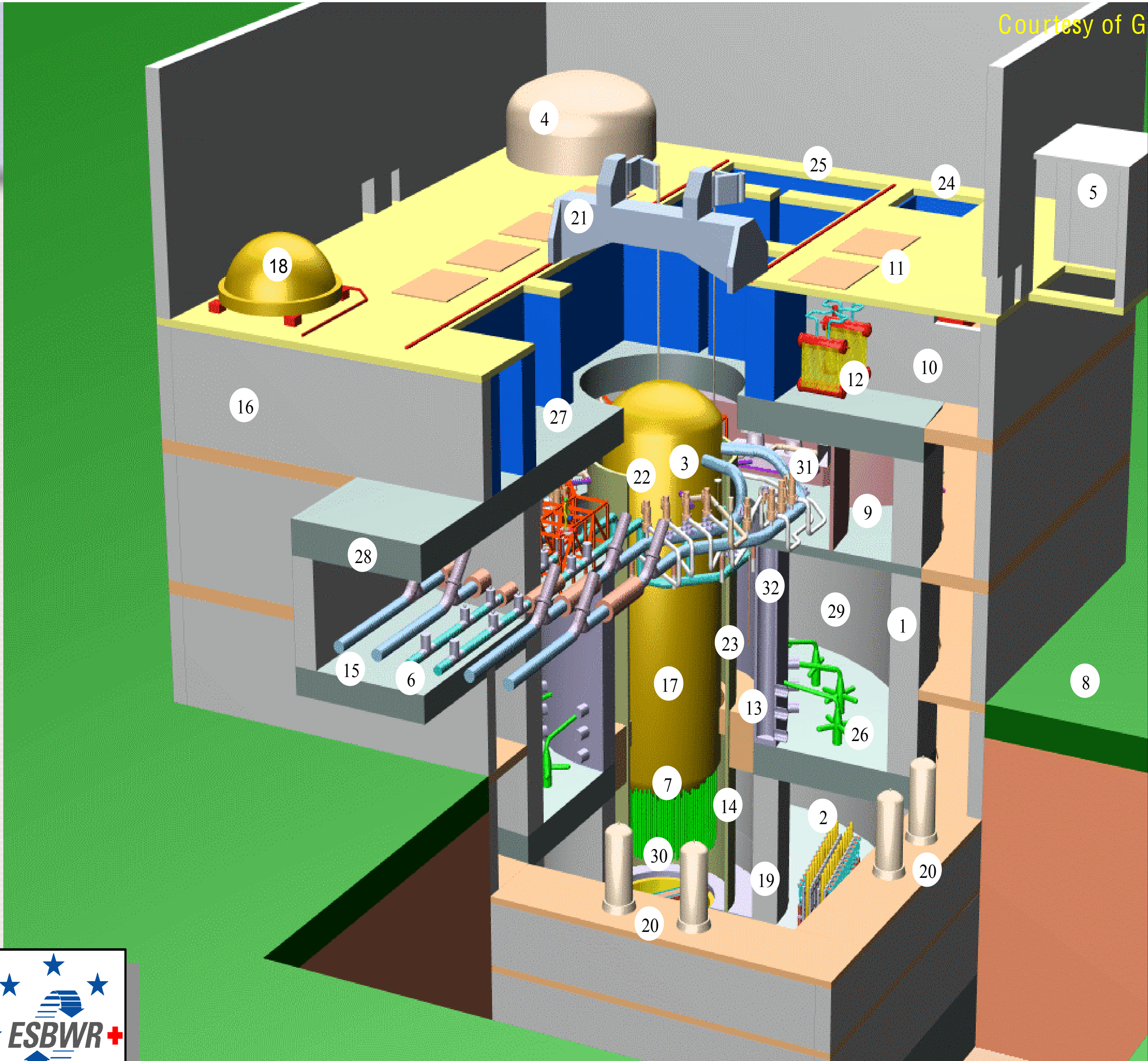
IAEA

ESBWR Spent Fuel Storage Pool

Courtesy of GE Hitachi Nuclear Energy



Courtesy of GE Hitachi Nuclear Energy



1. Containment
2. **CRD HCU's**
3. Depressurization Valve
4. Drywell Head (storage)
5. External Equipment Removal Hatch
6. Feedwater Lines
7. Fine-Motion Control Rod Drives (FMCRD)
8. Grade Level (variable elevation)
9. Gravity Driven Cooling System (GDCS) Pool
10. Isolation & Passive Cooling (IC/PCC) Pools
11. IC/PCC Pool Cover
12. Isolation Condenser
13. LOCA Vents
14. Lower Drywell
15. Main Steam Lines
16. Reactor Building
17. Reactor Pressure Vessel (RPV)
18. RPV Head (storage)
19. RPV Pedestal
20. Reactor Water & Shutdown Cooling System
21. Refueling Machine
22. Safety Relief Valves (SRV)
23. Shield Wall
24. Spent Fuel Cask Pit
25. Spent Fuel Storage Pool
26. SRV Quenchers
27. Steam Dryer/Separator Storage Pool
28. Steam Tunnel
29. Suppression Pool
30. Under vessel Servicing Platform
31. Upper Drywell
32. Vent Wall

A Simplified 1380 MWe BWR

France

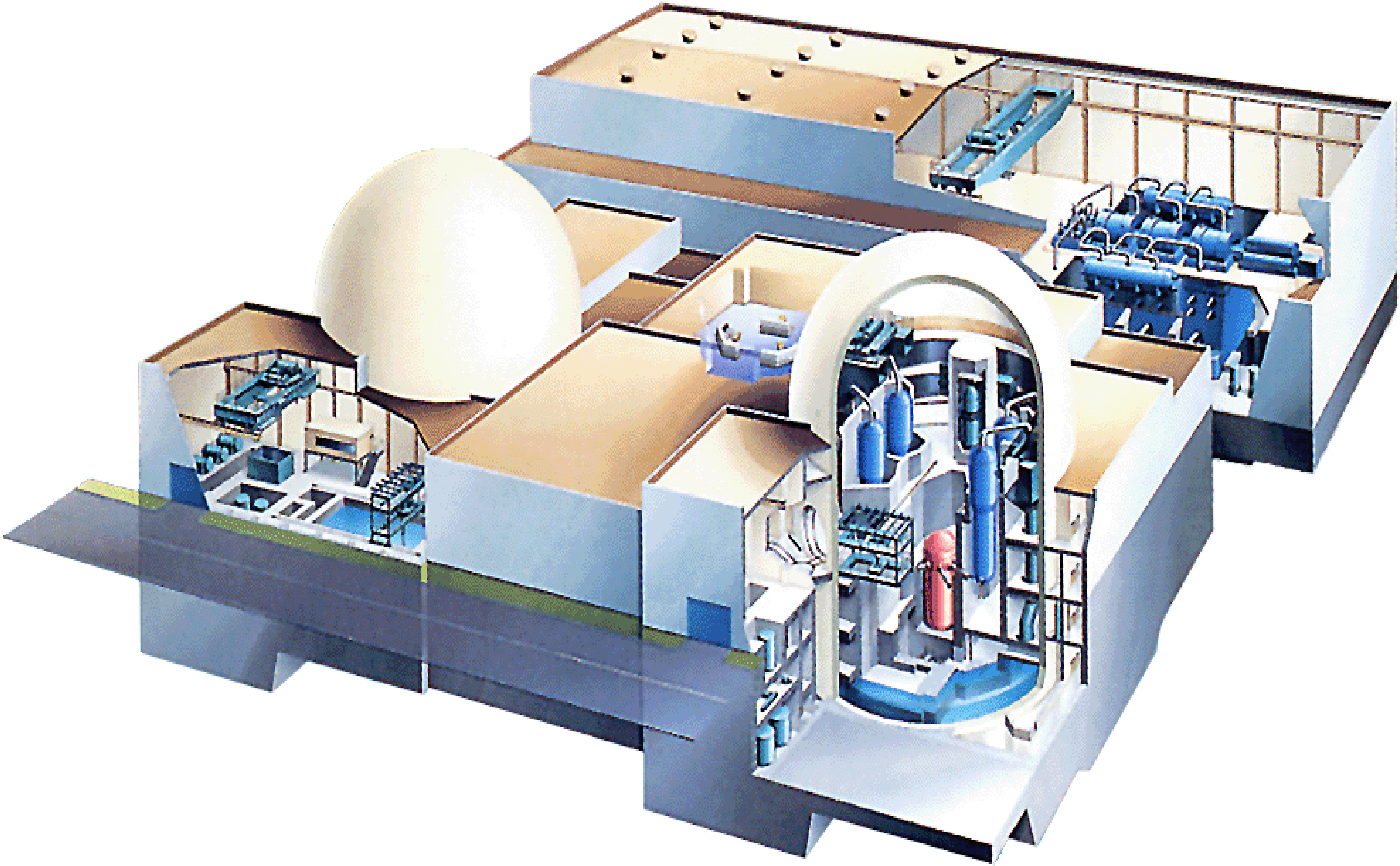
US-APWR and **EU-APWR**

Mitsubishi Heavy Industries, Ltd.

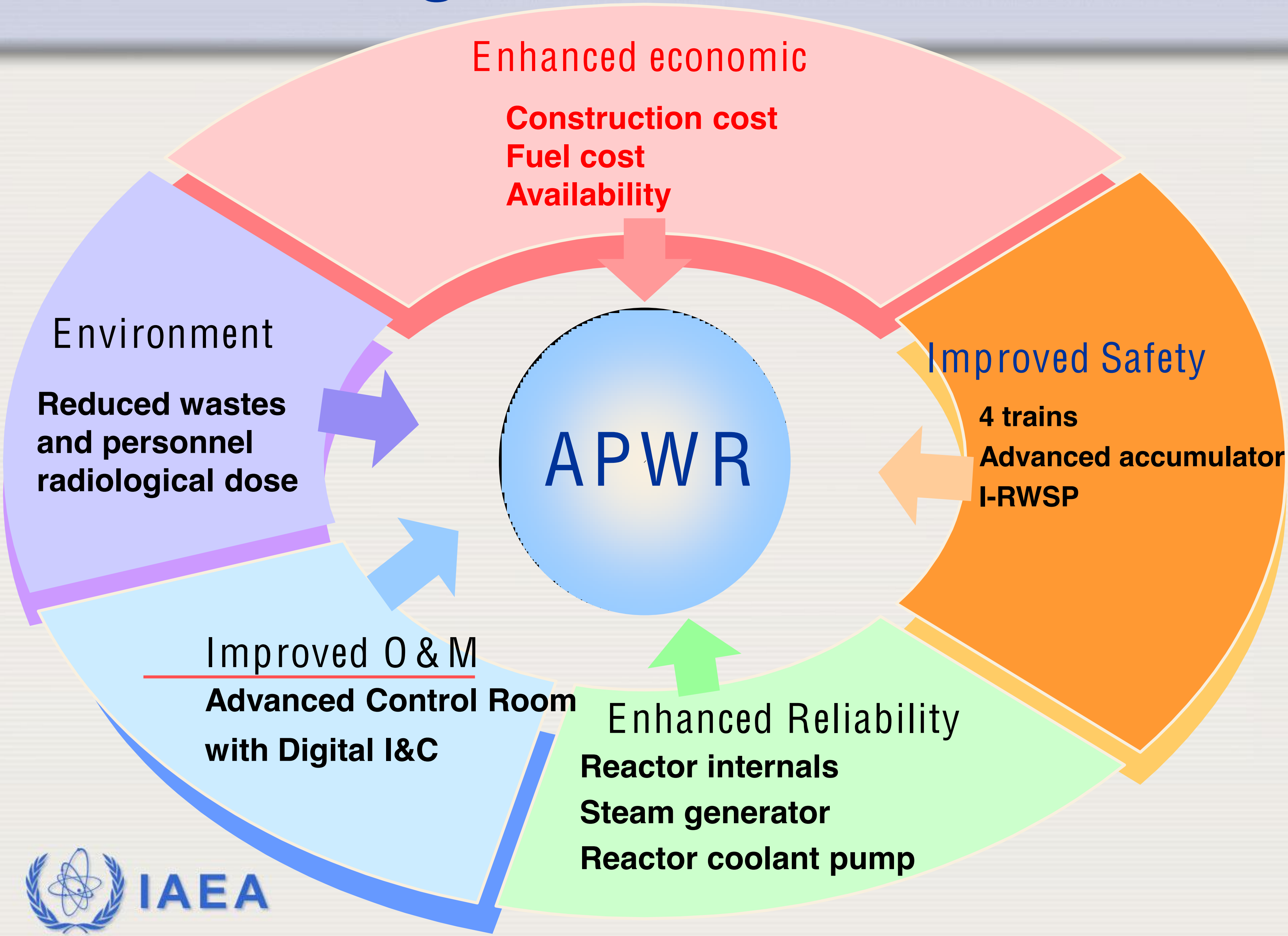
IAEA

Advanced Pressurized Water Reactor

Courtesy of Mitsubishi Heavy Industries



APWR Design Characteristics

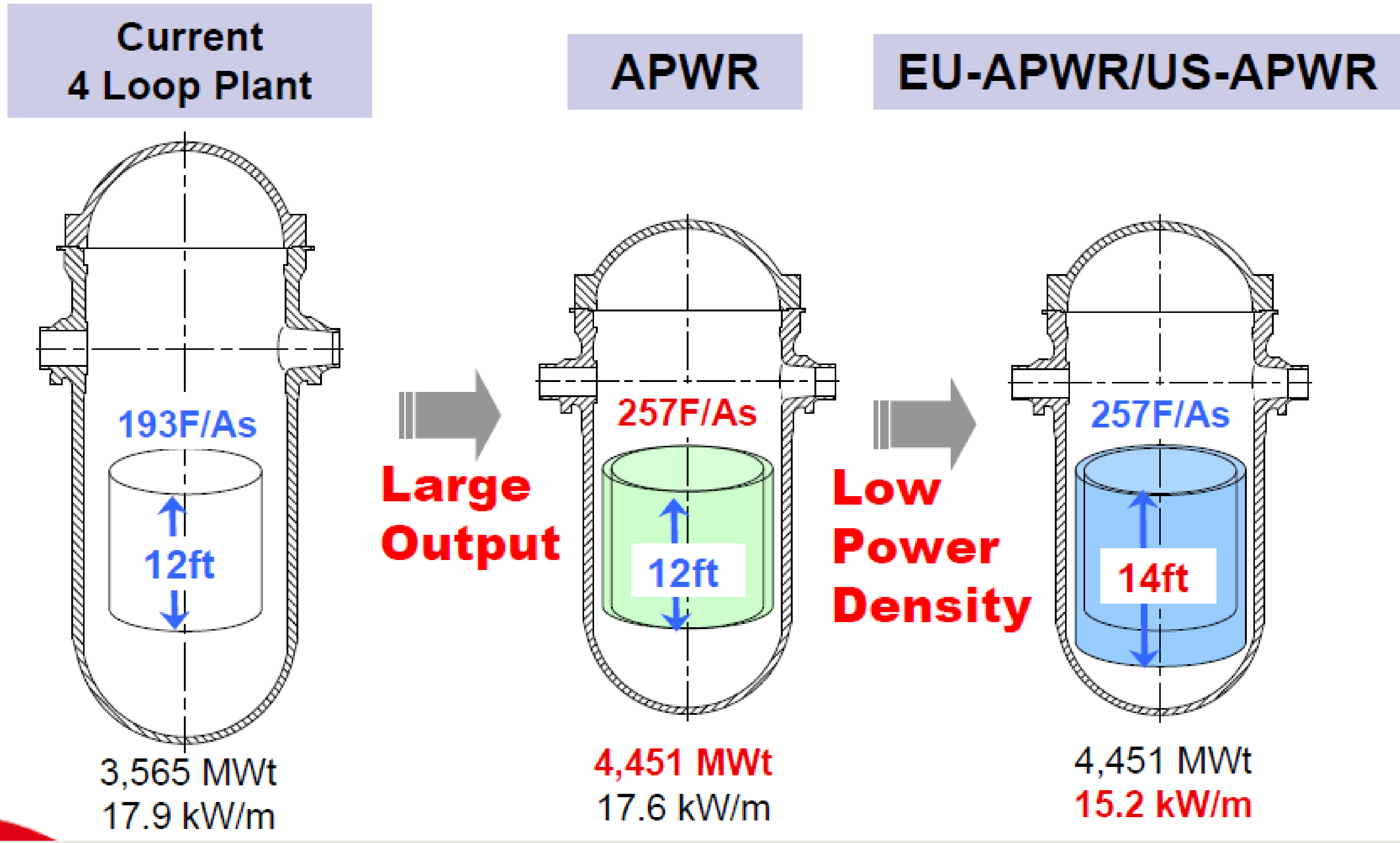


IAEA

APWR Technical Specification

ITEM		APWR	Current 4 Loop
Main Specifications	Electric Output	1,538 MWe	1,180 MWe
	Core Thermal Output	4,451 MWt	3,411 MWt
	Fuel Assembly Type, Number	17×17, 257	17×17, 193
	Radial Core Support	Neutron Reflector	Baffle/Former
	SG heat transfer area	6,500 m^2	4,879 m^2
	Coolant Flow	25,800 m^3/h/loop	20,100 m^3/h/loop
	Engineered Safety Features	4 Divisions	2 Divisions
	Steam Turbine	TC6F54	TC6F44
	I & C	Full-Digital	Digital (Partially)
Reliability / Safety	Design Life Time	60 years	40 years
	Core Damage Frequency	approx.1/10	Base
Operation / Maintenance	Occupational Dose	0.2 man SV/y	0.4 – 5 man SV/y
	Radioactive Waste	60 Drums/y	140 Drums/y
	Min. Operators in MCR	1	2
Uranium Saving	MOX Loading	1 / 3 – 1 / 1 Core	1 / 4 – 1 / 3 Core
Economy	Primary build.volume /KWe	20 %less	Base

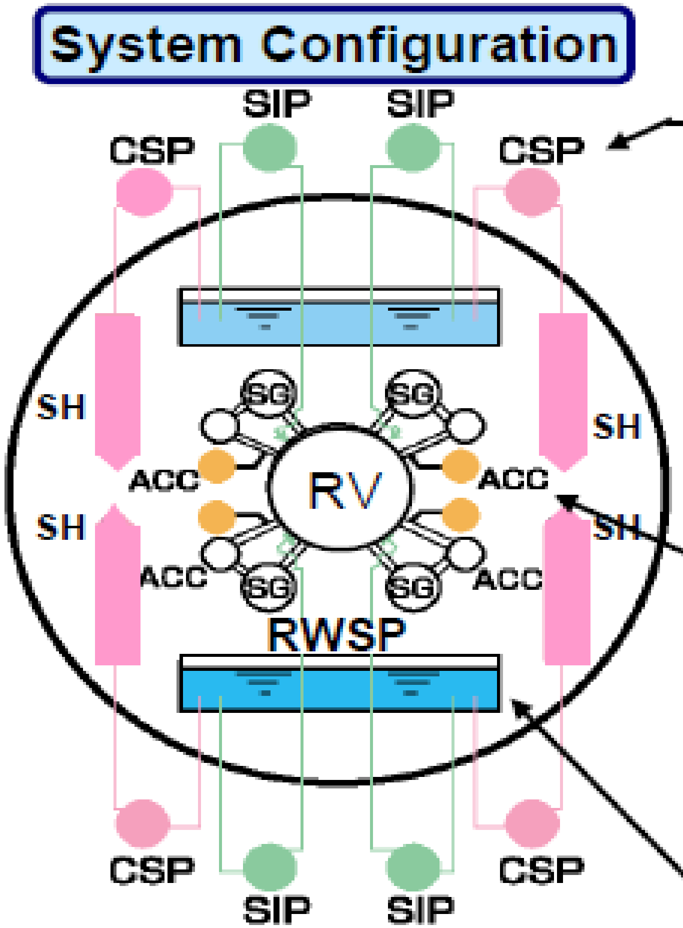
Evolution from 4-loop PWR to APWRs



IAEA

APWR's ECCS Configuration

Courtesy of MHI, Japan



4 train ;DVI*1 design for SIP
→ Independent and Redundant
→ Simplified configuration
50%-capacity pumps

Advanced Accumulator
→Passive flow switch
Combination of conventional accumulator and LHSIP

In-containment RWSP
→ Low CDF*2
1/10 of current 4-loop plant

ACC : Advanced Accumulator
SIP : Safety Injection Pump
LHSIP: Low Head SIP
CSP : Containment Spray Pump
SH : Spray Header
RV : Reactor Vessel
RWSP: Refueling Water Storage Pit

*1: Direct Vessel Injection

*2: Core Damage Frequency

APWR's Safety Design Items

Item	US-APWR	EU-APWR
Design Extension Conditions	■Not required in the US ■Alternate AC (non-safety)	■ATWS, multiple SGTR, MSLB+SGTR and SBO) and complex sequences ■Additional Diverse systems EBS, ACCWS, AAC(two trains)
Grace Period	■ 10minutes	■ 30 minutes
Fire protection System and layout design	■ assuming fire coincident with OLM	■ assuming fire coincident with a single failure and OLM
Air plane crash	■ Base	■ Assuming Larger Plane

APWR Plant Parameters

		Current 4 Loop	APWR	EU-APWR/US-APWR
Electric Output		1,180 MWe	1,538 MWe	1,700 MWe Class
Core Thermal Output		3,411MWt	4,451 MWt	4,451 MWt
Steam Generator	Model	54F	70F-1	91TT-1
	Tube size	7/8"	3/4"	3/4"
Reactor Coolant Pump	Model	93A-1	MA25(60 Hz)	MA25(50Hz)/MA25(60Hz)
Turbine	LP last-stage blade	44 inch	54 inch	74 inch

- **APWR**
  - ✓ 1,538MWe output is achieved by large capacity core and large capacity main components such as SG, RCP, turbine, etc.
- **EU-APWR/US-APWR**
  - ✓ 1, 700MWe output is achieved from a higher efficiency than APWR.
    - Same core thermal output with APWR
    - High-performance, large capacity steam generator
    - High-performance turbine

APWRs Safety System and I&C

			Current 4 Loop	APWR	EU-APWR US-APWR
Safety Systems	Trains	Electrical	2 trains	2 trains	4 trains
		Mechanical	2 trains	4 trains	4 trains
	Systems	HHSI pump	100% × 2	50% × 4(DVI)	50% × 4(DVI)
		LHSI pump	100% × 2	-	-
		ACC	4	4 (Advanced)	4 (Advanced)
	RWSP		Outside CV	Inside CV	Inside CV
Containment Vessel			PCCV	PCCV	PCCV
I & C	Control Room		Conventional	Full Digital	Full Digital
	Safety I&C		Conventional		
	Non-Safety I&C		Digital		

France and Japan

ATMEA1

ATMEA

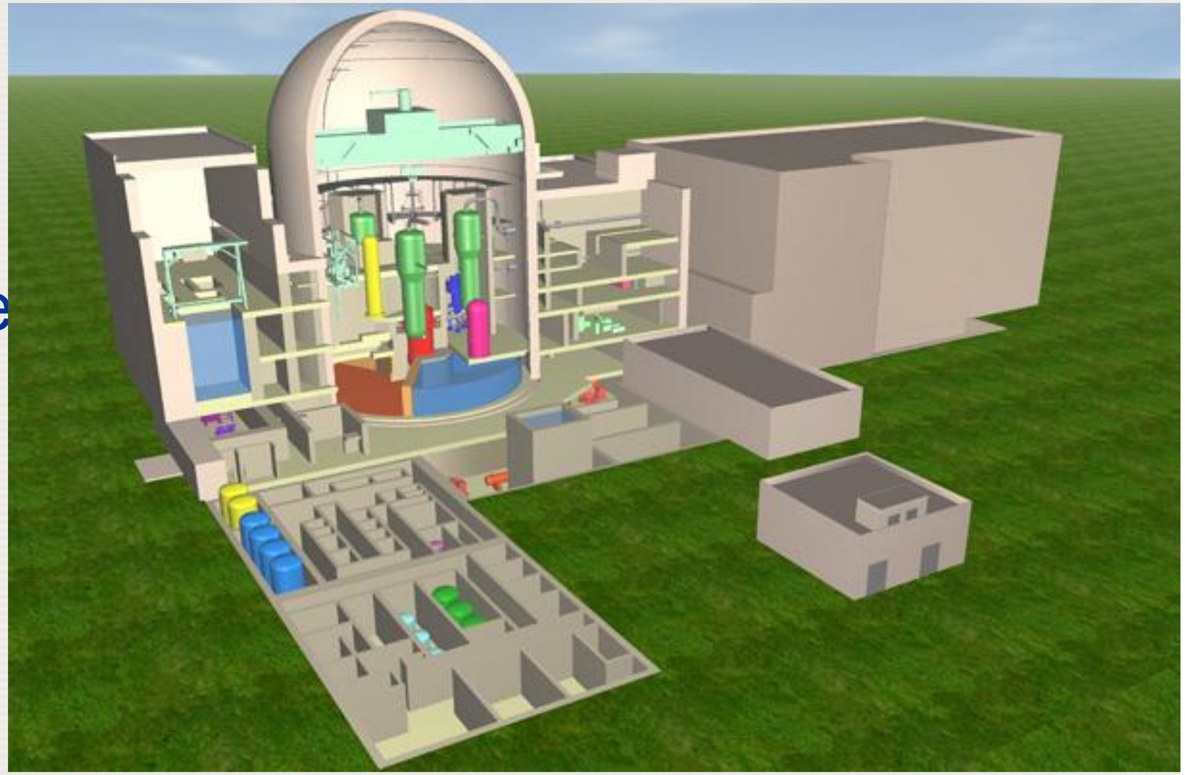
An AREVA and MHI Company

IAEA

ATMEA1

Courtesy of ATMEA, France

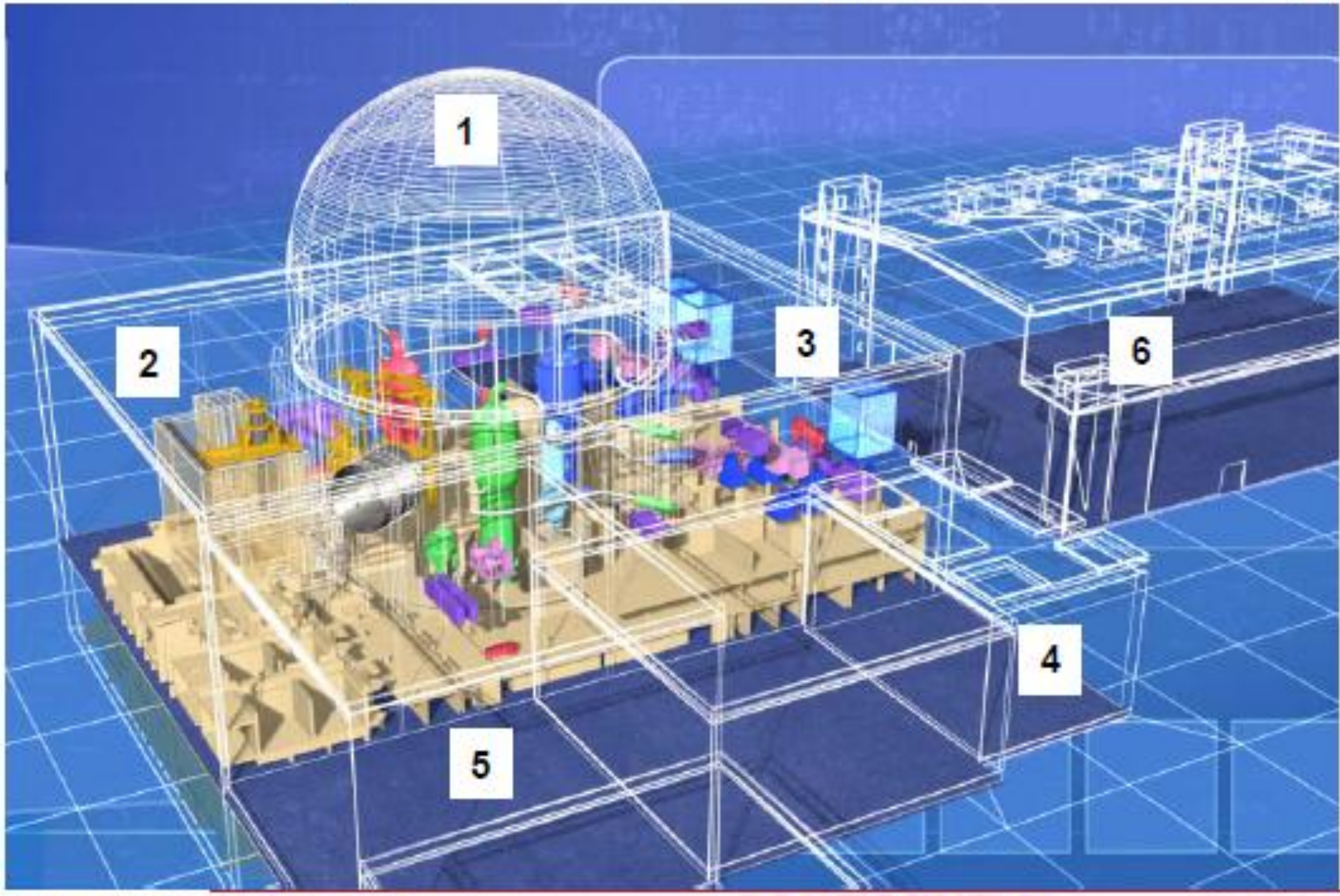
- 1100 MWe, 3 loop plant
- Combines AREVA & Mitsubishi PWR technologies
- Relies on active safety systems & includes core catcher
- Design targets:
  - 60 yr life
  - 92% availability
  - 12 to 24 month cycle; 0-100% MOX



IAEA

ATMEA1 Technical Description

Reactor Type	3-Loop PWR	Safety System	3-Train reliable active system with passive features
Electrical output	1100 – 1150 MWe (Net)	Severe Accident Management	Core catcher Hydrogen re-combiners
Core	157 Fuel Assemblies	Resists airplane crash	Pre-stressed Concrete Containment Vessel
Steam Pressure	More than 7 MPa	I&C	Full Digital



1. Reactor Building
2. Fuel Building
3. Safeguard Building
4. Emergency Power Building
5. Nuclear Auxiliary Building
6. Turbine Building

ATMEA

Operating Performance
Operating Parameters

	ATMEA1	Notes
Electrical output	1,100 – 1,150 MWe	Obtain 10% more electric output vs thermal output, which results in less cost and waste
Thermal output	3,150 MWth	
Height / Number of FAs	17x17- 4.2m / 157 FAs	Standard configuration
Fuel enrichment	< 5.0 wt%	Same as operating PWRs
Max. fuel assembly burn-up	62GWd/t	Max. achievable BU with UO2 enrichment less than 5%
Linear heat rate	17.5 kW/m	Low LHR contributes lowering fuel cost
T hot (best estimate)	326deg-C	High temperature within the range of our experiences
Cooling flow rate (BE)	24,800 m3/h/loop	Maximized flow rate within the range of our experiences
Thermal margin	>15 %	Satisfy URD requirement
Steam Generators	Around 8,000 m2 with axial economizer	Efficient and experienced design
Steam pressure (BE)	7.3 (MPa)	One of the world highest
Net efficiency	35 – 37 %	Site dependent

Canada

EC6 & ACR1000

CANDU Energy

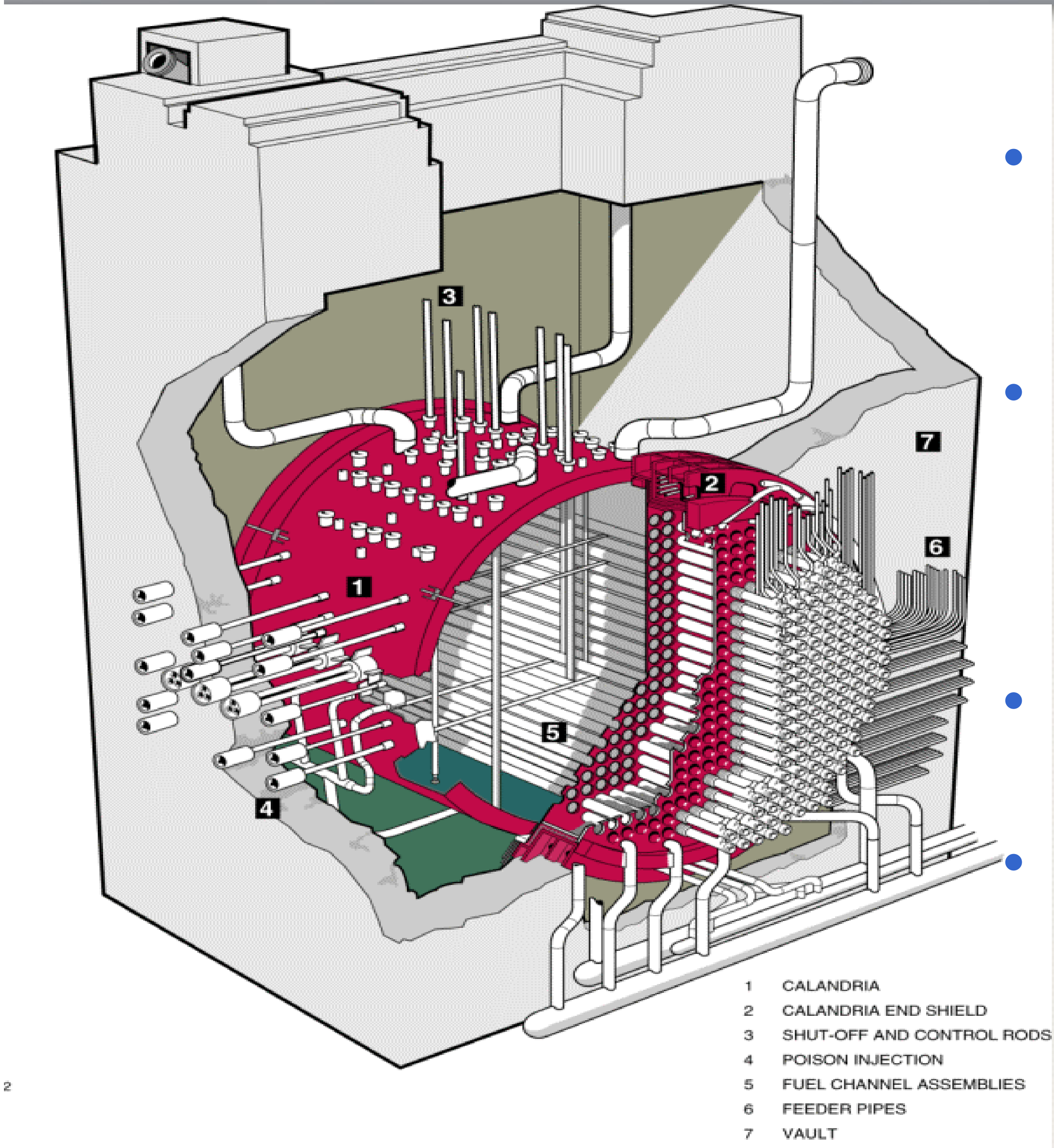
Flexibility of CANDU Design

- Natural uranium fuel → D_2O / heavy water cooled, necessary
- Slightly enriched 0.9~1.2%
- Can use recovered uranium
- Can use DUPIC (Direct Use of spent PWR fuel In CANDU)
- Can MOX (Mixed uranium and plutonium Oxide)
- Can use Pu/LWR waste-fuel
- Can use Thorium
- Can use CANFLEX bundle and LWR fuel bundles

IAEA

Reactor Assembly

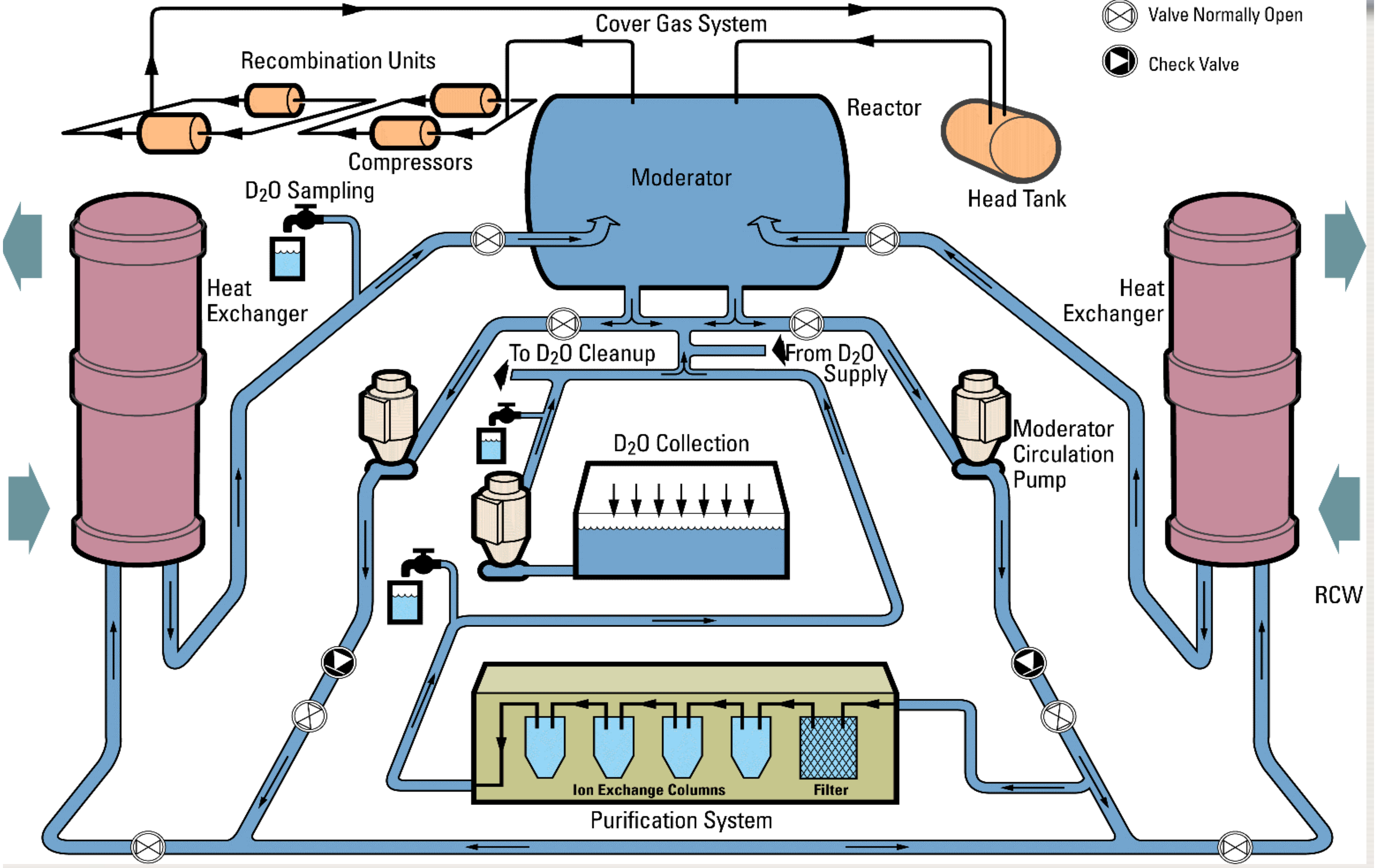
Courtesy of AECL, Canada



- Fuel channels and moderator inside the CALANDRIA
- CALANDRIA located inside concrete VAULT filled with water, also acts as SHIELDING.
- END-SHIELDS filled with water and steel pebbles.
- CALANDRIA has low temperature and pressure

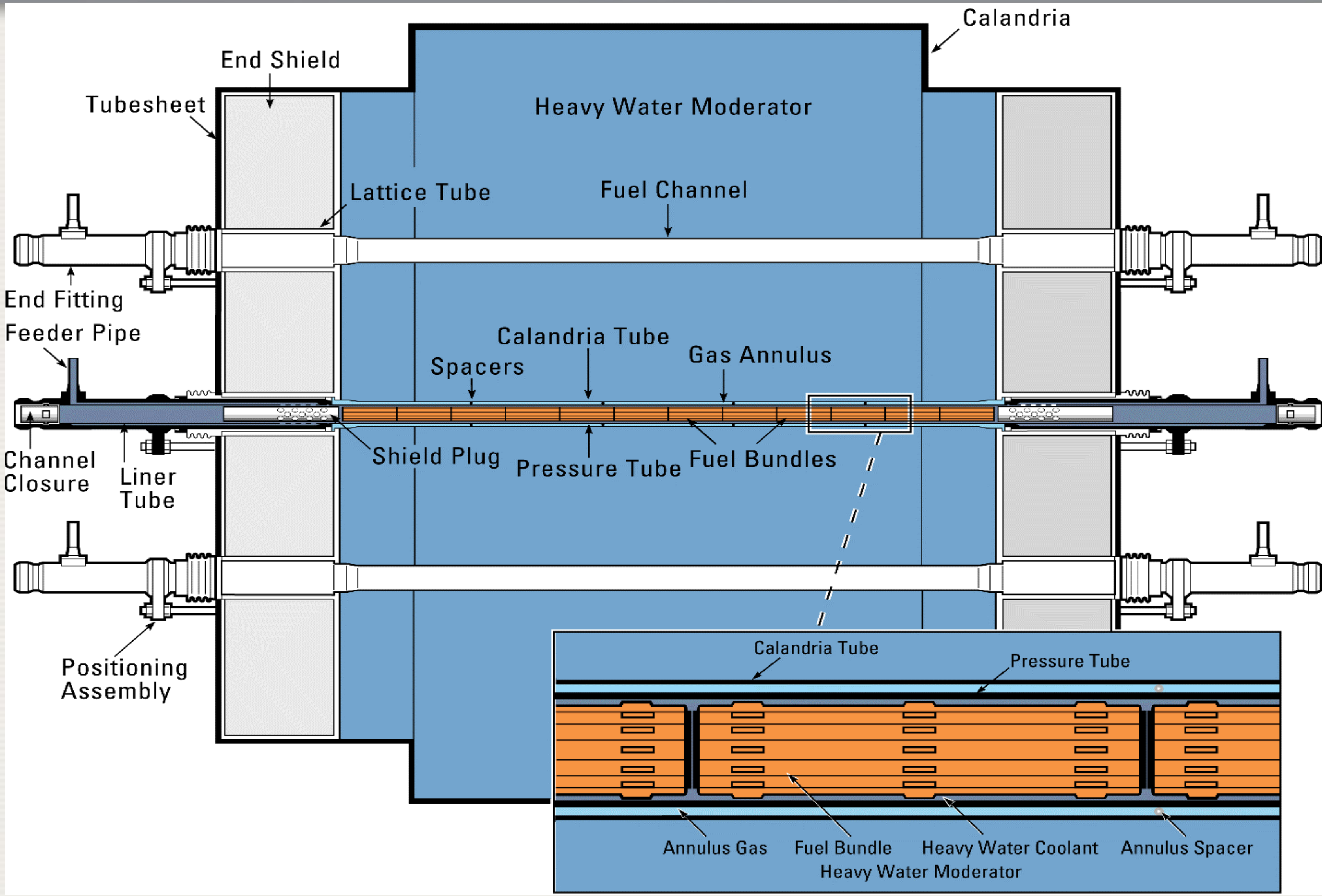
Moderator System

Courtesy of AECL, Canada



CALANDRIA & Fuel Channel Concept

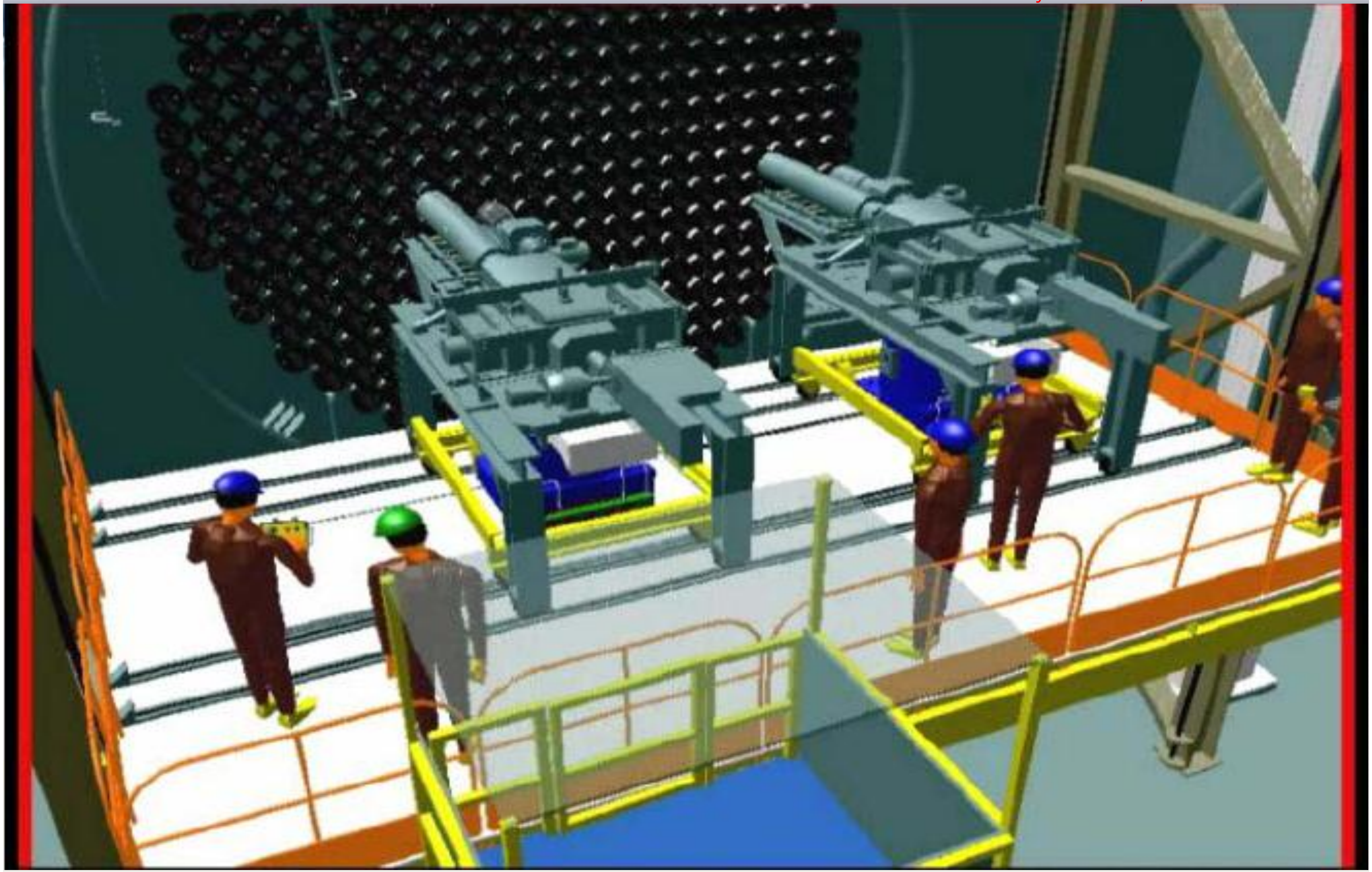
Courtesy of AECL, Canada



IAEA

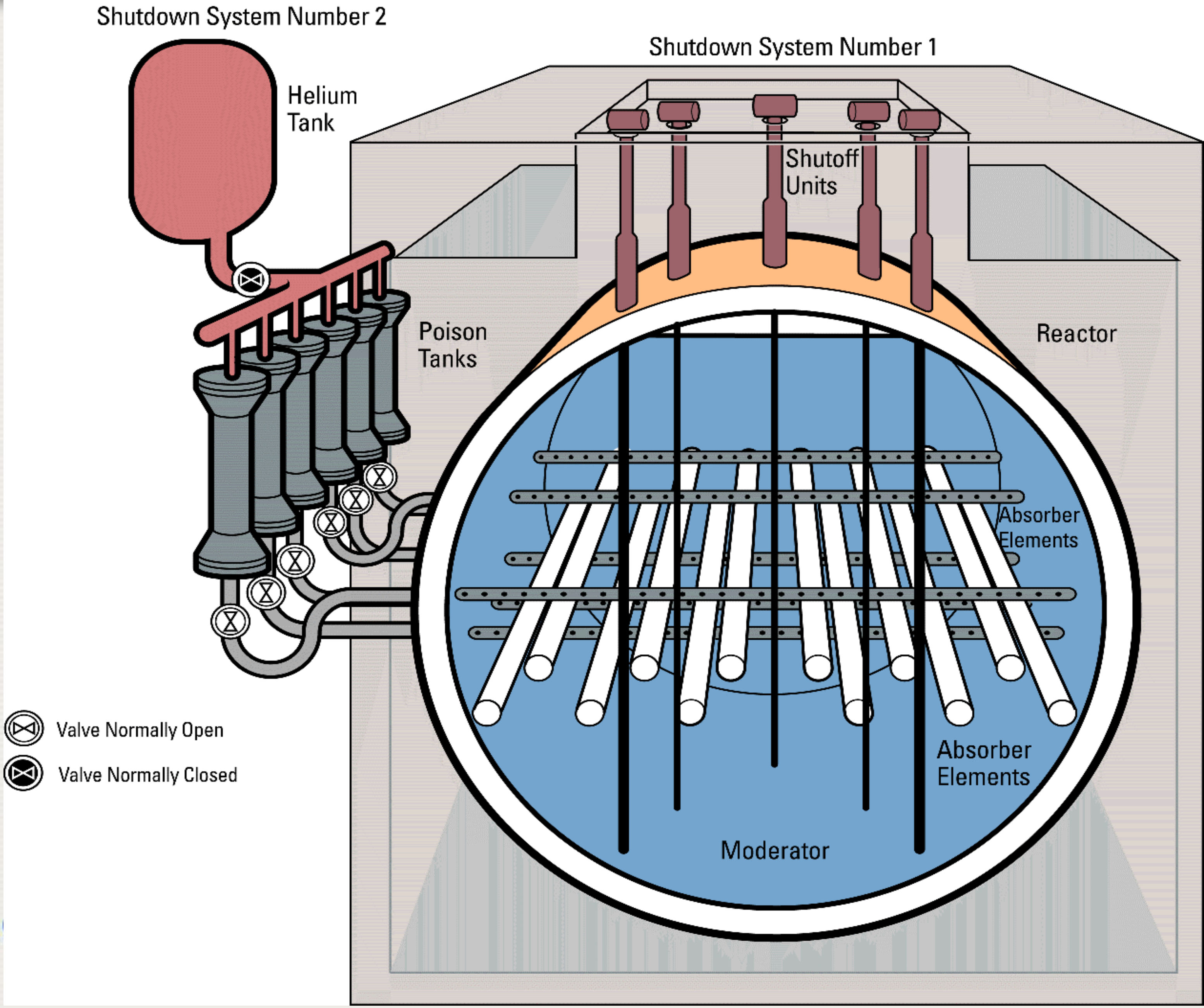
On-power Refueling

Courtesy of A E C L, Canada



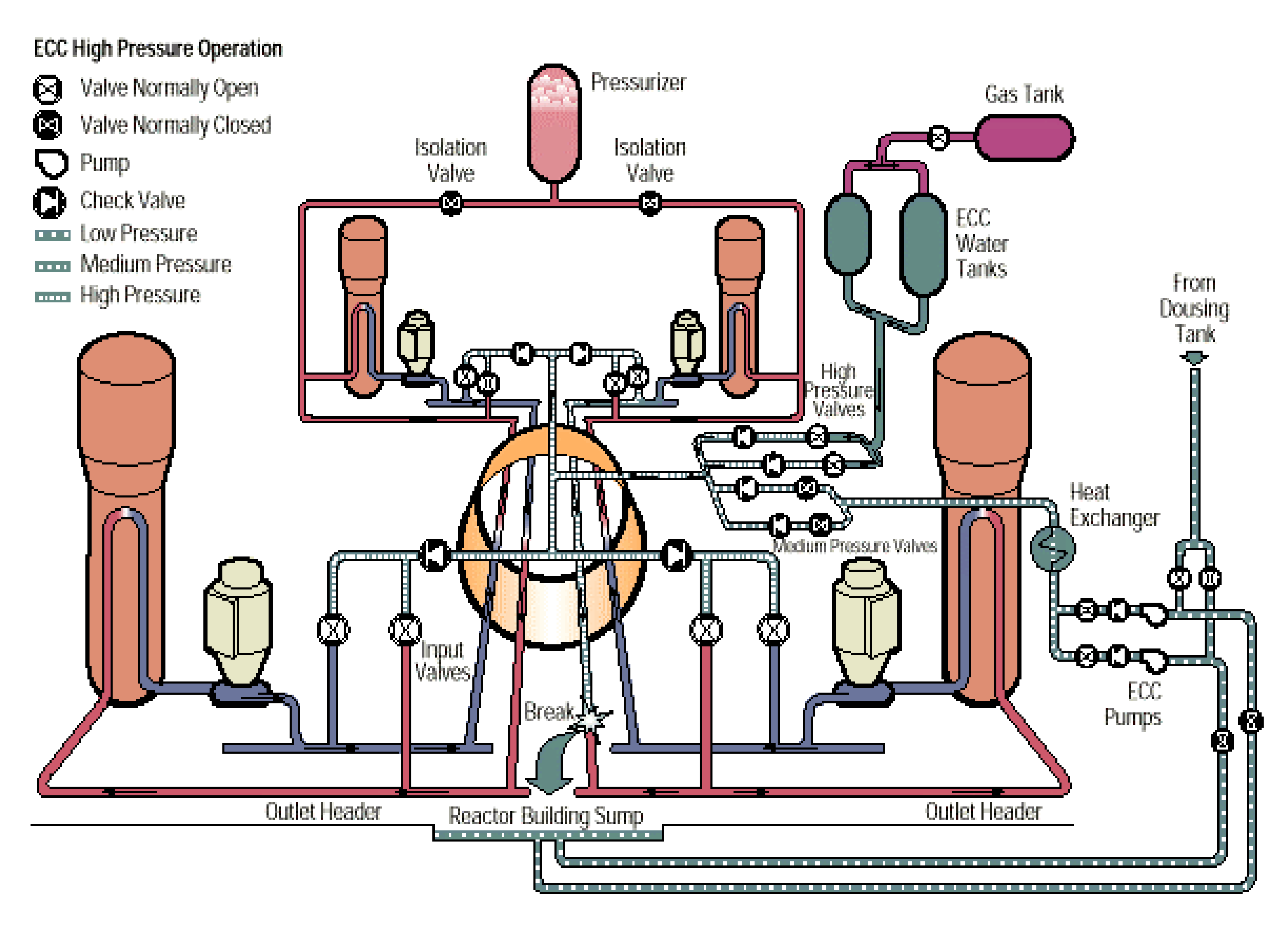
Shutdown Systems

Courtesy of AECL, Canada



Emergency Core Cooling System

Courtesy of A E C L, Canada



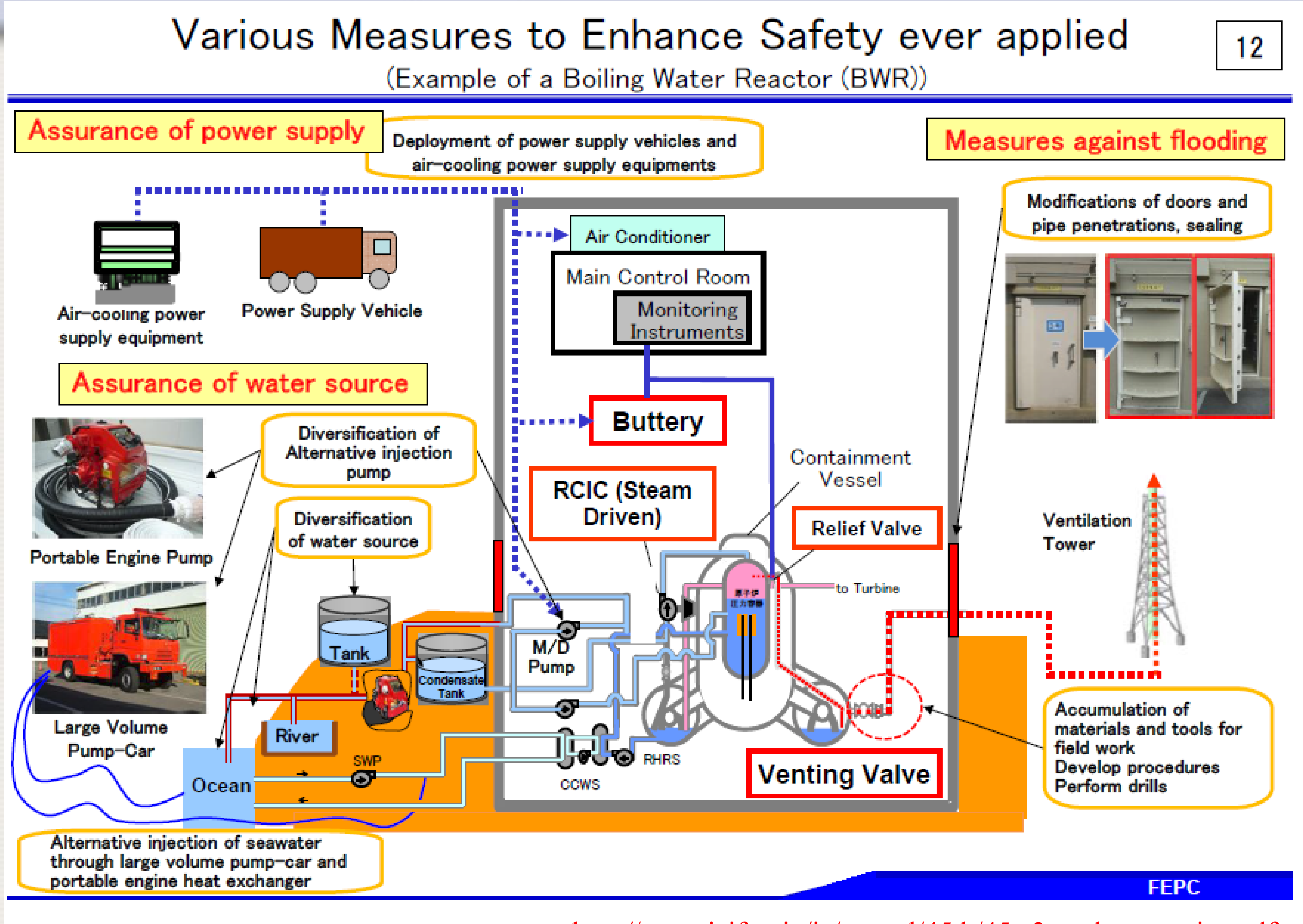
IAEA

Identified issues from the Fukushima Daiichi Nuclear Accident

- Review extensive scenario of Design Basis Accident (DBA) → Multiple external initiating events and common cause failures
- Station blackout mitigation
- Ultimate heat sink for core and containment cooling in post severe accident
- Reliability of emergency power supply
- Optimization of the grace period (i.e. operator coping time)
- Enhanced containment hydrodynamic strength
- Hybrid passive and active engineered safety features
- Safety viability of multiple-modules – first of a kind engineering
- Accident management, emergency response capability and costs
- Seismic and cooling provisions for spent fuel pool
- Hydrogen generation from steam-zirconium reaction; recombiner system
- Environmental impact assessment and expectation
- Control room habitability in post accident transient

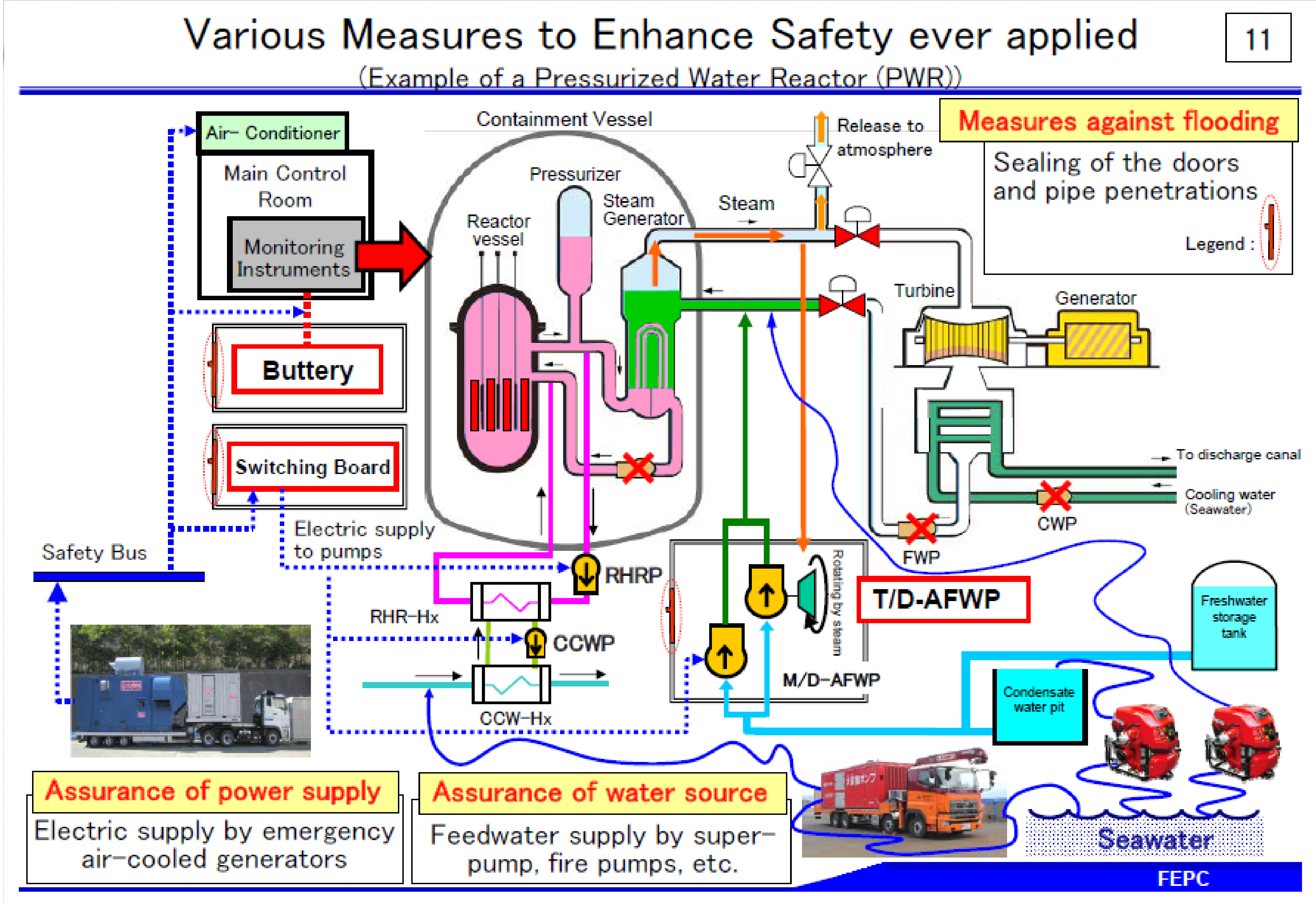
IAEA

Safety Improvement of Operating Reactors



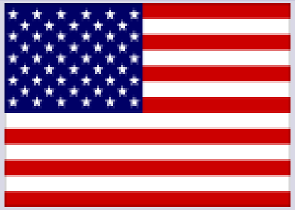
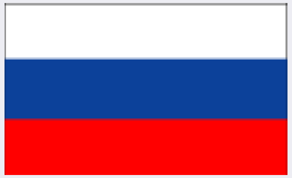
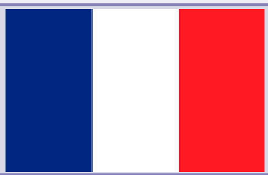
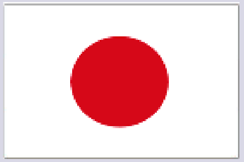
http://www.jaif.or.jp/ja/annual/45th/45-s2_makoto-yagi_e.pdf

Safety Improvement of Operating Reactors

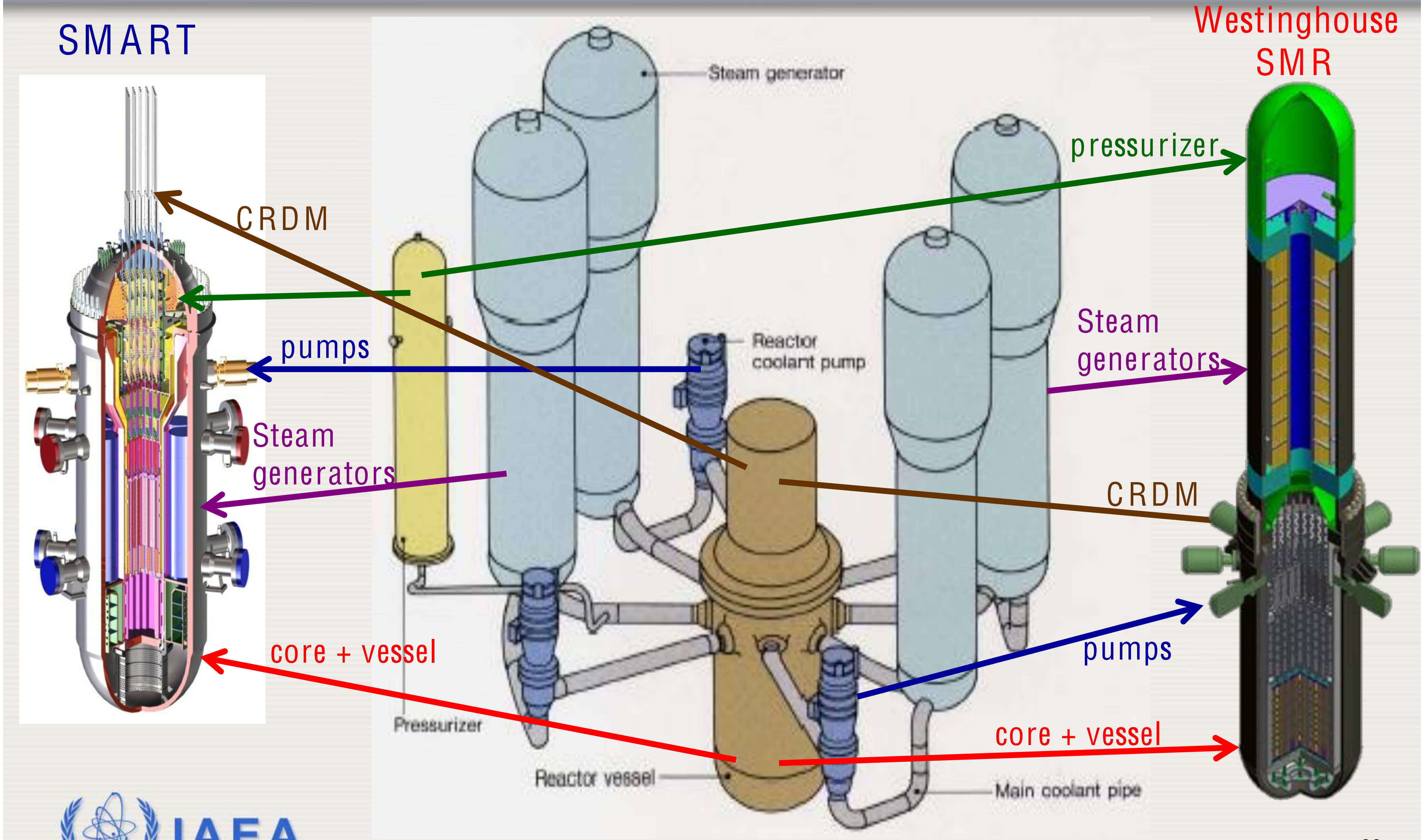


http://www.jaif.or.jp/ja/annual/45th/45-s2_makoto-yagi_e.pdf

What's New in Global SMR Development?

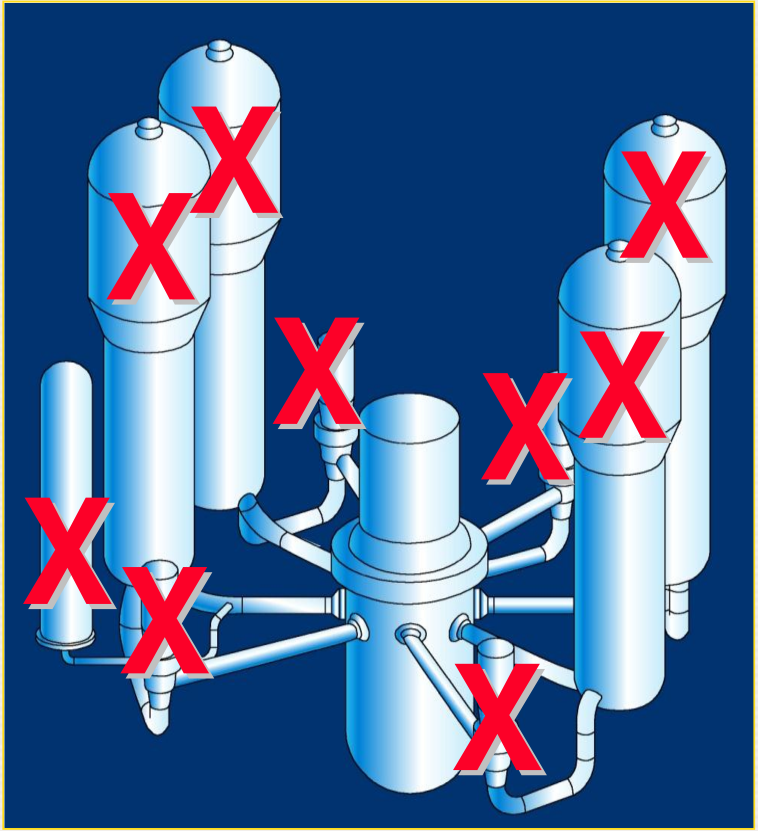
	**SMART**	On 4 July, the Korean Nuclear Safety and Security Commission issued the Standard Design Approval for the 100 MWe **SMART** – the first iPWR received certification.
	**NuScale** **mPower** **W-SMR** **Hi-SMUR**	US-DOE funding of 452M$/5 years for two (2) out of the four (4) US competing iPWR based SMRs. Some have utilities to adopt in specific sites
	**KLT-40s** **SVBR-100** **SHELF**	2 modules marine propulsion-based barge-mounted **KLT-40s** are in construction, 90%; The lead-bismuth eutectic cooled **SVBR-100** deployed by 2018, **SHELF** seabed-based started conceptual PWR-SMR design
	**Flexblue**	DCNS originated **Flexblue** capsule, 50-250 MWe, 60-100m seabed-moored, 5-15 km from the coast, off-shore and local control rooms
	**CAREM-25**	Site excavation for **CAREM-25** was started in September 2011, construction of a demo plan starts soon in 2012
	**4S**	Toshiba had promoted the **4S** for a design certification with the US NRC for application in Alaska and newcomer countries.
	**HTR-PM** **ACP-100**	2 modules of **HTR-PM** are under construction; CNNC developing **ACP-100** conceptual design

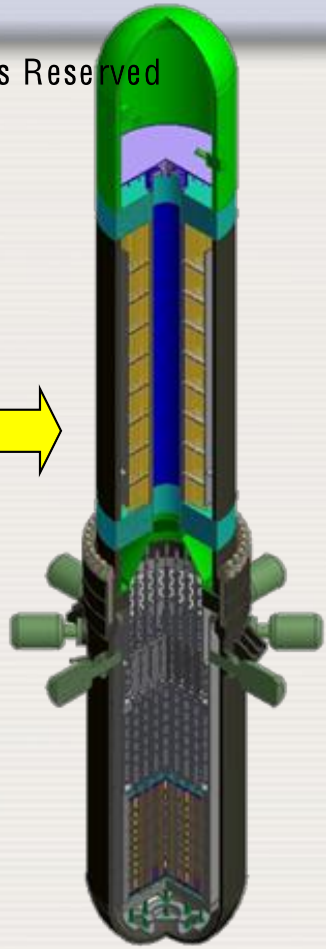
Concept of Integral PWR based SMRs

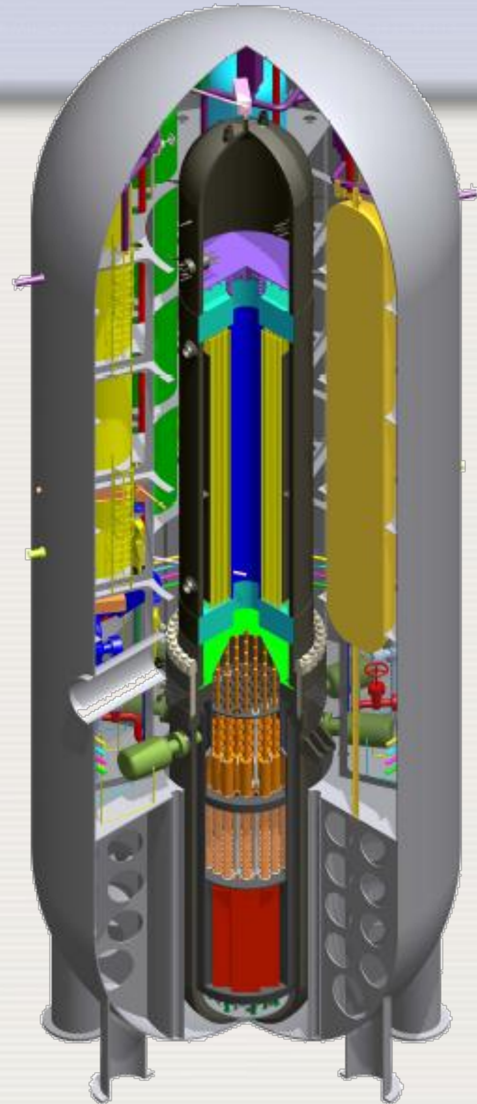


IAEA

Integral Primary System Configuration



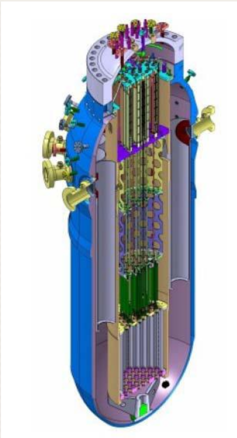




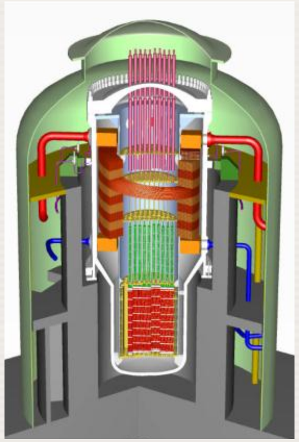
Benefits of integral vessel configuration:

- eliminates loop piping and external components, thus enabling compact containment and plant size → reduced cost
- Eliminates large break loss of coolant accident (improved safety)

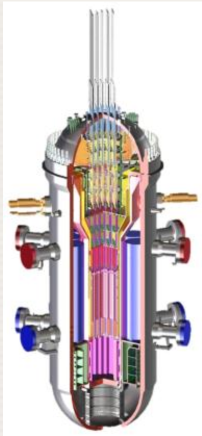
Light Water Cooled SMRs



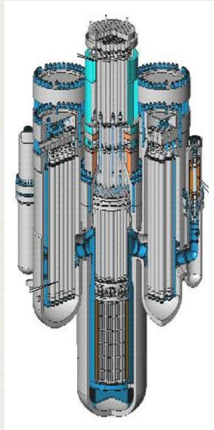
CAREM-25
Argentina



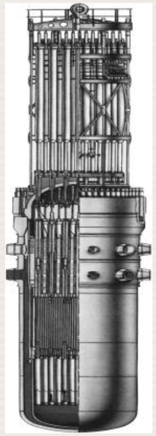
IMR
Japan



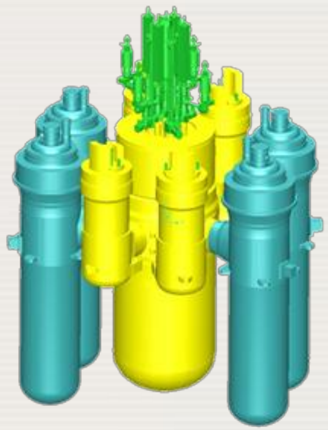
SMART
Korea, Republic of



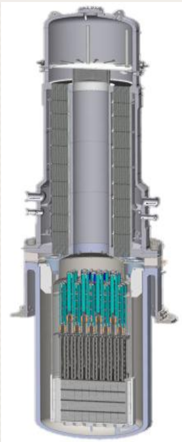
VBER-300
Russia



WWER-300
Russia



KLT-40s
Russia

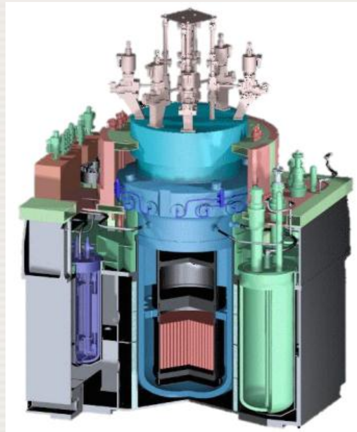


mPower
USA

NuScale
USA

Westinghouse
SMR - USA

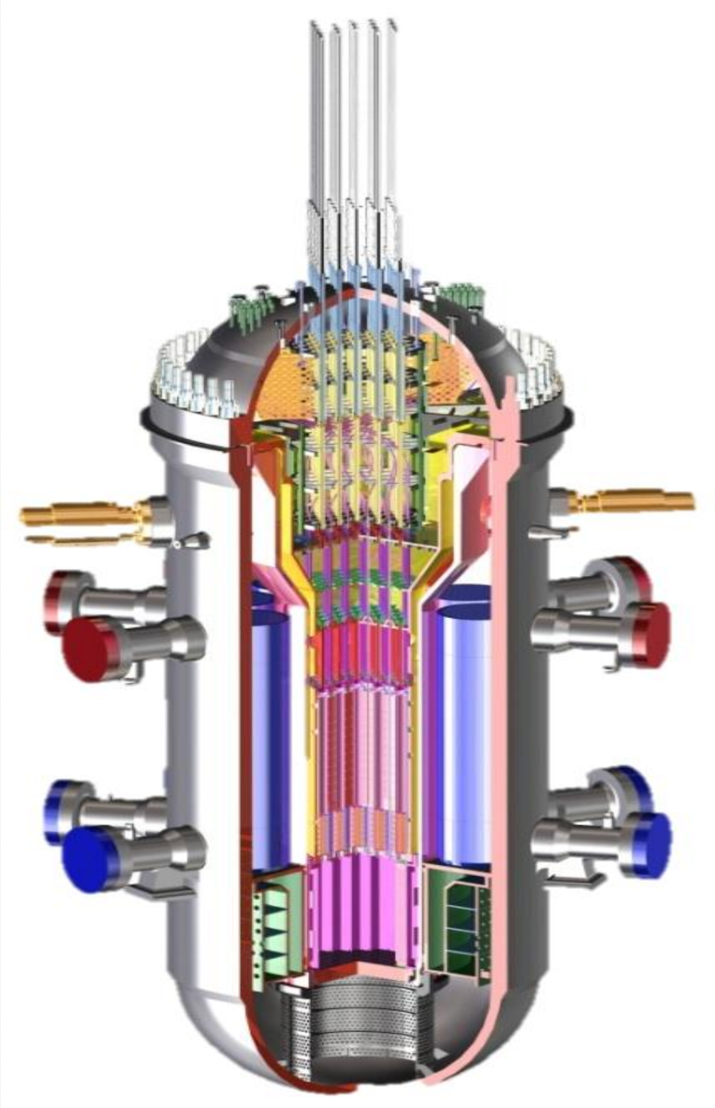
CNP-300
China, Peoples Republic of



ABV-6
Russia

IAEA

SMR for Near-term Deployment
SMART



- **Full name**: System-Integrated Modular Advanced Reactor
- **Designer**: Korea Atomic Energy Research Institute (KAERI), Republic of Korea
- **Reactor type**: Integral PWR
- **Coolant/Moderator**: Light Water
- **Neutron Spectrum**: Thermal Neutrons
- **Thermal/Electrical Capacity**: 330 MW(t) / 100 MW(e)
- **Fuel Cycle**: 36 months
- **Salient Features**: Passive decay heat removal system in the secondary side; horizontally mounted RCPs; intended for sea water desalination and electricity supply in newcomer countries with small grid
- **Design status**: Standard Design Approval just granted on 4 July 2012

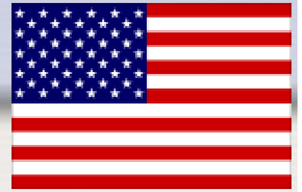
IAEA

SMART – Safety Systems

- **Inherent Safety**
  - No Large Break : vessel penetration < 2 inch
  - Large Primary Coolant Inventory per MW
  - Low Power Density (~2/3)
  - Large PZR Volume for Transient Mitigation
  - Low Vessel Fluence
  - Large Internal Cooling Source (Sump-integrated IRWST)
- **Engineered Safety Features**
  - Passive Residual Heat Removal System (50 % x 4 train)
    - Natural Circulation
    - Replenish-able Heat Sink (Emergency Cooling Tank)
  - Safety Injection System (100 % x 4 train)
    - Direct Vessel Injection from IRWST
  - Shutdown Cooling System (100 % x 2 Train)
  - Containment Spray System (2 Train)
- **Severe Accident Management**
  - In-Vessel Retention and ERVC
  - Passive Hydrogen Control (PARs)

IAEA

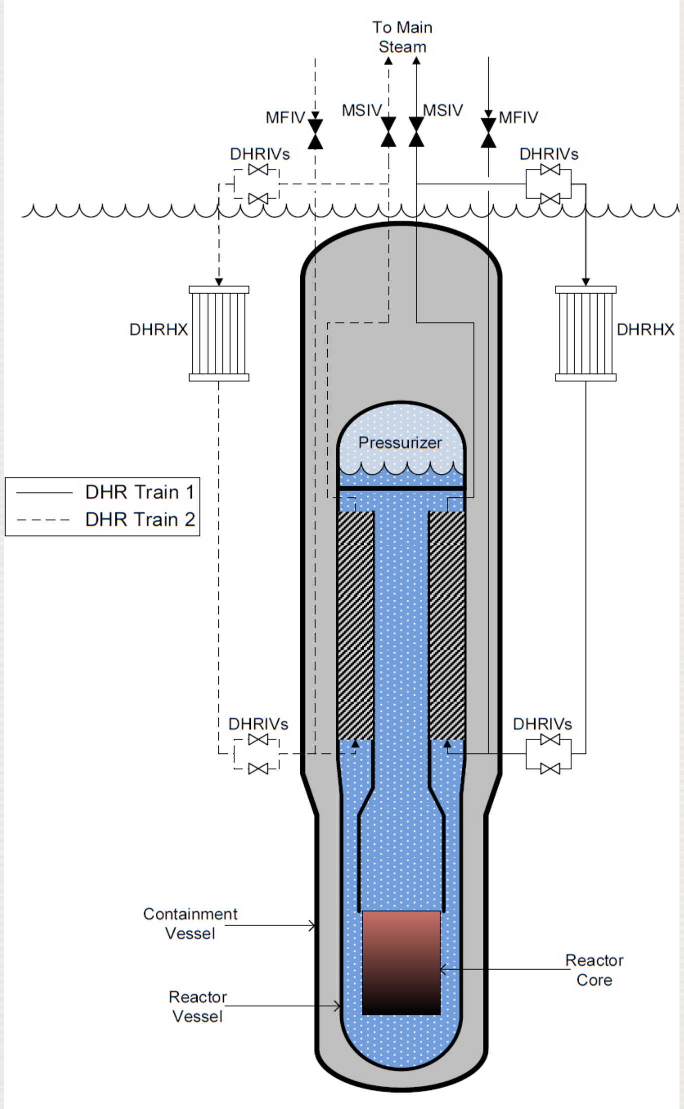
SMR for Near-term Deployment NuScale



- **Full name**: NuScale
- **Designer**: NuScale Power Inc., USA
- **Reactor type**: Integral Pressurized Water Reactor
- **Coolant/Moderator**: Light Water
- **Neutron Spectrum**: Thermal Neutrons
- **Thermal/Electrical Capacity**: 165 MW(t)/45 MW(e)
- **Fuel Cycle**: 24 months
- **Salient Features**: Natural circulation cooled; Decay heat removal using containment; built below ground
- **Design status**: Design Certification application expected in 4th Quarter of 2013

IAEA

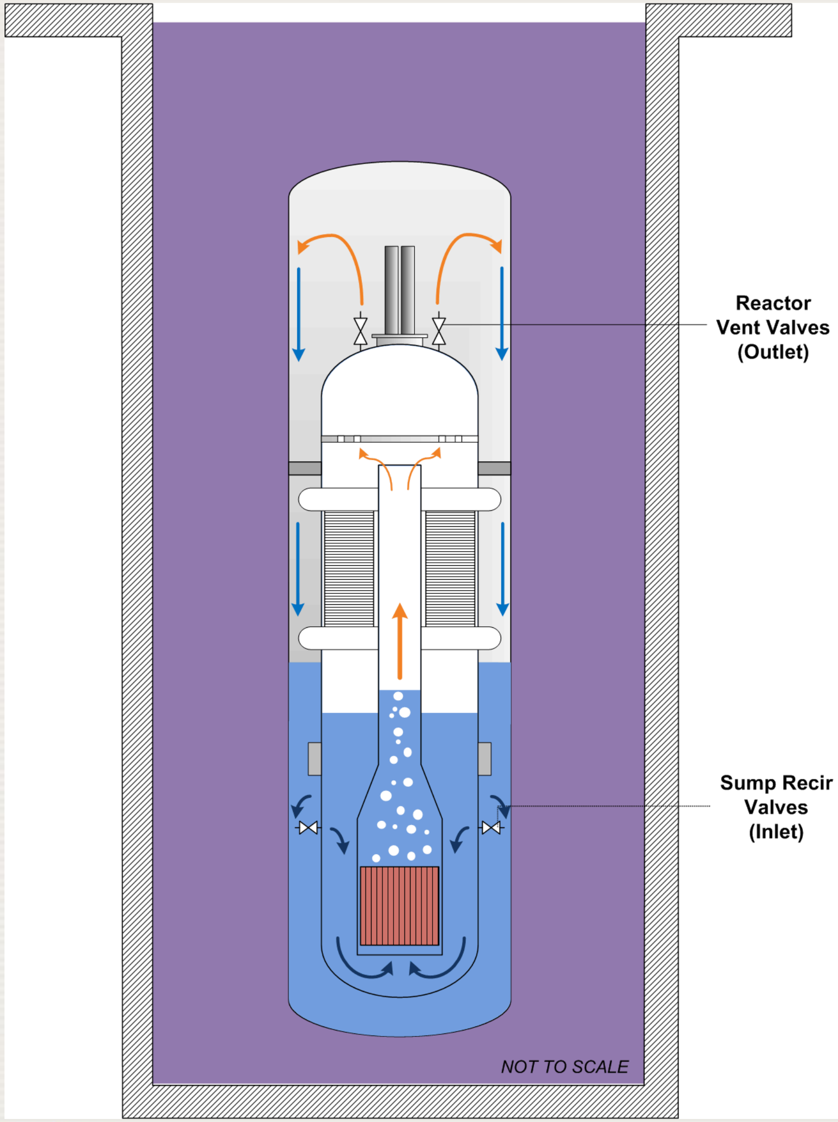
NuScale - Decay Heat Removal System



- Two independent single-failure-proof trains
- Closed loop system
- Two-phase natural circulation operation
- DHRS heat exchangers nominally full of water
- Supplies the coolant inventory
- Primary coolant natural circulation is maintained
- Pool provides a 3 day cooling supply for decay heat removal

IAEA

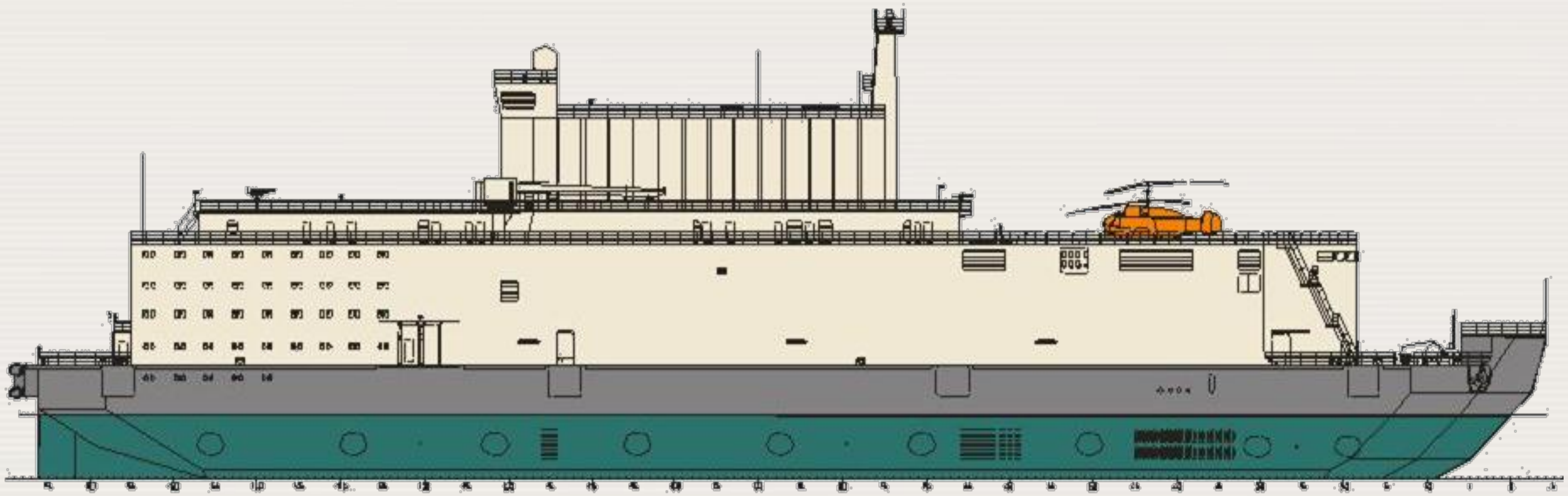
NuScale - Decay heat removal using Containment



- Provides a means of removing core decay heat and limits containment pressure by:
  - Steam Condensation
  - Convective Heat Transfer
  - Heat Conduction
  - Sump Recirculation
- Reactor Vessel steam is vented through the reactor vent valves (flow limiter)
- Steam condenses on containment
- Condensate collects in lower containment region
- Reactor Recirculation Valves open to provide recirculation path through the core
- Provides 30+ day cooling followed by indefinite period of air cooling.

IAEA

Main Engineering Characteristics of KLT-40s FNPP



TYPE - SMOOTH-DECK NON-SELF-PROPELLED SHIP

LENGTH, m	140,0
WIDTH, m	30,0
BOARD HEIGHT, m	10,0
DRAUGHT, m	5,6
DISPLACEMENT, t	21 000
FPU SERVICE LIFE, YEARS	40

IAEA

IAEA Response to the Global Trend

- **Project 1.1.5.5:**
  Common Technologies and Issues for SMRs
- **Objective:** To facilitate the development of key enabling technologies and the resolution of enabling infrastructure issues common to future SMRs
- **Activities (2012 – 2013):**
  - Formulate roadmap for technology development incorporating safety lessons-learned from the Fukushima accident
  - Review newcomer countries requirements, regulatory infrastructure and business issues
  - Define operability-performance, maintainability and constructability indicators
  - Develop guidance to facilitate countries with planning for SMRs technology implementation

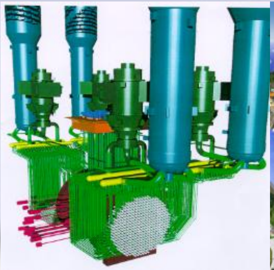
Summary and Conclusion (1)

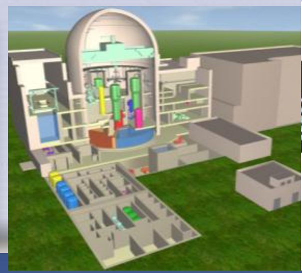
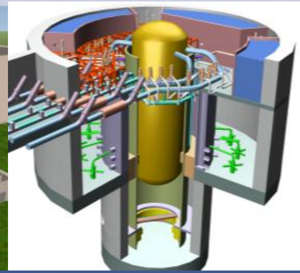
- Many options are realistic choice for countries
  - *Heavy water, boiling water and pressurized water reactors*
  - *SMR, high temperature or fast reactors have niche applications*
- Many of them have been:
  - *Endorsed by User Requirements (EUR, URD, etc)*
  - *Certified by licensing authorities in several countries*
  - *Built and operated for many years in various countries… or*
  - *… in the process of being built*
- Size and “provenness” requirements may limit choices for newcomer countries
- All options have multiple sources for lifetime spare parts
- Some options are better than others for fuel supply

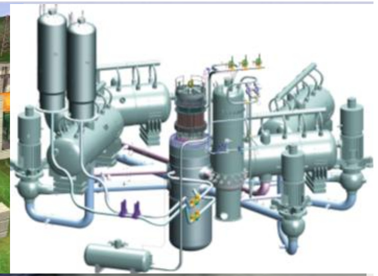
IAEA

Summary and Conclusion (2)

- Capacity building in newcomer countries is crucial, in which IAEA plays important and neutral roles
- Main issues:
  - Economic competitiveness
  - Proven technology or commercial availability
  - Some suppliers design for home country (SMR)
  - Capability of newcomer countries to assess supplier claims
- Needs to address lessons-learned from the Fukushima accident into the design development and plant deployment
- Needs to enhance the important roles of research reactor utilization for capacity building in newcomer countries embarking on nuclear power programs, through inter-regional training and expanding nuclear engineering curriculum through IAEA assistance and facilitation.





… Thank you for your attention.

IAEA

For inquiries, please contact:

Dr. M. Hadid Subki <M.Subki@iaea.org>