US-APWR

2nd Pre-Application Review Meeting

September 26, 2006

Mitsubishi Heavy Industries, Ltd.

Today's Meeting Objectives

- To propose the review meeting and related Topical Reports submittal schedule in accordance with the revised submittal date for the US-APWR design certification application
- To provide a more extensive technical discussion of the US-APWR to demonstrate the design is similar to current PWR plants already licensed by the NRC
  - ✓ To discuss safety design philosophy based on enhancing a proven, safe design
  - ✓ To explain the US-APWR design features compared with US current PWR plant and to identify the enhanced features of the proven US-APWR design
  - ✓ To identify the computer codes and methodology used for safety analysis

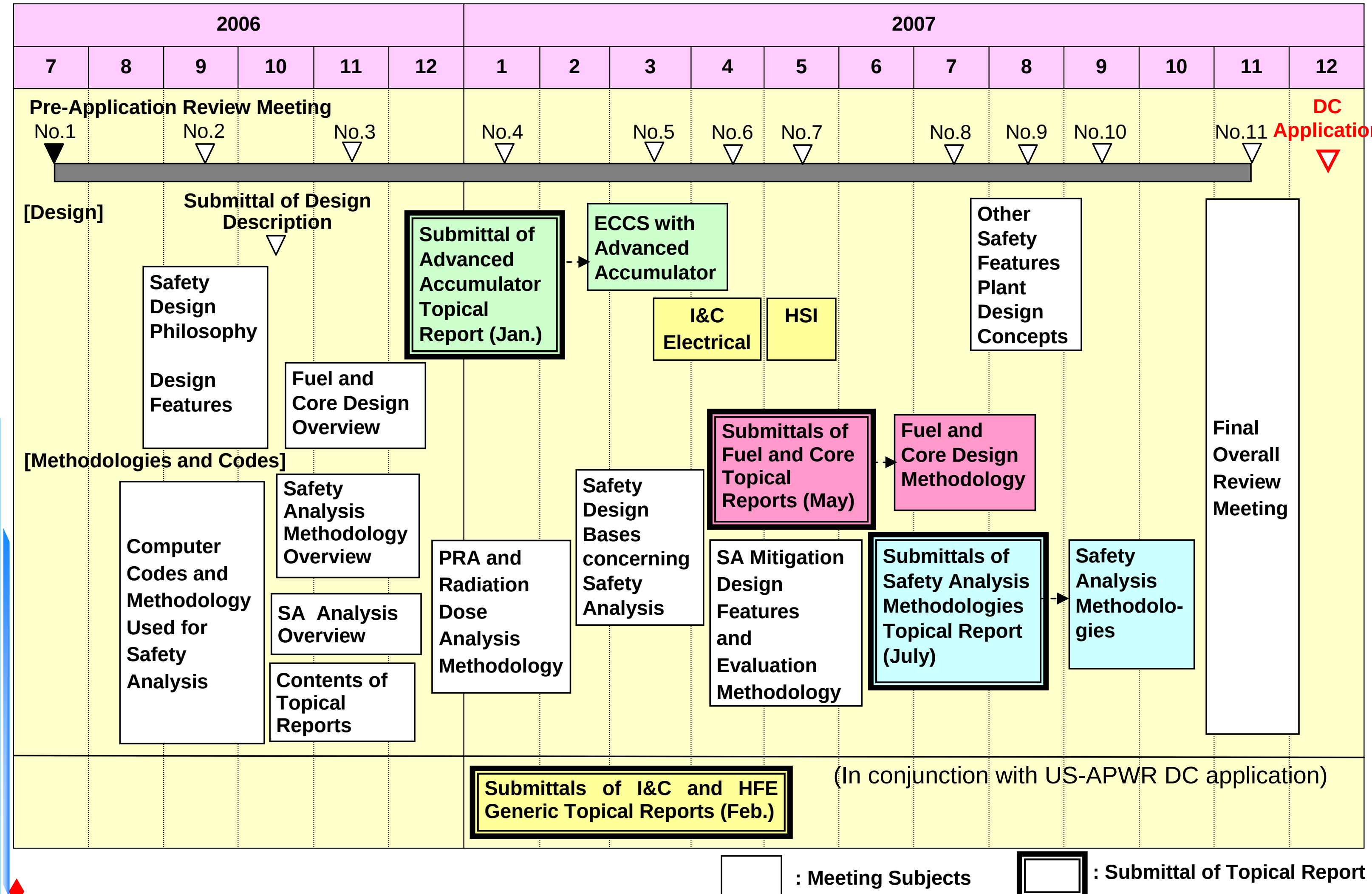
CONTENTS

1. Revised Pre-Application Review Meeting Plan and Topical Report Submittals Plan
2. Safety Design Philosophy
3. US-APWR Design Features
4. Computer Codes and Methodology Used for Safety Analysis
5. Next Meeting

1. Revised Pre-Application Review Meeting Plan and Topical Report Submittals Plan

MHI intends to submit the US-APWR Design Control Document in December, 2007

Pre-Application Review Meeting and Topical Report Submittals Schedule



Revised Pre-Application Review Meeting Plan

Meeting Subject	Meeting Objectives	Date
- Mitsubishi Nuclear Activities - US-APWR Overview	- Introduce MHI nuclear activities and US-APWR	July 2006
- Revised Pre-Application Review Meeting Plan - Topical Report Submittals Plan - Safety Design Philosophy - US-APWR Design Features - Computer Codes and Methodology Used for Safety Analysis	- Propose the review meeting and topical report schedule - Provide a more extensive technical discussion to demonstrate the design is similar to US current PWR plant	September 2006
- Fuel and Core Design Overview - Safety Analysis Methodology Overview - Severe Accident Analysis Methodology Overview - Contents of Topical Reports	- Explain needs of fuel and core topical reports - Explain codes used in analysis for transient and accident - Identify the form and contents of the topical reports	November 2006

Revised Pre-Application Review Meeting Plan

Meeting Subject	Meeting Objectives	Date
- Probabilistic Risk Assessment Methodology - Radiation Dose Analysis Methodology	-Explain codes, database and key assumption of the PRA -Explain codes and key assumption of the Radiation dose analysis methodology	January 2007
- Safety Related Features (ECCS, EFWS, Containment Vessel) - Accumulator with flow damper Topical Report - Including safety Design Bases concerning Safety Analysis	- Provide a more specific discussion on safety related features - Discuss the NRC comments after the NRC staff has read the topical report	March 2007
- Electrical and I & C System Design Criteria - Overall I & C and Safety System	-Provide plant-wide architecture and system level design and function of safety related I & C - Discuss the Digital I & C referring Platform generic topical report	April 2007

Revised Pre-Application Review Meeting Plan

Meeting Subject	Meeting Objectives	Date
- Human System Interface - Other Important I & C Systems -Safety Grade 1E Electrical Power System - Severe accident mitigation design features and evaluation methodology	-Provide HSI concept and system level design of other important I & C and safety electrical system - Discuss the HSI referring HFE generic topical report - Explain schematic system description and their effectiveness evaluation method	May 2007
- Mitsubishi Fuel System Design Criteria and Methodology Topical Report - Mitsubishi Thermal Design Methodology Topical Report	- Discuss the NRC comments after the NRC staff has read the topical report	July 2007

Revised Pre-Application Review Meeting Plan

Meeting Subject	Meeting Objectives	Date
- Other Safety Features (Safety shutdown system, SBO mitigation system etc.) - Plant Design Concepts (Plant Layout, Separation Criteria, Steel & Concrete Structure)	- Provide a more specific discussion on other safety features and plant design concepts	August 2007
- Safety Analysis Methodologies Topical Report	- Discuss the NRC comments after the NRC staff has read the topical report	September 2007
- Final Overall Review Meeting	- Summarize the NRC comments and discussion	November 2007

Topical Report Submittals Plan

Topical Report	Date
Accumulator with flow damper	January 2007
Mitsubishi Fuel System Design Criteria and Methodology	May 2007
Mitsubishi Thermal Design Methodology	May 2007
Safety Analysis Methodology (LOCA)	July 2007
Safety Analysis Methodology (Non-LOCA)	July 2007

I & C Generic Topical Report (In conjunction with US-APWR DC application)	Date
Digital I & C Platform	February 2007
Human Factor Engineering Process	February 2007

2. Safety Design Philosophy

Contents

- **Basic Concept of Safety Design**
- **The U. S. Regulatory Requirements**
- **Defense in Depth**
- **Measures Improving Safety against Internal Events**
- **Measures against Fire and External Events**

Basic Concept of Safety Design

- **Compliance with the U.S. regulatory requirements, guidance, and industry codes and standards**
- **Design concept**
  - ✓Use of proven, accepted technologies with improvements to enhance safety
  - ✓Enhanced safety design
    - Highly reliable prevention function
    - Well-established mitigation systems with active safety functions and passive safety functions
    - Functions against beyond design basis accidents

Basic Concept of Safety Design

➢ Approach

- ✓ Deterministic design approach based on the principle of "Defense in Depth"
- ✓ Probabilistic risk assessment used as an additional method

➢ Target of safety design

- ✓ Core Damage Frequency (CDF) and Large Release Frequency (LRF)
  - CDF less than 10^{-5}/reactor-year
  - LRF less than 10^{-6}/reactor-year

The U.S. Regulatory Requirements

- **List of the main U.S. Regulatory requirements regarding safety design**
  - ✓10 CFR 50 :Domestic licensing of production and utilization facilities
  - ✓10 CFR 100 :Reactor site criteria
  - ✓NUREG- 0800 :Standard Review Plan
  - ✓Regulatory Guides, etc.

Defense in Depth

Design of structures, systems and components of US-APWR based on the concept of defense in depth

Objective of defense	Basic concept	
	Essential criteria	Design approach
Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation	• Careful selection of materials and use of qualified fabrication processes • Margins in the design of systems and plant components • Utilization of operating experience
Detection of failures, control of abnormal plant states and accidents within the design basis	• Protection systems • Engineered safety features with critical support systems	•Redundancy •Separation
Control of beyond design basis accidents	Supplemental measures and accident management	Diverse measures against the design basis accidents

Measures Improving Safety against Internal Events

Defense level	Features of measures against internal events
Prevention of abnormal operation and failures	• Enhanced reliability of reactor coolant pressure boundary • Improved maintenance using 4 train safety systems • Enhanced reliability during shut down operation • Reduction of operator actions • Enhanced reliability of I & C systems
Detection of failures, control abnormal plant states and accidents within the design basis	• Enhanced reliability of shut down capability • Enhanced reliability of emergency core cooling systems • Enhanced reliability of containment cooling system • Enhanced reliability of support systems
Control beyond the design basis accident	• Enhanced measures against station blackout • Enhanced measures against Interfacing systems' LOCA • Measures against common mode failures in digital safety system • Enhanced measures against severe accidents after core damage

(Prevention of abnormal operation and failures)

- **Enhanced reliability of reactor coolant pressure boundary**
  - ✓ Vessel head : Alloy 690 at vessel head nozzle
    : T-cold at vessel head plenum temperature
  - ✓ Reduction of neutron fluence to reactor vessel using neutron reflector
- **Improvement of maintenance**
  - ✓ Enhanced safety during on line maintenance using 4 train safety systems

(Prevention of abnormal operation and failures)

- **Enhanced reliability during shutdown operation**
  - ✓ Shortening duration of mid loop operation
  - ✓ Automatic interlock to isolate the letdown line below mid loop water level
- **Reduction of operator actions**
  - ✓ Computerized control room with enhanced operability
- **Enhanced reliability of I &C systems**
  - ✓ Redundant digital control systems with enhanced reliability

(Detection of failures, Control abnormal plant states and accidents within the design basis)

➢ Enhanced reliability of shutdown capability

✓Reactor protection system

- 4 train system with 4 train reactor trip breakers

✓Cold shutdown with safety components

- Emergency core cooling system
- Emergency letdown line
- Safety depressurization and vent system

Measures Improving Safety against Internal Events

(Detection of failures, Control abnormal plant states and accidents within the design basis)

- **Enhanced reliability of Emergency Core Cooling System (ECCS) function**
  - ✓ Advanced accumulator and 4 train high head safety injection system
  - ✓ Elimination of switchover of ECCS suction by installation of the refueling water storage pit inside the containment (IRWSP)
- **Enhanced reliability of containment cooling system**
  - ✓ 4 train containment spray system
  - ✓ Elimination of switchover of spray pump suction by installation of IRWSP

(Detection of failures, Control abnormal plant states and accidents within the design basis)

➢ Enhanced reliability of support systems

- ✓ 4 train electric power system, component cooling water system, service water system, etc.
- ✓ Installation of high reliability emergency gas turbine generators

Measures Improving Safety against Internal Events

(Control of beyond design basis accident)

- **Station blackout**
  - ✓ A diverse AC power source
- **Interfacing systems' LOCA**
  - ✓ Upgrading piping of residual heat removal system
- **Common mode failures in digital safety system**
  - ✓ Diverse actuation functions (reactor trip, turbine trip, emergency feed water system initiation)
- **Severe accidents after core damage**
  - ✓ Measures to reduce hydrogen detonation, molten core concrete interaction, high pressure melt ejection

➢ Fire and external events

- ✓Fire
- ✓Seismic
- ✓Tornado / Hurricane
- ✓Missiles (internal and external)

(Fire Protection)

- **Compliance with the U.S. regulatory requirements**
- **Design based on defense in depth concept**

⇩

Physical separation in 4 train safety system

Measures against Fire and External Events

(Design for natural phenomena)

Natural Phenomena*	Design condition	Remarks
Earthquake	PGA 0.3 G (Peak Ground Acceleration)	0.3 G covers the most of the US sites.
Tornado	Wind velocity 300 mph	RG 1.76, SRP 2.31
Hurricane	Wind velocity 145 mph	ASCE 7-98 (American Society of Civil Engineers)

* : Based on the 10CFR Part 50 Appendix A Criteria 2

Measures against Fire and External Events

(Missile Protection)

Missiles	Design for the structures and components*	Remarks
Pressurized components, high-energy piping and rotating equipment	i) Locate in a missile-proof structure ii) Separate redundant systems or components	SRP 3.5.1.1 SRP 3.5.1.2
Turbine disk (or internal structure) fragment	Probability of unacceptable damage $< 10^{-7}$ / year	SRP 3.5.1.3 RG 1.115
Tornado missiles	Seismic category I structures to withstand the effect of tornado missiles.	SRP 3.5.1.4

* : based on General Design Criterion (GDC) 2 and 4 of Appendix A to 10 CFR 50

3. US-APWR Design Features

Contents

3.1 Main Specifications of US-APWR

Main Specifications of US-APWR

Parameters	US-APWR	Japanese APWR	US Current 4 Loop Plant*
Gross electric output (MWe)	1,700 class	1,538	1,219
Core thermal output (MWt)	4,451	4,451	3,565
Number of loops	4	4	4
Coolant pressure (psia)	2,250	2,250	2,250
Coolant temperature (Hot leg) (deg.F)	617	617	620
Number of fuel assemblies	257	257	193
Fuel rod lattice	17 x 17	17 x 17	17 x 17
Active fuel length (ft)	14	12	12
Average linear heat rate (kW/ft)	4.6	5.3	5.7
Number of RCCAs	69	69	53
Thermal design flow (gpm/loop)	112,000	113,600	93,600
Steam generator heat transfer area (ft^2/SG)	91,500	70,000	55,000
PZR volume (ft^3)	2,900	2,300	1,800
Design life (years)	60	60	40

*Callaway NPP

3.2 Fuel and Core Design

Core/Fuel/RI Design Feature Overview

- ➢**Based on proven, verified and/or approved technology**
- ➢**Two Mitsubishi-developed method subject to Pre-application review.**

Design Area	Design Features			Design Methodology
	Flexible Operation	Enhanced Economy	Improved Reliability	
Nuclear	•**Lower power density** •High Gadolinia content	•**Neutron Reflector**	-	**PARAGON/ANC**
Thermal Hydraulics	-	-	•**RVH cooling** •**Neutron Reflector cooling**	***WRB-2 / VIPRE-01***
Fuel	•High Gadolinia content •Large Plenum Volume •14ft active length	•High density pellet •Zircaloy-4 grid	•Grid fretting resistant design •**ZIRLO™*** •Bottom nozzle with built-in filter	***FINE***
Reactor Internals	•**Enlarged Core Barrel** •LCP/LCS integration	-	•**Neutron Reflector** •Upper mounted ICIS	-

NOTE : Proven(in operating plant)
Verified(by test or analysis) or Approved(by NRC)
Subject to Pre-application Review by NRC

*ZIRLO™:Trademark of Westinghouse Electric Co.

Nuclear Design Features

➢To improve economy and safety

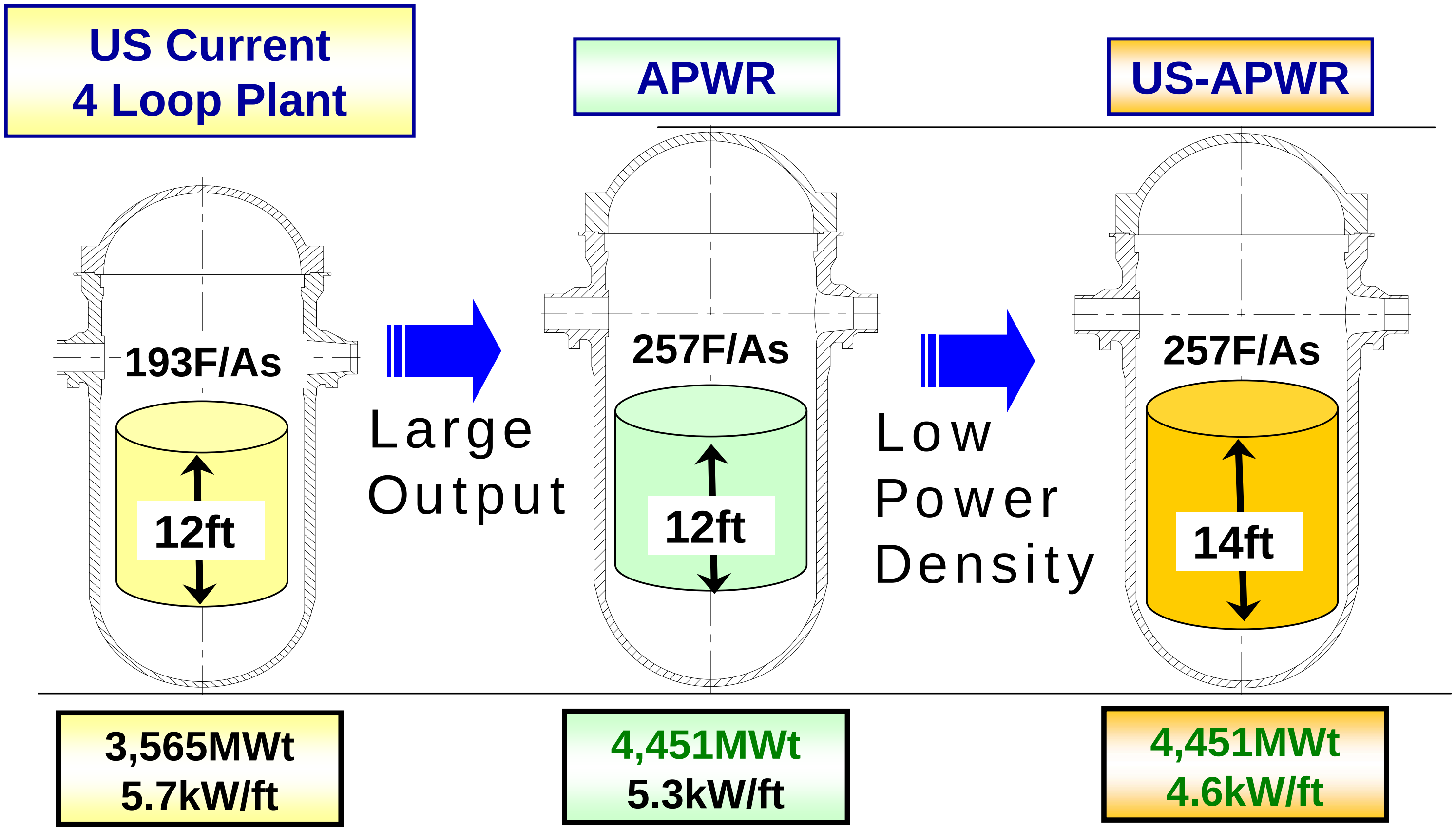
- ✓Large thermal output for scaling benefit
- ✓Low power density for flexible operation with large thermal margin
- ✓ Neutron reflector for neutron economy

➢High reliability with proven technology

- ✓ Fuel rod array - 17x17
- ✓ Reactivity control systems
- ✓ Power distribution monitoring and control systems
- ✓ Nuclear design codes

Nuclear Design Features

➢Large thermal output and low power density



Nuclear Design Features

➢Low power density for flexible operation

✓ Longer cycle operation for a given cycle burnup

- 24-month 2-batch equilibrium cycles sustainable with
  - U235 enrichment < 5wt%
  - Maximum rod burnup $\leq$ 62GWd/t

✓ Improved load follow capability

- F_{dH} ~1.7 F_Q ~2.6 with large thermal margins

➢Steel Neutron Reflector

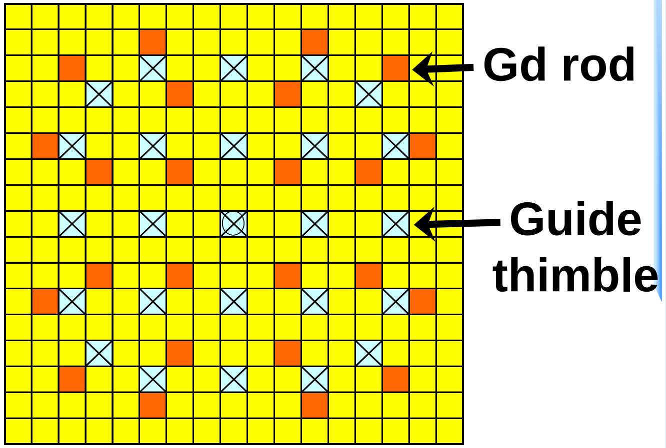
✓ Reduce neutron leakage to enhance neutron economy

✓ Reduce reactor vessel irradiation

Nuclear Design Reliability

➢ Fuel and absorbers

- ✓ 264 fuel rods in 17X17 array
- ✓ Ag-In-Cd absorber for control rods
- ✓ Discrete borosilicate glass BP rods
- ✓ Gd integral fuel rod
  - Appx.10wt% Gadolinia
  - Utilized in Japan



24 Gd rod allocation

➢ Power distribution control & monitoring

- ✓ Ex-Core Nuclear Instrumentation System (NIS)
- ✓ In-Core Instrumentation System (ICIS)
- ✓ Optional On-line Power Distribution Monitoring System (OPDMS) can give real-time monitoring function

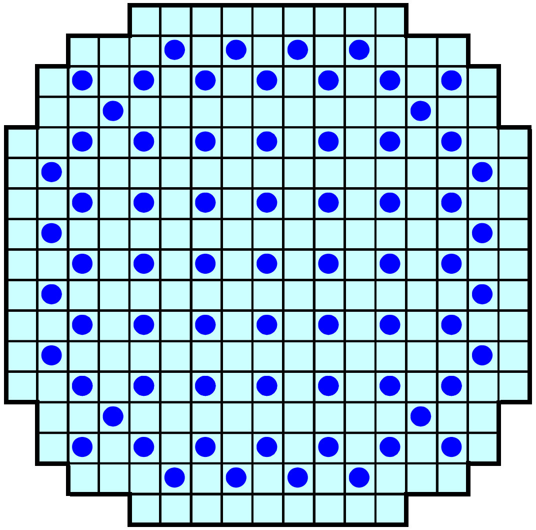
Nuclear Design Reliability

➢ Negative reactivity feedback

- ✓ Doppler feedback against rapid reactivity insertion
- ✓ Moderator temperature coefficient with negative feedback effect during operation

➢Current control mechanisms

- ✓ 69 RCCs for fast reactivity transients and hot shutdown
- ✓ Soluble boron system for cold shutdown



●: 69 control rod assemblies

Nuclear Design Reliability

- **Nuclear design codes approved by the NRC**
  - ✓ PARAGON/ANC for nuclear design
  - ✓ Joint development with Westinghouse for 20 years
- **Significant experience in core design**
  - ✓ More than 400 cycles' core design in PWRs
  - ✓ 2.0-4.95wt% in U235 enrichment
  - ✓ 6-10wt% in Gadolinia content
- **Validation performed for**
  - ✓ Neutron reflector effect by criticality experiments
  - ✓ Large core size effect on Xe oscillation by calculation

Thermal / Hydraulics

➢ Design Bases

- ✓ To prevent DNB during Condition I & II
  - Design criteria are determined based on 95x95 basis
- ✓ To prevent fuel centerline melting during Condition I & II

➢ Design features

- ✓ Low power density core design allows larger safety margins than conventional plants for DNB and fuel centerline melting.
- ✓ Thermal Design Flow is determined considering 10% of SG tube plugging.
- ✓ The core bypass flow is determined considering,
  - Reactor Vessel Head temperature kept as T-cold
  - Neutron Reflector cooling

Thermal / Hydraulics

- DNB design codes based on NRC approved methodologies
  - ✓WRB-2 DNB correlation
    - •Approved by NRC
    - •Applicable to MHI fuel design
  - ✓VIPRE-01 subchannel analysis code
    - •Developed by EPRI and approved by NRC
    - •Modified introducing design DNB correlation
  - ✓ Revised Thermal Design Procedure (RTDP)
    - •Approved by NRC

Fuel Design Features

- **Enable flexible core operation**
  - ✓Adoption of Gadolinia content
- **Enhance Fuel Economy**
  - ✓Increase total content of UO_2
- **Improve Reliability**
  - ✓To prevent grid fretting
  - ✓To prevent debris fretting
  - ✓To prevent fuel rod corrosion

Fuel Design Features

Enable Flexible Core Operation

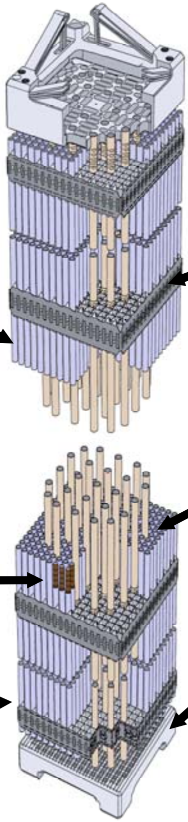
Fuel Active Length : 14ft
Fuel Rods Array : 17x17

Higher Gadolinia Content Pellet (10wt%)

Enhance Fuel Economy

Higher Density Pellet (97%T.D.*1)

Zircaloy-4 Grid (for Neutron Economy)



High Reliability

Grid Fretting Resistant Design
(Shorter Span Length*2 with 11 grids & Grid Spring Design)

Corrosion Resistant Cladding Material (ZIRLO™)

Anti-debris bottom nozzle with built-in filter

*1 T.D. :Theoretical Density
*2 compared with conventional 14ft Fuel

Fuel Design Specification

Fuel Assemblies	
Fuel Rods Array in Fuel Assembly	17 x 17
Number of Fuel Rods per Fuel Assembly	264
Number of Control Rod Guide Thimbles	24
Number of in-core Instrumentation guide tube	1
Number of Spacer Grids	11
Fuel Rods	
Outside Diameter	0.374 in.
Cladding Thickness	0.022 in.
Active Fuel Length	13.8ft
Reload Fuel Enrichment	Max. 5 wt%
Gadolinia Content	Max. 10 wt%
Pellet Density	97 % T.D.
Materials	
Cladding	ZIRLO™

Fuel Design Verification

Based on verified and proven design

- **Fuel Rod is designed using FINE Code (Developed by Mitsubishi)**
  - ✓ FINE Code is verified by significant irradiation data and out-of-pile data

Fuel Design in US-APWR	Operating PWR
14 ft. F/A with 11Grid	14ft. F/A : 3 XL / 4 XL* Span Length : Japanese Plants
10% Gd Content Pellet	Japanese Plants
97%T.D. Pellet	
ZIRLO™	US & Japanese Plants
Built-in filter	Japanese Plants

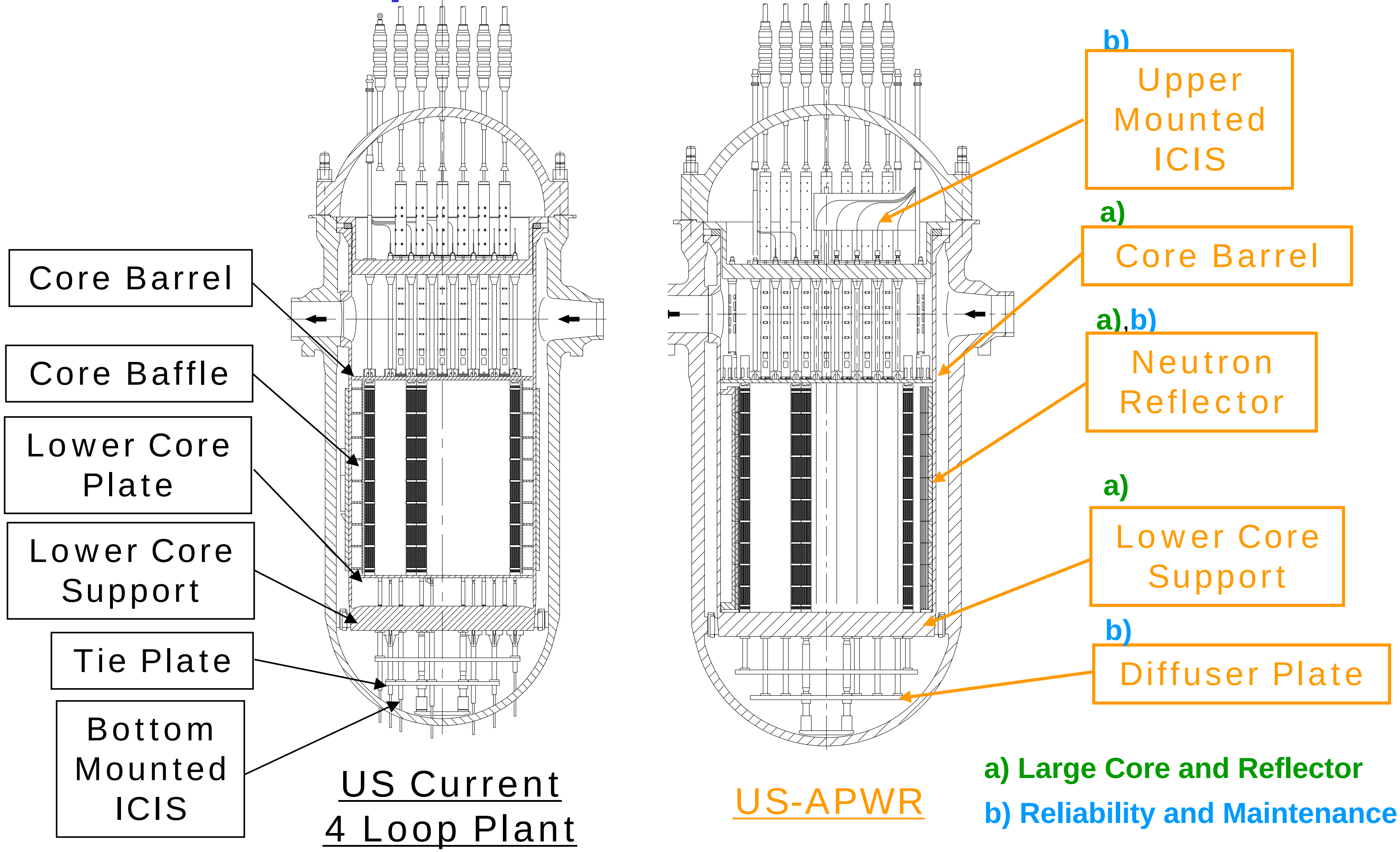
*South Texas 1 & 2

Reactor Internals Design

- **To accommodate the large core with improved neutron economy**
  - ✓ Large diameter of Core Barrel for 257F/A's
  - ✓ Integration of Lower Core Plate and Lower Support for 14ft F/A.
  - ✓ Neutron Reflector
- **To enhance reliability and maintainability**
  - ✓ Neutron Reflector
  - ✓ Upper mounted In-Core Instrumentation System and Simplified Diffuser Plate.

Reactor Internals Design

Comparison of main structures



a) Large Core and Reflector

b) Reliability and Maintenance

Reactor Internals Design

➢ Based on a design verified by operation, test or regulatory approval

- ✓ The Core Barrel design has been verified by the scale model flow test
- ✓ The other features have been proven in other operating plants or previously approved in U.S. Design Certification process

US- APWR modification items from conventional 4loop	**Operating PWR**	**Design Certification**
Core Barrel for 14 ft F/A	3XL / 4XL*	AP1000
LCS(LCP/LCS integration)	3XL / 4XL*	AP600 / AP1000
Neutron Reflector	-	AP600
Upper Mounted ICIS	2loop (C/E) 4 loop (SIEMENS)	AP600 / AP1000
Diffuser Plate	-	AP600 / AP1000

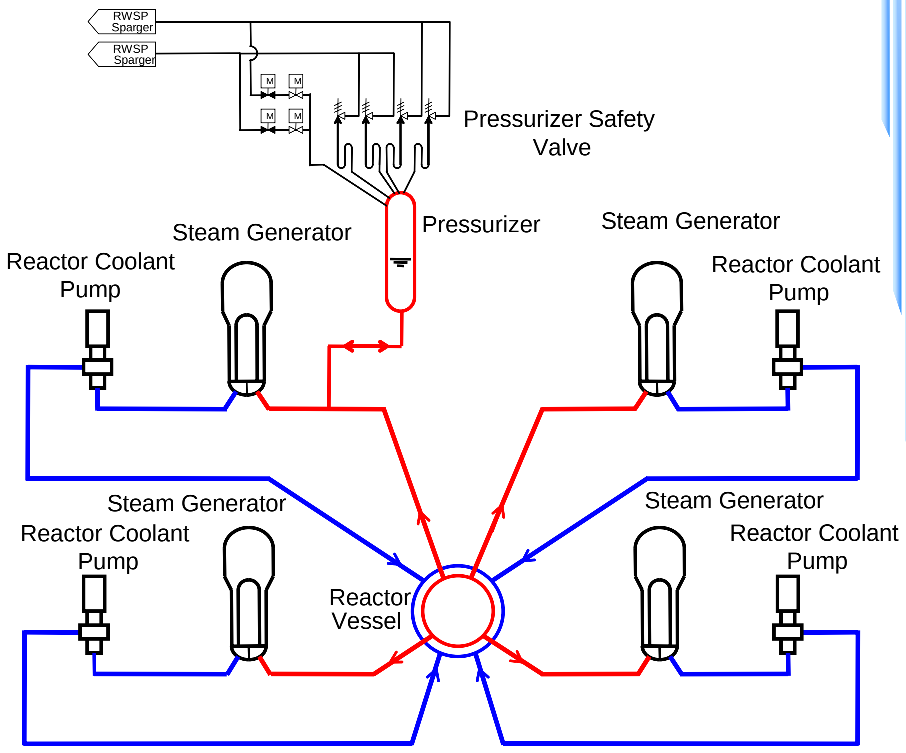
*South Texas1&2

3.3 Fluid System Design

Reactor Coolant System

➢ Design concept of the RCS

- ✓ Basic configuration is the same as current operating 4 loop plants proven by long term operating experience and enhanced reliability
- ✓ Large main components with large thermal output and high efficiency
- ✓ Enhance the plant controllability with large volume of the Pressurizer



Reactor Coolant System

➢ Larger main components

- ✓ Larger diameter and height of Reactor Vessel with enhanced reliability
- ✓ Larger heat transfer area in SG contributes high efficiency due to high steam pressure
- ✓ Larger reactor coolant flow rate of RCP with 8000 HP motor

➢ Enhance the plant control

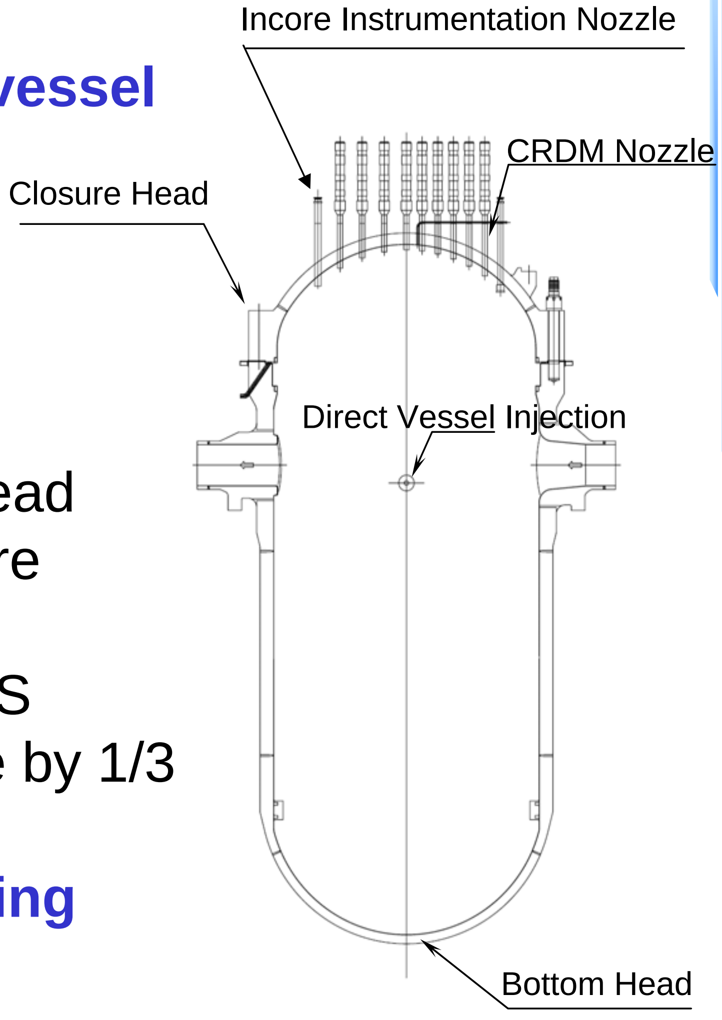
- ✓ Larger volume of the Pressurizer assures greater margin for the transients

Specifications	US-APWR	US Current 4 Loop Plant	Ratio
Core thermal output	4,451MWt	3,565MWt	1.25
SG Heat transfer area	91,500ft^2	55,000ft^2	1.66
Reactor Coolant Flow	112,000gpm	93,600gpm	1.20
Pressurizer Volume	2,900ft^3	1,800ft^3	1.61

Reactor Vessel

Design Features

- **Higher PWSCC resistance of vessel head penetration**
  - ✓ Material ; Alloy 690
  - ✓ Temperature ; T-cold
- **Improved reliability and maintainability**
  - ✓ No penetrations at bottom head and reduced number of Incore instrumentation nozzles by adopting Upper Mounted ICIS
  - ✓ Reduction of neutron fluence by 1/3 using the neutron reflector
- **Reduction of weld lines by using integrated forging**



Steam Generator

Design Features

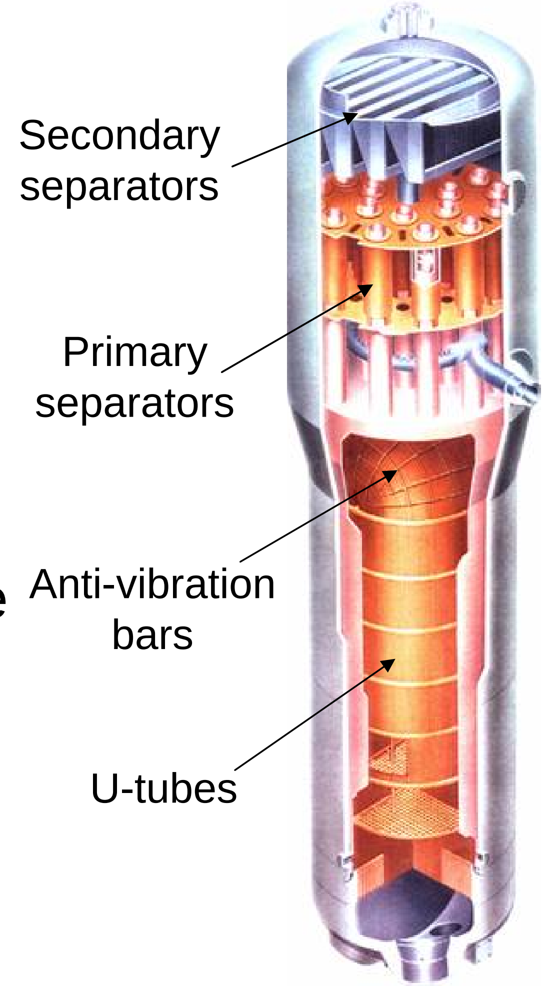
➢Primary separator

✓High performance of moisture separation
Moisture carry over at SG outlet: less than 0.1%

➢Anti-vibration Bar

✓Sets of 5 V-shaped AVBs with 10 support points of the outer most tube

Tube material	Alloy 690
Tube OD	0.75 inch
Tube arrangement	triangular
Tube pitch	1 inch
Heating surface	91,500 ft^2



Reactor Coolant Pump

➢Improved hydraulic performance

✓Large capacity and high efficiency by improved impeller and diffuser

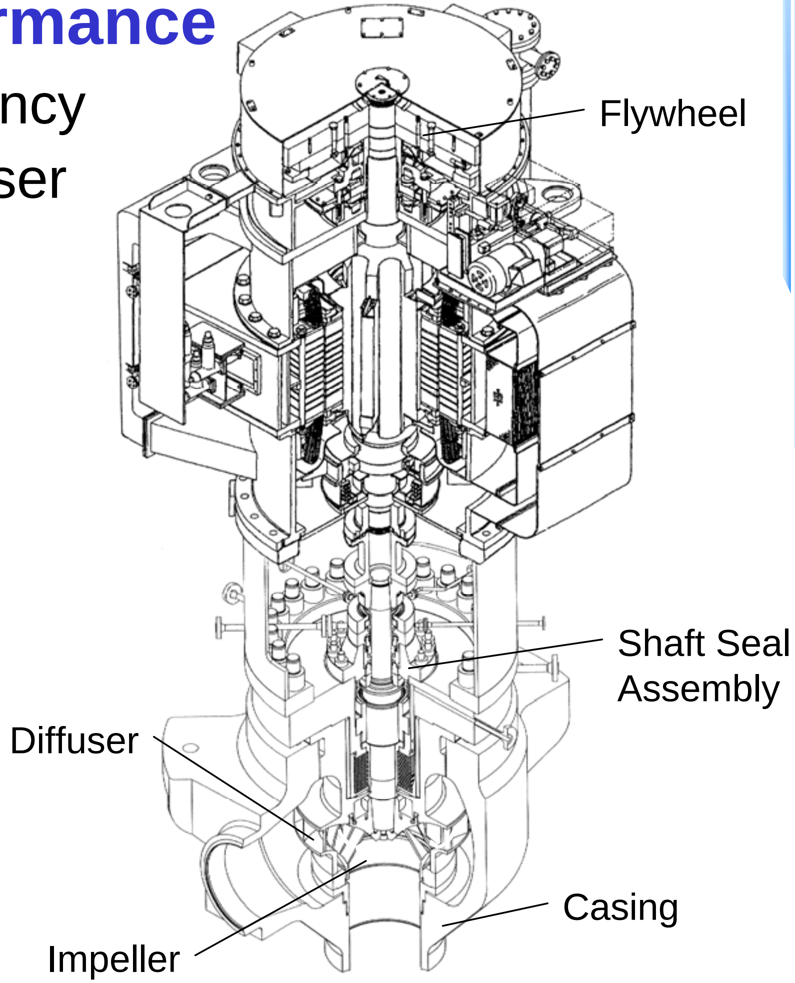
- Pump Efficiency : Over 85%
- Flow Rate : 112,000 gpm/loop
- Head : Approx. 310 ft

➢Advanced seal

✓Stabilization of No.1 seal leak-off characteristics

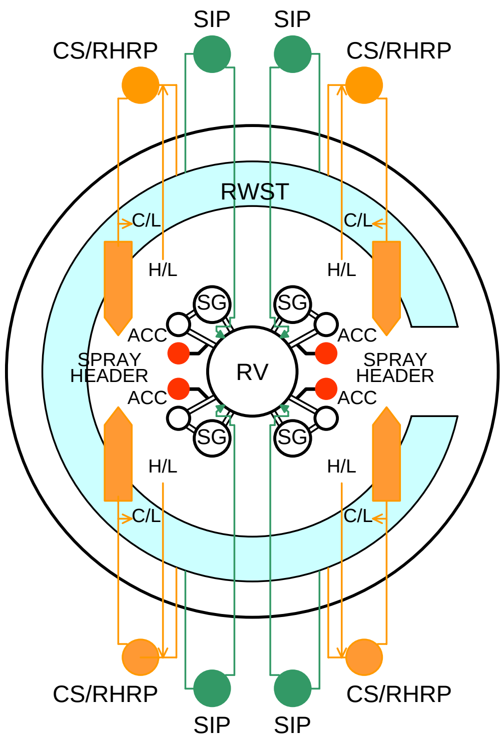
✓Extension of seal life

✓Countermeasure to station-blackout



Model 100A RCP

Design concept of ECCS and CSS/RHRS



➢ Basic design concept

- ✓ Achieve high reliability with simplified systems
- ✓ Introduce On Line Maintenance assuming single failure

➢ High Reliability

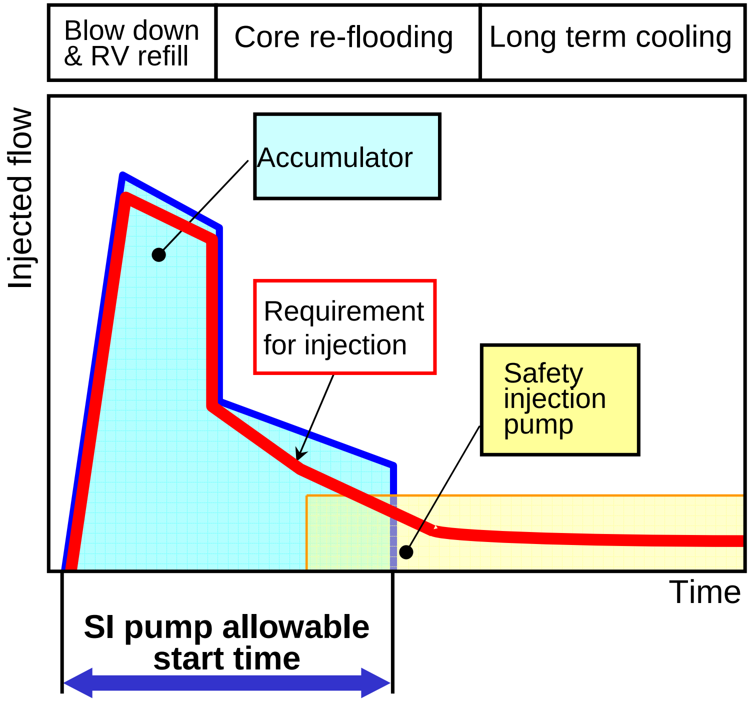
- ✓ 4 Train Configuration
  (50% x 4 for large break LOCA)
- ✓ In-containment RWSP
  (eliminate recirculation switchover)

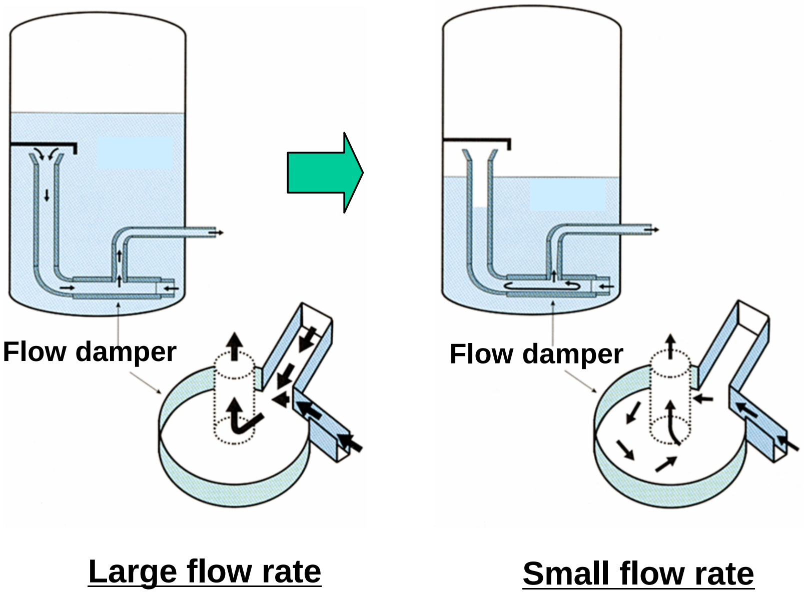
➢ Simplification

- ✓ Advanced accumulators
  (Integrated function of low head injection system)
- ✓ ECCS train includes only an accumulator and high head injection system
- ✓ Direct vessel injection
  (no inter-connection between trains)
- ✓ Common use of CSS and RHRS

Accumulator with flow damper

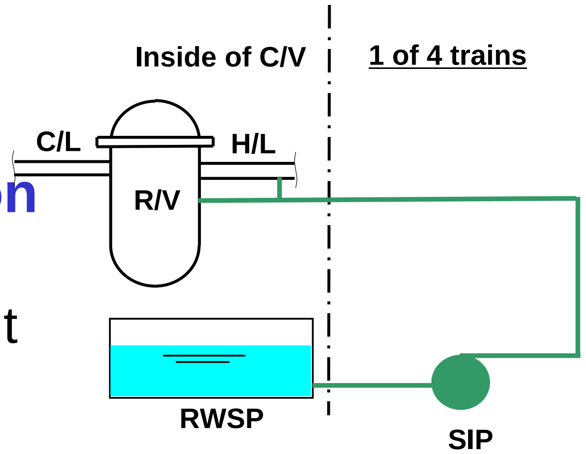
- Automatic switching of injection flow rate by flow damper
- Integrated function of low head injection system
- Long accumulator injection time allows more time for safety injection pump to start





Design feature of high head injection system

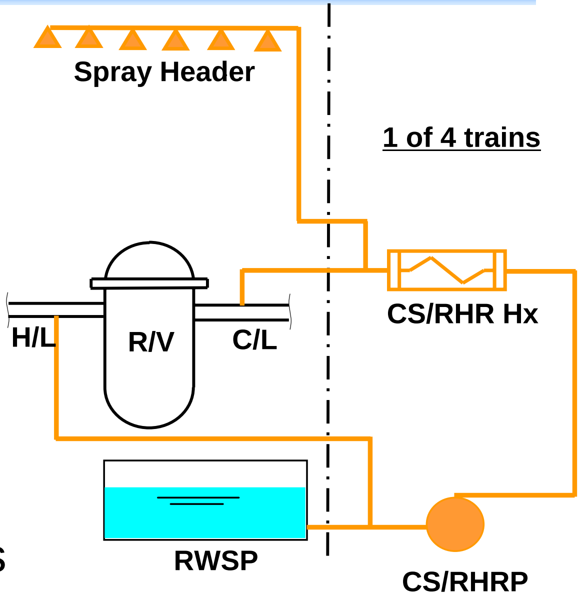
- **4 independent trains without interconnections between trains**
- **Sufficient capacity of safety injection pumps**
  - ✓ Meets the safety injection requirement in core reflooding stage



Item	US-APWR	US Current 4 Loop Plant	Reason and/or Advantage
Train	4 train	2 train	•Enhance the reliability •Achieve OLM under single failure
High head Injection	DVI 4 SIP	Loop injection 2 SIP + 2CH/SIP	•No interconnection between trains
Refueling Water Storage Pit	Inside CV	Outside CV	•Eliminate recirculation switchover

Design Feature of CSS/RHRS

- **Common use of CSS and RHRS**
- **CSS and RHRS are not required for operation at same time, therefore no affect on safety**
  - ✓ Normal operation: Stand by as CSS
  - ✓ LOCA: Operated as CSS
  - ✓ Plant shut down: Operated as RHRS
  - ✓ Pump suction and discharge valves are interlocked for prevention of human errors



Item	US-APWR	US Current 4 Loop Plant	Reason and/or Advantage
Containment Spray	4 CS/RHRP 4 CS/RHR Hx	2 CSP 2 CS Hx	• Enhance the reliability • Achieve OLM under single failure
Residual Heat Removal	Integrated to CSS	Integrated to LHSIS 2 RHRP 2 RHR Hx	• Simplify safety system configurations

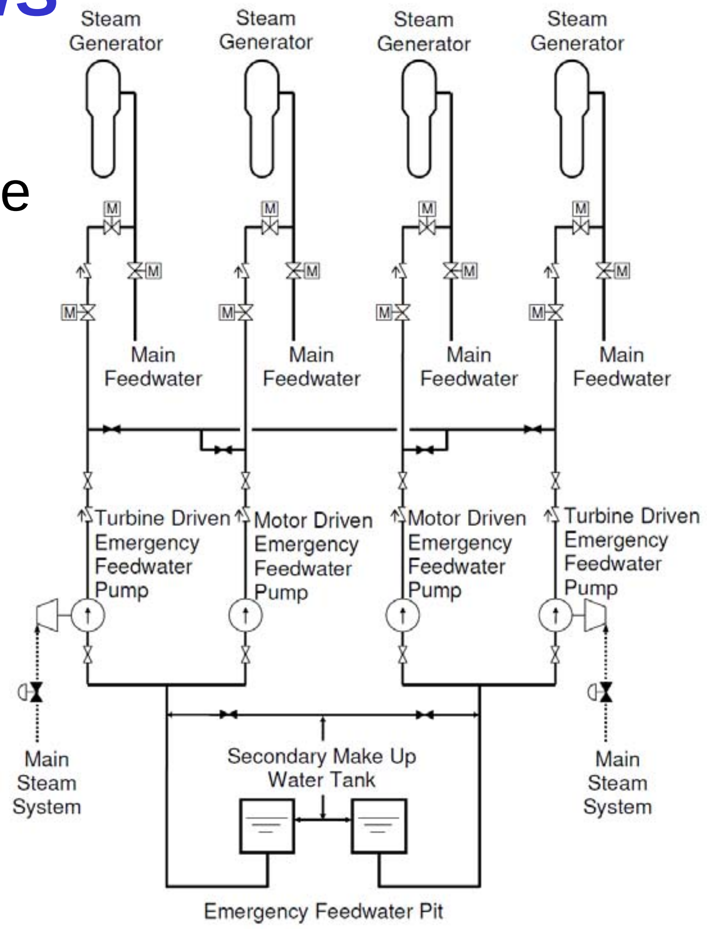
Emergency Feedwater System

➢Design concept of the EFWS

- ✓Achieve high reliability with simplified systems
- ✓Introduce On Line Maintenance assuming single failure

➢Feature of the EFWS

- ✓Independent 4 train system
- ✓2 safety grade water sources
- ✓Diverse power sources for the pumps
- ✓Cross connection inlet and outlet of the pumps (normally isolated)



Emergency Feedwater System

- **4 train configuration**
  - ✓ 4 pumps with diverse power sources
    - 2 motor-driven pumps
    - 2 turbine-driven pumps
  - ✓ Cross connection inlet of the pumps allows On Line Maintenance (OLM)
- **2 safety grade independent feedwater sources**
  - ✓ Two 50% capacity emergency feedwater pits
  - ✓ Cross connection inlet of the pumps backs up each feedwater source

Item	US-APWR	US Current 4 Loop Plant	Reason and/or Advantage
System Configuration	4 train	2 train	A pump is allowed OLM under the single failure
Emergency Feedwater Pump	M/D EFWP: 2 T/D EFWP: 2	M/D EFWP: 2 T/D EFWP: 1	Diverse power sources
Emergency Feedwater Source	2	1	2 independent pits (backup available)

CCWS & ESWS

(Component Cooling Water System & Essential Service Water System)

➢ Design concept

- ✓ CCWS and ESWS constitute a safety cooling chain
- ✓ Achieve high independency and high reliability
- ✓ Allows On Line Maintenance assuming single failure

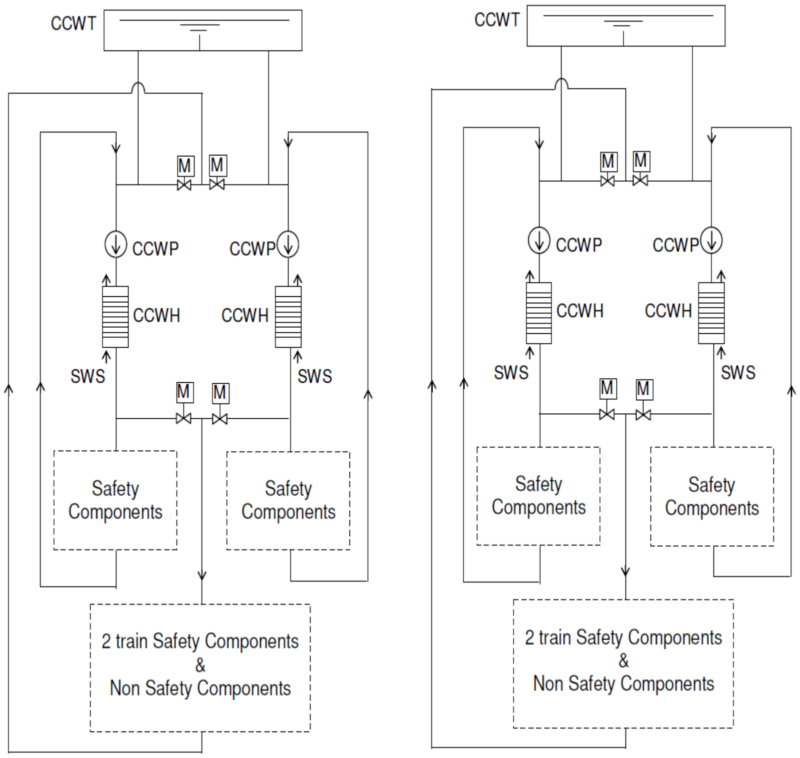
Component Cooling Water System

➢Component Cooling Water System

✓ 4 safety train configuration

✓ Completely separated into 2 independent sections even in normal operation

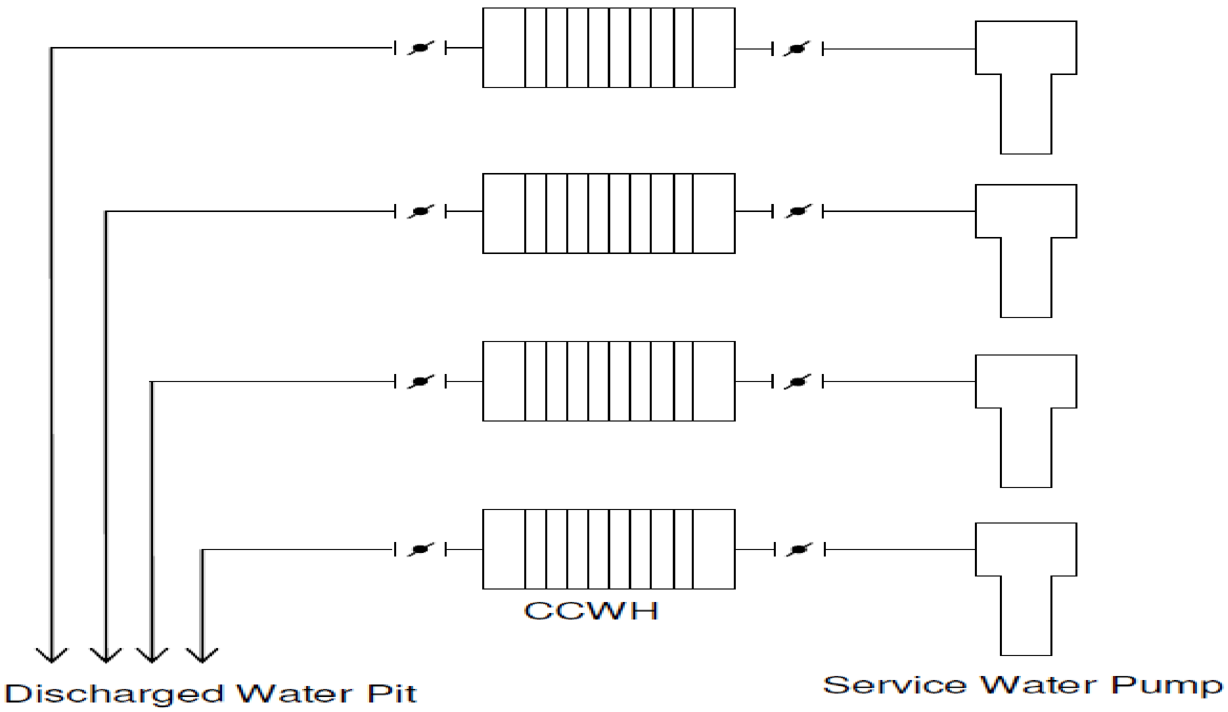
- OLM available in each train
- 2 train safety components (e.g.;SFP Hx.) are supplied with cooling water from 2 of 4 safety trains



Essential Service Water System

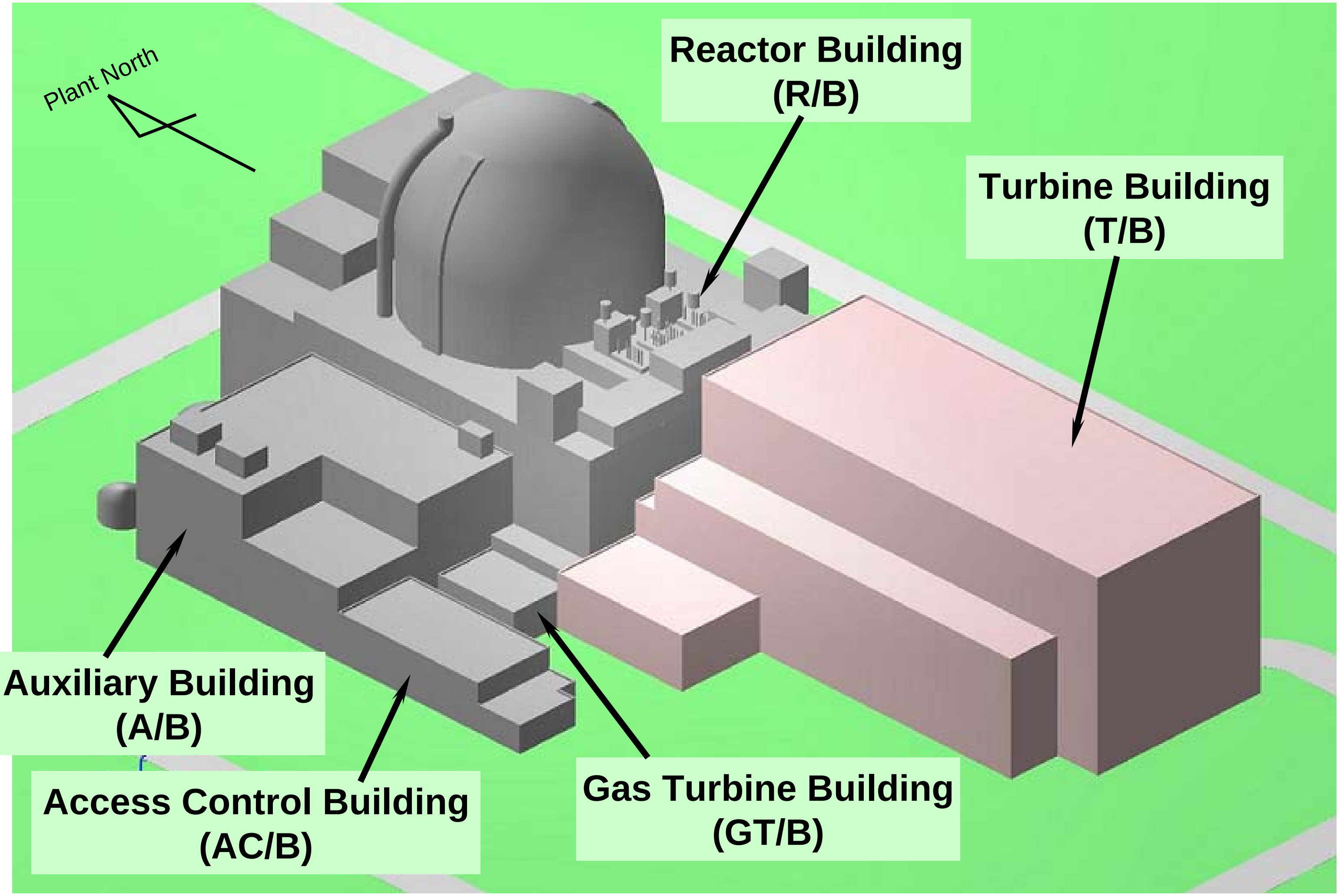
➢Essential Service Water System

- ✓Completely independent 4 train configuration
- ✓Raw water cooling only for the CCW Hx.



3.4 General Arrangement

Arrangement of Main Power Block



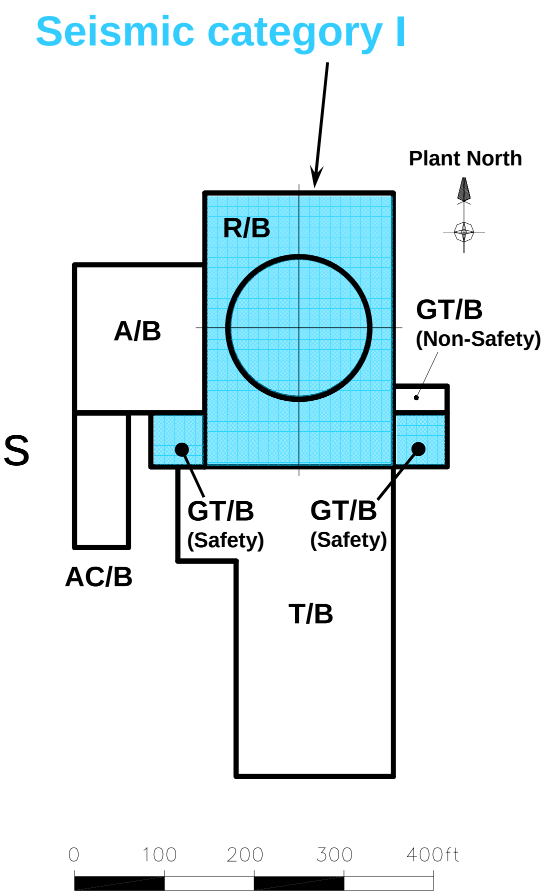
Arrangement of Main Power Block

➢ Reactor Building (R/B)

- ✓ Containment vessel
- ✓ In-containment RWSP
- ✓ Safety-related Pumps and Hxs
- ✓ Safety-related Electrical, I&C and HVAC
- ✓ Fuel Handling and Storage Facilities

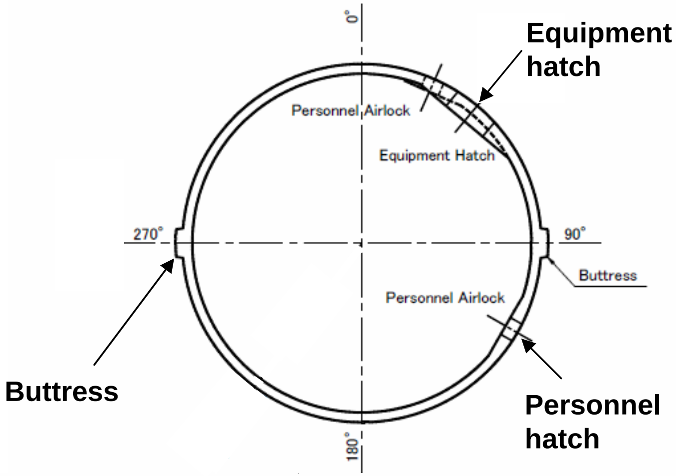
➢ Safety Related Gas Turbine Building (GT/B)

- ✓ The safety-related gas turbine generators

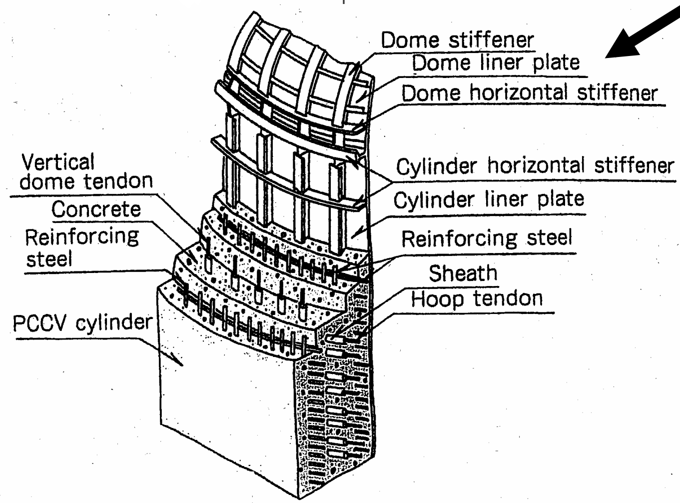


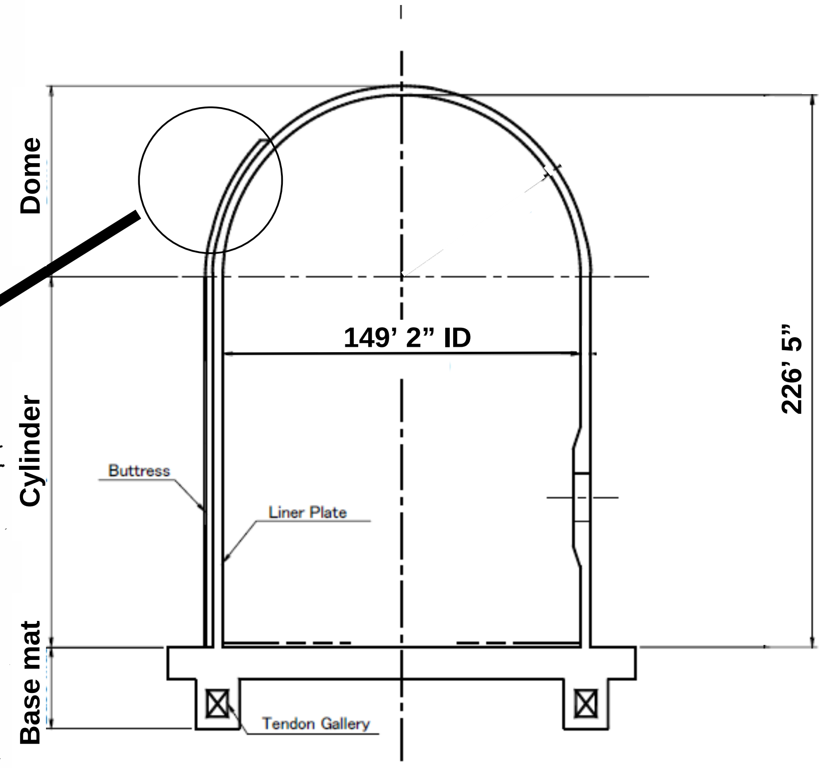
Containment Vessel Design

➢Robust and reliable pressure vessel with steel liner



Design Pressure	68 psig
Maximum Allowable Leakage Rate	0.1 percent / day

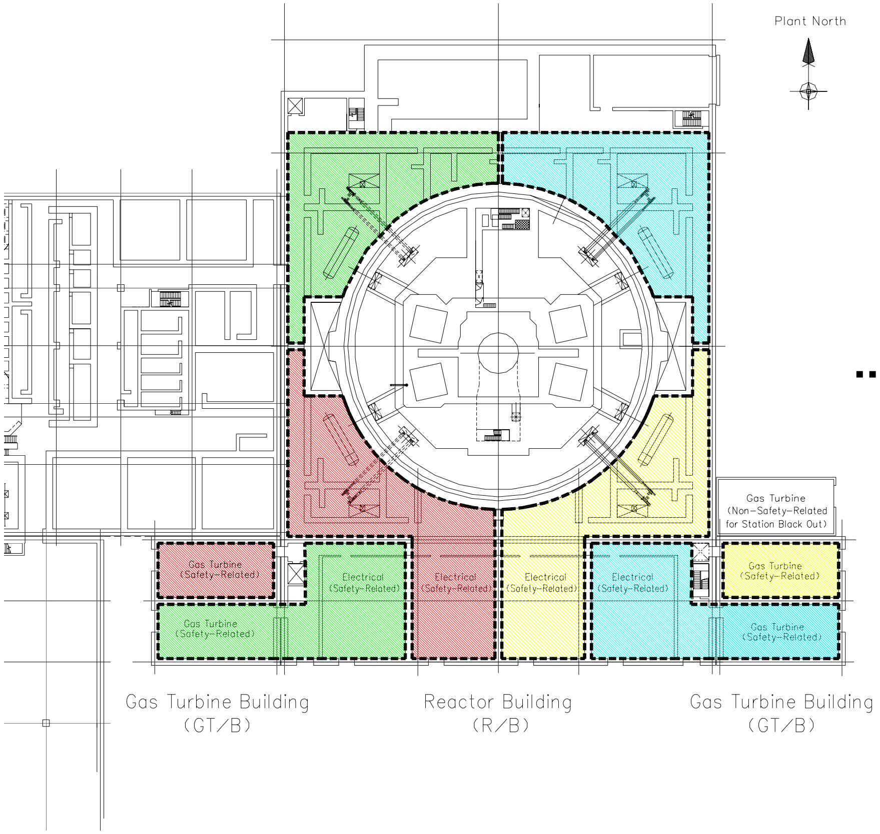




Separation of Safety System

➢ Separation of 4 train safety related systems

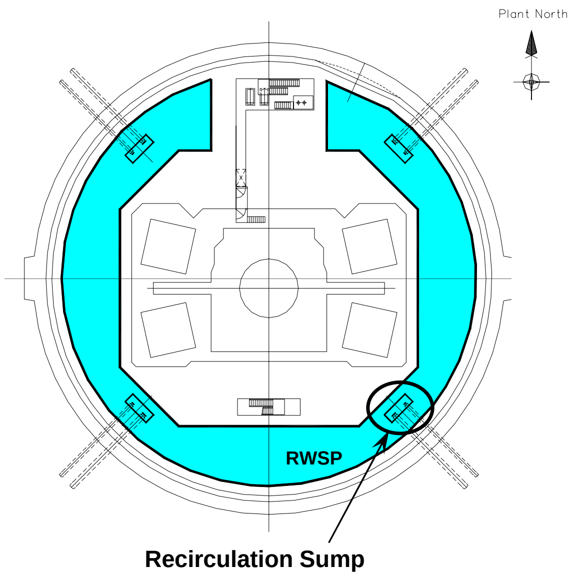
✓ **Separated by physical barrier**

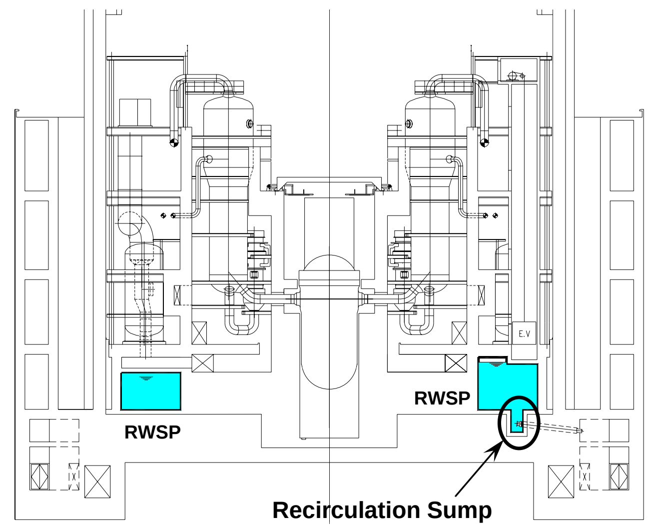


……· : physical barrier

In-containment Refueling Water Storage Pit

- ✓ Located at the lowest part of containment
- ✓ Provides a continuous suction source for both the safety injection and the CS/RHR pumps
  (Eliminates the switchover of suction source)
- ✓ 4 recirculation sumps are installed





3.5 I & C and Electrical System

History of Digital Application

- **Introduced in operating plants for**
  - ✓ Improvement of plant safety and availability
  - ✓ Improvement of operability and monitoring capability
  - ✓ Proper balance of costs and performance
- **Compliance with US codes & standards**
- **Conservative phase-in of digital technology**
  - ✓ Five plants each with average 10 years operation
  - ✓ Applied to all non-safety I&C, 50 applications per plant
  - ✓ Over 20 million hours total operating experience
  - ✓ No system malfunction caused by S/W or H/W failure
- **Excellent Non-Safety history now allows digital application to Safety and HSI System**

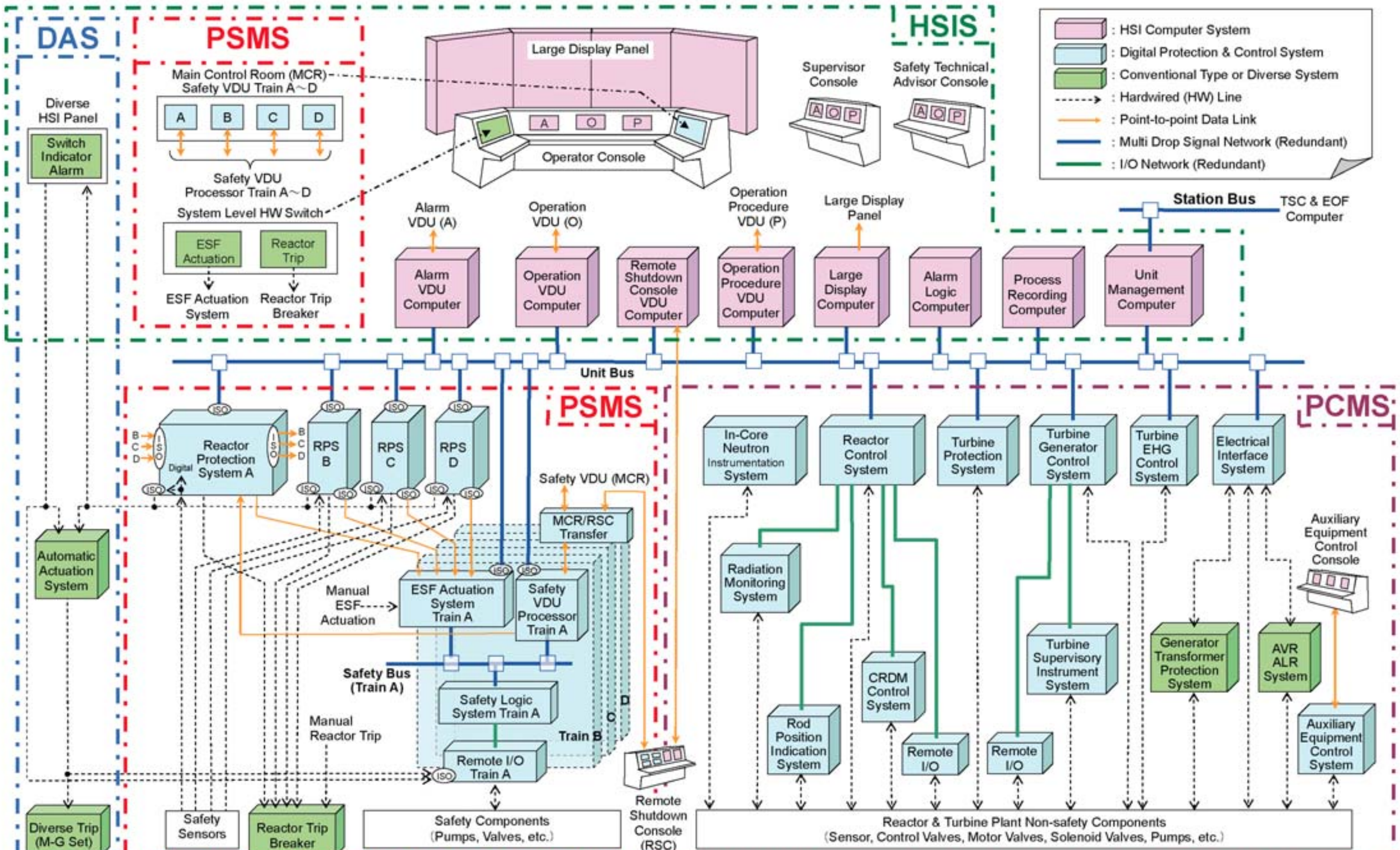
Main Features

- **Full Digital I&C System**
  - ✓ Soft HSI for all control and monitoring
  - ✓ Minimum inventory of Fixed Position & Conventional HSI
  - ✓ Micro-processor based safety and non-safety systems
  - ✓ Multiplexed communication including class 1E signals
- **Based on Defense-in-Depth & Diversity concept**
- **4 train redundant Safety System configuration**
- **Redundant configuration for Non-Safety Systems**
- **Maximum standardization with diverse back-up Non-Safety System for CMF**

Application Plan

- **DCD of US-APWR will include followings;**
  - ✓ Plant-wide architecture
  - ✓ System level design of Safety related I&C systems
  - ✓ HSI concept and HFE (Human Factor Engineering) process
  - ✓ Detailed Design Acceptance Criteria and ITAAC
- **Provide Generic Topical Report of Digital Platform in conjunction with US-APWR DC application**
  - ✓ Detail design of S/W and H/W for Digital Platform
  - ✓ Detail design process and architecture of Safety System
- **Provide Generic Topical Report of HFE Process in conjunction with US-APWR DC application**
  - ✓ Human Factor Engineering (HFE) process
  - ✓ Detail HSI Design and V&V results by US operator

I&C Overall Architecture



DAS : Diverse Actuation System   PSMS : Protection and Safety Monitoring System   HSIS : Human System Interface System   PCMS : Plant Control and Monitoring System

Safety I&C Design Features

- **PSMS: Protection & Safety Monitoring System**
  - ✓ Reactor Trip and ESF Actuation Functions
  - ✓ Monitoring of the Safety Critical Parameters (RG 1.97)
  - ✓ Control of ESF, Safe Shutdown and Important Interlocks
- **4 train redundancy for all Safety Systems**
- **Electrical, physical, functional, communication and data isolation to accommodate**
  - ✓ Signal interface between safety systems
  - ✓ Signal interface safety to non-safety systems
  - ✓ Non-safety HSI for safety functions with back-up safety HSI
- **Integrity of software per BTP-14 and BTP-19**
- **Qualified platform with extensive experience**

Non-safety I&C Design Features

➢ PCMS: Plant Control and Monitoring System

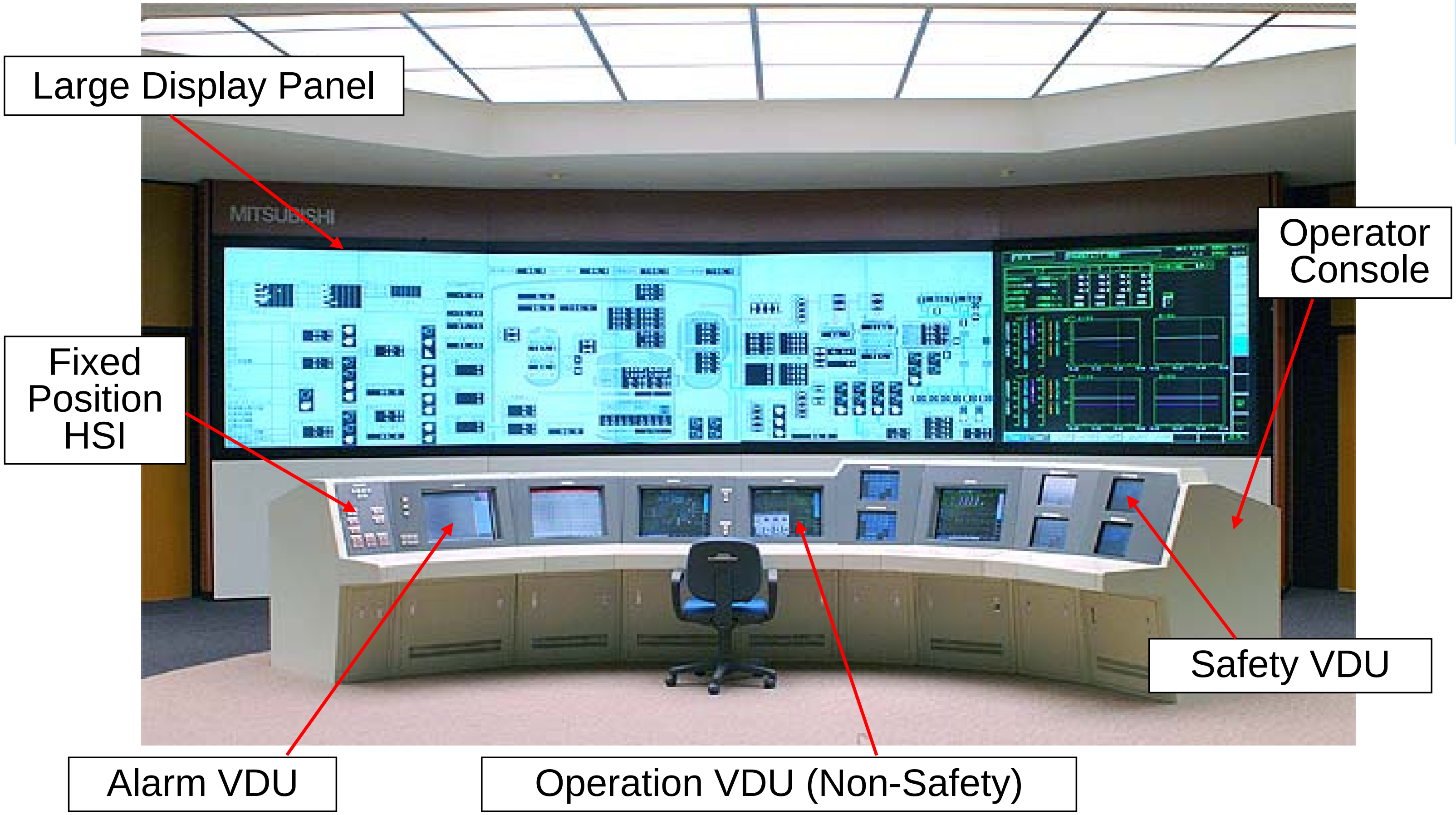
- ✓ Reactor, Turbine / Generator, BOP control & monitoring
- ✓ Redundant and fault tolerant configuration
- ✓ Functional isolation of shared sensor signals from PSMS with Automatic Signal Selector
  (for compliance with 10CFR 50 Appendix A GDC 24)

➢ DAS: Diverse Actuation System

- ✓ Countermeasure for CMF in the digital safety system
  (according to BTP-19 for Software CMF & 10CFR 50.62 for ATWS)
- ✓ Diverse automation: Reactor Trip, Turbine Trip, EFWS
  (based on actuations required within 10 minutes of event)
- ✓ Diverse manual controls: SI, CV Isolation, etc.
  (diverse component level actuation for critical safety functions)
- ✓ Diverse monitoring: Safety critical parameters
  (diverse signal processing and HSI)

HSI System

Computerized Main Control Room



HSI System Design Features

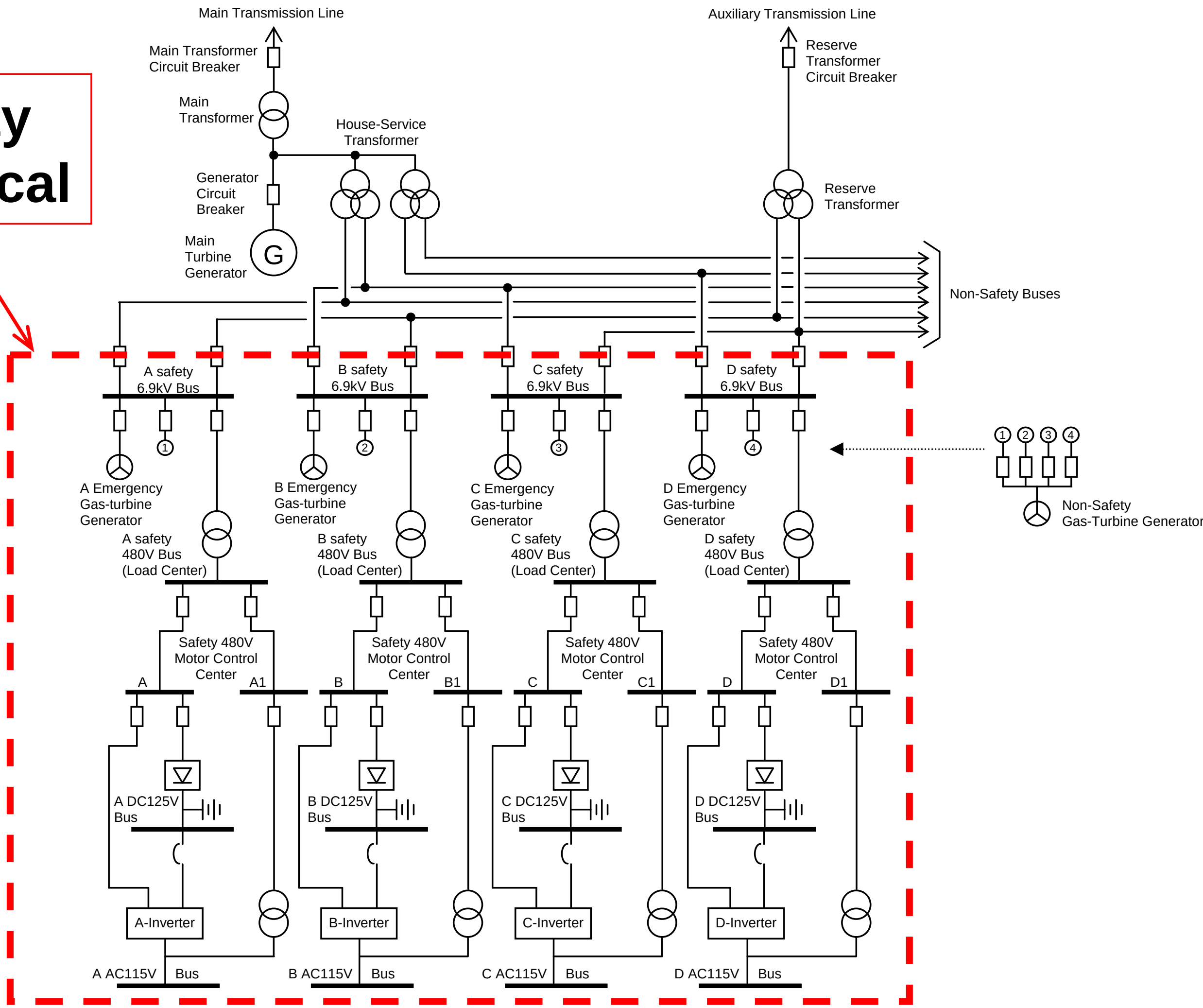
- **Functions of HSI System (Human System Interface System)**
  - ✓ Fully computerized video based approach
  - ✓ Integrated display for monitoring and soft control
  - ✓ Dynamic alarm prioritization & computerized procedures
- **Development Process (NUREG 0711 Program Model)**
  - ✓ Task analysis and human factor design
  - ✓ Step by step prototyping and V&V by plant operators
- **Staffing**
  - ✓ Operator, Supervisor & Safety Technical Advisor
- **Main features**
  - ✓ All operations are available from non-safety VDUs
  - ✓ All safety operations from back-up safety VDUs
  - ✓ Minimum inventory of Fixed Position HSI
    - Conventional HSI for manual system level actuation (RG 1.62)
    - Critical Functions & Bypass or Inoperable Status (RG 1.47)

Electrical System Design Features

- **2 offsite transmission systems**
- **4 train onsite Safety Power Systems**
- **4 train Safety Gas-Turbine Generators (GT/G)**
  - ✓ Starting time requirement is extended with GT/G
  - ✓ Higher reliability and easier maintenance than DG
- **4 train Safety Batteries**
  - ✓ 2 hours capability for loss of AC power
  - ✓ Supplies power to equipment that must be energized prior to starting time requirement of GT/G
- **One non-safety GT/G alternate AC power**
  - ✓ Supplies power to equipment required during SBO
  - ✓ Starting time requirement : 5 minutes
  - ✓ Manually connected to safety bus at SBO initiation

Electrical System Diagram

Safety Electrical



4. Computer Codes and Methodology Used for Safety Analysis

Contents

- **Basic Concepts of Methodology Selection**
- **Nuclear, Thermal & Hydraulic and Fuel Design**
- **Safety Analysis**
- **Summary of Report Submittal Plan**

➢The US-APWR design

- ✓The US-APWR design based on conventional U. S. PWR
- ✓No need for new method

➢Approach for code and methodology selection

- ✓Maximum use of methods already accepted by the NRC
- ✓Correlation of technical parameters related to the US-APWR unique design features, such as advanced accumulator

➢Nuclear Design

✓PARAGON/ANC

- Approved by the NRC

➢Thermal and Hydraulic Design

✓WRB-2 : DNB correlation

✓VIPRE-01 : Subchannel analysis

- Approved by the NRC

✓RTDP : Revised thermal design procedure

- Approved by the NRC

✓Topical report regarding the combination of the techniques will be submitted in May 2007

➢Fuel Design

✓FINE

- Developed by MHI

Safety Analysis

➢Large Break LOCA

✓WCOBRA/TRAC + Statistical Treatment

- Approved by the NRC
- US-APWR features
  - Advanced accumulator
- The applicability to the US-APWR and the model difference from the approved one will be reported and submitted

➢Small Break LOCA

✓RELAP5 + Appendix-K model (conservative model)

- Widely used in the U.S.A.
- US-APWR Features
  - Advanced accumulator
- The applicability to the US-APWR and the conservatism for the evaluation model will be reported and submitted

Safety Analysis

➢Containment Response

✓GOTHIC

- Significant experience for licensing analysis in the U.S.A.

➢Mass and Energy Release

✓SATAN-VI, WREFLOOD

- Approved by the NRC
- US-APWR Features
  - Advanced accumulator model
- The applicability to the US-APWR and the model difference from the approved one will be reported and submitted

Safety Analysis

➢Non-LOCA

✓MARVEL : Plant response

- Approved by the NRC
- Model improvement
  - 2-Loop ➡ 4-Loop
  - Built in RCP model

✓TWINKLE : Core response after reactivity insertion events

- Approved by the NRC

✓VIPRE-01 : DNB & fuel transient analysis

- Same as the thermal-hydraulic design

➢Dose Evaluation

✓RADTRAD, PWR-GALE, etc.

- Substantial licensing experience in the USA

Summary of Report Submittal Plan

Design	Code / Correlation / Methodology	NRC Approved Topical Report	Submittal Date Plan
Nuclear design	PARAGON/ANC	WCAP-16045-P-A	N/A
Thermal & Hydraulic design	WRB-2	To be reviewed for MHI fuel by NRC	May, 2007
	VIPRE-01	NP-2511-CCM-A (Rev.4)	
	RTDP	WCAP-11397-P-A	N/A
Fuel design	FINE	To be reviewed by NRC	May, 2007

Summary of Report Submittal Plan

Event	Code / Methodology	NRC Approved Topical Report	Submittal Date Plan
Large Break LOCA	WCOBRA/TRAC + Statistical Treatment	WCAP-16009-P-A	July, 2007
Small Break LOCA	RELAP5 + App-K	To be reviewed by NRC	
Mass and Energy Release	SATAN-VI WREFLOOD	WCAP-10325-P-A	
Containment Response	GOTHIC	N/A	N/A (Substantial licensing experience in the USA)
Non-LOCA	MARVEL TWINKLE VIPRE-01	WCAP-8844 WCAP-7979-P-A NP-2511-CCM-A, Rev. 4	July, 2007
Dose Evaluation	RADTRAD PWR-GALE, etc.	N/A	N/A (Substantial licensing experience in the USA)

5. Next Meeting

Agenda for Next Meeting

- **Fuel and Core Design Overview**
- **Safety Analysis Methodology Overview**
- **Severe Accident Analysis Methodology Overview**
- **Contents of Topical Reports**