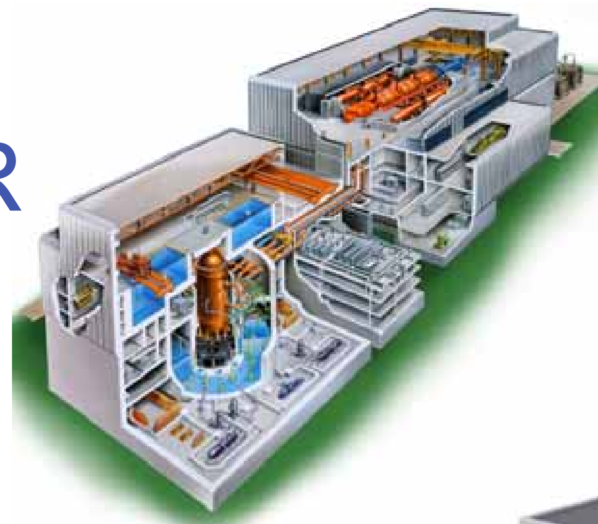
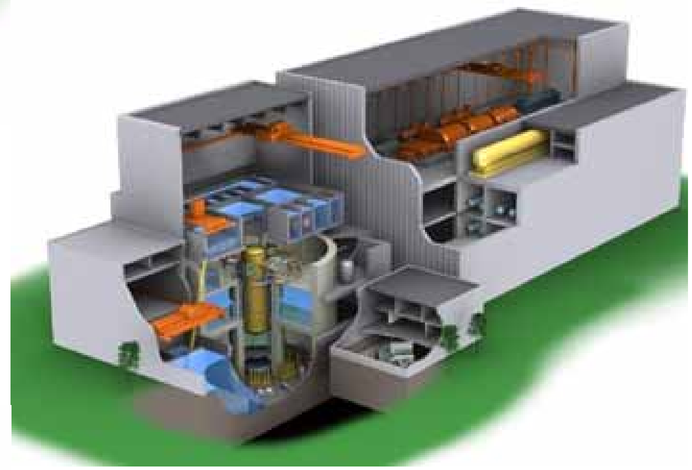
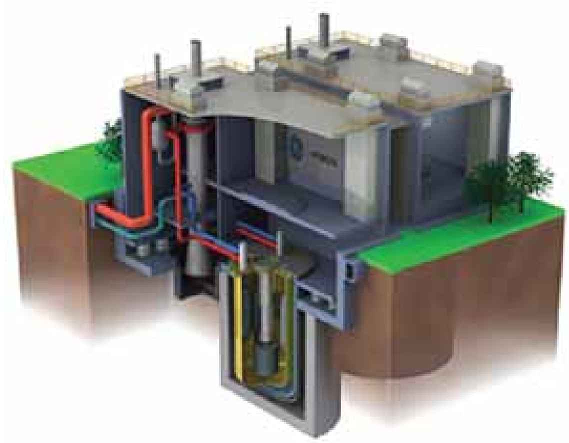
ABWR, ESBWR and PRISM





Craig Sawyer
Chuck Boardman
March 26, 2015



Current Nuclear Products of GE-Hitachi Nuclear Energy (GEH)

- Advanced Boiling Water Reactor (ABWR)
  - Developed jointly by GE, Hitachi and Toshiba with support of Japanese BWR utilities
  - First Generation III reactor
- Economic Simplified Boiling Water Reactor (ESBWR)
  - Developed by GEH with support of European and US utilities and the Dept. of Energy (DoE)
  - A Generation III+ reactor
- Power Reactor Innovative Small Module (PRISM)
  - Development support from the DoE
  - A Generation IV reactor

BWR Overview

- Operates under saturated conditions
  - Over 50 years of operational experience
  - Operating Pressure is nominally 7.2 MPa (1040 psia) with saturation temperature ~ 287 ºC (550 ºF)
  - Direct Cycle
    - Saturated Steam
    - Quality at exit is greater than 99.9%
  - Evolution

Comparison of GE designed BWRs

Parameter	BWR/4-Mk I (Browns Ferry 3)	BWR/6-Mk III (Grand Gulf)	ABWR	ESBWR
Power (MWt / MWe)	3293*/1098	3900/1360	3926/1365	4500/1590
Vessel height / diameter (m)	21.9/6.4	21.8/6.4	21.1/7.1	27.7/7.1
Fuel Bundles (number)	764	800	872	1132
Active Fuel height (m)	3.7	3.7	3.7	3.0
Power density (kw/l)	50	54.2	51	54
Recirculation pumps	2 (large external)	2 (large external)	10 (RPV mounted)	zero
Number of CRDs / type	185/LP	193/LP	205/FM	269/FM
Safety system pumps	9	9	18	zero
Safety Diesel Generator	2	3	3	zero†
Core damage freq./yr	1E-5	1E-6	1E-7	1E-8
Safety Bldg Vol (m^3/MWe)	120	170	180	135

* Before power uprate
† ESBWR has 2 non-safety diesel generators

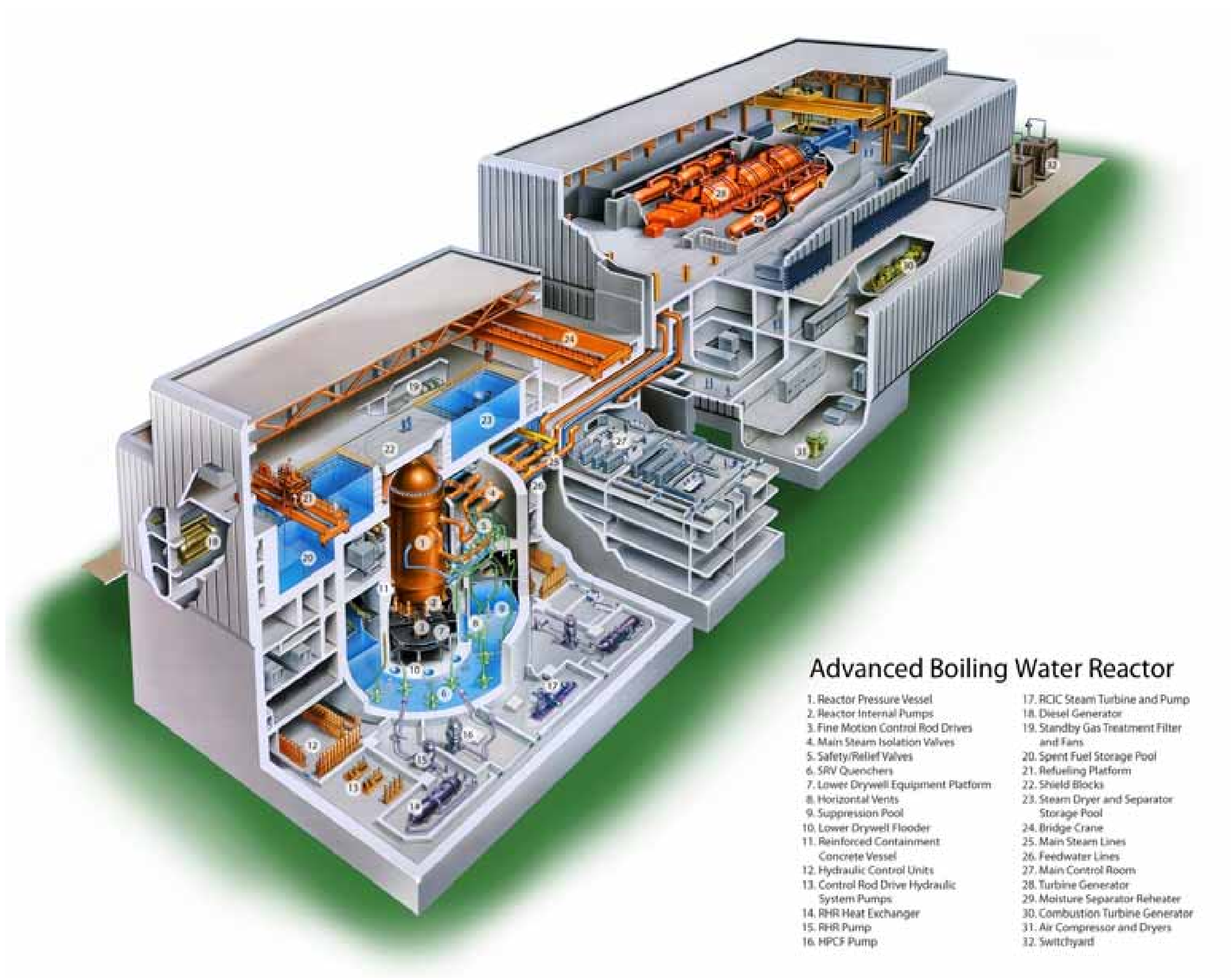
ABWR Standard Plant Basic Parameters

- 3,926 Megawatt Core Thermal Power*
  - ~1,365 Megawatt electric gross
  - Nominal summer conditions
- Reactor Internal Recirculation Pumps (RIP)
  - No recirculation piping
  - Canned rotor pumps
- 3 Divisions of Active Safety Systems (ECCS)
  - High pressure and low pressure pumps in each division
  - Core and/or containment heat removal in each division
  - At least 72 hours operators hands-off capability, limited by size of diesel fuel tanks and ultimate heat sink

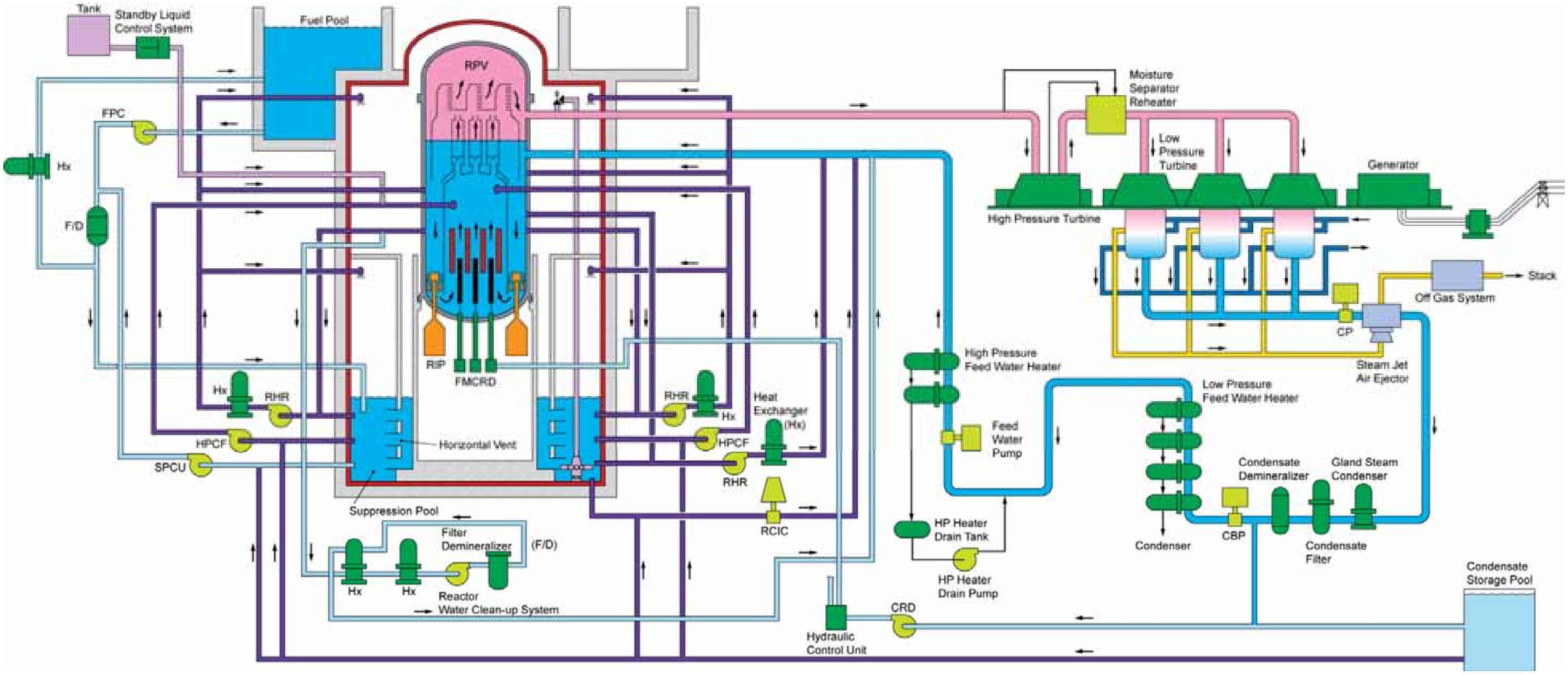
---

* Power upratable to 4300 MWt

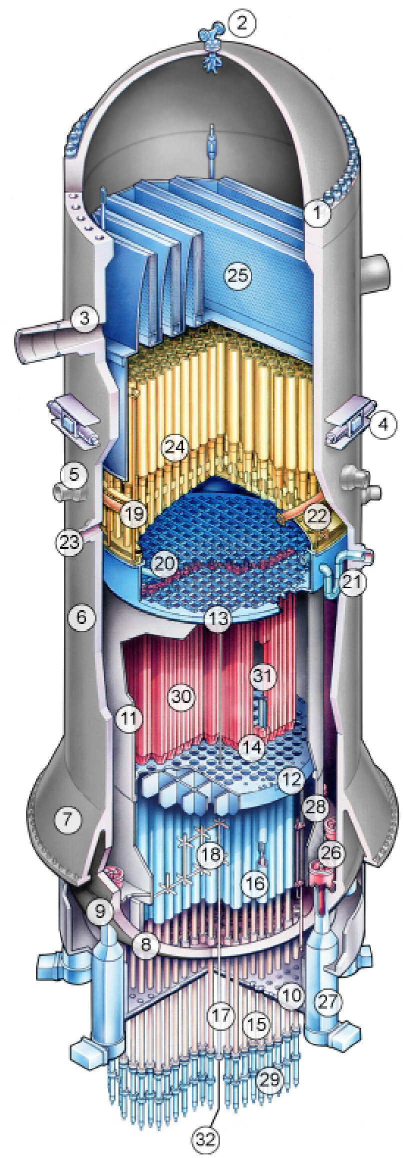
ABWR 3D Cutaway



ABWR Overall Flowchart

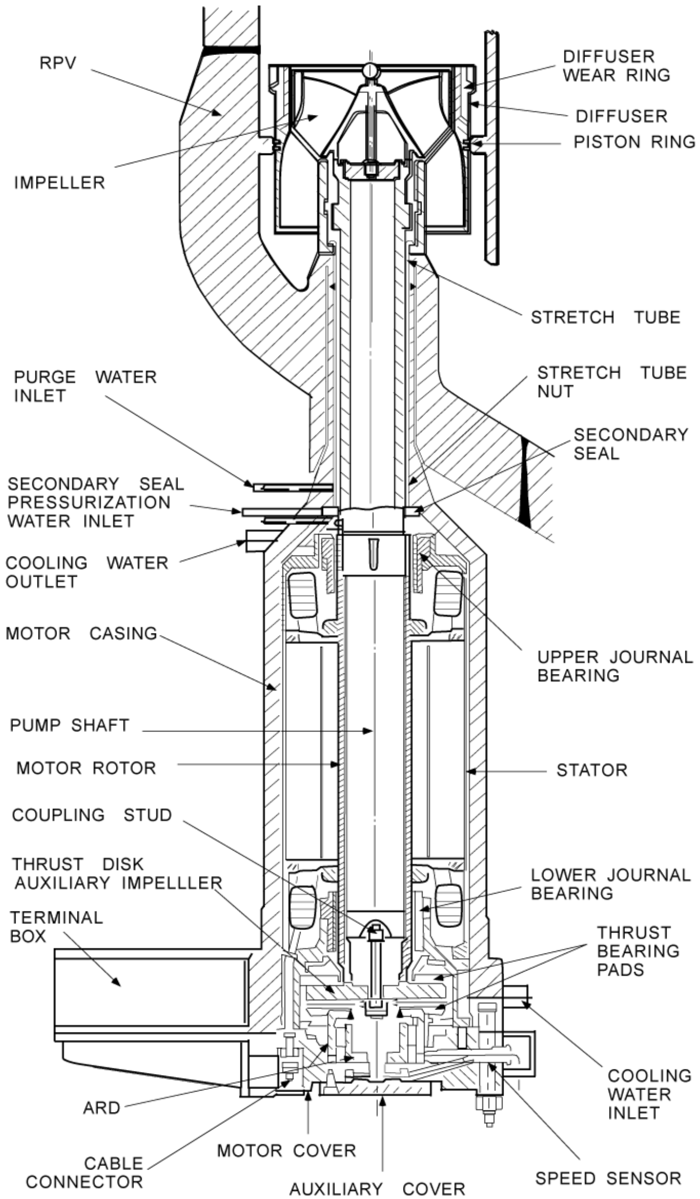


ABWR RPV Assembly



1 - Vessel flange and closure head
2 - Vent and head spray assembly
3 - Steam outlet flow restrictor
4 - RPV stabilizer
5 - Feedwater nozzle
6 - Forged shell rings
7 - Vessel support skirt
8 - Vessel bottom head
9 - RIP penetrations
10 - Thermal insulation
11 - Core shroud
12 - Core plate
13 - Top guide
14 - Fuel supports
15 - Control rod drive housings
16 - Control rod guide tubes
17 - In-core housing
18 - In-core guide tubes and stabilizers
19 - Feedwater sparger
20 - High pressure core flooder (HPCF) sparger
21 - HPCF coupling
22 - Low pressure flooder (LPFL)
23 - Shutdown cooling outlet
24 - Shroud head and steam separator assembly
25 - Steam dryer assembly
26 - Reactor internal pumps (RIP)
27 - RIP motor casing
28 - Core and RIP differential pressure line
29 - Fine motion control rod drives
30 - Fuel assemblies
31 - Control rods
32 - Local power range monitor

Reactor Internal Pump (RIP)



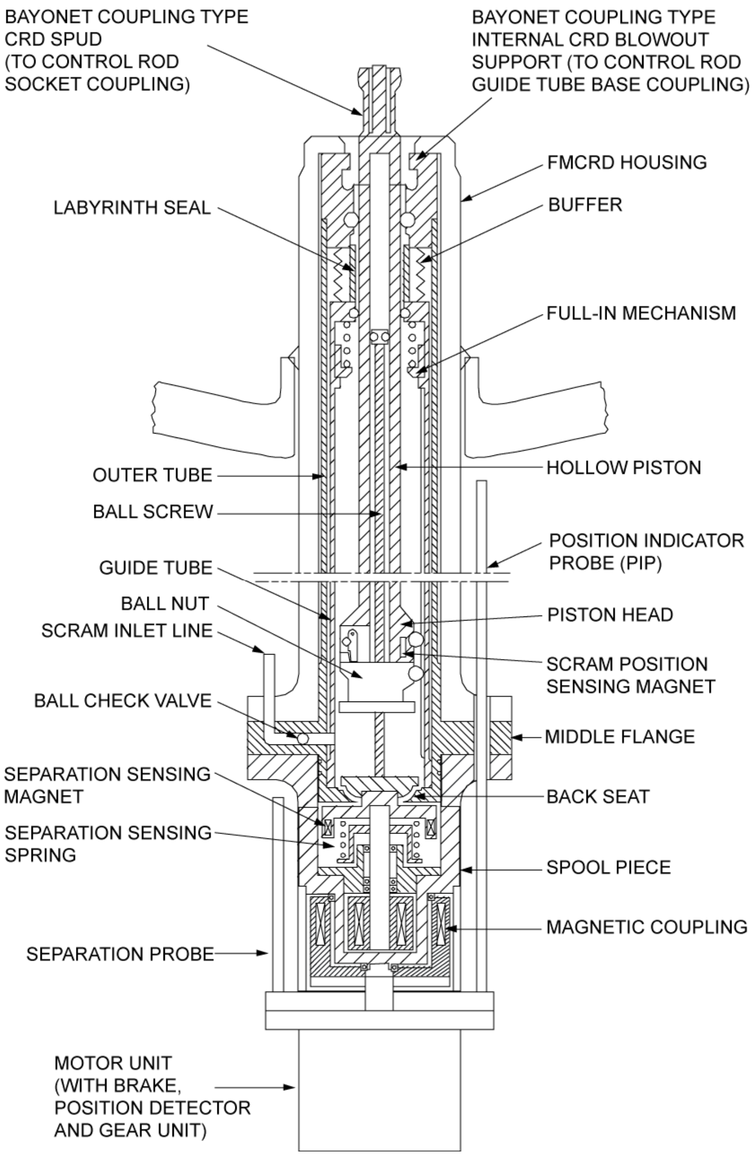
Key design features:

- Ten pumps can provide up to 110% flow
  - Full flow with one pump out of service
- Wet induction motor, seal-less design
- Continuous purge with clean water
- Impellers and motors removable without reactor draining
- Back seating shaft and blowout restraint hangers provide redundant LOCA prevention
- Solid-state, adjustable frequency speed control
- Multiple power supplies reduces probability of significant flow loss

Key benefits:

- Eliminates the external recirculation loops
  - Compact containment design
  - No large nozzles below core
    - safer/ECCS optimized
  - Reduced In-service inspections
    - less occupational exposure
- Less pumping power required
- Flexible operation

Fine Motion Control Rod Drive (FMCRD)



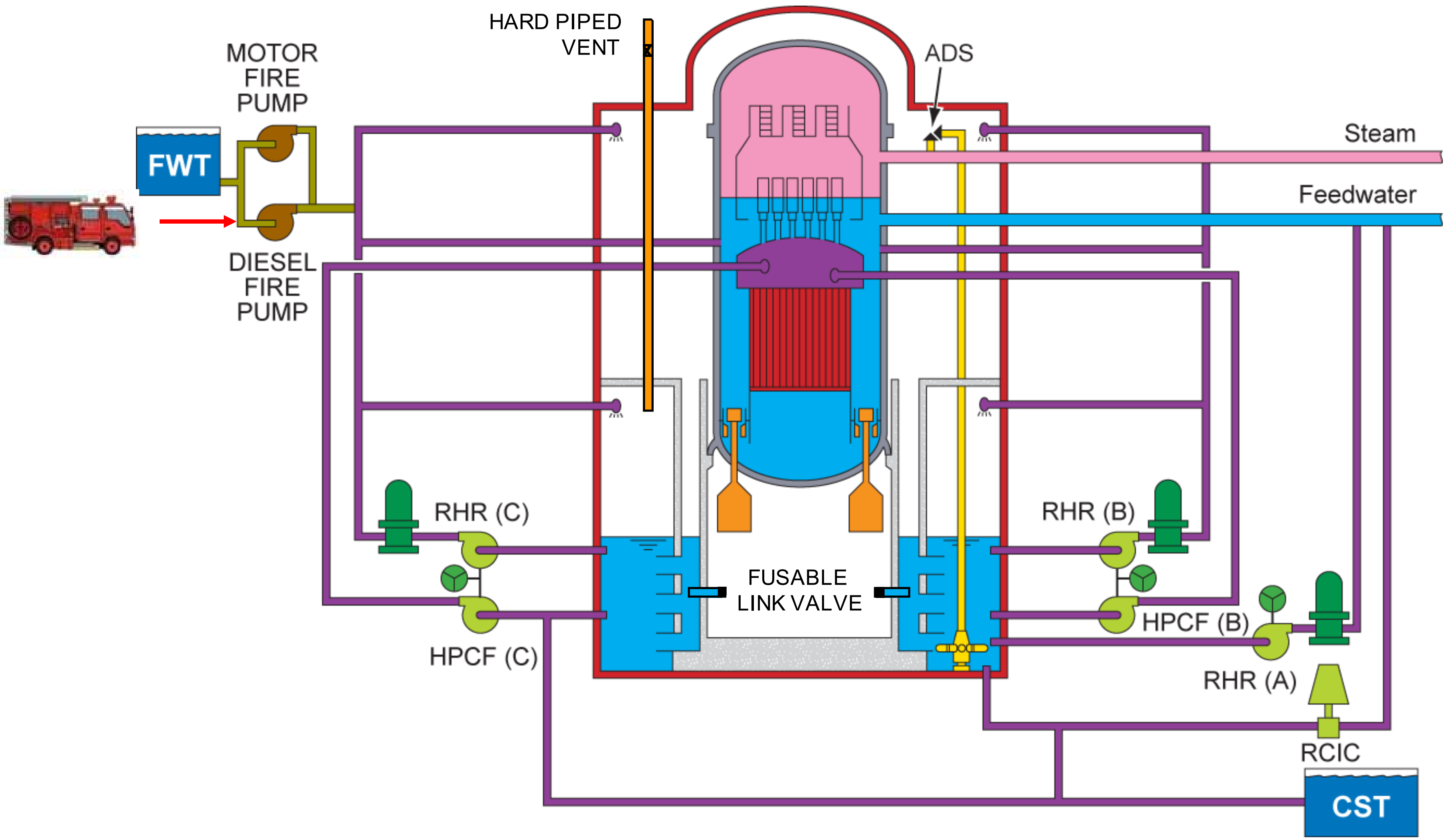
Key Design Features

- Electro-hydraulic design
- Two drive units per Hydraulic Control Unit (HCU)
- No scram discharge volume
- Ganging of up to 26 rods is possible
- Internal restraint to prevent blowout (No external restraints required)
- Clean water purge flow
- Capability to detect drive/blade separation
- Electro-mechanical brake to prevent rod runout on pressure boundary failure

Key benefits:

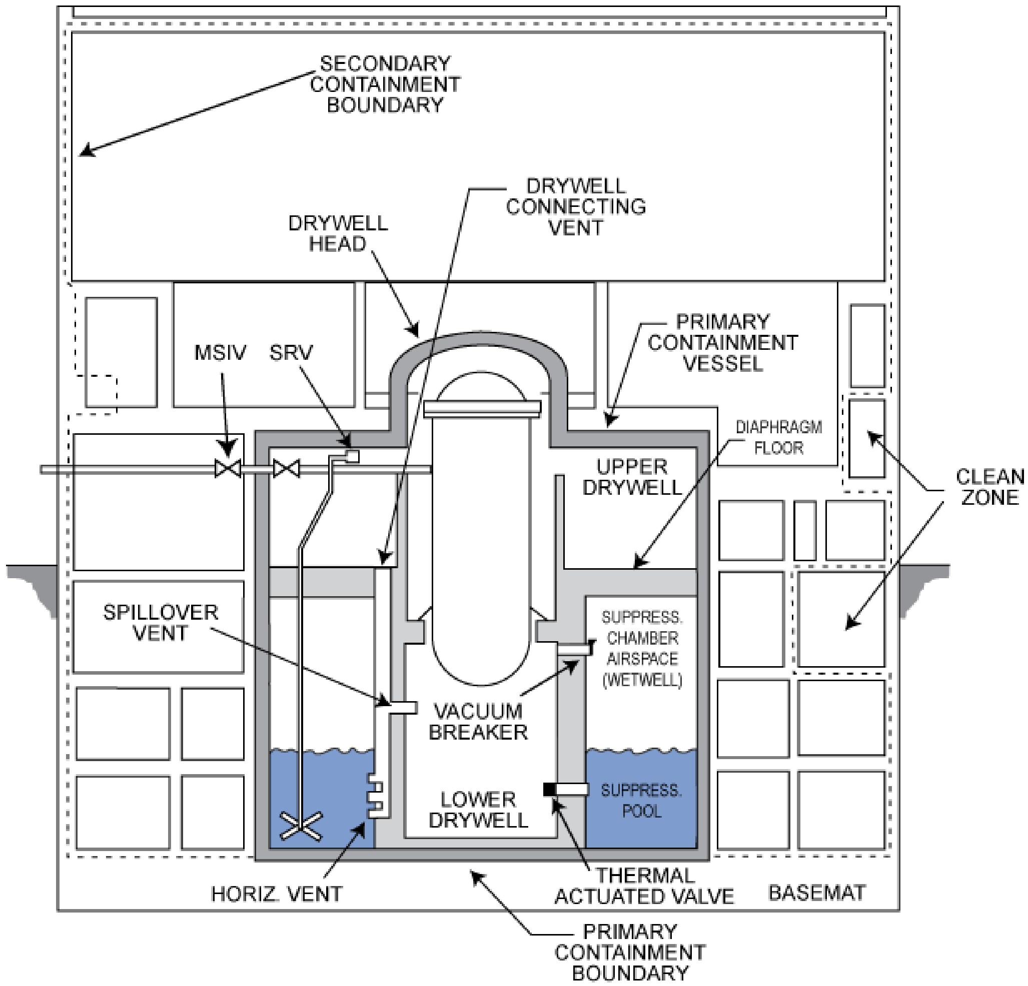
- Provides fine rod motion during normal operation
  - Small power changes
- Improved startup time and power maneuvering
- Diverse shutdown capability
  - Hydraulic with electrical backup
- Reactivity accidents eliminated
  - Rod drop and rod ejection
- Inspections reduced from 30 to 2 drives per outage
  - less occupational exposure

ABWR ECCS

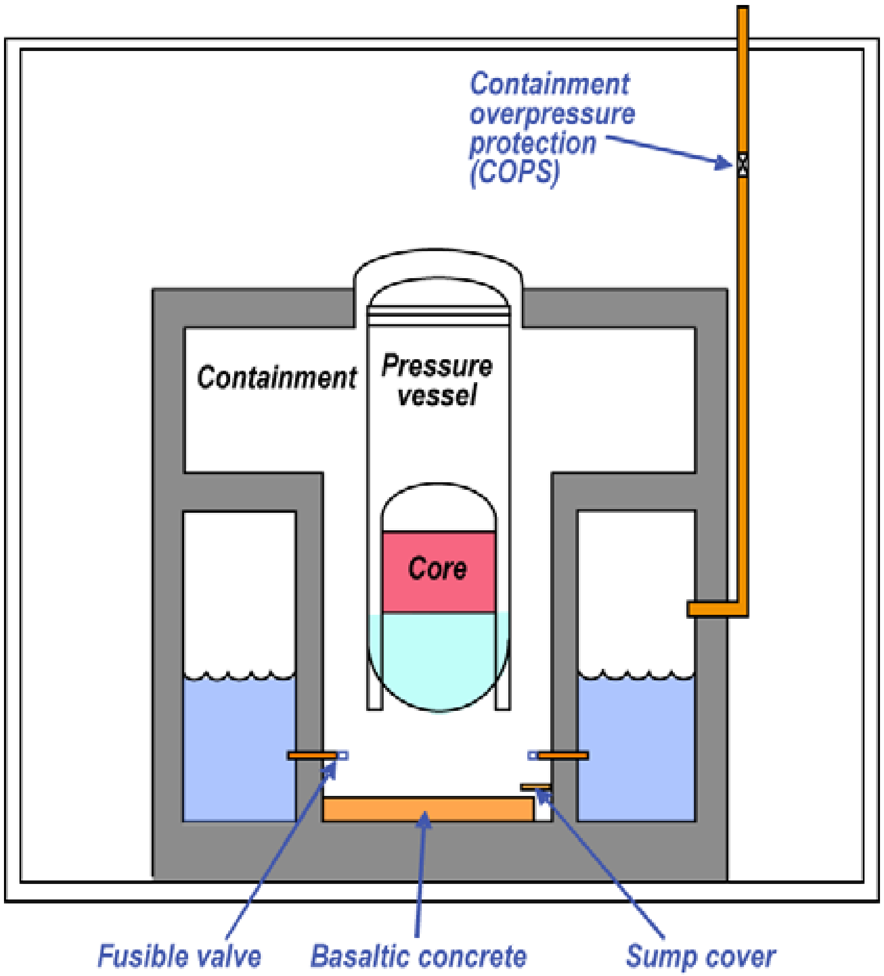


HARD PIPED
VENT
MOTOR
FIRE
PUMP
FWT
DIESEL
FIRE
PUMP
ADS
Steam
Feedwater
RHR (C)
HPCF (C)
FUSABLE
LINK VALVE
RHR (B)
HPCF (B)
RHR (A)
RCIC
CST

ABWR Reactor Building & Containment



ABWR Severe Accident Mitigation Features



ABWR passive features which mitigate severe accidents:

- Inerted Containment
- Lower Drywell flood capability
- Lower Drywell special concrete & sump protection
- Suppression pool - fission products scrubbing & retention
- Containment overpressure protection via rupture disks

Operator Options

- AC Independent water addition via fire pumps

ABWR Summary

- First Generation III reactor
- First Commercial Operation in 1996
- Design Certified in USA in 1997
- Four Units Built and Operating in Japan*
- One Unit Under Construction in Japan
- Two Units Under Construction in Taiwan
- Two Units Nearing Construction-Operating License (COL) Submission in USA

Lungmen 1 & 2
Under Construction

Kashiwazaki-Kariwa 6 & 7

Hamaoka 5

Shika 2

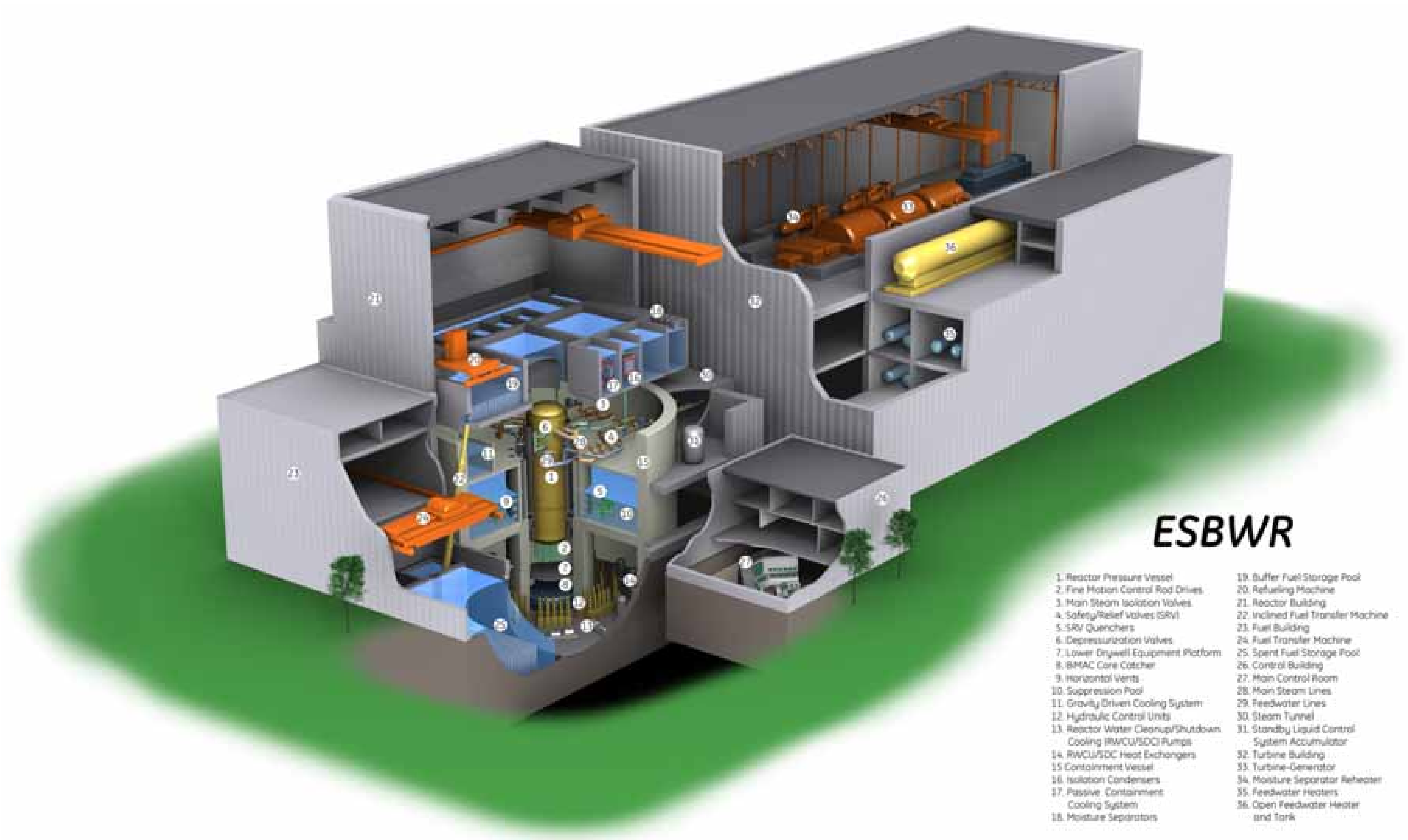
Shimane 3
Under Construction

* Currently shut down for post-Fukushima safety review

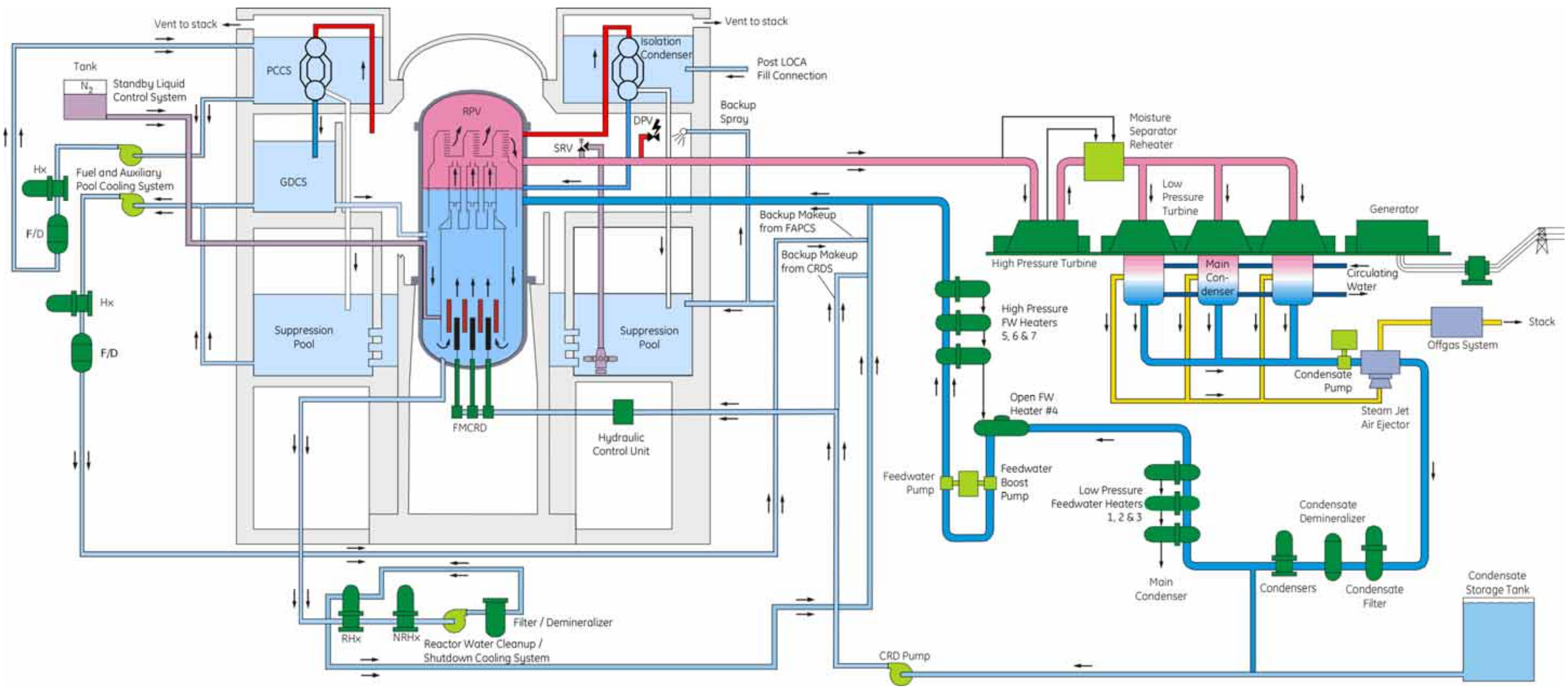
ESBWR Basic Parameters

- 4,500 Megawatt core thermal power
  - ~1,590 Megawatt electric gross
  - Nominal summer conditions
- Natural circulation
  - No recirculation pumps
- Passive safety systems
  - 4 divisions
  - 72 hours passive capability
    - » Simple actions to extend, such as recharge batteries and refill PCCS pools

ESBWR 3D Cutaway

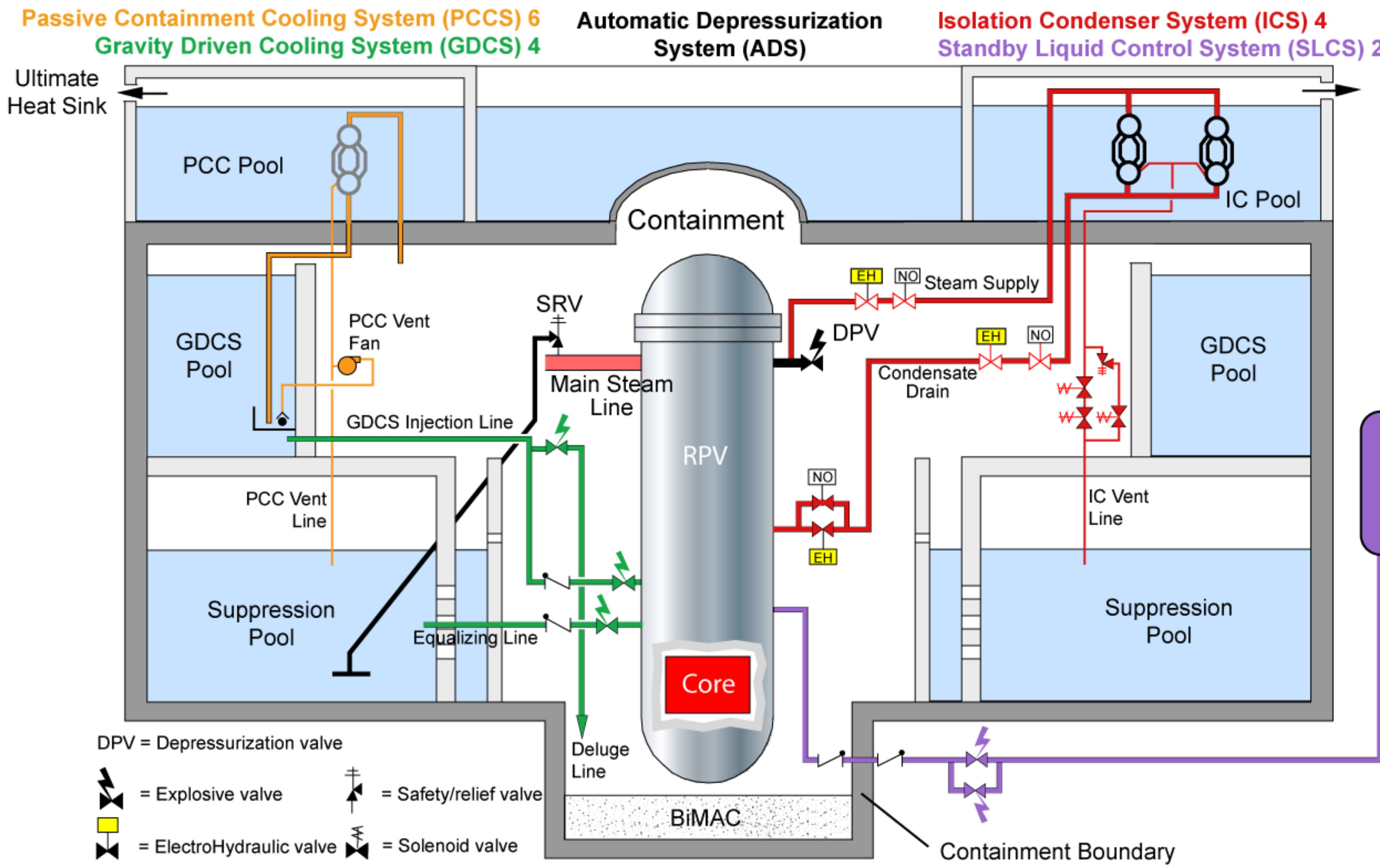


ESBWR Overall Flowchart



ESBWR Reactor Pressure Vessel

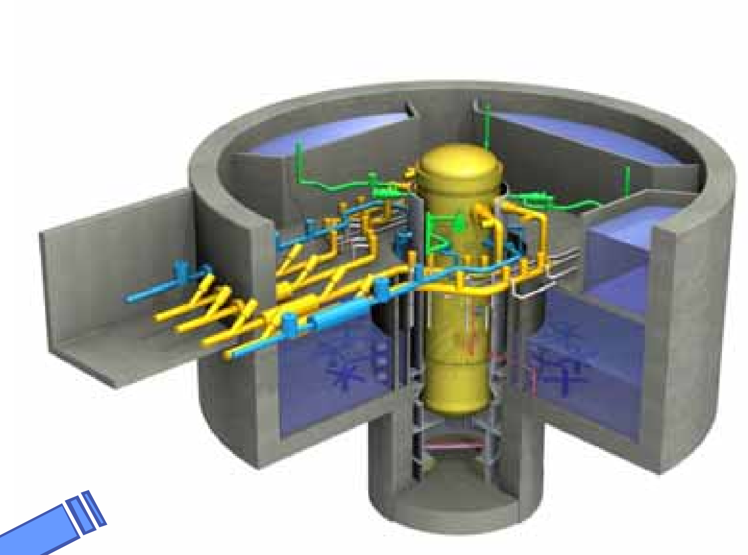
ESBWR Passive Safety



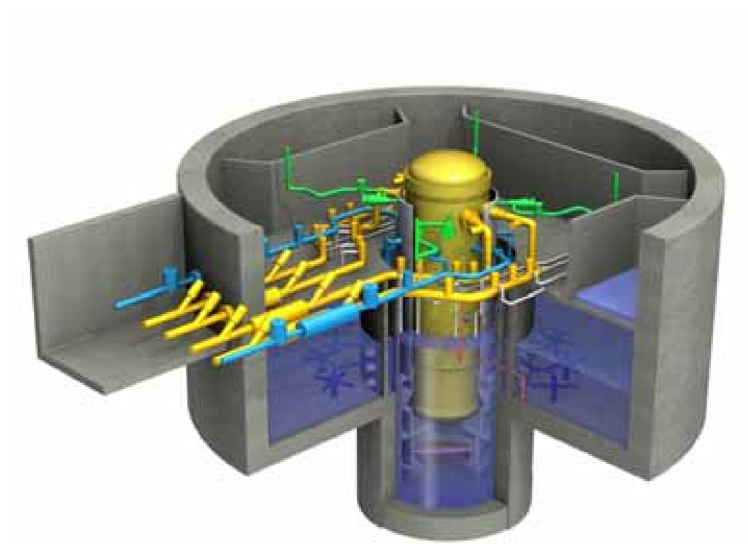
Gravity Driven Cooling System …

Simple design
Simple analyses

Extensive testing
Large safety margins



Before

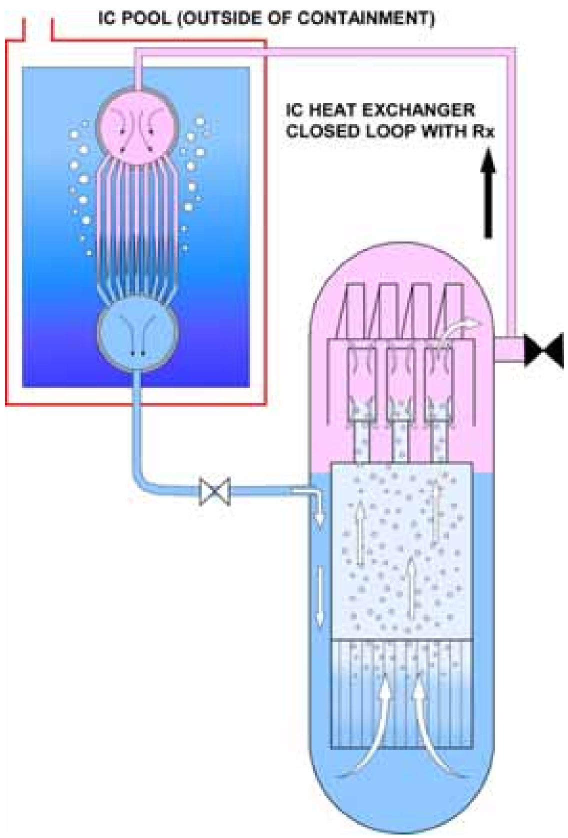


After

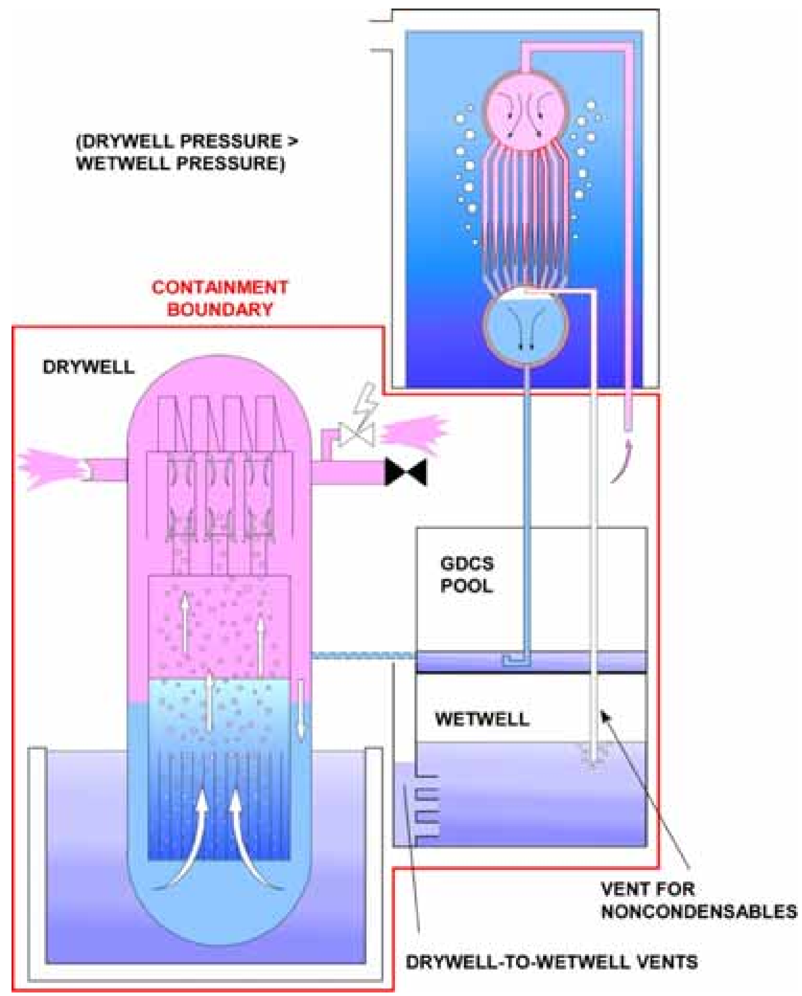
Gravity driven flow keeps core covered

Passive Safety Systems …

Isolation Condenser System



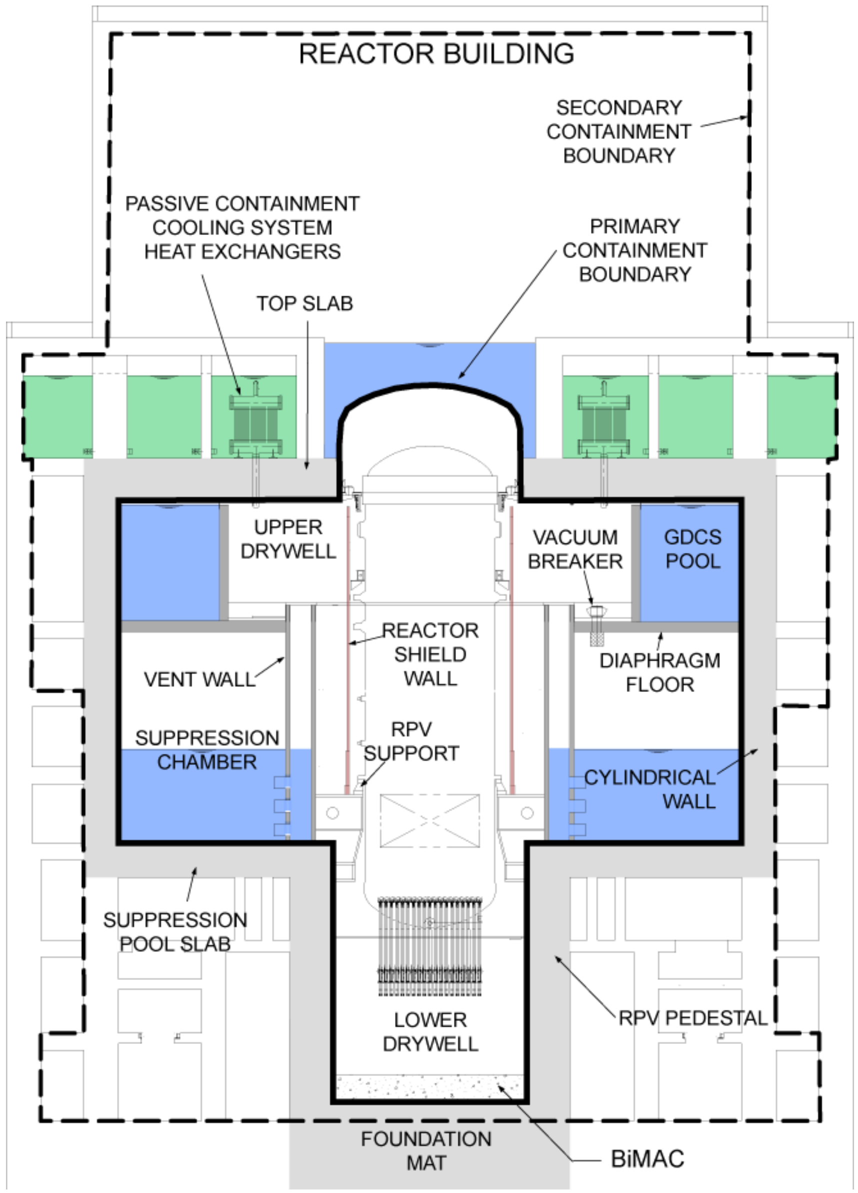
Passive Containment Cooling



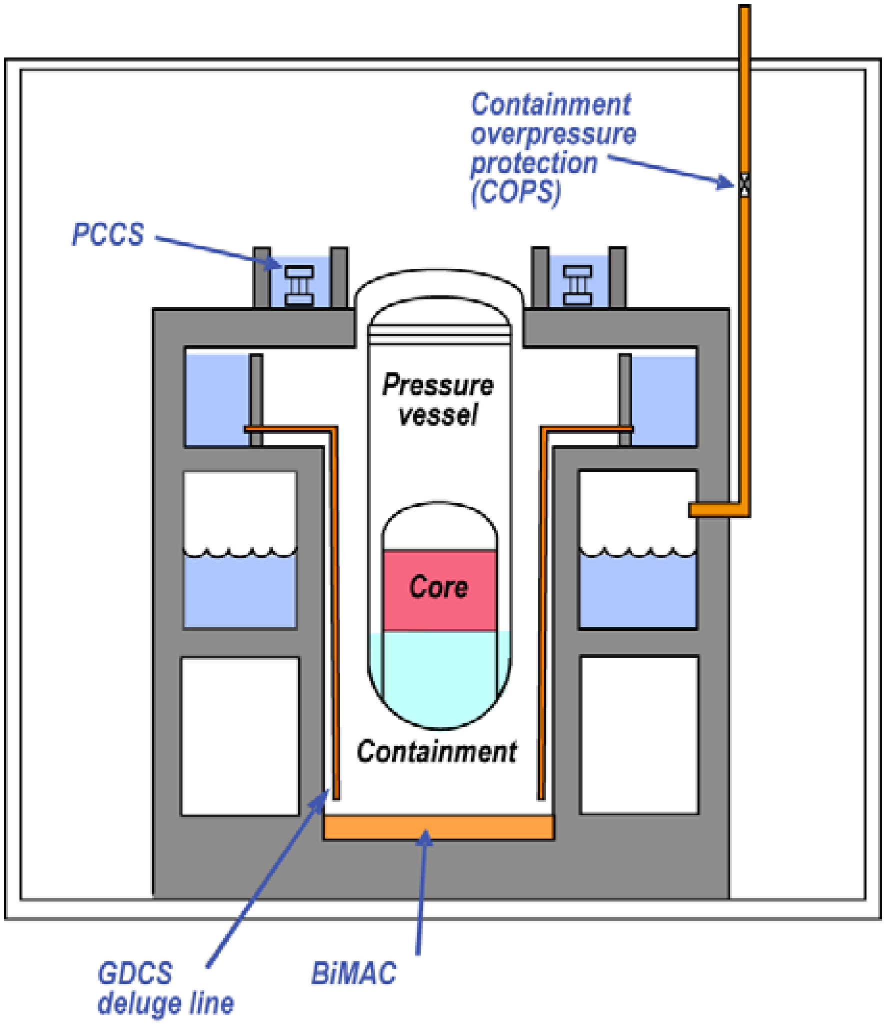
Other ESBWR New Features

- Control Rod Drive Hydraulic System (CRDHS, non-safety)
  - Modified the standard BWR system to allow RPV water injection as a non-safety grade backup
- RWCU and shutdown cooling mode of RHR combined
  - High pressure decay heat removal from RPV possible
- Fuel and Auxiliary Pool Cooling System (FACPS, non-safety)
  - Combines the separate cooling systems of previous plants
  - Provides a pathway for low pressure RPV injection using the suppression pool (SP) as source, or cross tie to a fire pump (FP)

ESBWR Containment Boundaries



ESBWR Severe Accident Mitigation Features



ESBWR passive features which mitigate severe accidents:

- Inerted Containment
- Lower Drywell flood capability
- Lower Drywell Basemat Internal Melt Arrest and Coolability (BiMAC)
- Suppression pool - fission product scrubbing & retention
- Passive Containment Cooling System (PCCS)
- Containment overpressure protection via rupture disks

Operator Options

- AC Independent water addition via fire pumps

ESBWR Summary

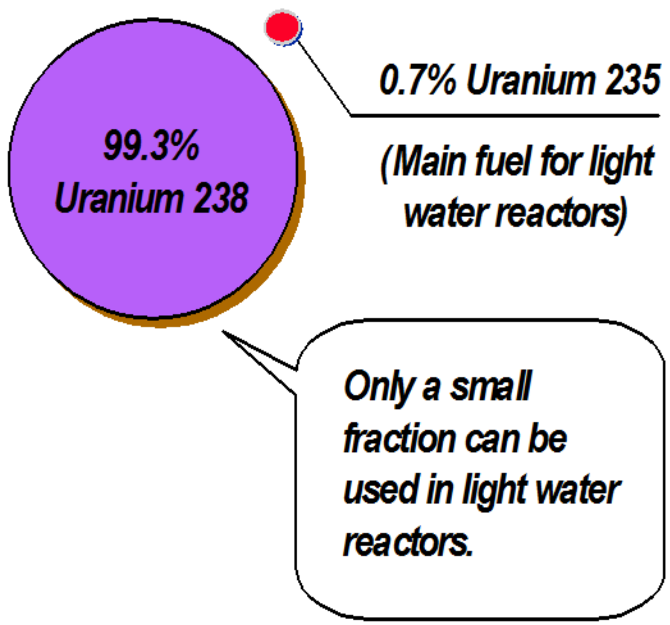
- ESBWR is GEH's latest evolution in BWR design
  - 4500 MWt/~1590MWe
  - Natural circulation
  - Passive safety features
  - Significant simplification
- ESBWR design certified by USNRC Oct 2014
- ESBWR chosen by Dominion (North Anna 3) and Detroit Edison (Fermi 3) as reference design in COL applications
  - DTE COL expected Mid 2015
  - Dominion COL expected 2016 or later

GEH Sodium Cooled Fast Reactor

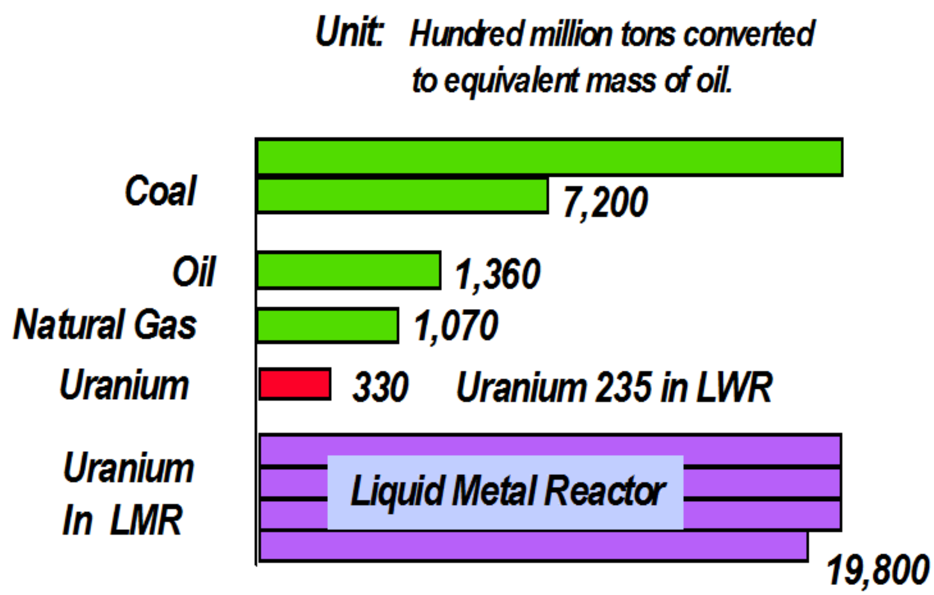
- Incentive/History of Fast Reactors
- Description of Super PRISM (S-PRISM)
- What is Next?

GEH Sodium Cooled Fast Reactor

Breakdown of Uranium

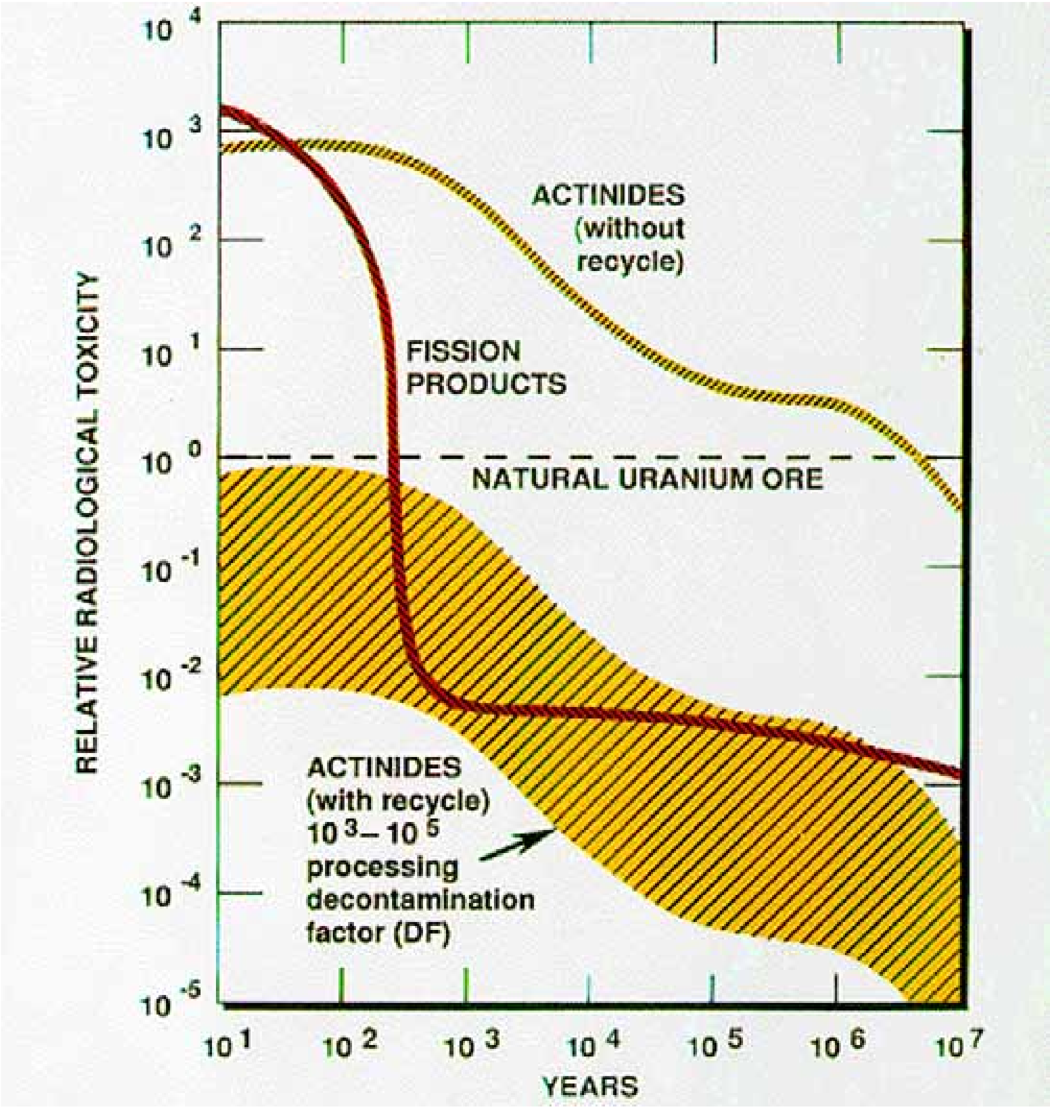


Assured Energy Reserves



Source JAPC using "Consolidated Energy Statistics" et al.

Time Phased Relative Waste Toxicity (LWR Spent Fuel)

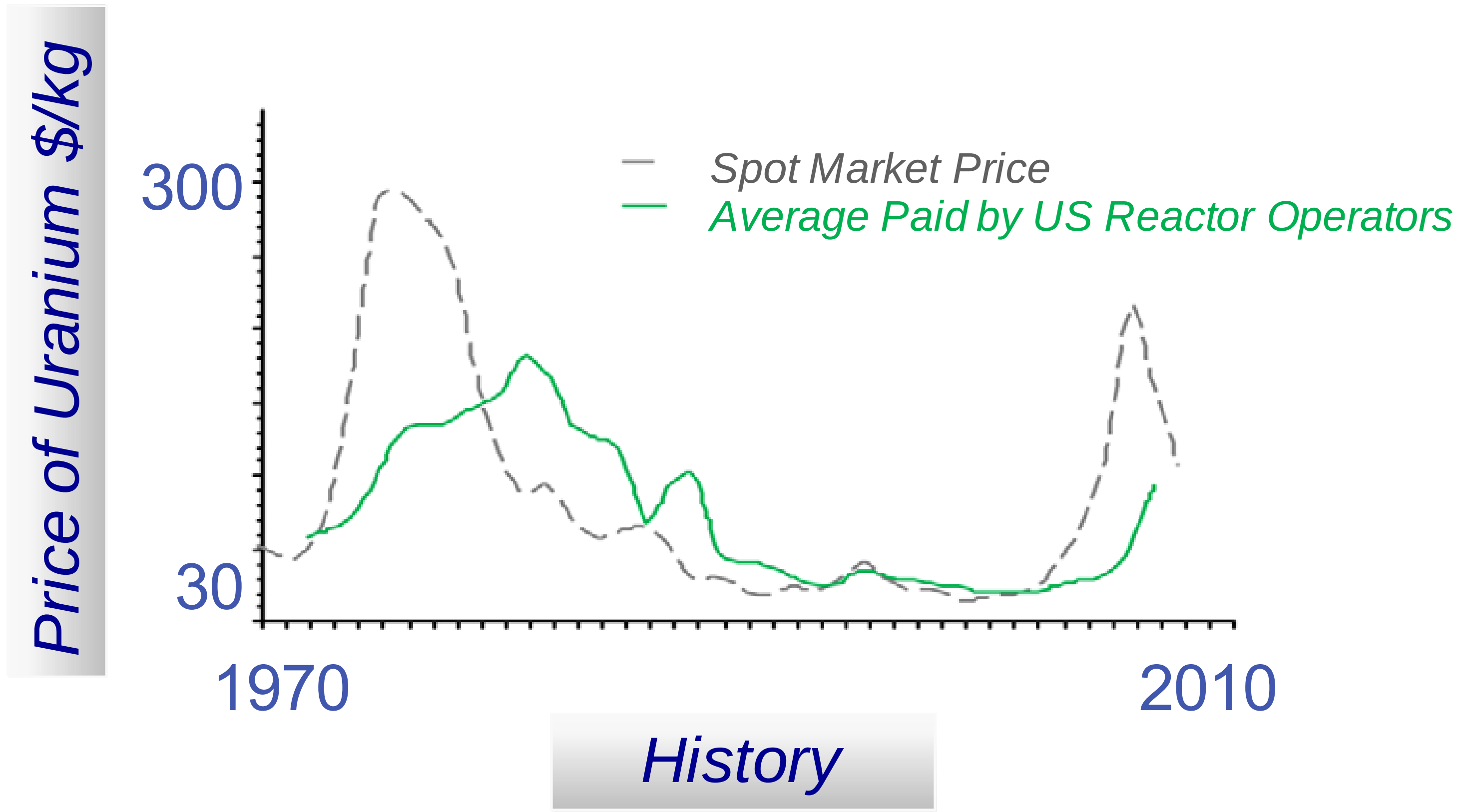


Impact of FRs on Repository Requirements by Year 2100

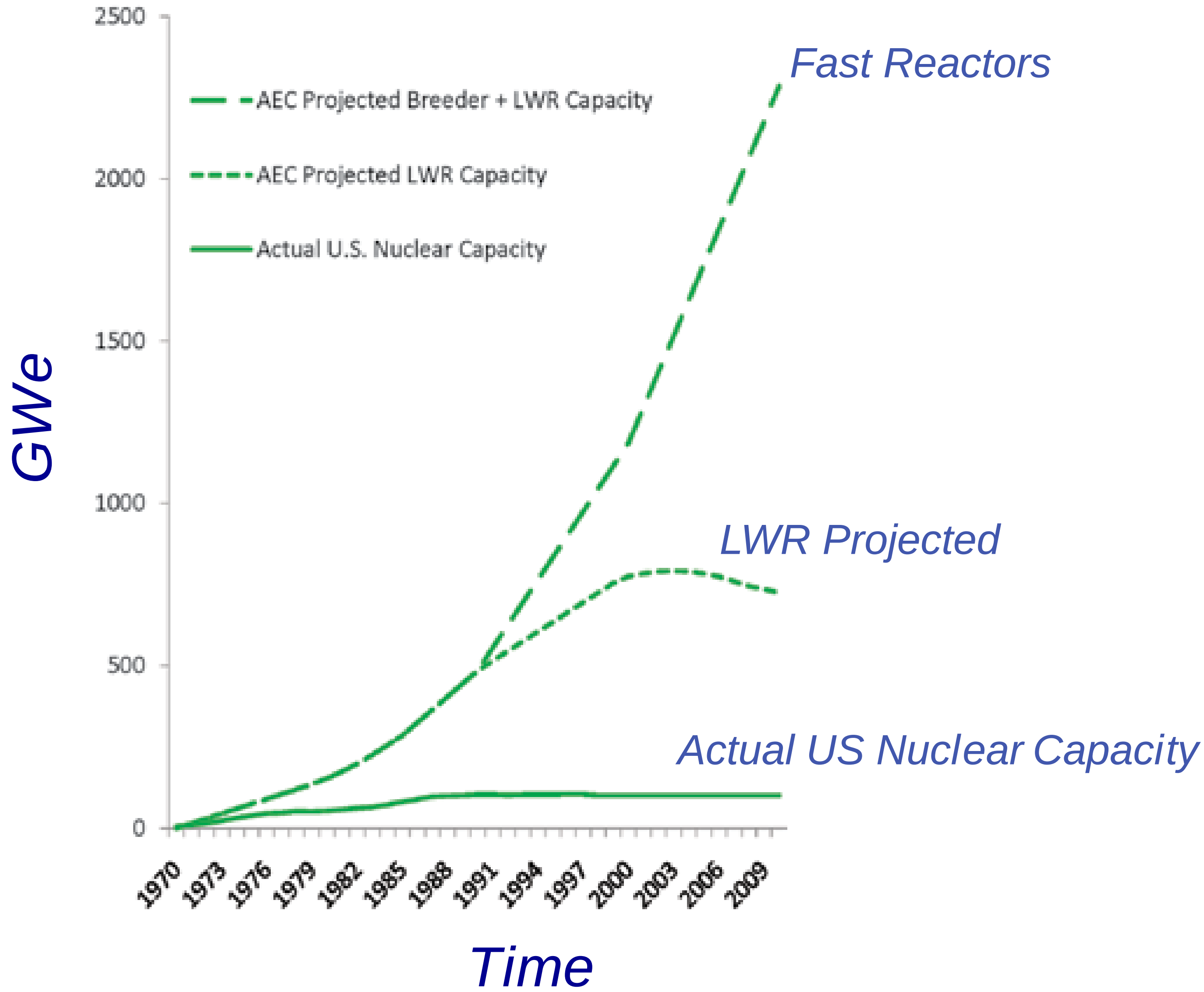
Nuclear Future	Existing License Completion	Extended License Completion	Continuing Level Energy Generation	Continuing Market Share	Growing Market Share 3.2% growth in nuclear power
Cumulative Spent Fuel (MTHM)	90,000	120,000	250,000	600,000	1,500,000
	New Reactors Only		Existing and New Reactors		
Fuel Management Approach	Number of Repositories Needed (at 70,000 MT each)				
Direct Disposal (current policy)	2	2	4	9	22
Limited Thermal Recycle with Expanded Repository Capacity	1	1	2	5	13
Recycle with Fast Reactors			1	1	1

Kathryn A. McCarthy – INL
Systems Analysis National Technical Director
Presentation to NERAC Gen IV subcommittee, May 2, 2005

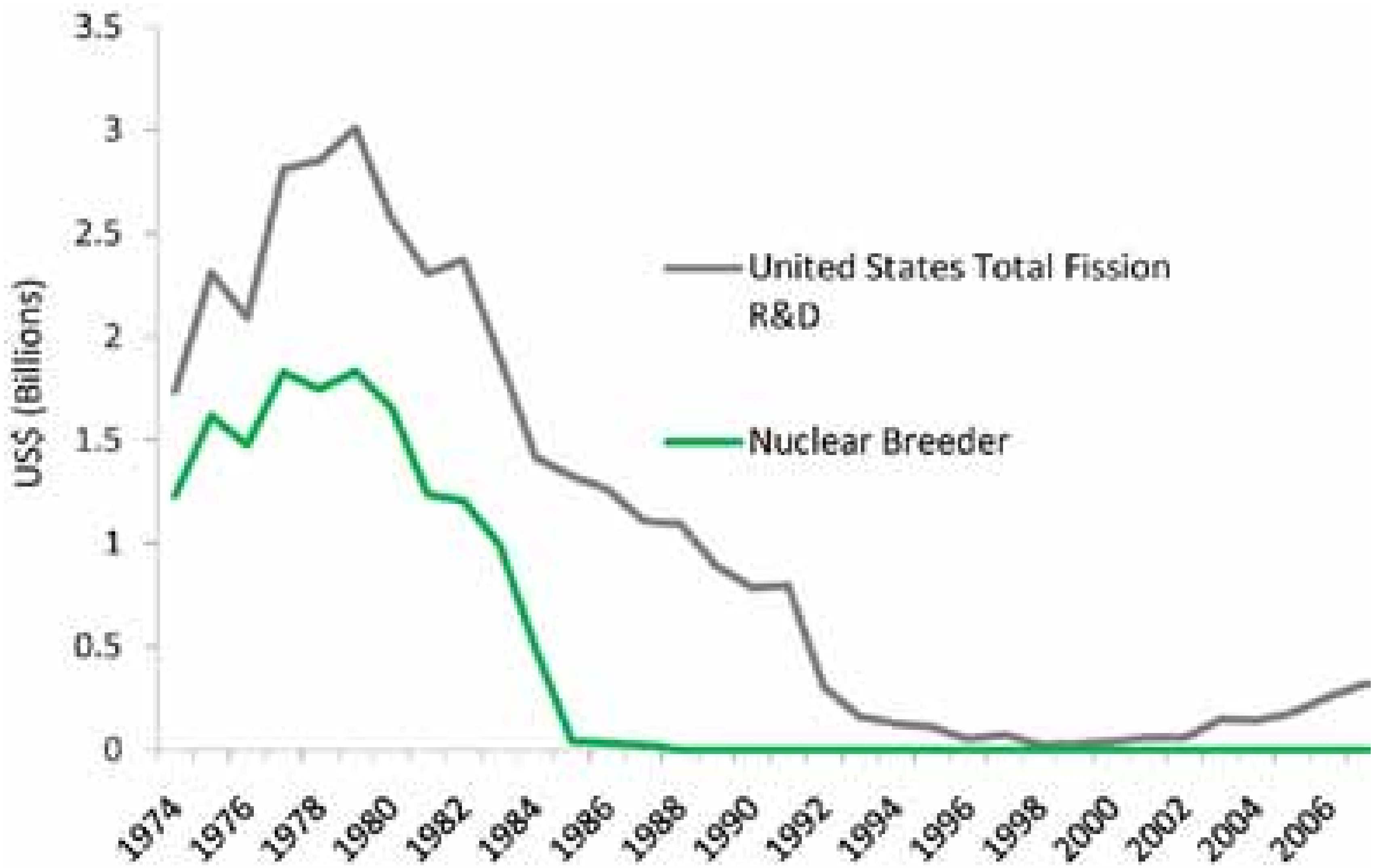
The Price of Uranium Since 1970



Nuclear Growth Projection - 1970



R&D by DOE



3.5
3
2.5
2
1.5
1
0.5
0
US$ (Billions)
United States Total Fission R&D
Nuclear Breeder
1974
1976
1978
1980
1982
1984
1986
1988
1990
1992
1994
1996
1998
2000
2002
2004
2006

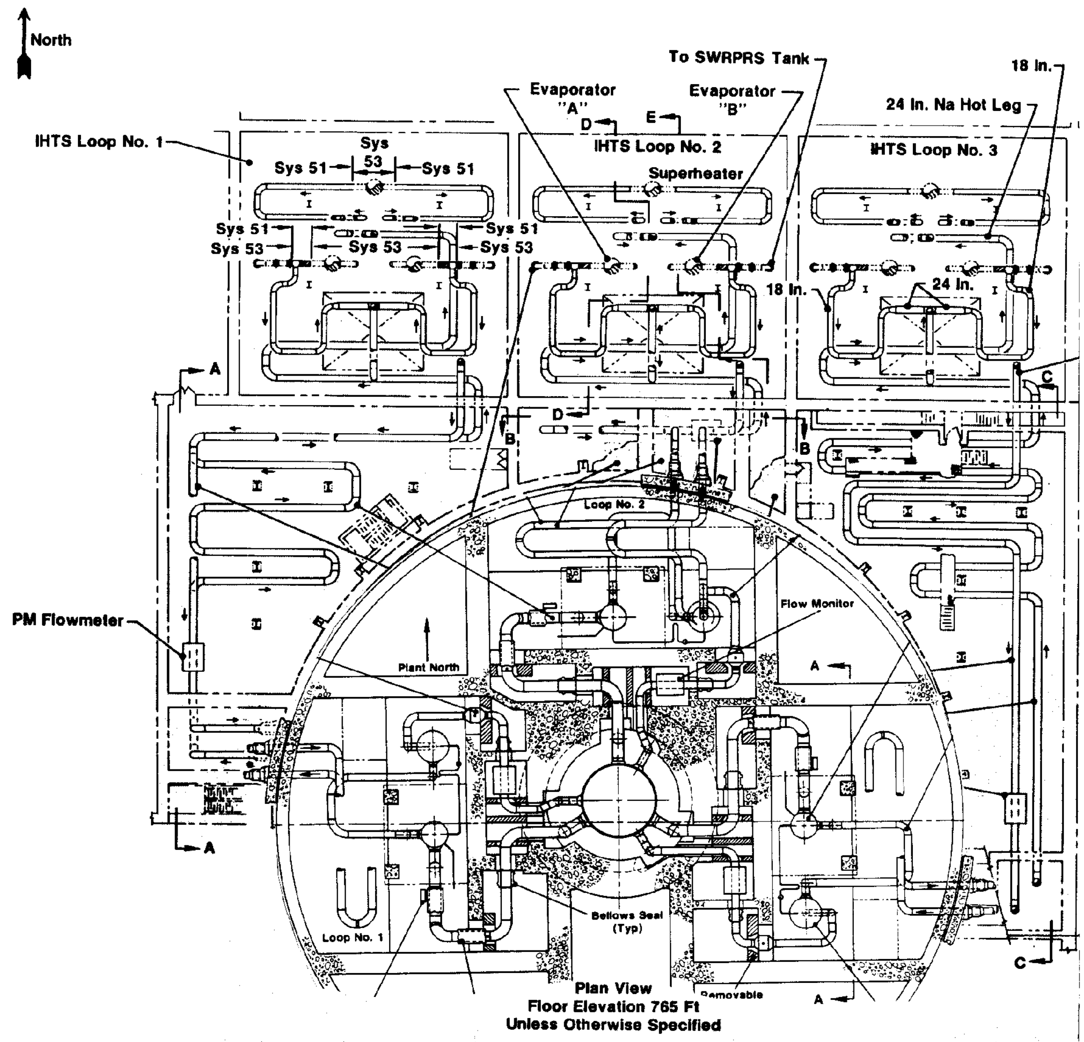
WORLD LIQUID METAL COOLED FAST REACTORS

NAME	LOCATION	PURPOSE	OPERA-TIONAL	SHUT-DOWN	POWER (MWt)	POWER (MWe)	FUEL	COOLANT
France								
Rapsodie	Cadarache	Test	1967	--	40	--	U02/Pu02	Na
Phenix	Marcoule	Prototype	1974	--	560	250	U02/Pu02	Na
SuperPhenix	Creys Malville	Demonstration	1985	1995	3000	1240	U02/Pu02	Na
INDIA								
FBTR	Kalpakkam	Test	--	--	42.5	12.4	(Pu-U)C	Na
ITALY								
PEC	Brasimone	Test	1981	--	120	--	U02/Pu02	Na
JAPAN								
Joyo	Oarai	Test	1978	--	100	--	U02/Pu02	Na
Monju	Ibaraki	prototype	1993	--	714	300	U02/Pu02	Na
UK								
DFR	Dounreay	Test	1963	1977	72	15	U-Mo	NaK
PFR	Dounreay	Prototype	1976	--	600	270	U02/Pu02	Na
USA								
Clemetine	Los Alamos	Research	1946	1953	0.025	--	Pu	Hg
EBR-1	Idaho	Research	1951	1963	1	0.2	Pu	Nak
Lampre	Los Alamos	Research	1959	1964	1	--	Pu	Na
EBR-2	Idaho	Test	1964	1992	62.5	20	U	Na
Enrico Fermi	Michigan	Test	1965	1972	200	61	U-Mo	Na
SEFOR	Arkansas	Test	1969	1972	20	--	U02/PuO2	Na
FFTF	Richland	Test	1980	1989	400	--	U02/PuO2	Na
Clinch River	Oak Ridge	Prototype	--	--	975	380	U02/PuO2	Na
USSR								
BR-2	Obninsk	Research	1956	--	0.1	--	Pu	Hg
BR-5	Obninsk	Test	1959	--	5	--	Pu	Na
BOR-60	Melekess	Test	1969	--	60	12	U02	Na
BN-350	Shevchenko	Prototype	1973	--	1000	150	U02/Pu02	Na
BN-600	Beloyarsk	Prototype	1980	--	1470	600	U02/Pu02	Na
BN-800	--	Demonstration	--	--	2100	800	U02/Pu02	Na
BN-1600	--	demonstration	--	--	4200	1600	U02/Pu02	Na
W. Germany								
KNK	Karlruhe	Test	1972	--	58	21	U02/Pu02	Na
SNR-300	Kalkar	Prototype	--	--	730	327	U02/Pu02	Na
SNR-2	Kalkar	demonstration	--	--	3420	1460	U02/Pu02	Na

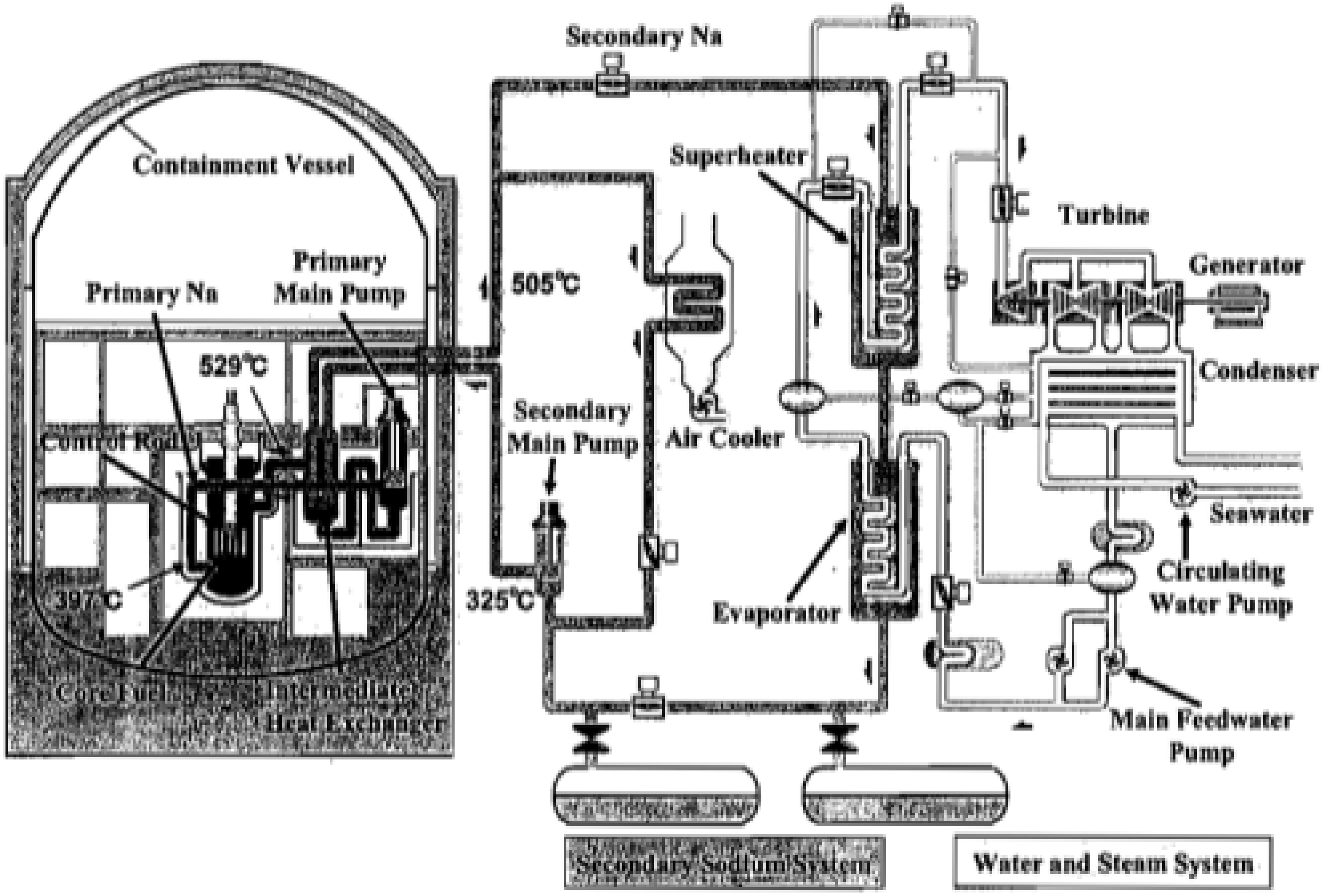
History

- Over 20 “Demonstration” Plants were built
- Initial FR Designs were Small and Complex
- “DEMONSTRATION” Plants ARE Expensive

Clinch River (three loop 985 MWt Fast Reactor)



Monju (three loop 714 MWt Fast Reactor)



GEH Sodium Cooled Fast Reactor

- Incentive/History of Fast Reactors
- Description of Super PRISM (S-PRISM)
- What is Next?

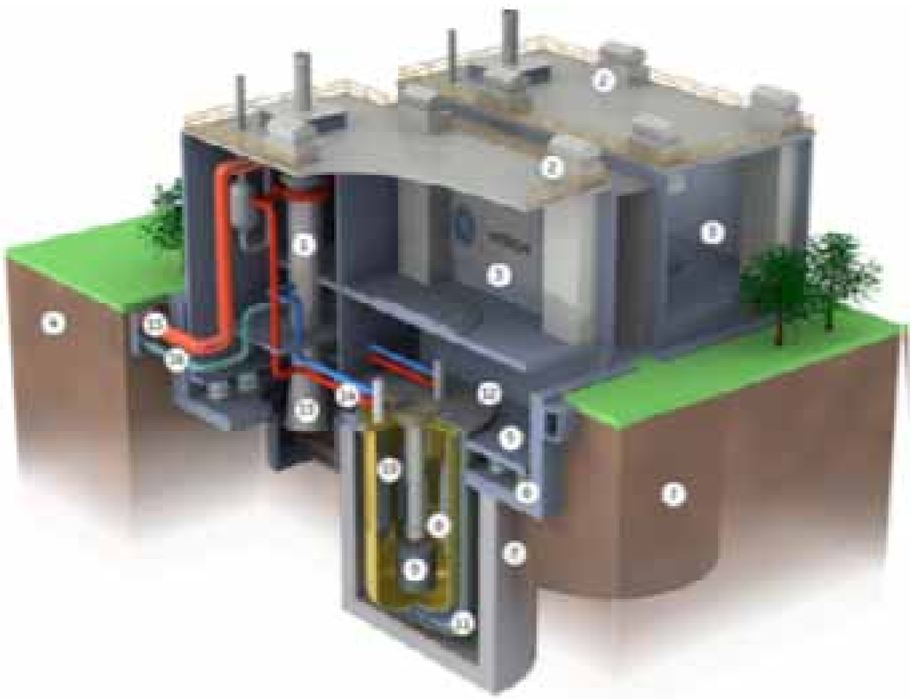
PRISM Basic Parameters

From GEH Web Site (*http://gehitachiprism.com/what-is-prism/how-prism-works***)**

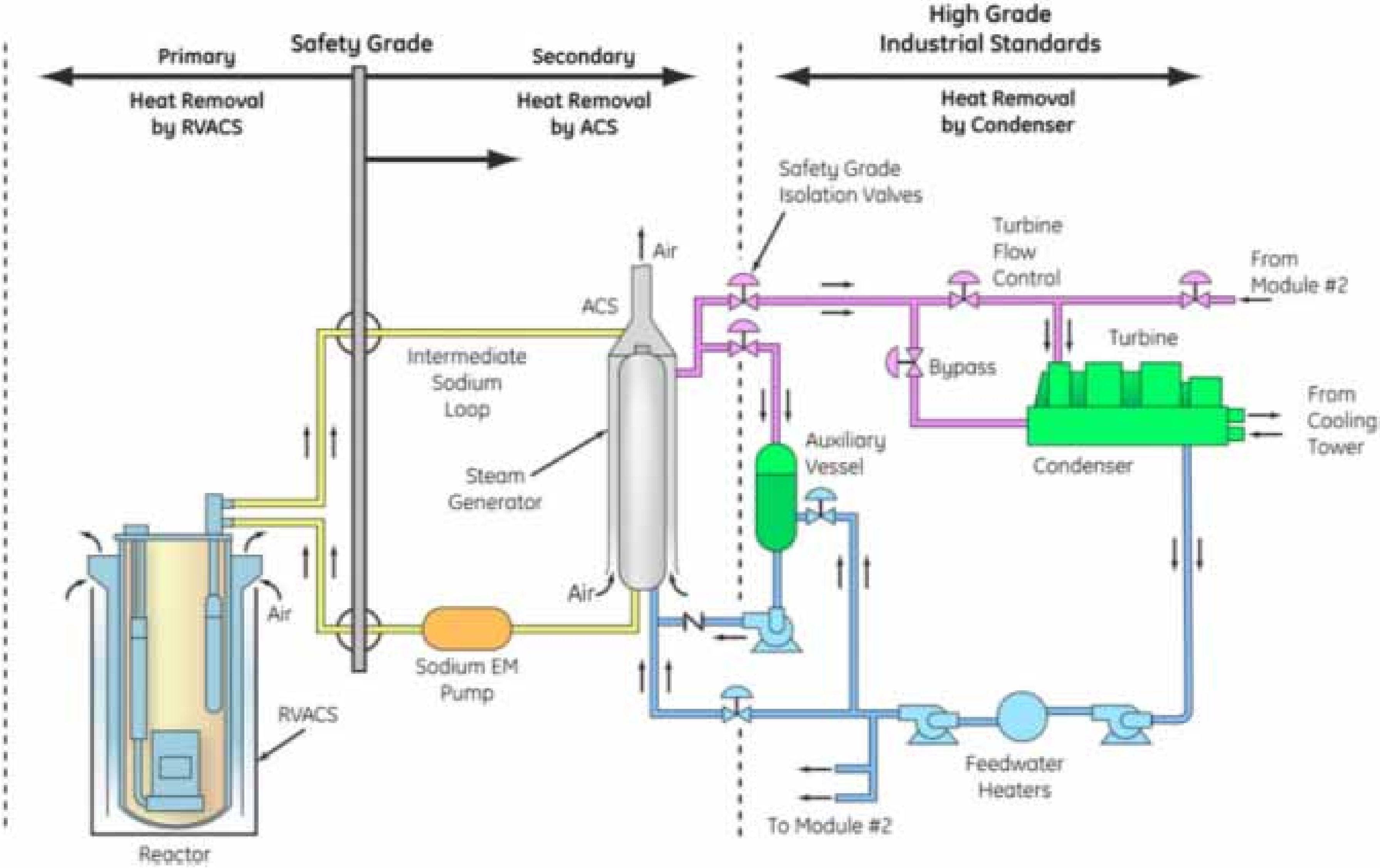
- PRISM is a small pool-type Sodium Cooled Fast Reactor.
- A non-radioactive intermediate loop that transfers heat from the Intermediate HX to the SG.
- Sodium loops are at low pressure; the steam loop is at 14.7MPa
- PRISM employs passive safety systems:
  - Decay Heat Removal
  - Spent Fuel Cooling
  - Post Accident Containment Cooling
- 840 MWt and 311 MWe.
- A Two Reactor Power Block produces 622 MWe
- A Six Reactor Site would produce 1866 MWe
- Net Efficiency, 37%

PRISM Power Block Cutaway

1. Steam Generator
2. Reactor Vessel Auxiliary Cooling Sys
3. Refueling Enclosure Building
4. Steam Tunnel To Turbine
5. Reactor Protection System Modules
6. Seismic Isolation Bearing
7. Reactor Module
8. Primary Electromagnetic Pump
9. Reactor Core
10. Intermediate Heat Exchangers
11. Lower Containment Vessel
12. Upper Containment Building
13. Sodium Dump Tank
14. Intermediate Heat Transfer System
15. Steam Outlet Piping
16. Feedwater Return Piping

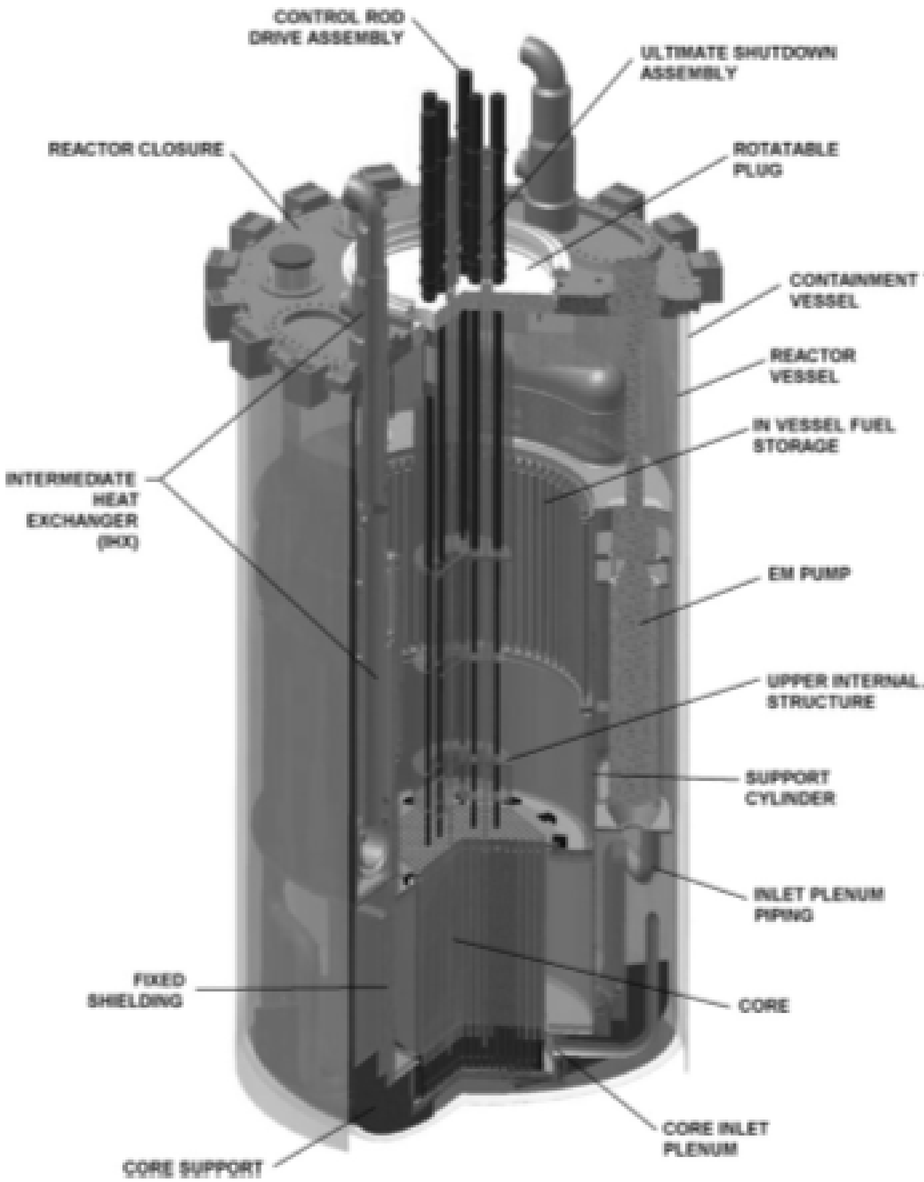


PRISM Overall Flow Chart

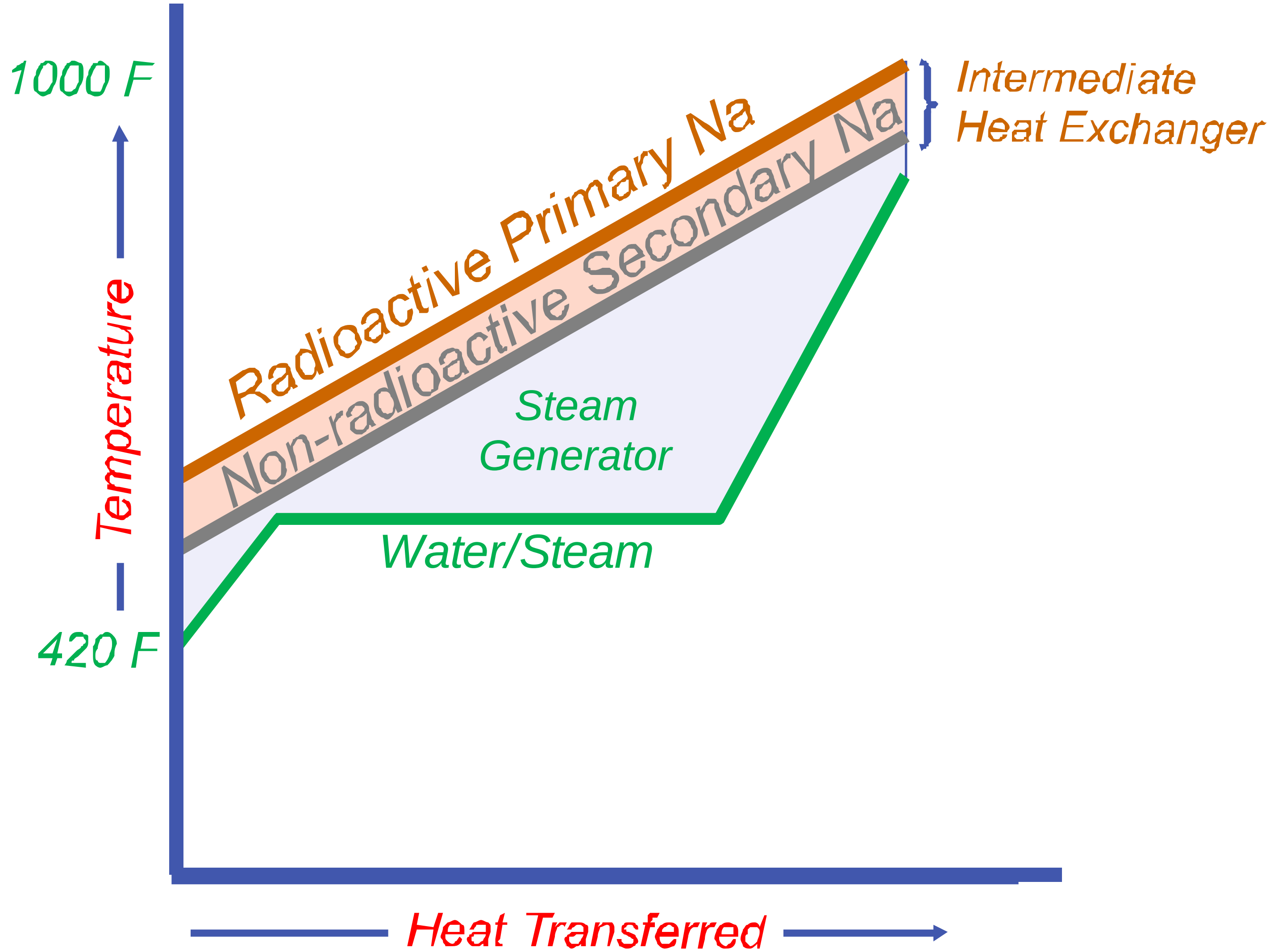


RVACS	ACS	Condenser

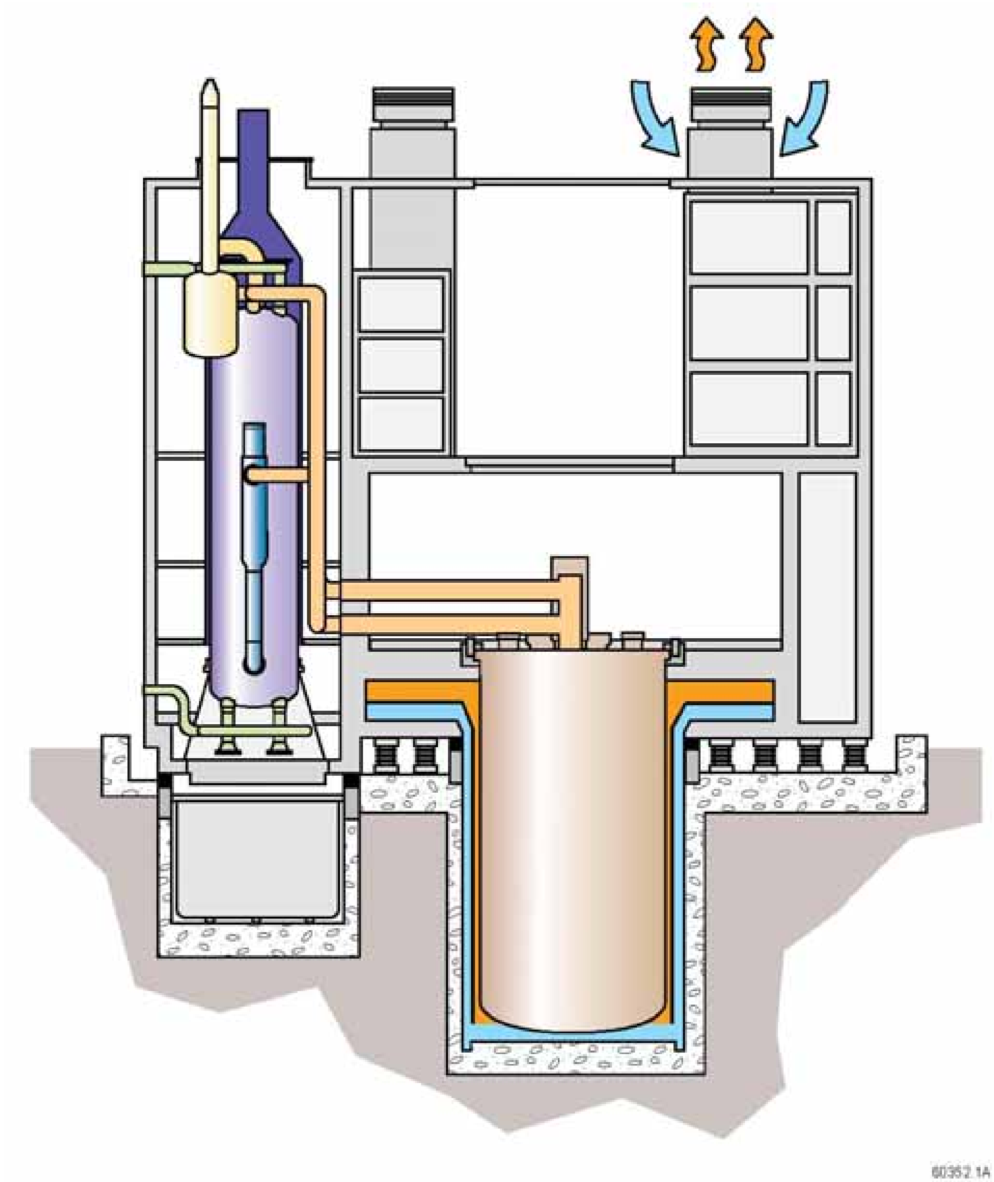
PRISM Reactor and Containment Vessel



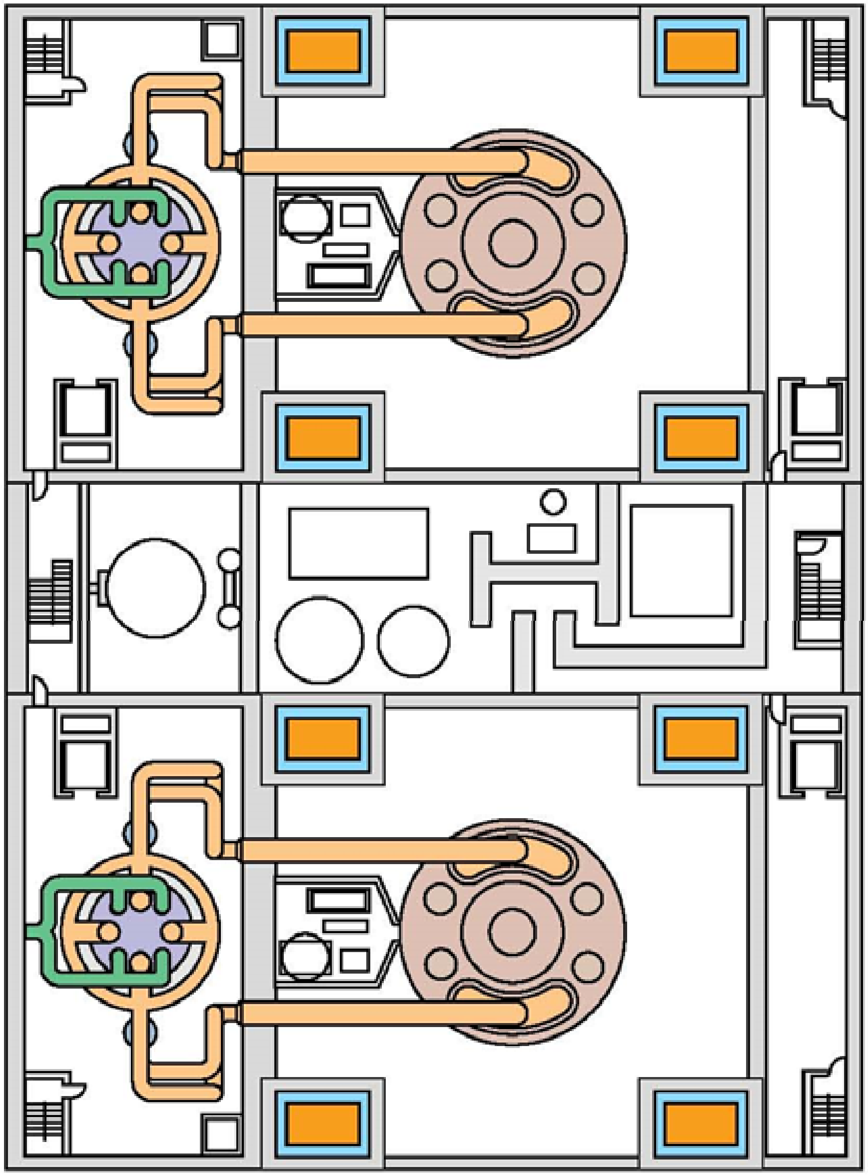
Heat Transport System



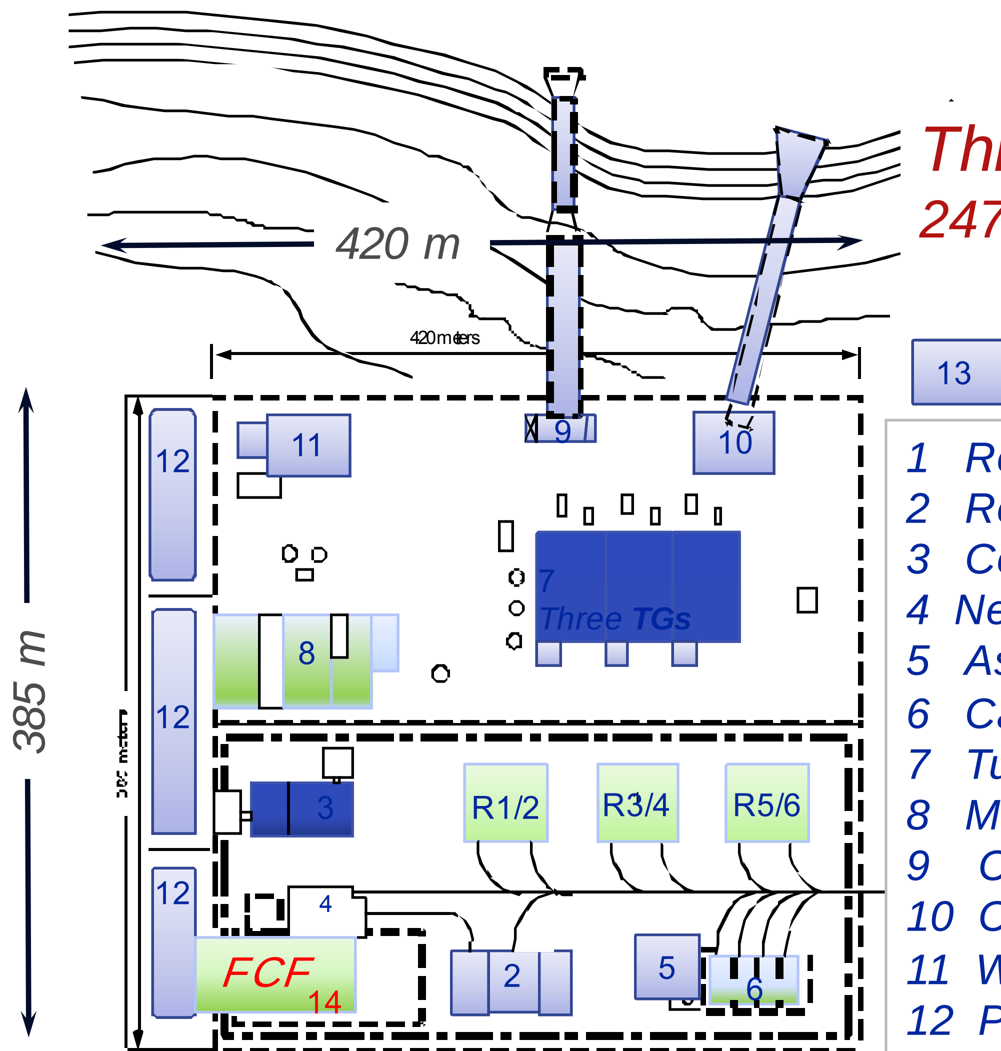
S-PRISM – One-On-One Arrangement



S-PRISM Power Block



S-PRISM – Compact Site with On Site Reprocessing

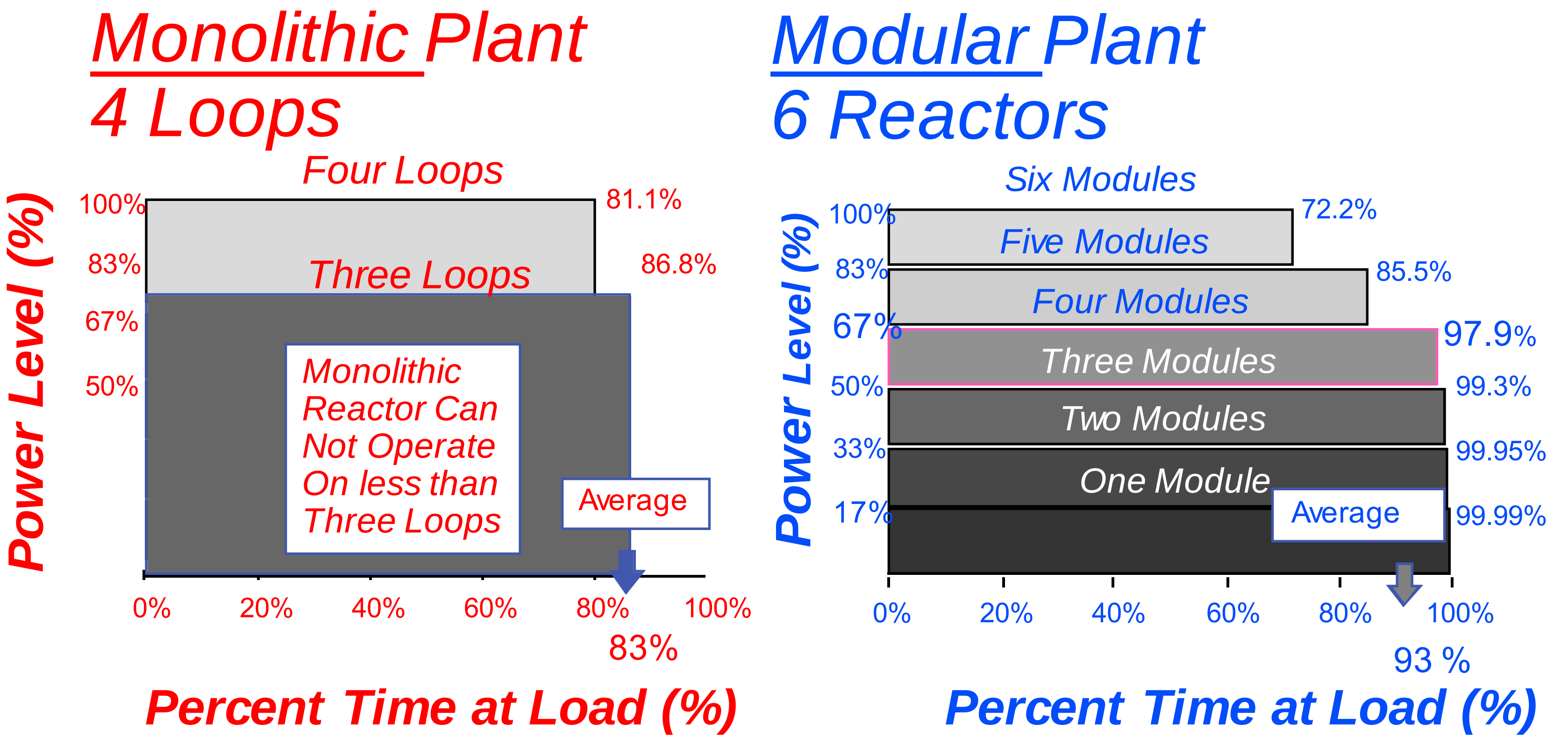


Three Power Block Plant

2472 MWe (2274 MWe net)

1 Reactor Building (2 NSSS/block)
2 Reactor Maintenance Facility
3 Control Facility
4 New and Spent Fuel Handling Facility
5 Assembly Facility
6 Cask Storage Facility
7 Turbine-Generator Facility
8 Maintenance Facility
9 Circulating Water Inlet Pump Station
10 Circulating Water Discharge
11 Waste Treatment
12 Parking Lot
13 Switch Yard
14 Optional Spent Fuel Recycle Facility

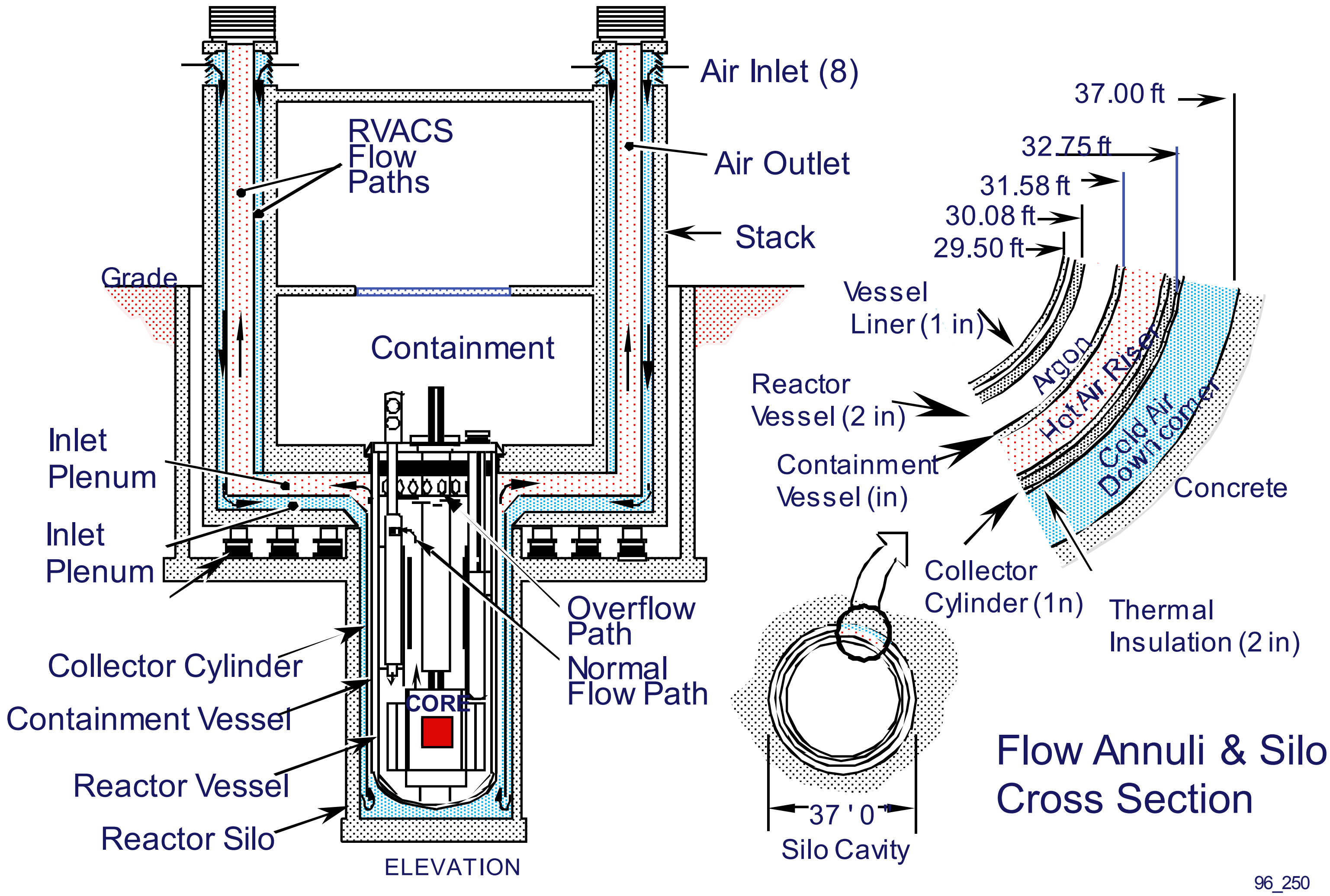
Modular vs Monolithic Availability Comparison



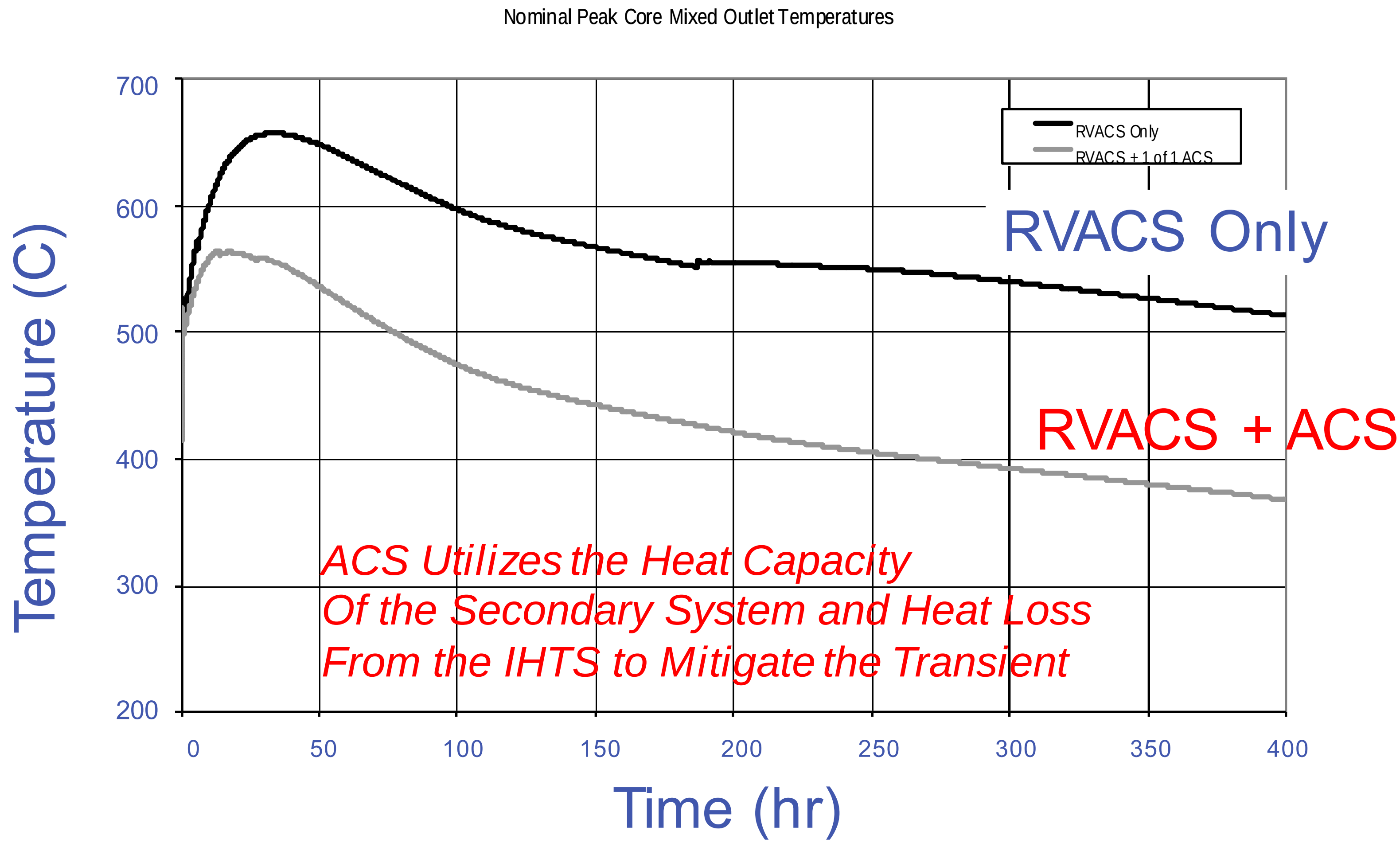
Ten Point Advantage Due To:

- *Modular Reactors Operate Independently of the others*
- *~ 98 % of the Time the Plant Output will Exceed 67%*

Passive Decay Heat Removal System



RVACS Cooling - Core Outlet Temperature



Safety Comparison, ALMR versus LWR

Function	ALMR	LWR
• Shutdown Heat Removal	Completely Passive	Active and Passive ECCS
• Post Accident Containment Cooling	Passive Air Cooling of Upper Containment	Redundant and Diverse
• Coolant Injection/Core Flooding	N/A	Redundant and Diverse
• Shutdown System	Primary 7/9 Rods (N-2) Ultimate Shutdown 2/3 Rods ATWS Capability	Most Rods Required Boron Injection
Emergency AC Power	<200 kWe from Batteries	~10,000 kWe

Postulated Accidents Have Been Assessed

- BDB Containment Accident
- BDB Core Melt Accidents
- Runaway Steam Generator Accident in a Large HCSG can be safely accommodated
  - Expected initial leaks
  - Passively Terminating SG Leaks of Any Size
  - BDB SG Leaks only used to size the relief system

Sodium Cooled Fast Reactors

- Incentive/History of Fast Reactors
- Description of Super PRISM (S-PRISM)
- What is Next?

What Is Next For the Fast Reactor

Stalled Programs:

- US (except for TP and GE Initiatives)
- UK
- Japan
- Germany

Moving Forward

- China
- Russia
- India
- France

GEH Proposal to the UK

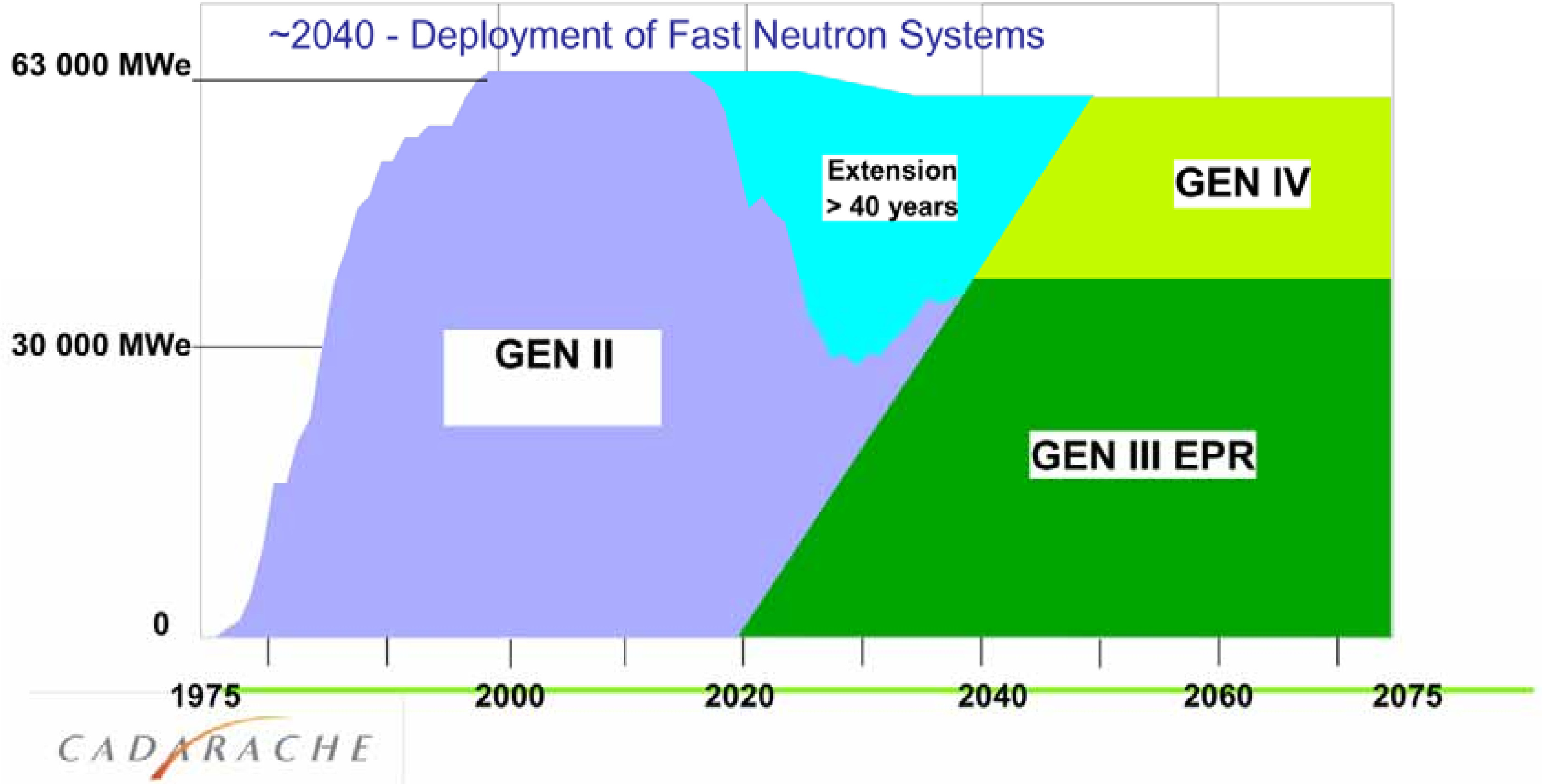
- Use PRISM Reactors to “Spike” excess plutonium to make it proliferation resistant.
- Two 311 MWe Reactors in a single power block.
- Some of the spiked fuel will be used to operate the reactors in a conventional 18-24 month fuel cycle to generate electricity.
- GEH working with UK to develop and assess the S-PRISM in competition with other approaches

Scenario for Future French Grid

Major part for PWR during 21e century

- PWR Gen II : extension of life duration (> 40 years)
- PWR Gen III/III+ : replacement of PWR by 2015 and operation during 21e century

~2040 - Deployment of Fast Neutron Systems



Direction of French Program

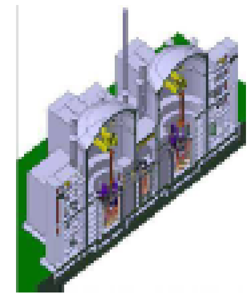
40 MWt

560 MWt
260 MWe
1965

3000 MWt
1200 MWe
1985

3600 MWt
1500 MWe
Design

1500 MWt
Prototype
2027



~3000 MWt
Commercial
Plant
2040

Latest News

UK Considers PRISM to be a Serious Contender

DOE has provided GEH with the necessary funds to develop an advanced insulation for the large high EMPs

July 2014, GEH and Iberdrola Nuclear Services Agreed to work together on a Long Term Solution to dispositioning the 100 metric tons of Civil Pu currently stored at the Sellafield Site where PRISM would be constructed.

November 2014, ANL will support GEH's effort modernize the Next Generation PRA for PRISM

DOE is Supporting an Up-Dated Safety Assessment

The latest multimillion-dollar investment from Uncle Sam will be used to provide an updated PRA with support from ANL.

1. ***The last assessment was conducted in the 1990's,a lot has happened in terms of global energy demand and safety requirements.***
2. ***For PRISM, a "modern probabilistic risk assessment," which considers how complex systems work together to quantify and characterize risk factors is needed.***
3. ***It's time to take a second look into the design limits and process metrics for the PRISM reactor now that it's 2015.***

Bibliography

- “The ABWR Plant General Description”, GE-Hitachi Company, July, 2007 http://www.ge-energy.com/content/multimedia/_files/downloads/ABWR%20General%20Description%20Book.pdf
- “The ESBWR Plant General Description”, GE-Hitachi Company, June, 2011 http://www.ge-energy.com/content/multimedia/_files/downloads/ESBWR_General%20Description%20Book.pdf
- Publically available documents for ABWR and ESBWR on NRC web site, using ADAMS searches
- “PRISM: A Competitive Small Modular Sodium-Cooled Reactor”, B. S. Triplett et. al., *Nuclear Technology*, Vol. 178, May 2012

Acronyms

ACIWA	AC Independent Water Addition
ACS	Auxiliary Cooling System
ADS	Automatic Depressurization System
BiMAC	Basemat-internal Melt Arrest Coolability
COPS	Containment Overpressure Relief System
CRDHS	Control Rod Drive Hydraulic System
DPV	Depressurization Valve
FAPCS	Fuel and Auxiliary Pools Cooling System
FMCRD	Fine Motion Control Rod Drive
GDCS	Gravity-Driven Cooling System
ICS	Isolation Condenser System
HCU	Hydraulic Control Unit
HPCF	High Pressure Core Flooder

Acronyms

LPFL	Low Pressure Flooder
PCCS	Passive Containment Cooling System
RCIC	Reactor Core Isolation Cooling
RIP	Reactor Internal Pump
RHR	Residual Heat Removal
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling System
RVACS	Reactor Vessel Auxiliary Cooling System
SRV	Safety/Relief Valve