MB

individual, his/her representative, or to those who have a need to access the information in performing assigned duties in the process of determining access to SGI. No individual authorized to have access to the information may re-disseminate the information to any other individual who does not have a need-to-know.

3. The personal information obtained on an individual from a criminal history records check may be transferred to another licensee if the gaining licensee receives the individual's written request to re-disseminate the information contained in his/her file, and the gaining licensee verifies information such as the individual's name, date of birth, social security number, sex, and other applicable physical characteristics for identification purposes.

4. The licensee shall make criminal history records, obtained under this section, available for examination by an authorized NRC representative, to determine compliance with the regulations and laws.

5. The licensee shall retain all fingerprint and criminal history records received from the FBI, or a copy, if the individual's file has been transferred, for three (3) years after termination of employment or determination of access to SGI. After the required three (3) year period, these documents shall be destroyed by a method that will prevent reconstruction of the information in whole or in part.

[FR Doc. E7–22574 Filed 11–16–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on December 5, 2007, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, December 5, 2007, 8:30 a.m. until 10 a.m.

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Officer, Mr. Sam Duraiswamy (telephone: 301–415–7364) between 7:30 a.m. and 4 p.m. (ET) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695).

Further information regarding this meeting can be obtained by contacting the Designated Federal Officer between 7:30 a.m. and 4 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: November 8, 2007.

Cayetano Santos,

Chief, Reactor Safety Branch.

[FR Doc. E7–22537 Filed 11–16–07; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Meeting of the ABWR Subcommittee; Notice of Meeting

The ACRS Subcommittee on ABWR will hold a meeting on December 5, 2007, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed to discuss General Electric Company proprietary information pursuant to 5 U.S.C. 552b(c)(4).

The agenda for the subject meeting shall be as follows:

Wednesday, December 5, 2007—12:30 p.m. until 5 p.m.

The Subcommittee will meet with representatives of the General Electric Company and the NRC staff to discuss the ABWR certified design, proposed amendment to the certified design, issues to be addressed through topical reports, issues to be addressed through Combined License Submittals (design centered working group), and the staff's review schedule. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Officer, Ms. Maitri Banerjee (telephone 301/415–6973) 5 days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on September 26, 2007 (72 FR 54695).

Further information regarding this meeting can be obtained by contacting the Designated Federal Officer between 7 a.m. and 4:45 p.m. (ET). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes to the agenda.

Dated: November 8, 2007.

Cayetano Santos,

Chief, Reactor Safety Branch.

[FR Doc. E7–22540 Filed 11–16–07; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

Request for Comments Concerning Compliance With Telecommunications Trade Agreements

AGENCY: Office of the United States Trade Representative.

ACTION: Notice of request for public comment and reply comment.

SUMMARY: Pursuant to section 1377 of the Omnibus Trade and Competitiveness Act of 1988 (19 U.S.C. 3106) ("section 1377"), the Office of the United States Trade Representative ("USTR") is reviewing and requests comments on: The operation, effectiveness, and implementation of and compliance with the following agreements regarding telecommunications products and services of the United States: the World Trade Organization ("WTO") Agreement; the North American Free Trade Agreement ("NAFTA"); U.S. free trade agreements ("FTAs") with Australia, Bahrain, Chile, Morocco, and Singapore; the Dominican Republic—Central America—United States Free Trade Agreement ("CAFTA–DR"); and any other FTA or telecommunications trade agreement coming into force on or before January 1, 2008. The USTR will conclude the review by March 31, 2008.

Advisory Committee on Reactor Safeguards
ABWR Subcommittee Meeting – ABWR Information Briefing
December 5, 2007
Rockville, MD

Proposed Agenda

ABWR Subcommittee Chair: Said Abdel-Khalik
Cognizant Staff Engineer: Maitri Banerjee mxb@NRC.GOV (301) 415-6973

Topics	Presenters	Time
Opening Remarks	S. Abdel-Khalik, ACRS	12:30 pm - 12:35 pm
Introductions	Mark Tonacci, NRO	12:35 pm – 12:45 pm
ABWR Technology Overview	Alan Beard, GE	12:45 pm – 2:15 pm
Break		2:15 pm – 2:30 pm
PRA Aspects	Dennis Henneke, GE	2:30 pm – 3:15 pm
Licensing and Operating Experience	Joe Savage, GE	3:15 pm – 3:45 pm
Staff Review of TRs and STP COLA	Mark Tonacci, NRO	3:45 pm – 4:00 pm
Committee discussion and Closing remarks	S. Abdel-Khalik, ACRS	4:00 pm – 4:30 pm

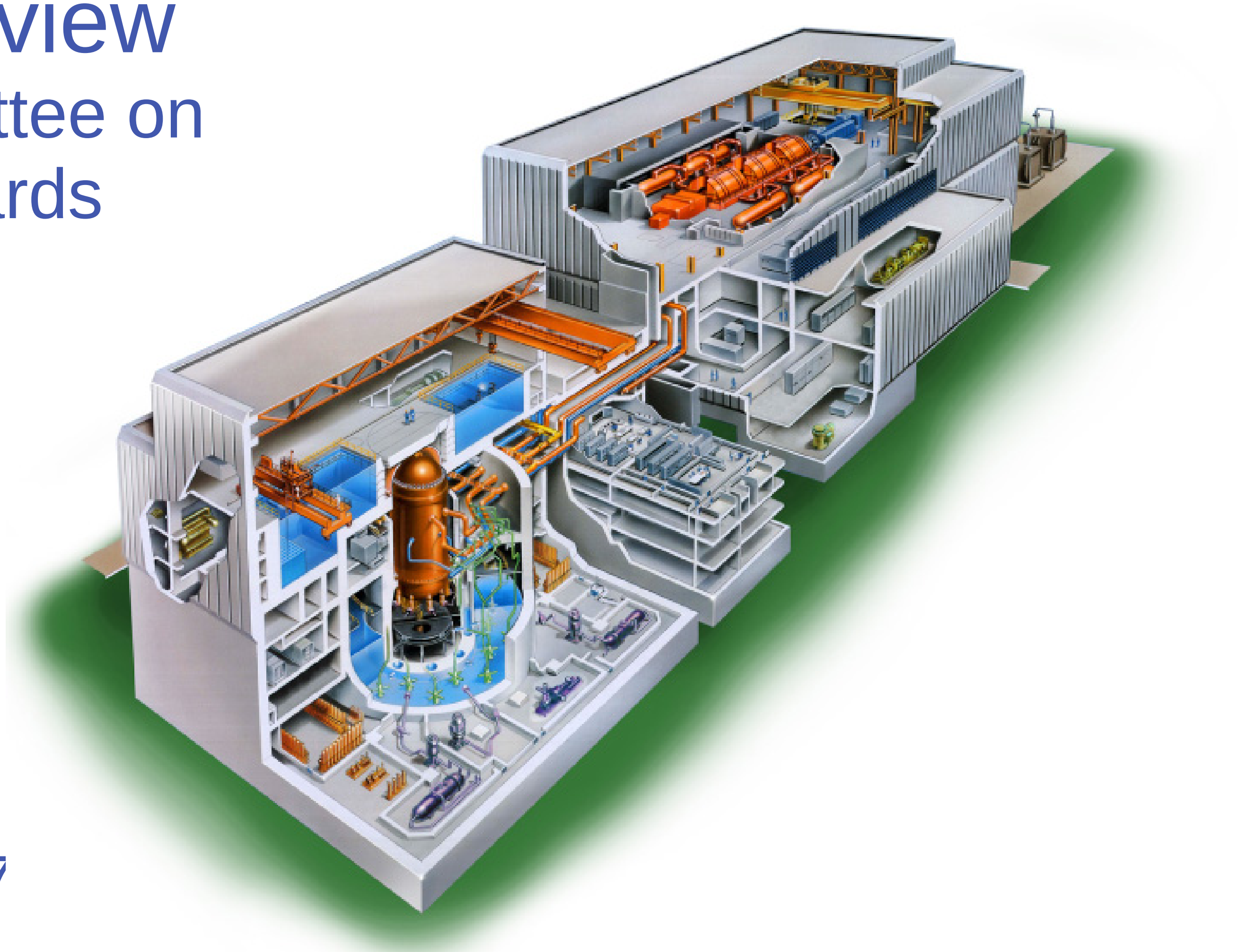
NOTE:

1. Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.

2. Number of copies of presentation materials to be provided to the ACRS - 35

ABWR Overview

Advisory Committee on Reactor Safeguards



J. Alan Beard

December 5, 2007

HITACHI

Outline

- BWR Design Evolution
- Design Improvements
- Containment
- Nuclear Steam Supply
- Engineered Safety Features
- Safety (Core Damage)
- PRA Summary

BWR Design Evolution

- Operates under saturated conditions
  - Over 40 years of operational experience
  - Operating Pressure is nominally 7.2 MPa (1040 psia) with saturation temperature ~ 287 °C (550 ºF)
  - Direct Cycle
    - » Saturated Steam
    - » Quality at exit is greater than 99.9%
      - Higher than most PWRs
  - Evolution

BWR Design Evolution (cont'd)

- Power is controlled by positioning control rods & varying core flow
  - Flow control in ABWR provides rapid power changes
  - No Boric Acid as moderator
- ABWR* are designed for 100% load rejection without reactor Scram
  - Standard USA ABWR designed for 33% Bypass
  - Can operate in "Island Mode" where licensed

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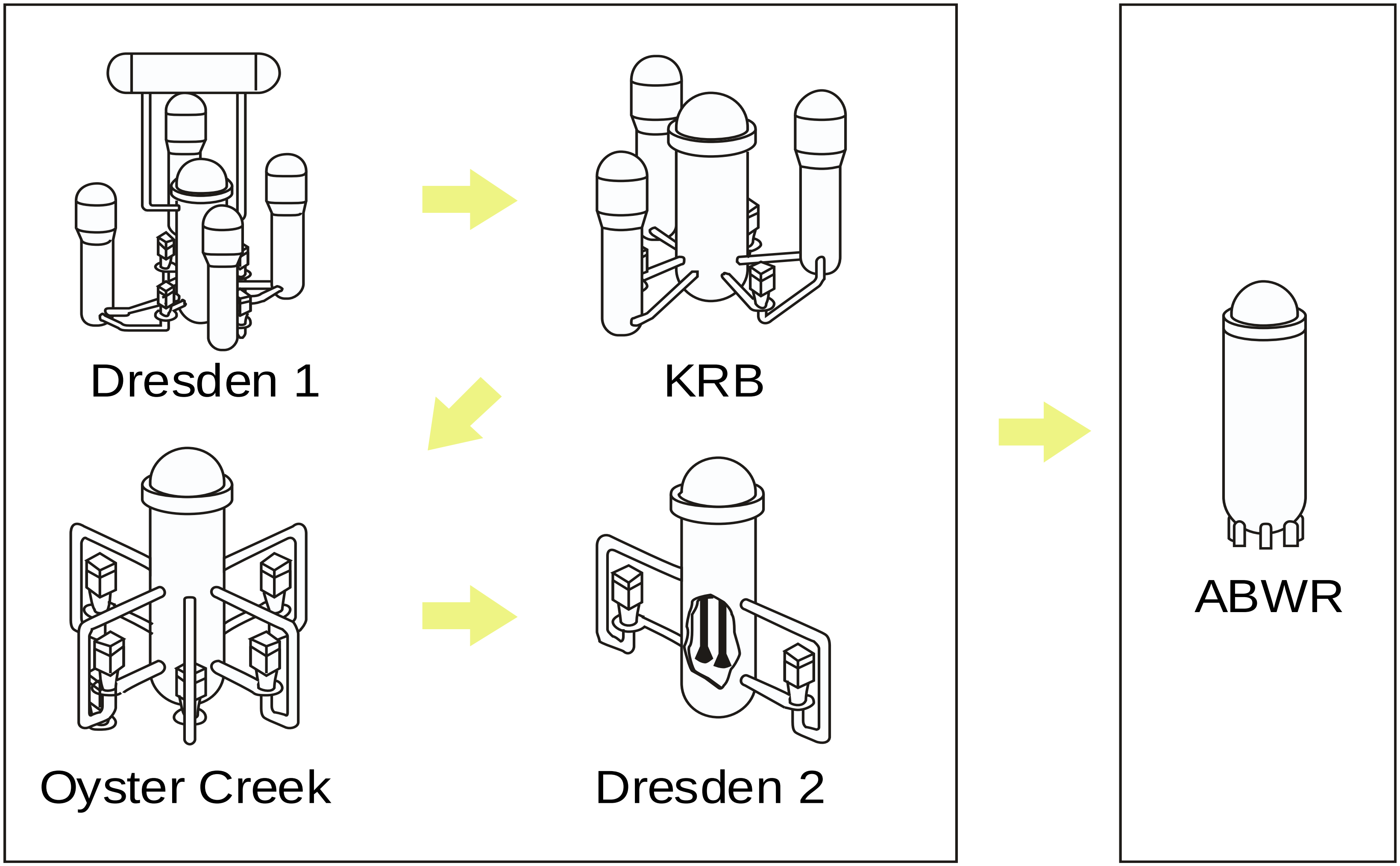
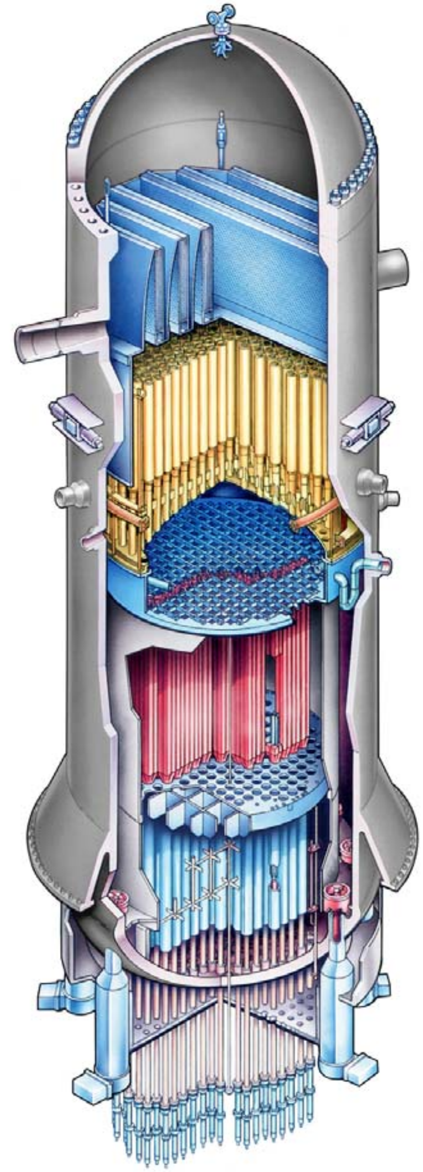
*Lungmen

BWR Evolution



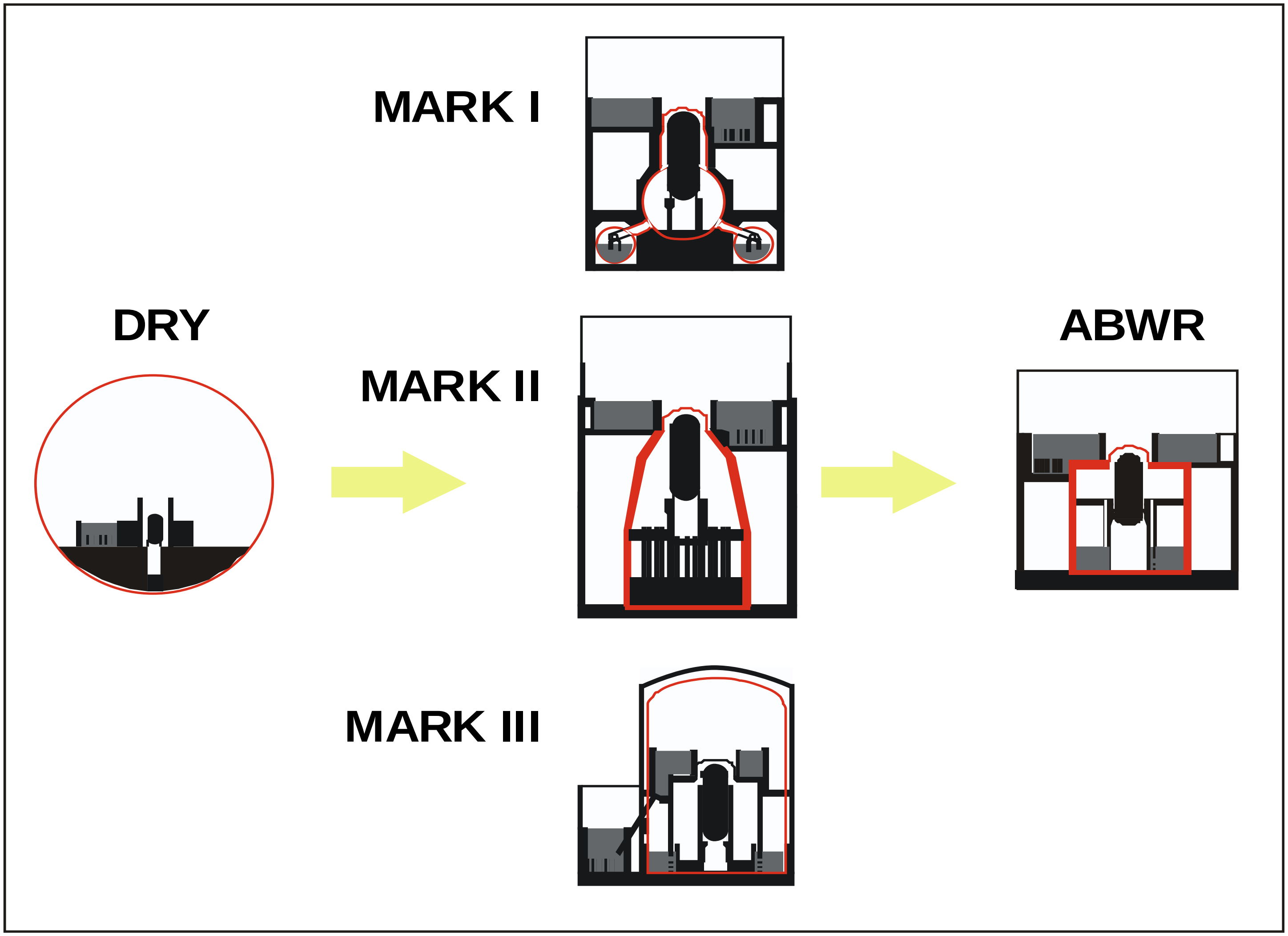
ABWR RPV Assembly



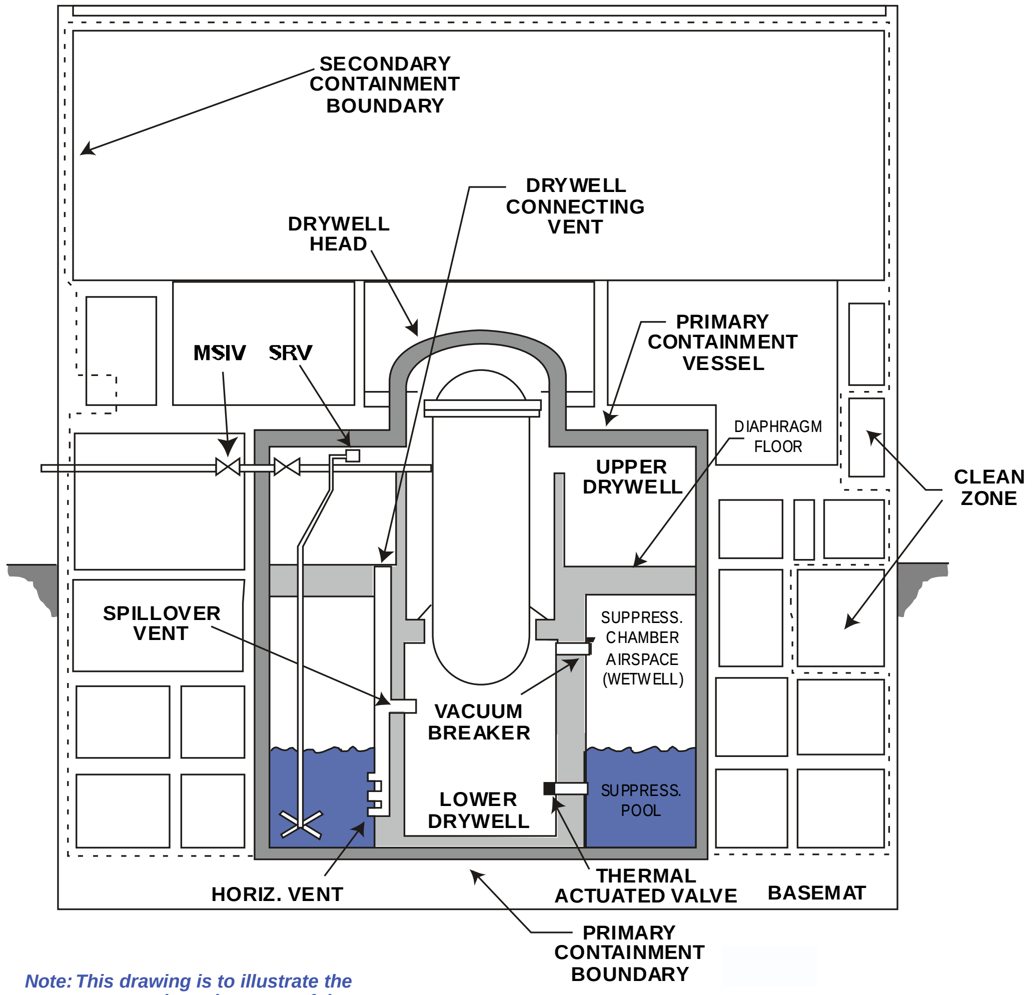
Pressure Suppression Containment

- Reinforced Concrete Containment Vessel
  - Steel Leakage Liner
- Consists of Two Major Elements
  - Drywell
    - » Upper and Lower
  - Wetwell
    - » Suppression pool and airspace
- Inerted with Nitrogen During Operation
- Steam released during accident or transient
  - Routed to Suppression Pool
  - Non-condensable gases are transferred to wetwell airspace

Primary Containment Evolution

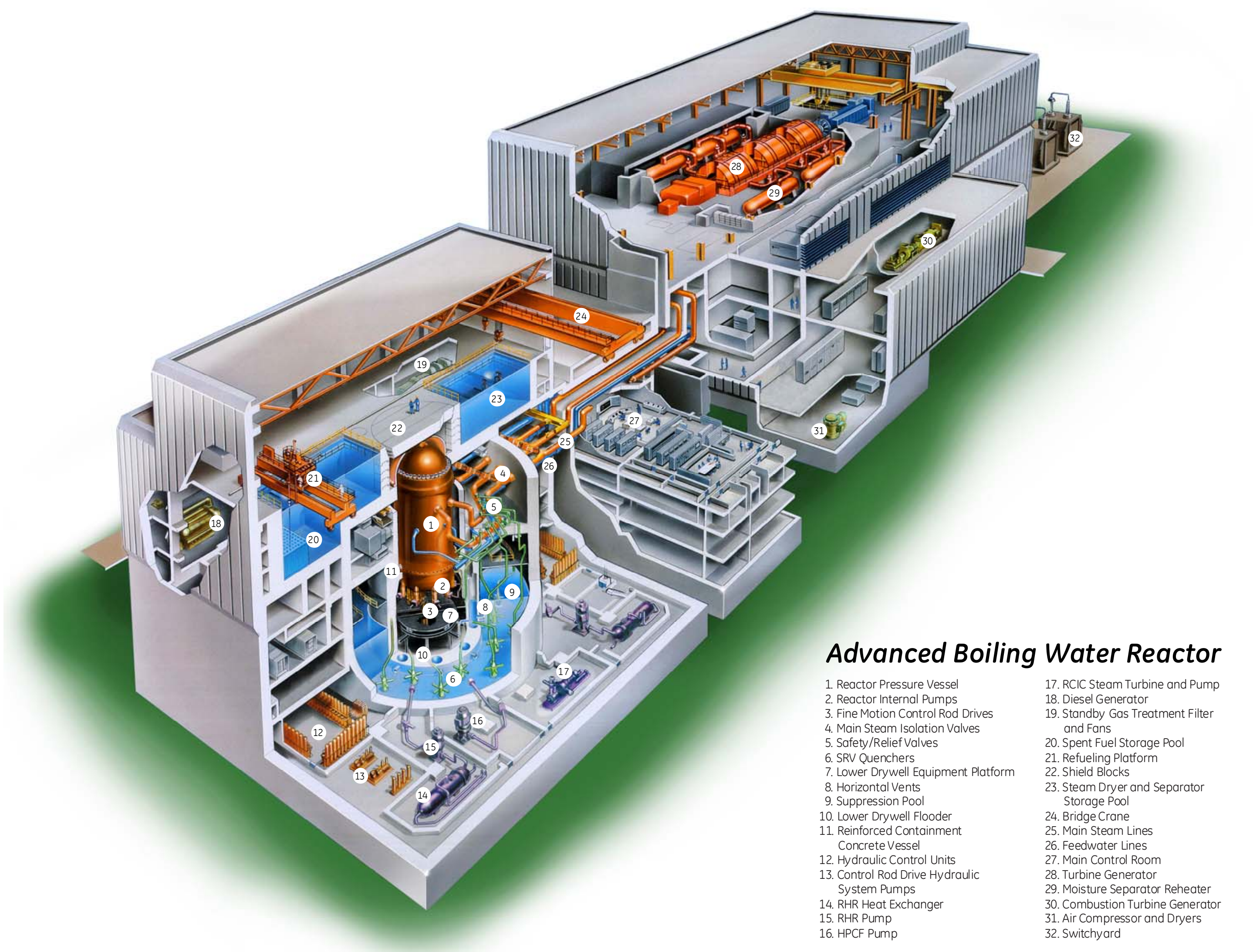


ABWR Reactor Building & Containment



Note: This drawing is to illustrate the scope and requirements of the design and is not intended to show the final detail.

ABWR 3D Cutaway



Advanced Boiling Water Reactor

- Licensed / Certified in 3 Countries
  - First Design Certified by NRC under Part 52
  - Generation III
- Four operating in Japan
- Several more under construction or planned
  - Japan's BWR for foreseeable future
- Power Level(s)
  - 3,926 MWt (1350 MWe net) US Certified
  - 4,300 MWt (1460 MWe net) FIN5 Offering

ABWR Basic Parameters

- 3,926 Megawatt Core Thermal Power
- ~1,365 Megawatt Electric Gross
  – For nominal summer conditions
- Internal Reactor Recirculation Pumps (RIP)
  – No recirculation piping
  – Canned Rotor Pumps
- 3 Divisions Safety Systems
  – At least 72 hours operators hands-off capability

ABWR Design Parameters

- Designed to bound most potential site in United States
  - Based on EPRI URD recommendations
    - » Extreme Wind
    - » Maximum & Minimum Temperature
    - » Seismic 0.3 g (all soils) in US (0.4 g in Taiwan)
    - » Tornado missiles
- Both 60Hz and 50 Hz

ABWR Site Parameters

- Tornado
  - » 483 km/hr (300 mph)
- Extreme Winds for Safety-Related Structures
  - » 197 km/hr (122 mph)
- Temperatures
  - » 0% exceedance
    - Maximum 46.1°C (115°F), 26.7°C (80°F) w.b. coincident (27.2°C; 81°F)
    - Minimum -40°C (-40°F)
  - » 1% exceedance
    - Maximum 37.8°C (100°F), 25.0 °C (77°F) w.b. (26.7°C; 80°F)
    - Minimum –23.3°C (-10°F)

ABWR Site Parameters

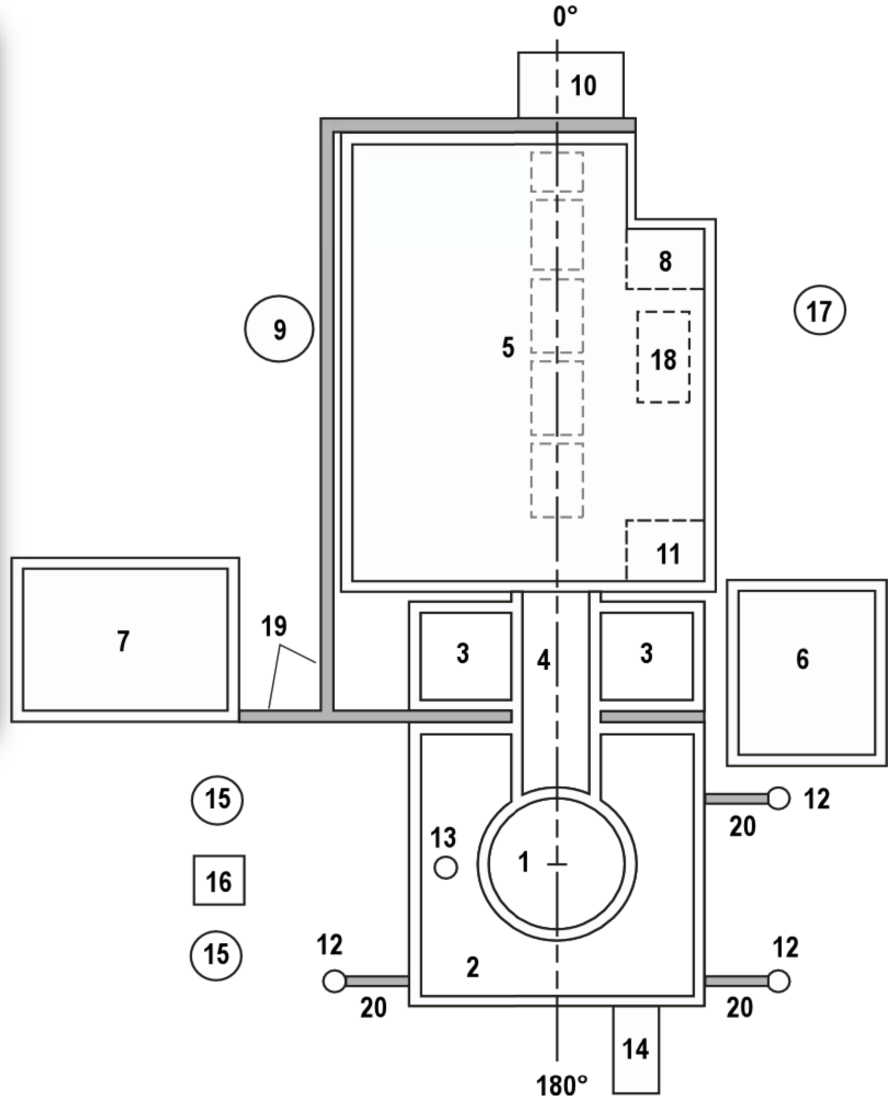
- Soil Bearing Capacity
  - 718 kPa
- Minimum Shear Wave Velocity
  - 300 m/s
- Maximum Site Flood Level
  - 30.5 cm (12 in) below grade
- Maximum ground water level
  - 61 cm (24 in) below grade

Site Specific Design Elements

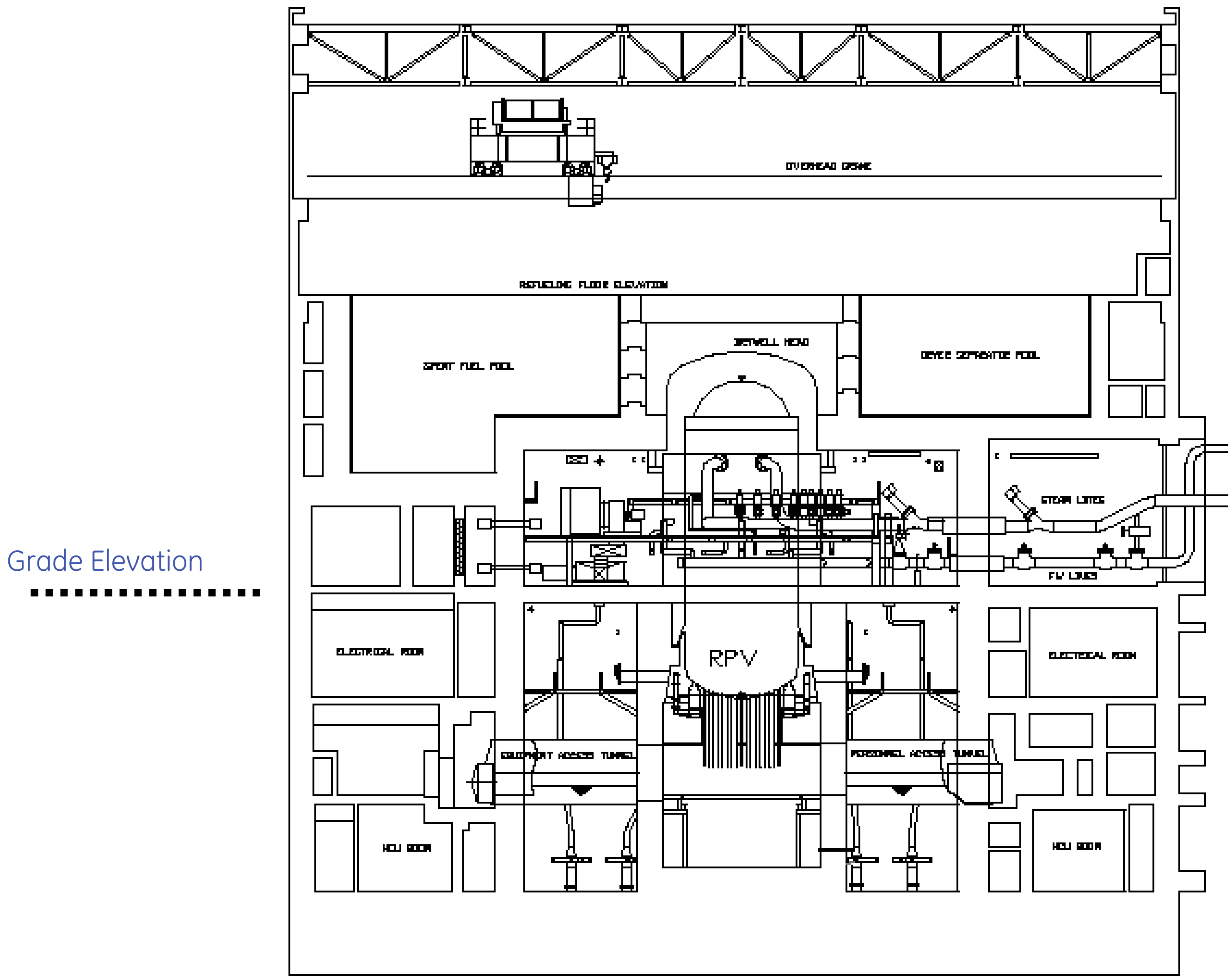
- Circulating Water System (Power Cycle Heat Sink)
- Ultimate Heat Sink
  - Reactor Service Water (RSW)
  - Safety-related
- Off-site electrical
- Make-up water
- Other site works

ABWR Site Plan

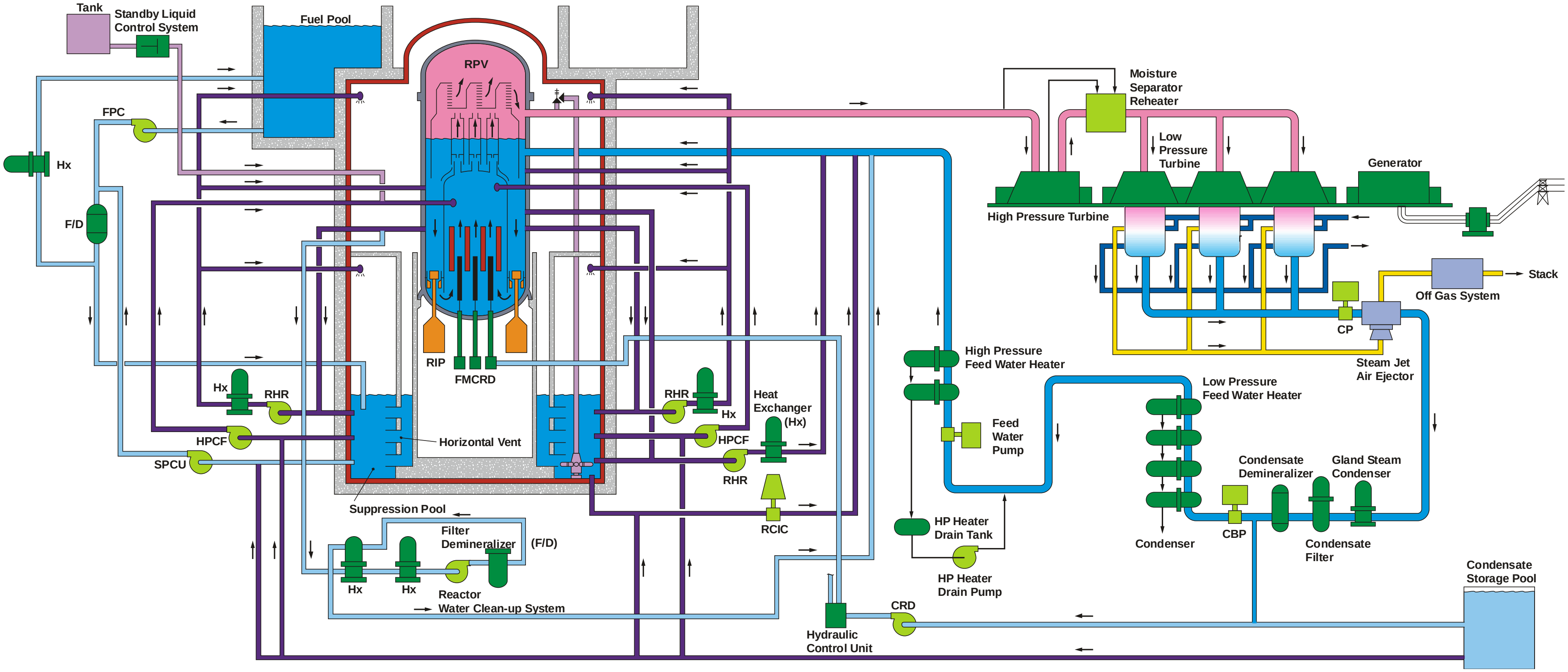
No.	FACILITY
1	REACTOR CONTAINMENT
2	REACTOR BUILDING
3	CONTROL BUILDING
4	MAIN STEAM / FEEDWATER TUNNEL
5	TURBINE BUILDING
6	SERVICE BUILDING
7	RADWASTE BUILDING
8	HOUSE BOILER
9	CONDENSATE STORAGE TANK
10	UNIT AUXILIARY TRANSFORMERS
11	NORMAL SWITCHGEAR
12	DIESEL OIL STORAGE TANK (3)
13	STACK
14	EQUIPMENT ENTRY LOCK
15	FIRE PROTECTION WATER STORAGE TANK (2)
16	FIRE PROTECTION PUMPHOUSE
17	BUNKER FUEL TANK
18	COMBUSTION TURBINE GENERATOR
19	RADWASTE TUNNELS RB, CB, TB
20	DG OIL TRANSFER TUNNEL (3)



ABWR Reactor Building Sectional



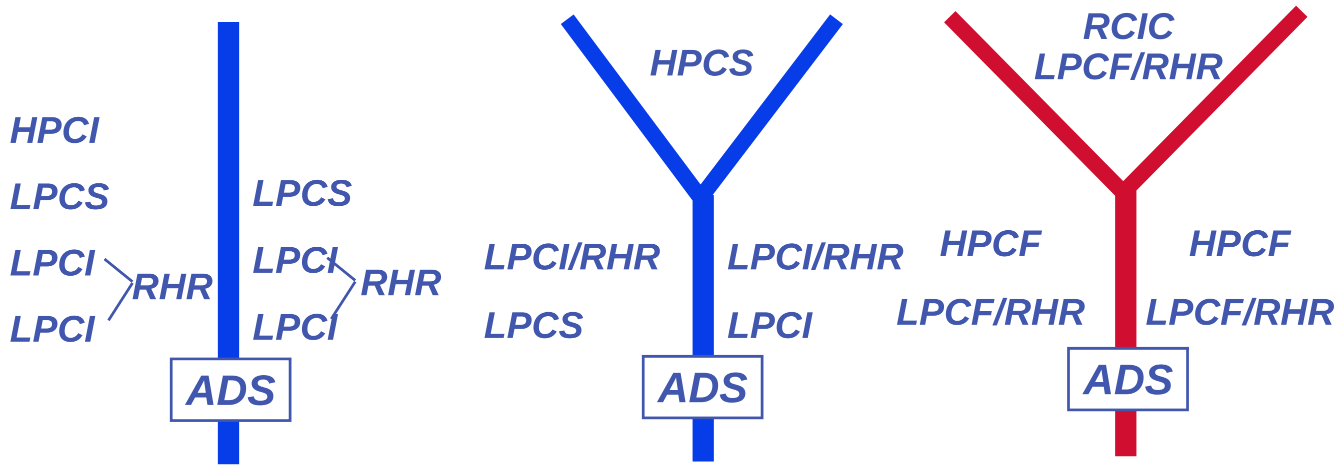
ABWR Overall Flowchart



Emergency Core Cooling

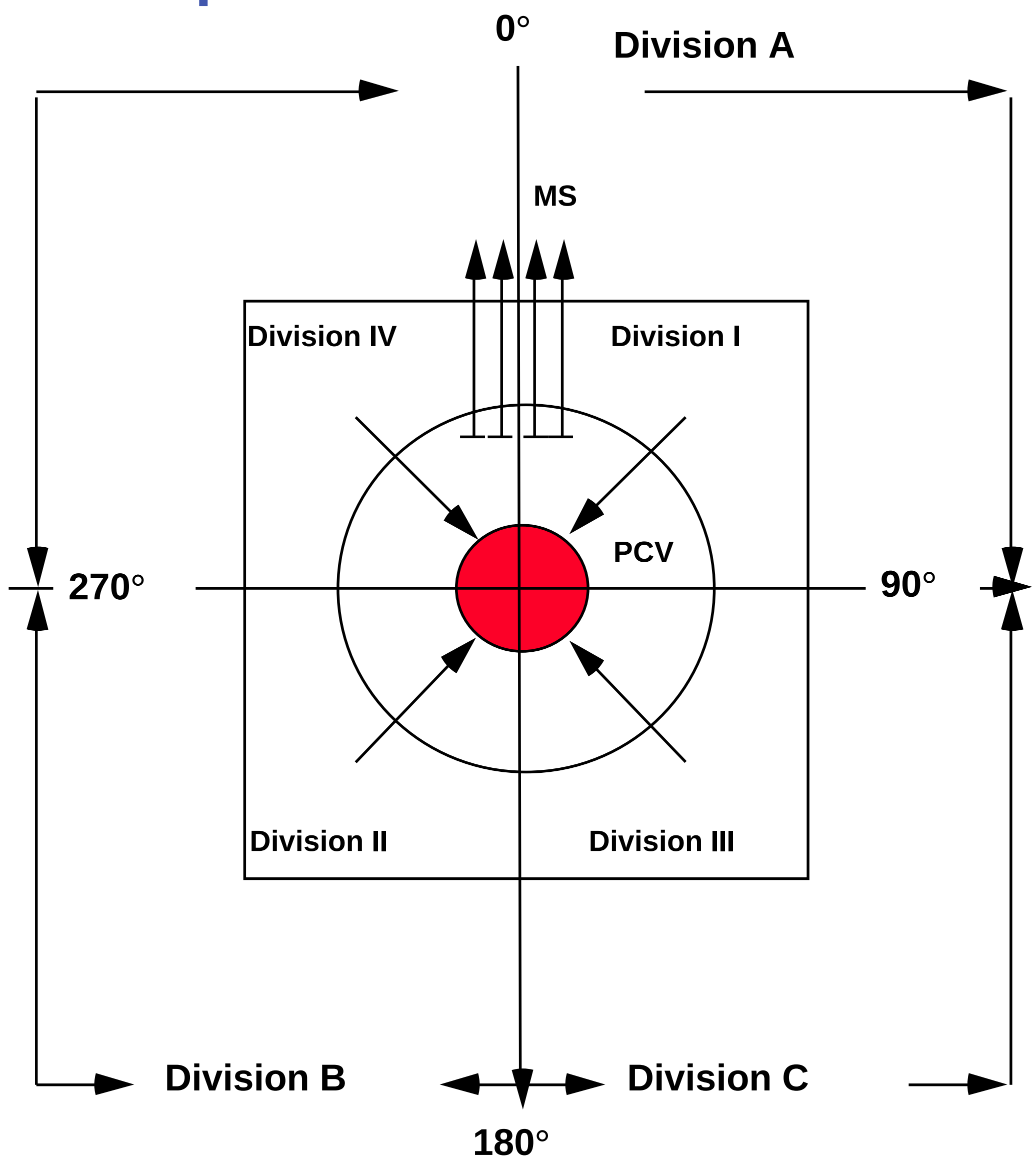
ECCS Systems Evolution

	Typical BWR/4	Typical BWR/5 BWR/6	ABWR



	Typical BWR/4	Typical BWR/5 BWR/6	ABWR
High pressure capacity	*5000*	*1900*	*2800*
Low Pressure capacity	*42000*	*29000*	*19000*
No. of large pipes below core	*12*	*12*	*0*
Peak clad temperature (App K), °F	*1600*	*1100*	*No uncovery*
Operator action	*Required*	*Required*	*Automated >72 hours*

Divisional Separation



Engineered Safety Features

- Redundancy and Diversity
  - Three Divisions each having high & low pressure pumps:
    - » High Pressure
      - Two Motor-driven High Pressure Core Flooder (HPCF)
      - One Steam-driven Reactor Core Isolation Cooling System (RCIC)
    - » Low Pressure
      - Automatic Depressurization System (ADS)
      - Residual Heat Removal
        - » Low Pressure Flooder Mode (LPFL)
        - » Suppression Pool Cooling
        - » Containment Spray

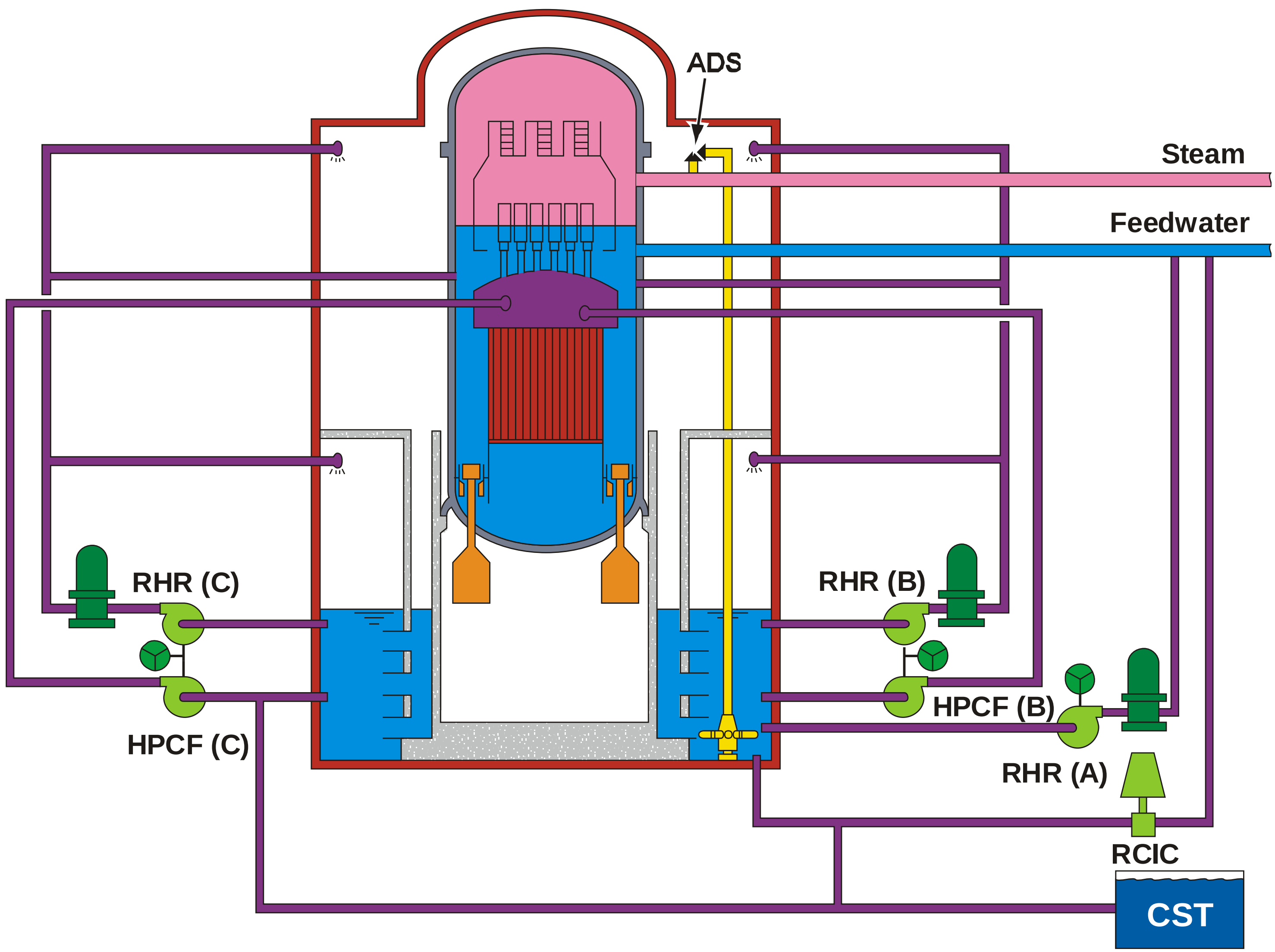
ABWR ECCS Improvements

- Three completely separate mechanical & electrical divisions
  - Core cooling
  - Heat removal
  - Emergency Diesel Generators
- Station BlackOut (SBO) addressed
  - Steam-driven RCIC
  - Combustion turbine-generator
  - Fire system cross-tie
- Automation of Suppression Pool cooling function
  - Heat exchangers always in the loop

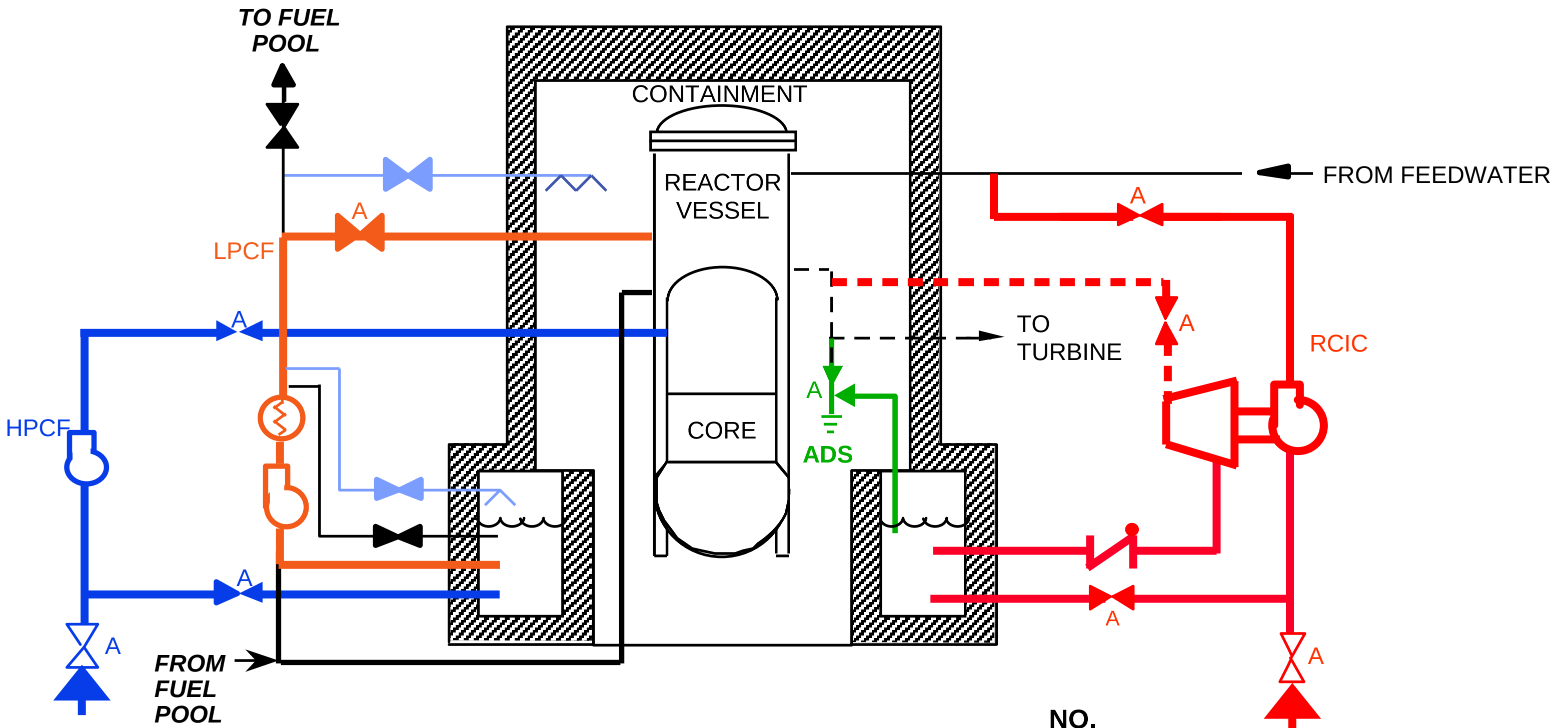
ABWR ECCS Improvements (cont'd)

- Elimination/transfer of complex modes
  - Reduced valves, pipes by one-third
- Significant capacity reduction
- Greatly reduced duty during transients
  - N-2 Capability at high pressure
- Improved small break response
  - Reduced needs for ADS
- No fuel uncovery for any pipe break
- Low pressure piping/equipment design pressure raised to 40% of operating pressure to resolve ISLOCA concerns

ABWR ECCS

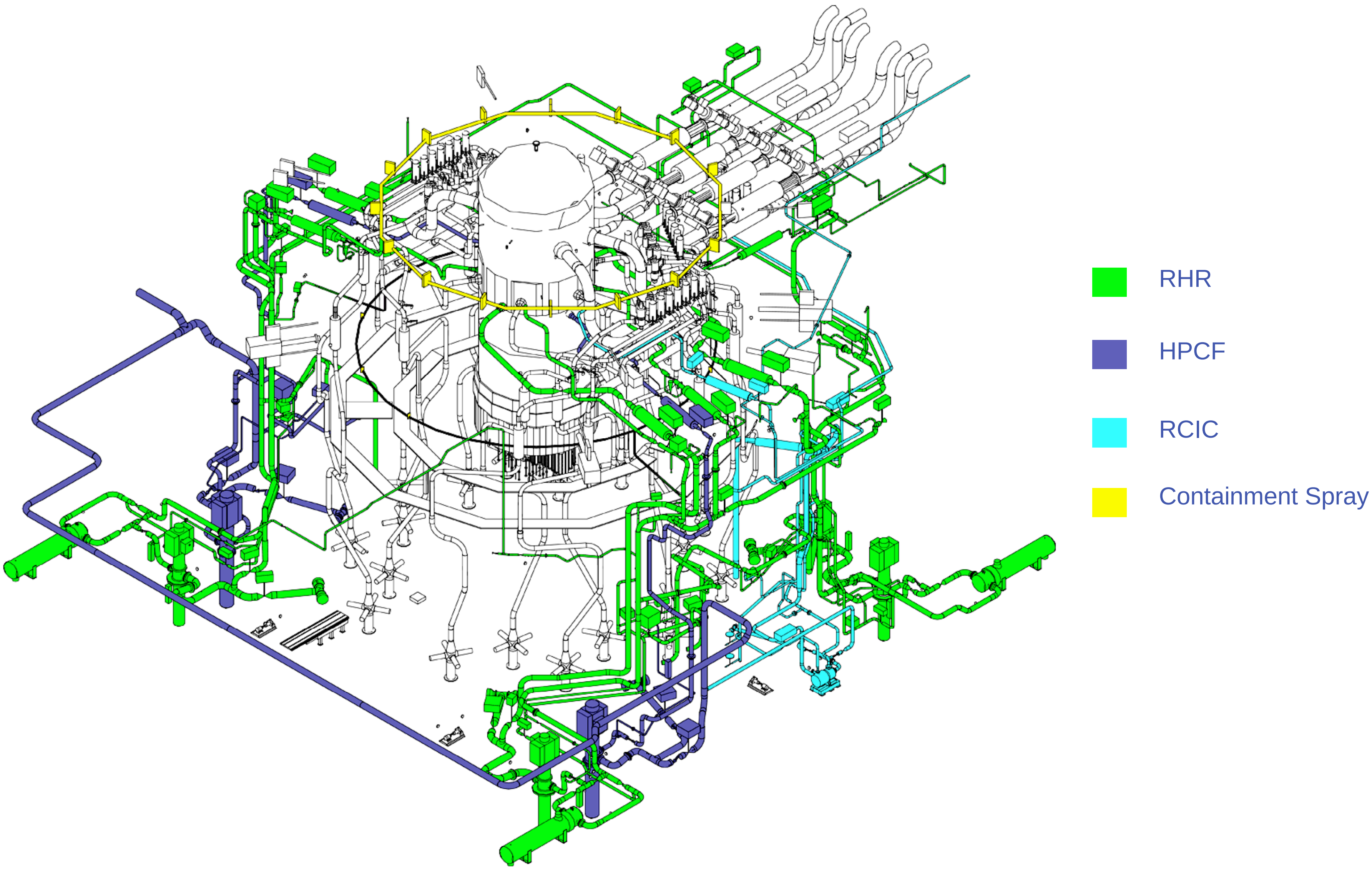


ABWR Emergency Core Cooling Systems



FUNCTION	TYPE	NO. DIVS.
• **HIGH PRESSURE CORE FLOODER**	AUTO	2
• **REACTOR CORE ISOLATION COOLING**	AUTO	1
• **AUTOMATIC DEPRESSURIZATION SYS.**	AUTO	2
• **LOW PRESSURE CORE FLOODER**	AUTO	3
• SUPPRESSION POOL COOLING	AUTO	3
• WETWELL SPRAY	MAN	2
• DRYWELL SPRAY	MAN	2
• **SHUTDOWN COOLING**	MAN	3
• FUEL POOL COOLING SUPPORT	MAN	2

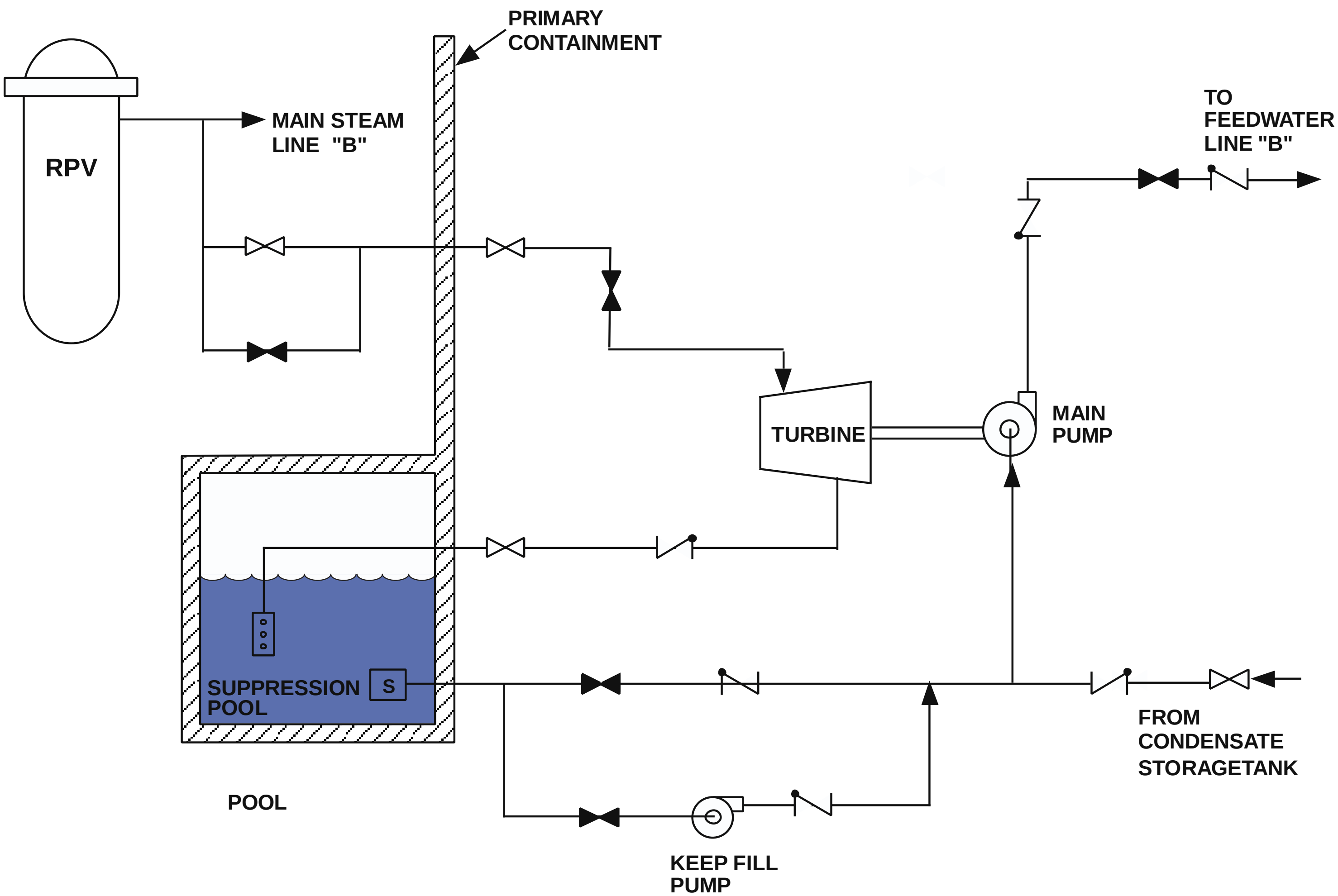
ABWR ECCS Piping



Reactor Core Isolation Cooling (RCIC)

- **Purpose:** Provide makeup water to RPV when it's isolated from FeedWater (FW) system. Also part of ECCS.
- Steam-driven High Pressure Pump
  - Flow is ~182 m^3 per hour (800 gpm)
    - » Provides sufficient makeup on loss of FW without need for any other makeup system
    - » Auto initiates at RPV Water Level 2
- AC independent system
  - Batteries for electrical operation
  - Steam for motive power
- Mitigates Station BlackOut (SBO) events
- 2 water sources
  - Suppression Pool (safety)
  - Condensate Storage Tank (preferred)

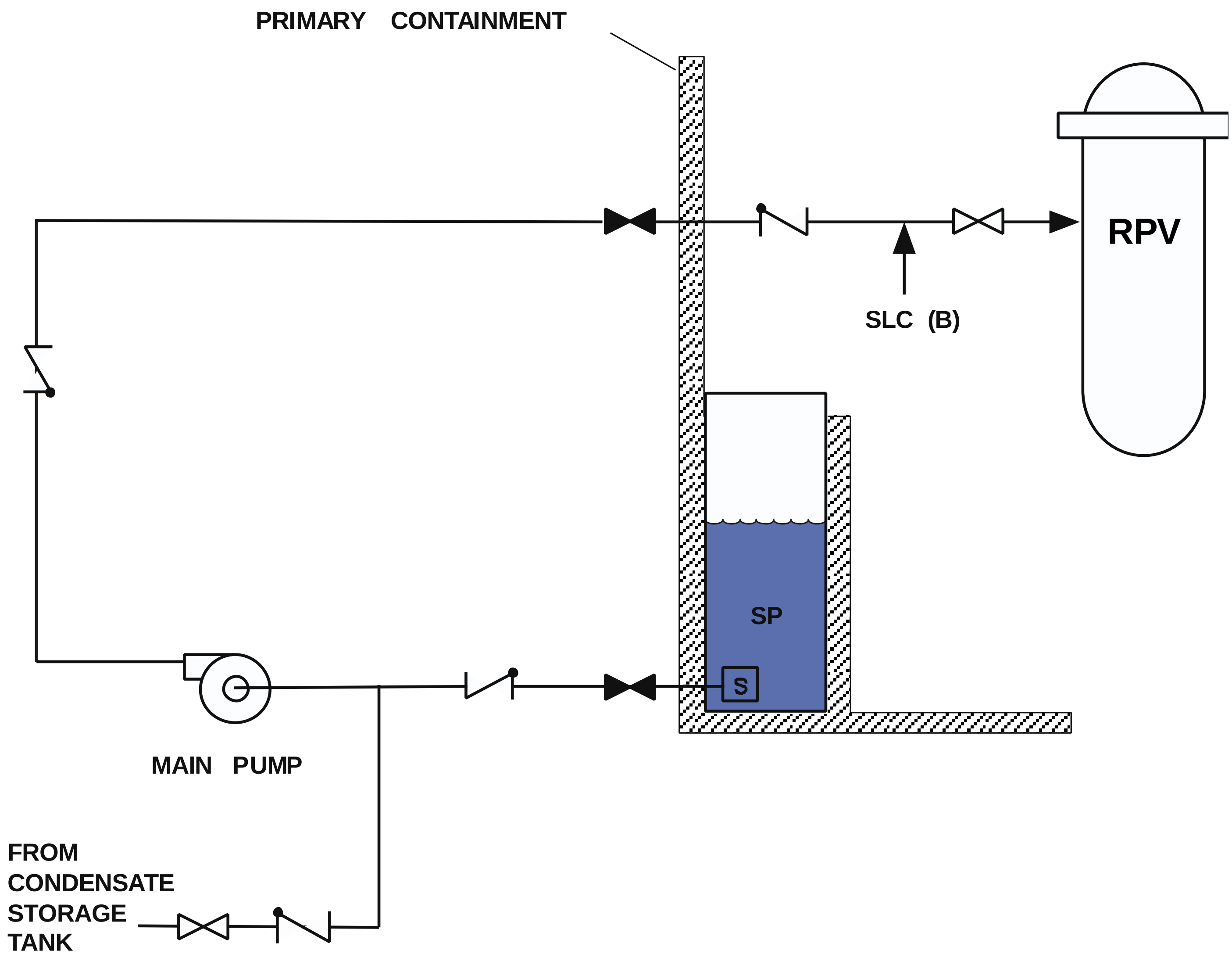
ABWR RCIC


PRIMARY
CONTAINMENT
MAIN STEAM
LINE "B"
RPV
TO
FEEDWATER
LINE "B"
TURBINE
MAIN
PUMP
SUPPRESSION
POOL
S
FROM
CONDENSATE
STORAGETANK
POOL
KEEP FILL
PUMP

High Pressure Core Flooder (HPCF)

- 2 Motor-driven High Pressure Pumps
  - Flow is ~ 182 m^3 per hour (800 gpm) at rated pressure
    - » Backs up RCIC for level transients
    - » Auto initiates at RPV Water Level 1.5
  - Flow is 727 m^3 per hour (3200) when vessel is depressurized
    - » Single pump operating ensures <u>no</u> core damage
- 2 water sources
  - Suppression Pool (safety)
  - Condensate Storage Tank (preferred)

ABWR HPCF


PRIMARY CONTAINMENT
RPV
SLC (B)
SP
S
MAIN PUMP
FROM
CONDENSATE
STORAGE
TANK

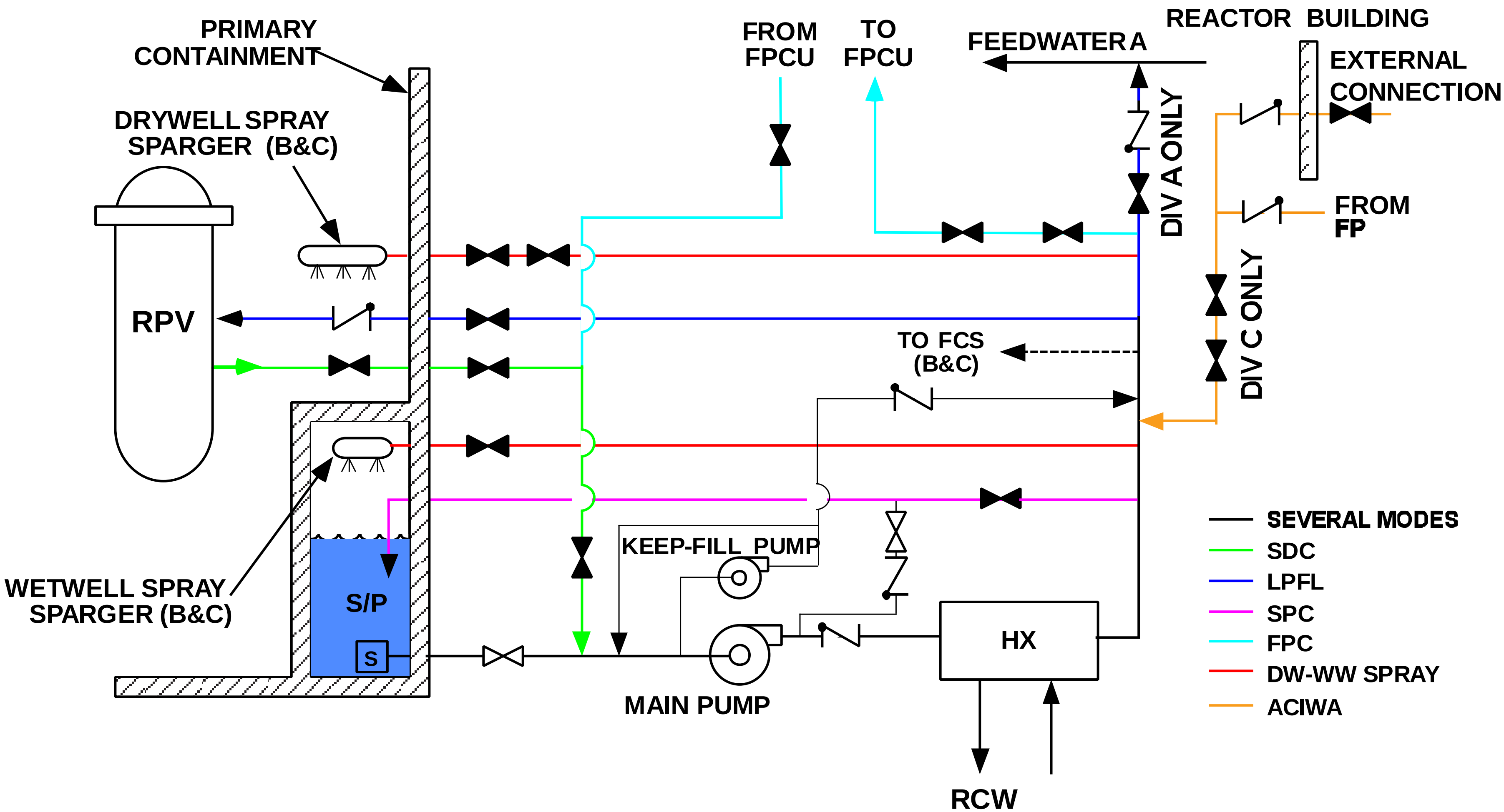
Residual Heat Removal (RHR)

- Six Different Modes of Operation
  - Safety-related modes
    - » Low Pressure Flooder (LPFL)
    - » Suppression Pool Cooling
    - » Containment Spray
  - Non-safety
    - » Shutdown Cooling
    - » Fuel Pool Cooling Support
    - » AC Independent Water Addition (Fire Water)

Residual Heat Removal (cont'd)

- Recirculates & cools water inside Primary Containment
- 3 Motor-driven Low Pressure Pumps
  - Flow is 954 m^3 per hour (4200 gpm) when vessel is depressurized
    - » Single pump operating ensures no core damage
- 1 water source
  - Suppression Pool (safety)

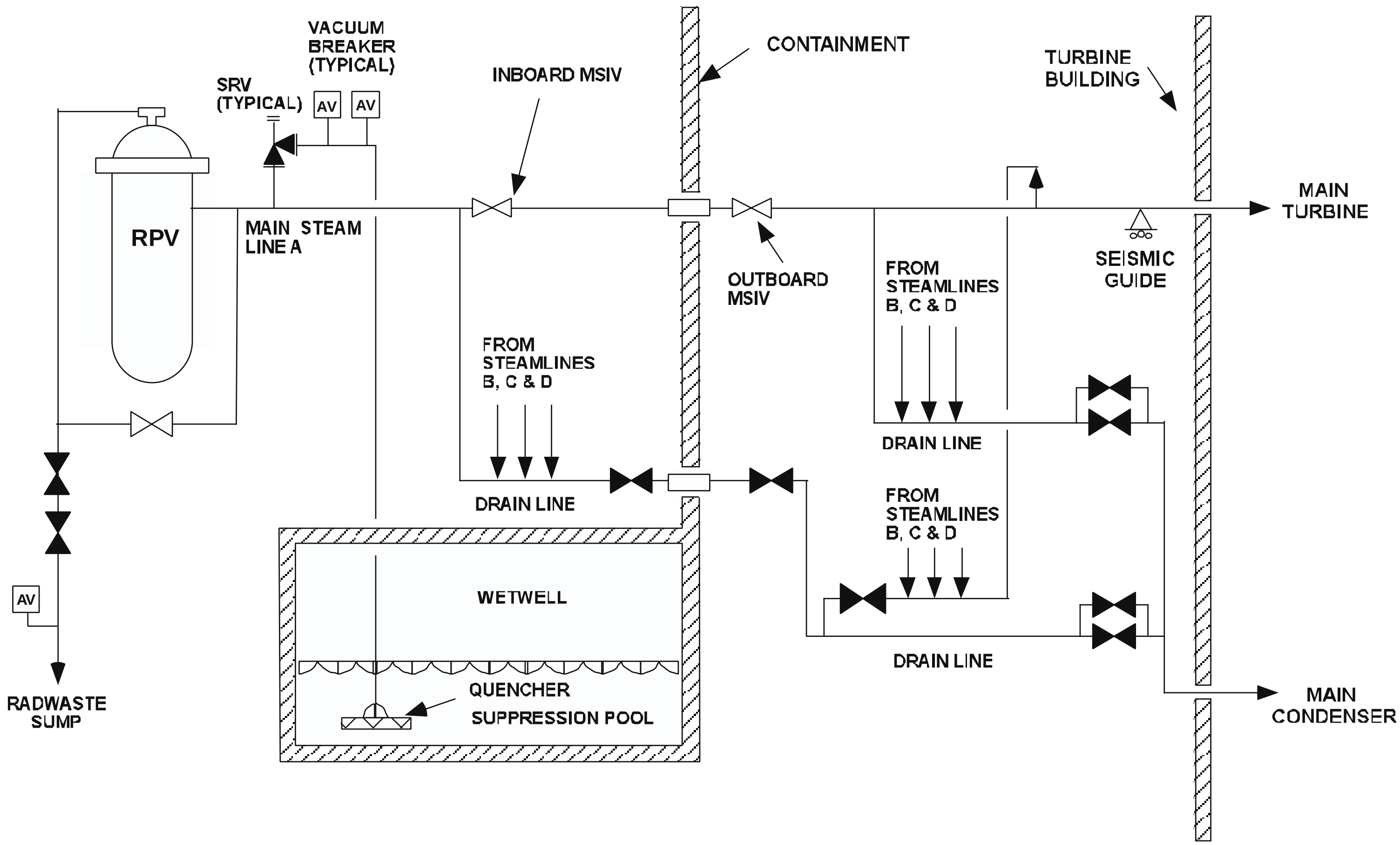
ABWR RHR


PRIMARY CONTAINMENT
DRYWELL SPRAY SPARGER (B&C)
RPV
WETWELL SPRAY SPARGER (B&C)
S/P
S
FROM FPCU
TO FPCU
FEEDWATER A
REACTOR BUILDING
EXTERNAL CONNECTION
DIV A ONLY
FROM FP
DIV C ONLY
TO FCS (B&C)
KEEP-FILL PUMP
MAIN PUMP
HX
RCW
SEVERAL MODES
SDC
LPFL
SPC
FPC
DW-WW SPRAY
ACIWA

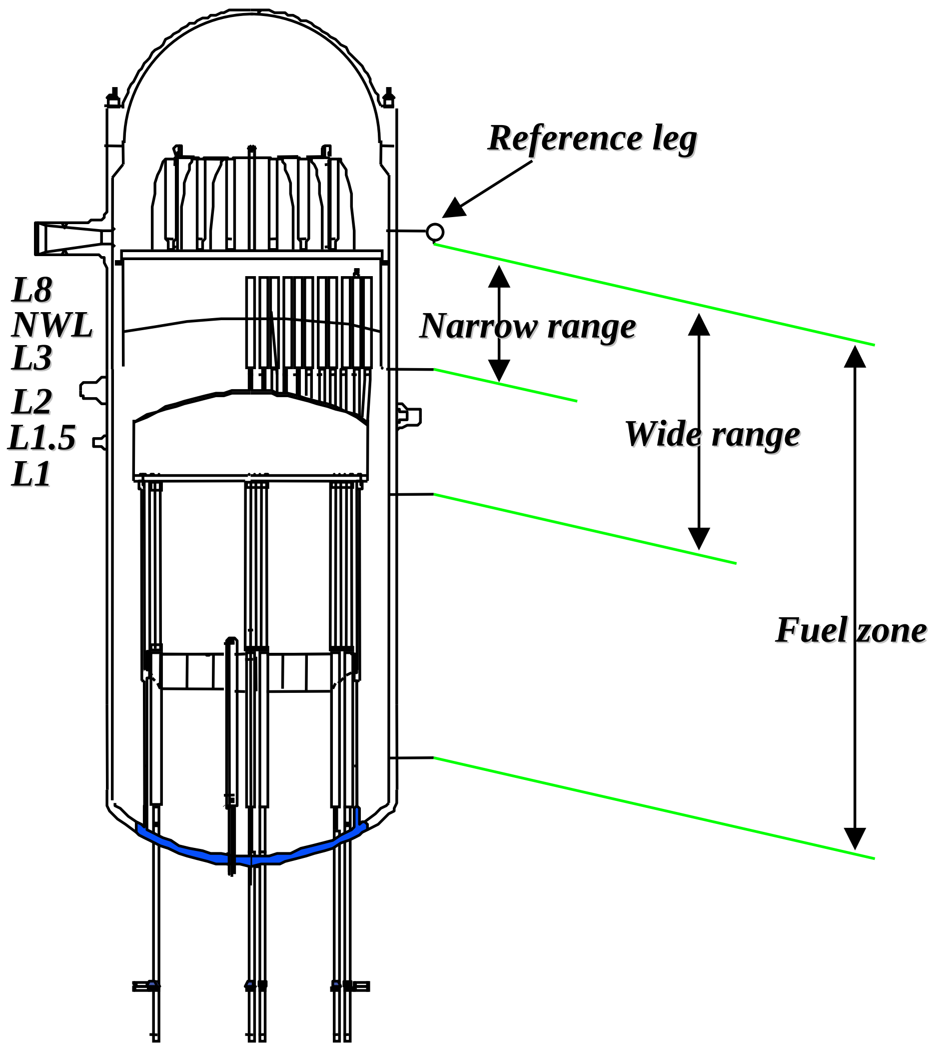
Automatic Depressurization System (ADS)

- 8 of 18 Safety Relief Valves (SRVs)
  - 2 SRVs on each Main Steam Line
  - Each SRV blowdowns to quencher in Suppression Pool
    - » Spring Safety mode for code pressure protection
    - » Externally actuated for Relief mode
      - Pressure transient mitigation

Automatic Depressurization System

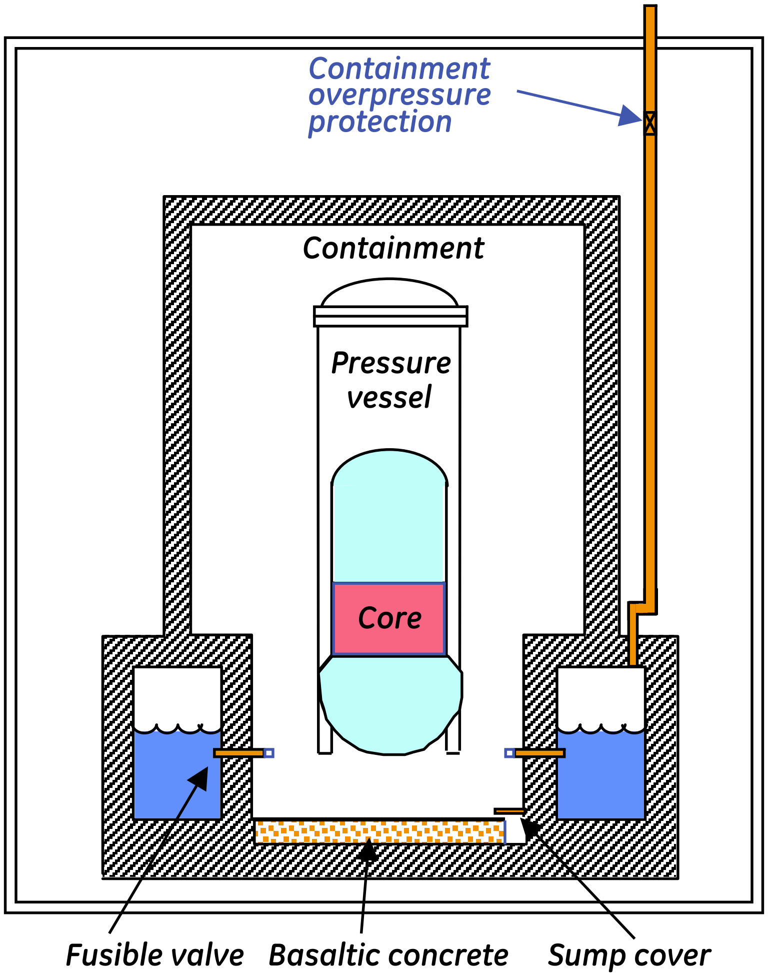


BWR Water Level Measurement



- **L8 - Turbine trip, MSIV close**
- **L3 - Scram**
- **L2 - RCIC start**
- **L1.5 - HPCF start**
- **L1 - Remaining ECCS start (i.e., LPFL, ADS)**

ABWR Passive Severe Accident Mitigation Features



ABWR passive features which mitigate severe accidents:

- *Inerted containment*
- *Lower drywell flood capability*
- *Lower drywell special concrete and sump protection*
- *Suppression pool - fission products scrubbing and retention*
- *Containment overpressure protection*

High degree of public protection

Reactor Building Cooling Water (RCW)
Reactor Building Service Water (RSW)

- **RCW Purpose:** Provide cooling to various systems in Nuclear Island

- **RSW Purpose:** Transfer heat from RCW HXs to Ultimate Heat Sink

Reactor Building Cooling Water (RCW)
Reactor Building Service Water (RSW)

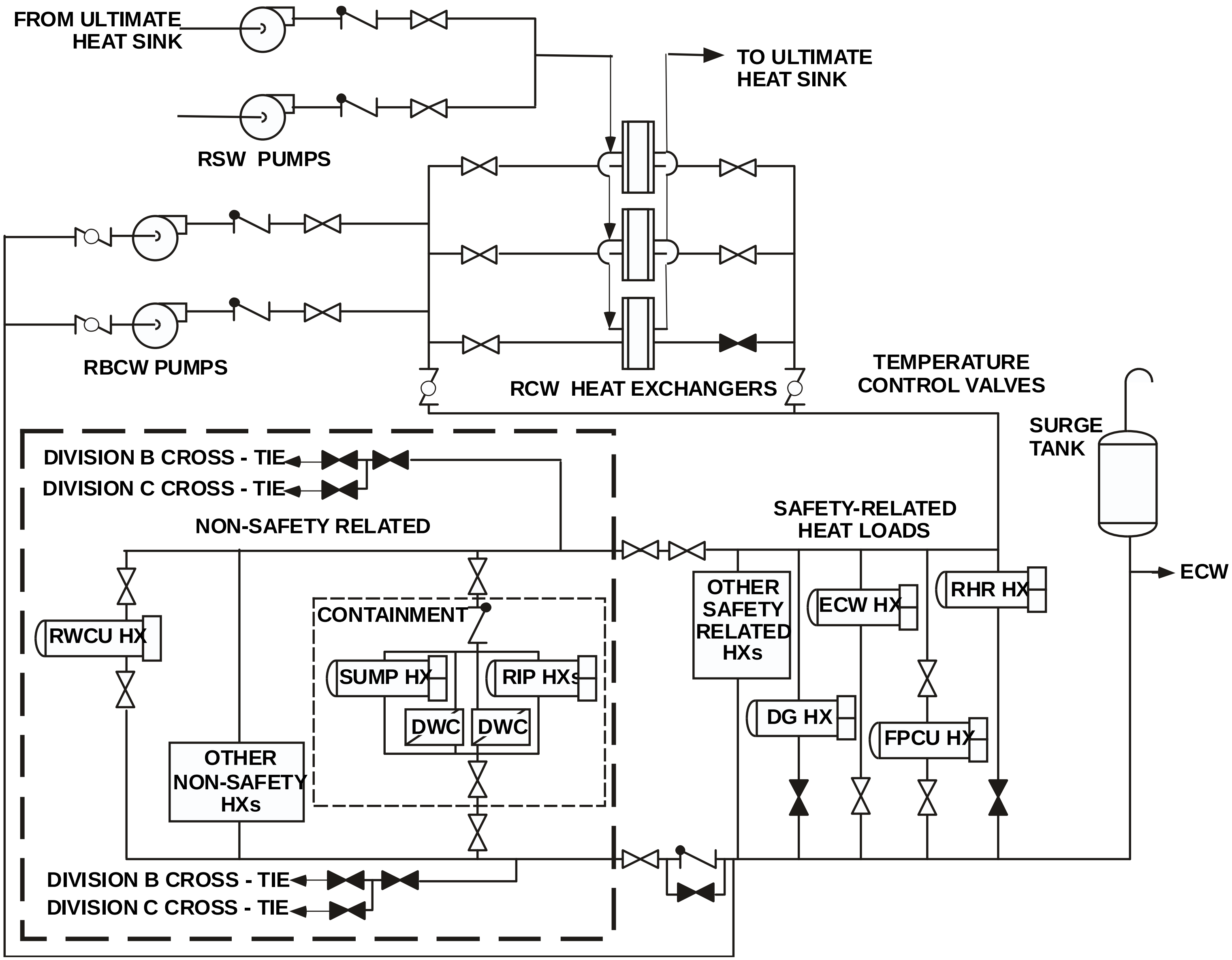
RCW

- Three separate safety divisions cool:
  - ECCS, EDGs, HVAC Emergency Chilled Water (HECW)
  - Non-safety systems: RIPs, RWCU, FPCU, DWC, etc.
    - » Isolated on LOCA Signal
  - Each division has HXs & two 50% Pumps
    - » Normally One Pump Operation
    - » 2nd Pump Auto Starts on LOCA Signal

RSW

- Each division has HXs & two 100% Pumps
- Flat Plate HXs for easier maintenance & better performance

ABWR RCW - RSW

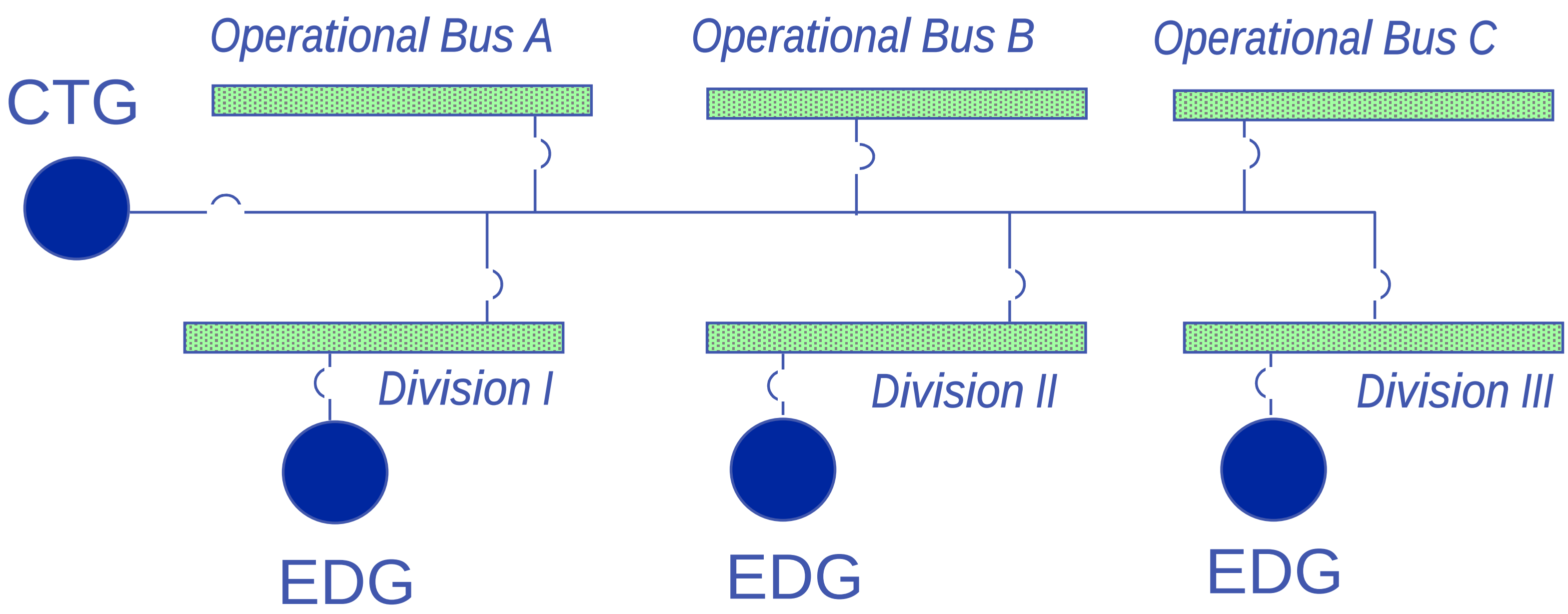


FROM ULTIMATE HEAT SINK
TO ULTIMATE HEAT SINK
RSW PUMPS
RBCW PUMPS
RCW HEAT EXCHANGERS
TEMPERATURE CONTROL VALVES
SURGE TANK
DIVISION B CROSS - TIE
DIVISION C CROSS - TIE
NON-SAFETY RELATED
SAFETY-RELATED HEAT LOADS
ECW
RWCU HX
CONTAINMENT
SUMP HX
RIP HXs
DWC
DWC
OTHER NON-SAFETY HXs
OTHER SAFETY RELATED HXs
ECW HX
RHR HX
DG HX
FPCU HX
DIVISION B CROSS - TIE
DIVISION C CROSS - TIE

ABWR On-Site AC Power

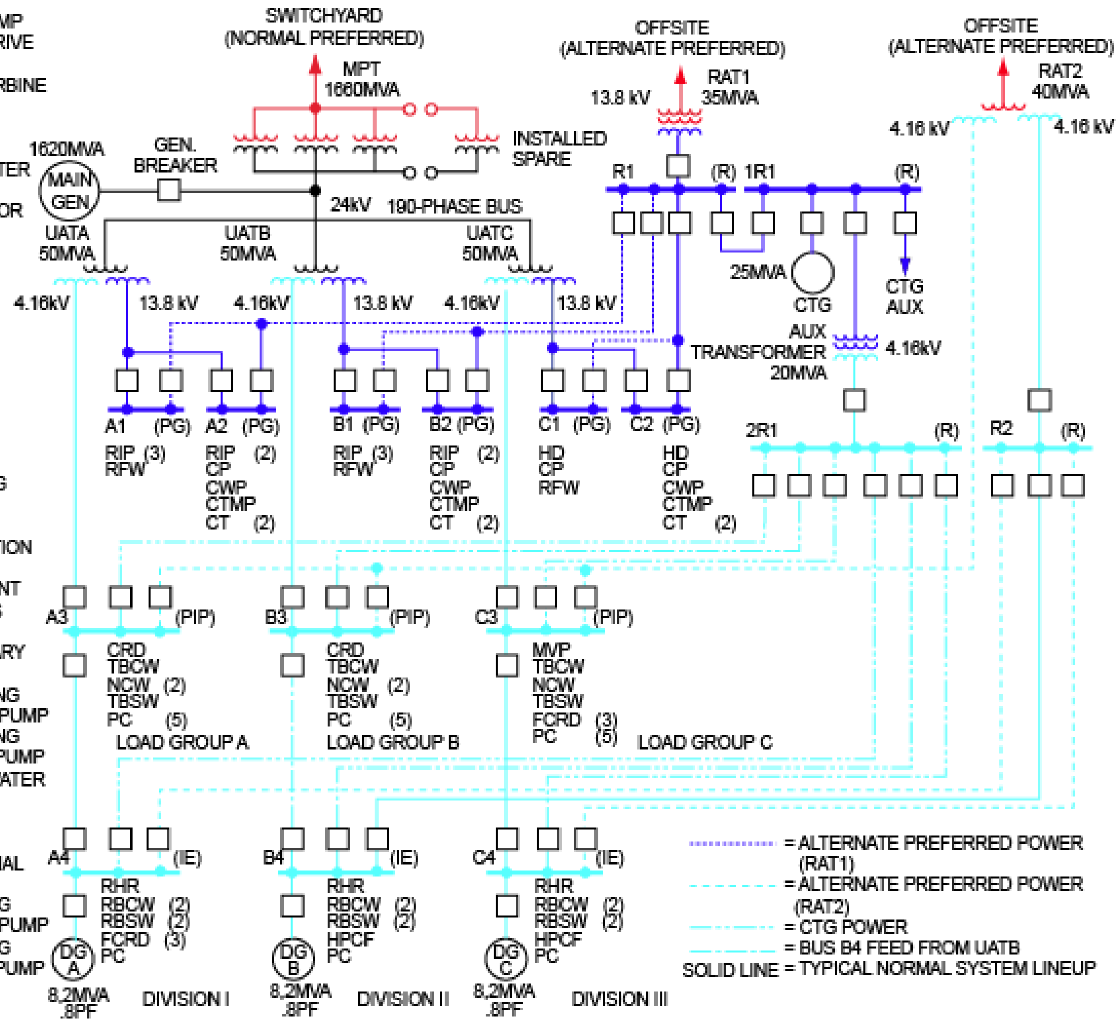
- Three (3) Safety-related Diesel Generators (EDG)
  - One (1) per division
  - ~7 MWe each
- One Combustion Turbine Generator
  - ~20 MWe
  - For the purposes of Station BlackOut (SBO) rule (10 CFR 50.63, CTG is classified as an Alternate AC Power Supply
  - Automatically starts
    - » Connects to PIP Busses
    - » Can be connected to the Safety-Related Busses

Alternate AC Power Supply



ABWR Electrical Distribution

CP = CONDENSATE PUMP
CRD = CONTROL ROD DRIVE PUMP
CTG = COMBUSTION TURBINE GENERATOR
CWMP = COOLING WATER MAKEUP PUMP
CWPH = CIRCULATING WATER PUMP HOUSE
DG = DIESEL GENERATOR
FMCRD = FINE MOTION CONTROL ROD DRIVE
FP = FIRE PUMP
HD = HEATER DRAIN PUMP
HPCF = HIGH PRESSURE CORE FLOODER PUMP
(IE) = CLASS IE BUS
MPT = MAIN POWER TRANSFORMER
NCW = NORMAL COOLING WATER CHILLER
PC = POWER CENTER
(PG) = POWER GENERATION BUS
(PIP) = PLANT INVESTMENT PROTECTION BUS
(R) = RESERVE BUS
RAT = RESERVE AUXILIARY TRANSFORMER
RBCW = REACTOR BUILDING COOLING WATER PUMP
RBSW = REACTOR BUILDING SERVICE WATER PUMP
RFW = REACTOR FEEDWATER PUMP
RHR = RESIDUAL HEAT REMOVAL PUMP
RIP = REACTOR INTERNAL PUMP
TBCW = TURBINE BUILDING COOLING WATER PUMP
TBSW = TURBINE BUILDING SERVICE WATER PUMP
UAT = UNIT AUXILIARY TRANSFORMER



Standby Liquid Control (SLCS)

- **Purpose:** backup to Control Rods to bring & maintain core sub-criticality (Cold Shutdown)
- Two 100% Motor-driven Positive Displacement High Pressure Pumps
  - Injects liquid neutron poison into RPV
    - » Sodium Pentaborate (enriched is optional)
    - » Enters RPV via HPCF B
- Either Control Rods or SLCS ensure reactor shutdown at cold conditions
- Reactor Water CleanUp system (RWCU) automatically isolates

SLCS Reactivity Requirements

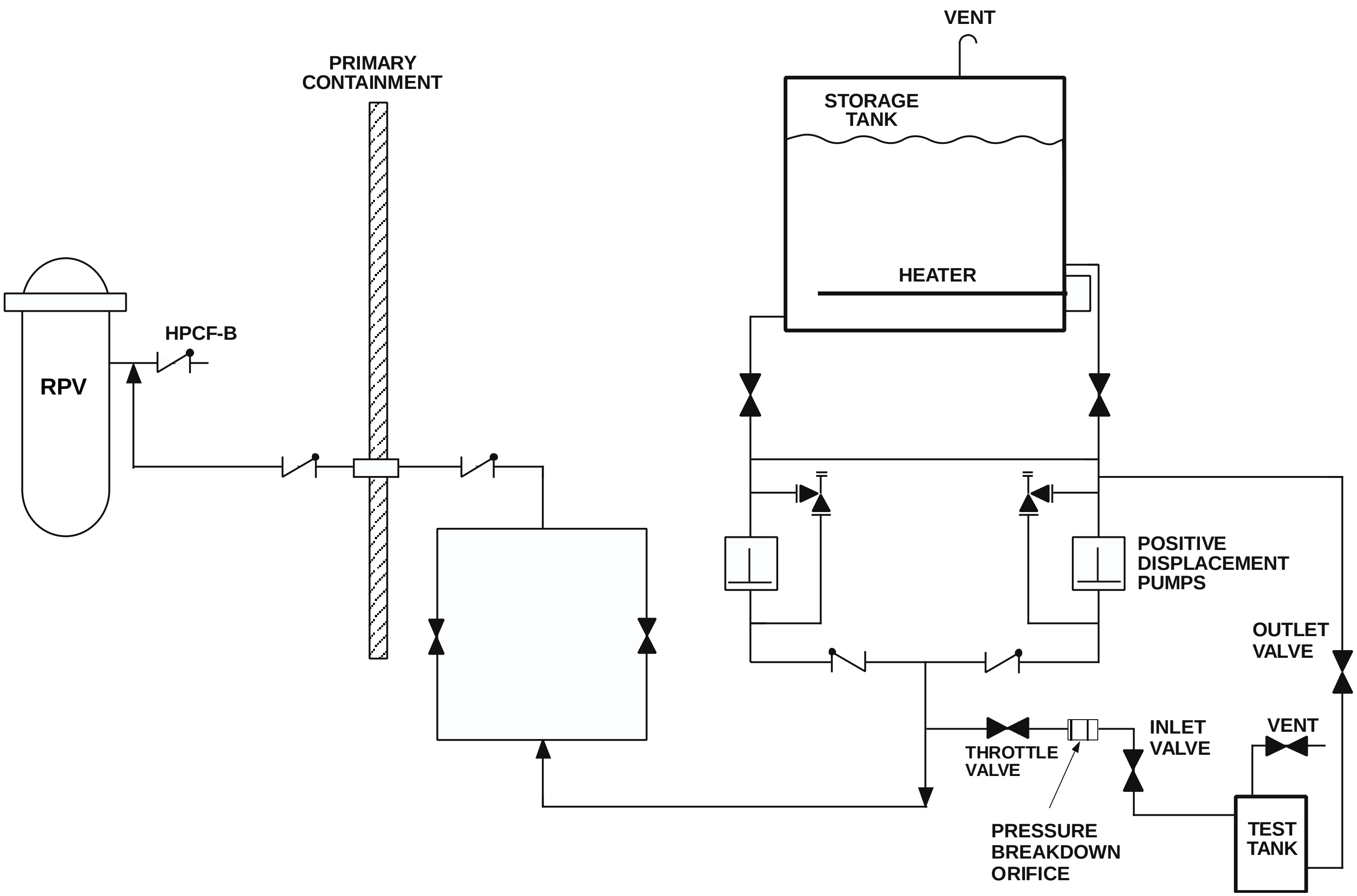
- To shutdown Rx with all Control Rods withdrawn.
- Must have enough negative reactivity to overcome:
  - Elimination of all steam Voids
  - Cool temperatures (~51.7°C; ~125 °F; water more dense & reduced Doppler effects)
  - Xenon free conditions
  - Dilution (to Residual Heat Removal (RHR) system)
  - Shutdown margin requirements

SLCS Initiations

- Manual from Main Control Room
  - Keylock switch for each division
- Automatic: Both divisions automatically initiate if Anticipated Transient Without Scram (ATWS) signal received
  - ATWS Signal: any of following conditions with 2 of 4 logic:
    - » High RPV Pressure (1125 psi); or low RPV water level (Level 2); or manual ARI/FMCRD run-in

    and
    - » Startup Range Neutron Monitor (SRNM) ATWS Permissive signal (i.e., 6% RTP or higher) for 3 minutes

ABWR SLCS


VENT
PRIMARY CONTAINMENT
STORAGE TANK
HEATER
HPCF-B
RPV
POSITIVE DISPLACEMENT PUMPS
OUTLET VALVE
INLET VALVE
VENT
THROTTLE VALVE
PRESSURE BREAKDOWN ORIFICE
TEST TANK

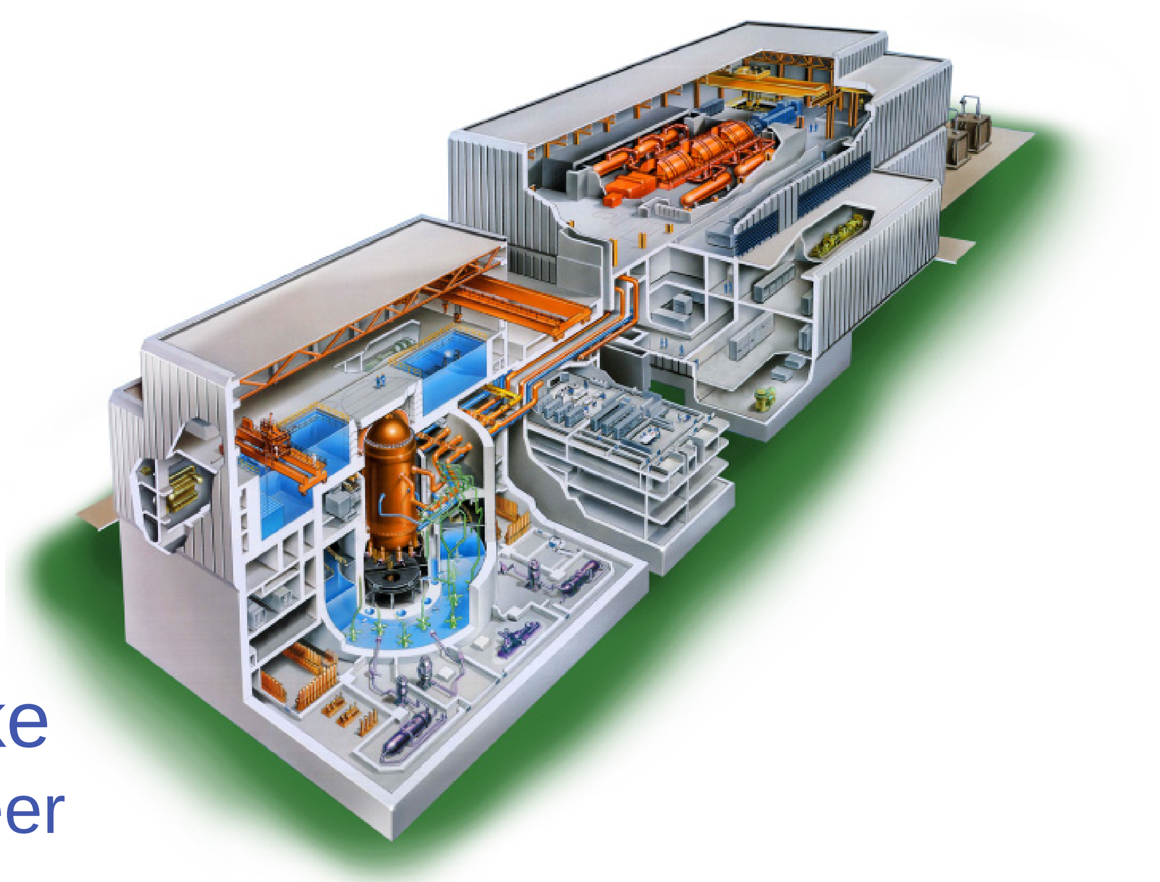
ABWR Safety Challenges Reduced

ATWS challenges reduced

- Prevention
  - Accumulator-driven Scram without Scram Discharge Volume
  - Alternate Rod Insertion (ARI)
    - » Diverse logic for Scram function
  - FMCRD electric run-in
- Automated mitigation
  - Recirculation pump trip (RPT)
    - » 6 on water level 2
    - » 4 on high reactor pressure or water level 3
    - » All on any scram or ARI
  - Feedwater runback
    - » High reactor pressure and SRNM ATWS permissive for 2 minutes
  - Boron injection

ABWR PRA Overview

Advisory Committee on Reactor Safeguards



Dennis W. Henneke

Principal PRA Engineer

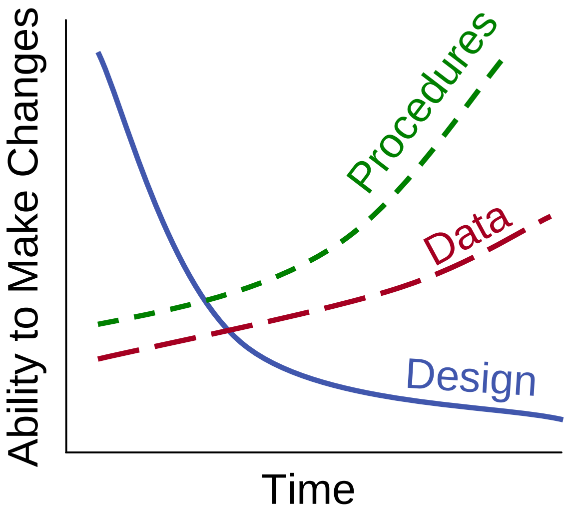
December 5, 2007

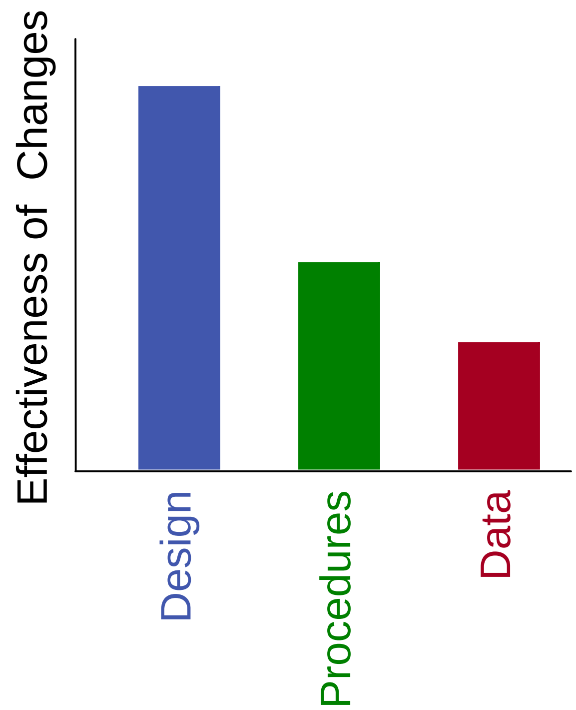
HITACHI

PRA For A New Reactor Design

- Determine risk management strategy
- Consider all aspects in the design
  - Core damage
  - Severe accidents
  - Internal and external events
- Design PRA provides a bounding assessment
  - Provides the safety case for the plant license
- Make risk assessment an integral part of the overall design process
- Updated PRA prior to fuel load

Three Chief Methods to Affect Risk





Using a PRA early provides maximum benefit

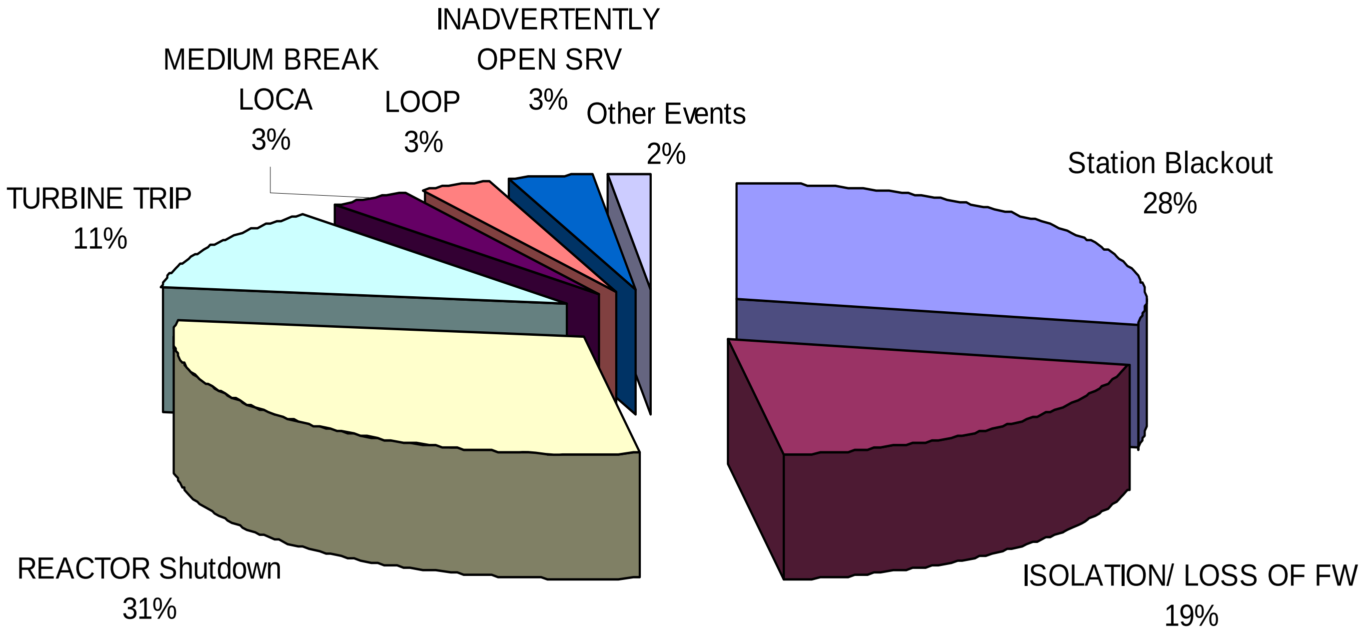
Features of ABWR PRA

- Level 1, 2, and 3 (Level 2, 3 for At-power internal events)
- Internal & External Events
- Full Power and Shutdown
- Seismic Margins
- Generic Data (conservative)
- Historical Initiating Event Frequencies (conservative)
- Parametric Uncertainty and Sensitivity Analysis
- Systematic Search for Key Modeling Uncertainties

Design Improvements using PRA Input

- Input to the design process;
  - Elimination of Recirculation Piping
    - Lower LOCA freq. not credited in PRA
  - Three-train design of ECCS
  - Credit for AC Independent Water Addition (Fire Water) (Level II)
  - Containment Overpressure Protection (COP) Relief
  - Combustion Turbine use and design
  - Use of the Lower Drywell Flooder
  - Emergency Procedure Guideline Improvements:
    - Accident Prevention: Seismic LOP Guidelines
    - Accident Mitigation: Fill Containment to bottom of Reactor Vessel for COPs operation.
  - Improved Aux. Shutdown Panel Operation
  - Internal Flood Controls and Instrumentation

Updated ABWR Core Damage Risk Profile



CDF ~ 2×10^{-7} yr^{-1}

At power, internal events

Risk Impact of Major Departures*

Departure or Change	Percent Change
Instrumentation Changes	0.30%
Power Distribution	-1.50%
RCIC Pump Design	-3.20%
UHS - Cooling Tower Fans (STP)	6.40%
LOOP/Rec (STP)	-11.80%
Final - Includes all Modifications	-8.80%

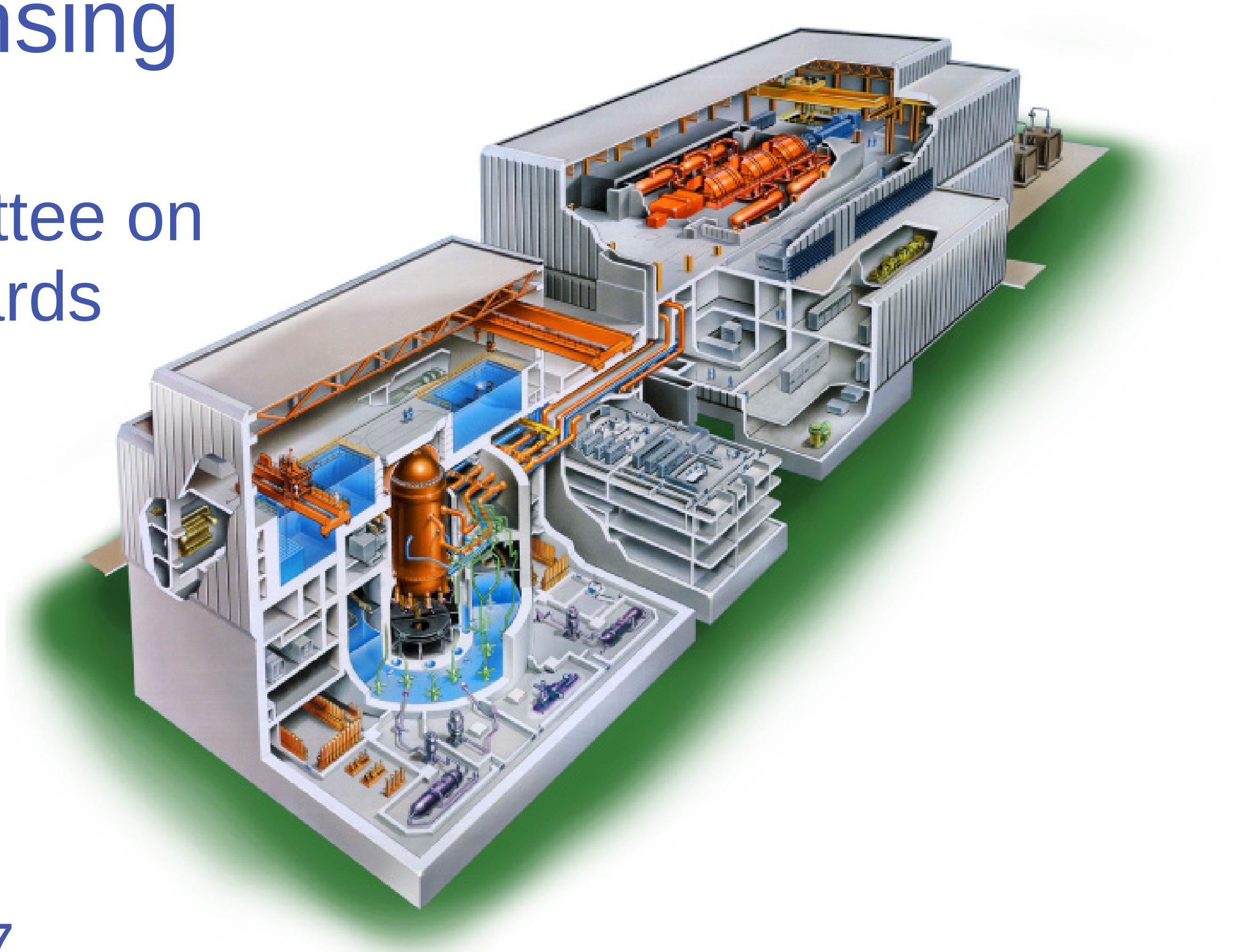
* Screening and qualitative evaluation of other departures.

Reliability Assurance Program Input

- The results of the PRA are reviewed to determine the appropriate reliability and maintenance actions that should be considered throughout the life of an ABWR plant so that the PRA remains an adequate basis for quantifying plant safety. These actions comprise a part of the plant's reliability assurance program (RAP).
  - Input from Level I/II, Shutdown PRA, External Events, etc.
  - Includes a list of important SSCs considered for periodic testing and/or PM, failure modes and important maintenance activities.
  - Update for STP COL, and accounts for external flooding issues.

ABWR Licensing Overview

Advisory Committee on Reactor Safeguards



J. Savage
Vice President,
ABWR Licensing
December 5, 2007

GE HITACHI

ABWR DCD Departures for STP 3 & 4 COLA

- SRV Setpoints

  Based on industry experience with SRVs; reduces weepage and maintenance

- ESF/RPS Control System Setpoint/Logic Changes

  Misc. detailed corrections, clarifications, and improvements

- Delete MSIV Closure on Hi Rad

  Existing regulatory/industry initiative reduces spurious trips

- Third Train of RHR to Fuel Pool

  Increases outage maintenance flexibility

- FWLB Mitigation

  Safety-related trip to ensure no overfilling/overpressurization of containment

- RCIC Monobloc Pump/Turbine

  Better, simpler hardware eliminates many negatives of prior designs

ABWR DCD Departures for STP 3 & 4 COLA

- Class 1E Undervoltage Shop Tests/ Breaker Coordination
  Substitute for in-situ testing and low voltage/low amperage clarification
- Additional Division of I&C Power
  Required to support fully developed safety-related logics
- Delete Recombiner
  Regulatory initiative in 10CFR50.44
- Control System Architecture & Technology
  Major design update of computer-based controls
- STP Site Parameters
  Rainfall and humidity slightly outside prior assumptions; site flood added
- Tier 2* Reg Guides and Codes & Standards Updates (10 documents)
  Updating selected regulatory and industry practices and requirements

ABWR DCD Departures for STP 3 & 4 COLA

- Design Departures (Tier 2) Requiring Tech Spec Changes – 13
- Changes of Intent to Tech Specs – 8
- Editorial Changes to Tech Specs – 90

Note: Full Tech Spec Improvement for later

ABWR DCD Departures for STP 3 & 4 COLA

- All screened/evaluated per Part 52 rules
- Total Tier 2 Departures – 118
- Most are updates and clarifications
- A few are regulatory-related (dual units)
- Some are site-specific adaptations (ultimate heat sink, RHR Hx heat capacity increase)
- Many are design detailing (updated suction strainers, delete FW turbidity monitor, equipment locations, etc)
- Radwaste Processing and Turbine-Generator Selection scopes were significant but non-critical to application

Licensing Topical Reports Referenced in STP 3&4 COLA

- Alternate RCIC Pumps
- Plant Procedures Development Plan
- Startup Administrative Manual
- APRM Oscillation Monitoring Logic
- Vibration Assessment Program
- Reactor Materials Surveillance Program
- Common Equipment and Structures
- Startup Test Specifications
- Plant Medium Voltage Electrical System
- Hydrogen Recombiner Requirements Elimination
- Life Cycle Management Program
- Stability Evaluation
- Containment Analysis

ABWR Presentation Summary

- ABWR design improves on operating US BWR designs:
  - Evolution of existing technology, incorporating lessons learned from the existing BWR fleet.
  - Improved ECCS design, electrical design, containment design, etc.
  - Very low overall risk of Core Damage and Release.
  - First Design Certified by NRC under Part 52.

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Protecting People and the Environment

ABWR Familiarization Briefing For The ACRS Subcommittee

Mark Tonacci, Senior Project Manager
NRO/DNRL/NGE2

December 5, 2007

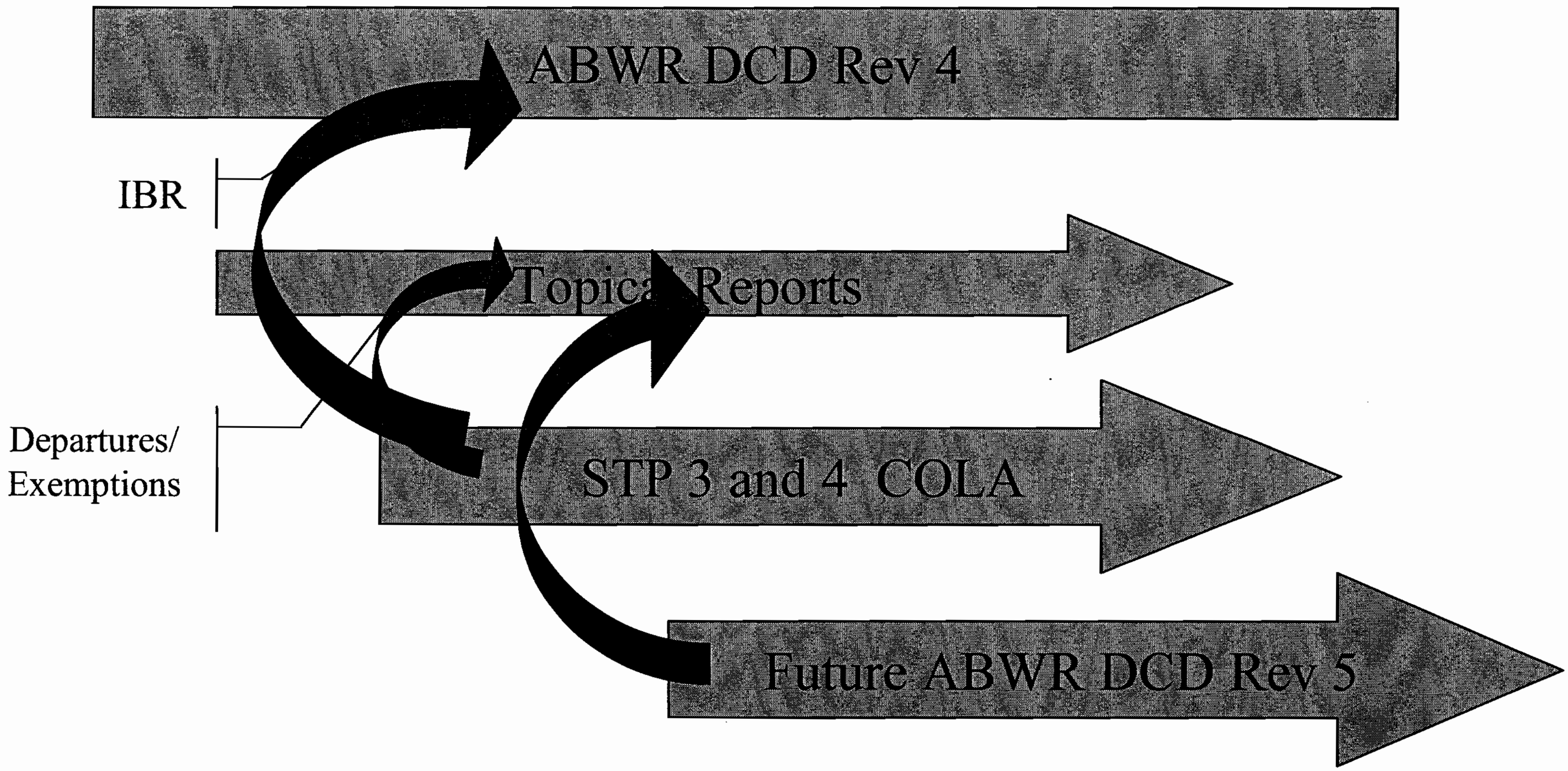
ABWR Familiarization Briefing
Agenda

- Introduction – Mark Tonacci
- ABWR Technology Overview:
  - Alan Beard: Hardware
  - Dennis Henneke: Risk Assessment
- ABWR Licensing
  - Joe Savage: GE-H Perspective
  - Mark Tonacci: NRC Perspective

ABWR Chronology

- ABWR Rev 4 Design Certified 1997
- 13 Topical Reports Submitted 12/06 – 9/07
  - Reports are on the GE-H Docket
- STP COLA Submitted 10/07
- STP COLA References the Topical Reports
- Design Certification Rulemaking Petition - Future

Relationship of the DCDs, Topical Reports and COLA



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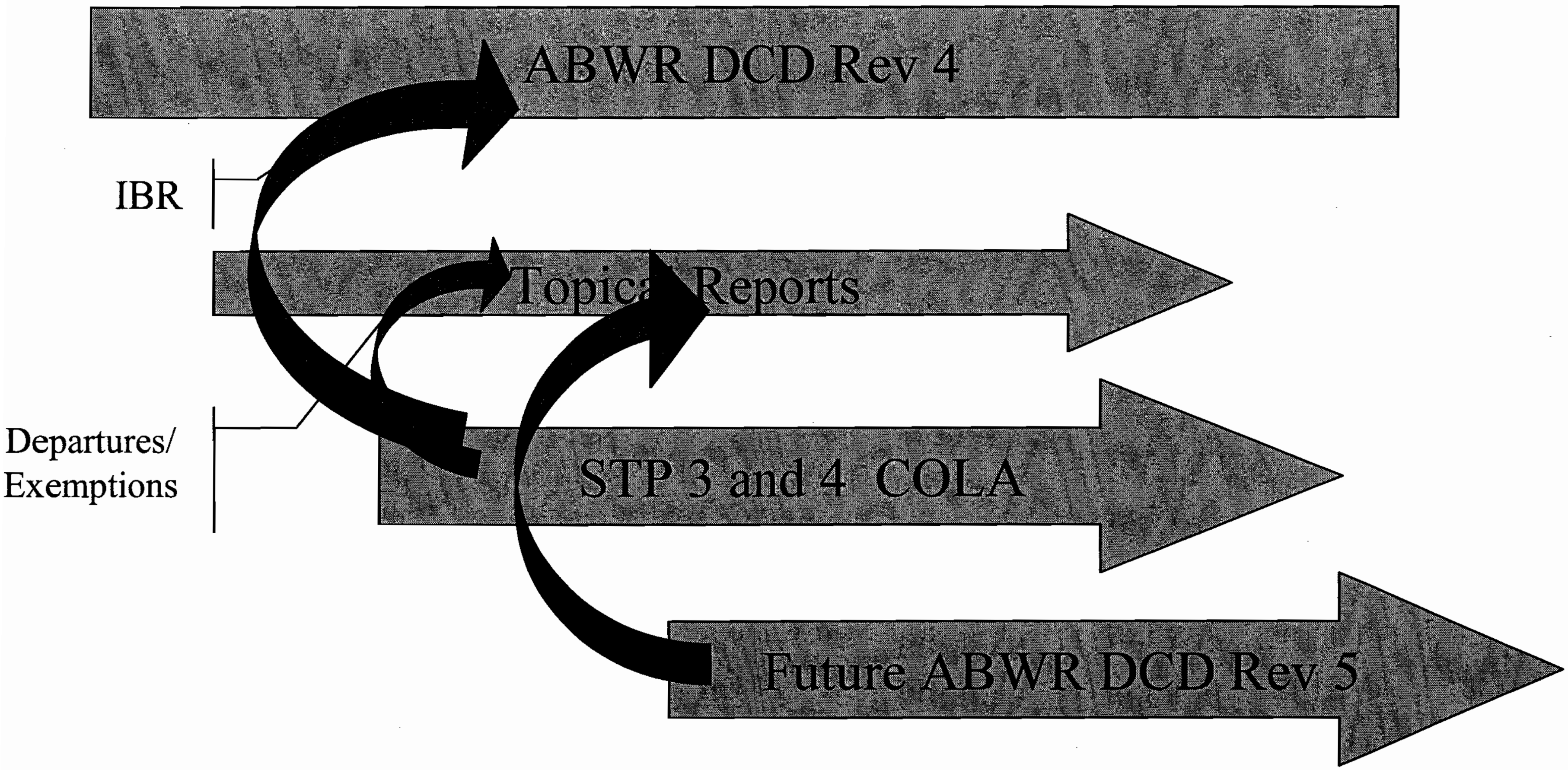
Protecting People and the Environment

ABWR Familiarization Briefing
Licensing Aspects

Mark Tonacci, Senior Project Manager
NRO/DNRL/NGE2

December 5, 2007

Relationship of the DCDs, Topical Reports and COLA



ABWR Topical Report Approval Process

- Process Will Be Very Similar to NRR Topical Report Approval
- RAIs and Safety Evaluations Prepared
- ACRS Review as Desired
- Management Approval as a Topical
- Reports Will Not Have Finality

What Is The Schedule?

Topical Reports

- Expect Some Late In 1st Quarter of 2008
- Will Coordinate To Present In Groups

STP COLA

- Two Rounds of Review by ACRS

How are COL Information Items Handled?

- Mostly for Operational Programs
- COL Applicant Must Address
- Some GE-H Topical Reports Address All or Portions of COL Information Items
- COL Information Item May Be Fully Addressed In COLA or Topical
- If Portions Are Still Open There Are 4 Options
- Construction Inspection to Close Issue

Other Questions?

Acronyms

ABWR	Advanced Boiling Water Reactor
ACRS	Advisory Committee on Reactor Safeguards
APRM	Average Power Range Monitor
DCD	Design Certified Document
COL	Combined Operating License
COLA	Combined Operating License Application
GE-H	General Electric - Hitachi
IBR	Incorporated By Reference
RG 1.206	Regulatory Guide 1.206 – Combined License Applications for Nuclear Power Plants
NRR	Office of Nuclear Reactor Regulation
RAIs	Requests for Additional Information
RCIC	Reactor Core Isolation Cooling
PRA	Probabilistic Risk Assessment
SE	Safety Evaluation
SRP	Standard Review Plan
STP	South Texas Project