US-APWR
Design Certification Application Orientation

1. US-APWR DCD Scope
2. Use of DAC in Completion of Standard Design
3. US-APWR Report Submittal Plan

January 15,16, 2008

Mitsubishi Heavy Industries, Ltd.

Presenter

US-APWR

Masayuki Kambara

Project Manager

New Plant Engineering

Mitsubishi Nuclear Energy Systems, Inc.

Contents

1. US-APWR DCD Scope
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1. US-APWR DCD Scope

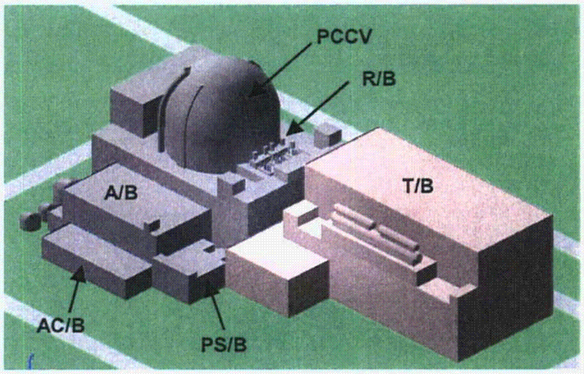
US-APWR DCD Scope for Buildings and Structures

Regulatory Position (10 CFR 52.47)

An application for certification of a nuclear power reactor design must provide an essentially complete nuclear power plant design except for site-specific elements such as the service water intake structure and the ultimate heat sink.

<u>Buildings within US-APWR DCD Scope</u>

- R/B : Reactor Building
- PCCV : Prestressed Concrete Containment Vessel
- PS/B : Power Source Buildings
- T/B : Turbine Building
- A/B : Auxiliary Building
- AC/B : Access Building



US-APWR DCD Scope for Buildings and Structures (cont'd)

Building and Structure (1)	Required by 10CFR52.47	Seismic Category	Safety related systems housed in	Tier 1							Tier 2				
				Building and Structures				Systems in the Building			Building and Structures			Systems in the Building	
				Description	Layout Drawing	Building Dimensions (2)	ITAAC	Description	Process Flow Diagram	ITAAC	Description	Layout Drawing	Building Dimensions (2)	Description	Process Flow Diagram
Reactor Building (incl. CV)	X	1	Yes	X	X(3)	X	X	X	X	X	X	X(3)	X	X	X
Power Source Building	X	1	Yes	X	X(3)	X	X	X	X	X	X	X(3)	X	X	X
Auxiliary Building	X	2	No	X	X(4)	-	X	X	X	X	X	X(5)	- (6)	X	X
Turbine Building	X	2	No	X	X(4)	-	X	X	X	X	X	X(5)	- (6)	X	X
Access Building	X	Non seismic	No	X	X(4)	-	X	X	-	X	X	X(5)	- (6)	X	X
Tunnel, Vault and Fuel Tank for Class 1E EPS (Seismic 1, ASME 3 piping)	X	1	Yes	X	X(4)	-	X	X	-	X	X	X(4)	- (7)	X	X

Note)
(1): Other buildings including site-specific elements such as the service water intake structure and the ultimate heat sink are not in the DCD scope.
(2): Building dimensions include wall/slab thickness and structural element locations.
(3): A general arrangement of the components in the building included.
(4): Only a plot (typical) plan i.e. a building arrangement at site included.
(5): A general arrangement of typical components (including radiation shielding wall thickness) for dose evaluation, fire hazard analysis, and flooding is provided.
(6): Structural dimensions will be finalized considering radiation shielding, fire hazard analysis, and flooding requirements by COL Applicant.
(7): Structural dimensions will be finalized based on the site specific arrangement by COL Applicant.

2. Use of DAC in Completion of Standard Design

Previous Applicability of DAC(SECY-92-053)

- Regulatory position
  - The use of DAC should be based upon a consideration of design areas affected by rapidly changing technologies, or design areas which as-built, or as-procured, information is not available at the time of DCD submittal.
  - DAC are to be objective (measurable, testable, or subject to analysis using pre-approved methods), and must be verified as a part of the ITAAC performed to demonstrate that the as-built facility conforms to the certified design.
- The NRC staff previously identified design areas using DAC
  - Piping design
  - I&C design
  - HFE design (Control room design)
  - Radiation shielding design
- The NRC staff may find justification for the use of DAC in other review areas for other designs.

Design areas that use DAC in US-APWR

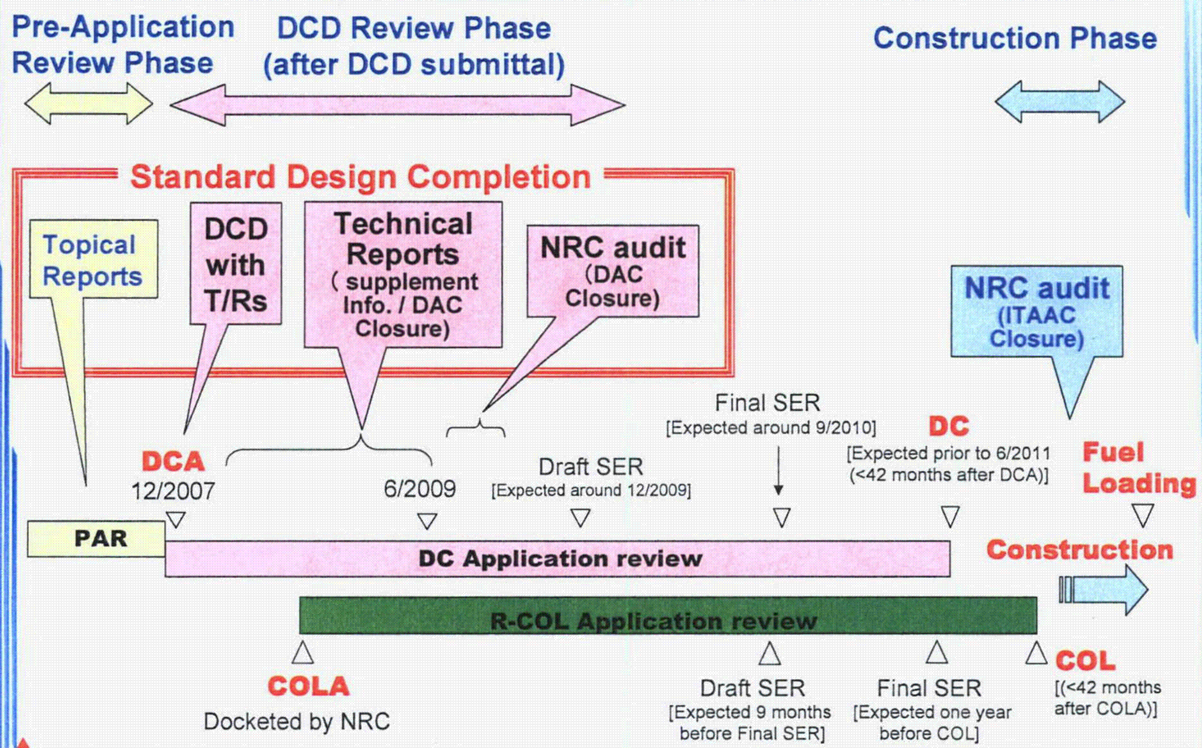
MHI is using DAC for the following design areas:

- Stress Analyses:
  - Piping: ASME Class 1, 2, 3
  - Components:
    - ASME Class CS
    - ASME Class 1, 2, and 3
    - Others (New and spent fuel storage racks)
  - Fuel Assemblies and RCC
- Human Factor Engineering (HFE) design

US-APWR Standard Design Completion

- Standard design completion will be accomplished and verified:
  - *Topical reports* [during Pre-application review phase]
  - *DCD with Technical reports* [Dec. 2007]
  - *Technical reports* [after the DCD submittal]
    - Will contain analyses and other information that supplement the materials included in the DCD and are incorporated by reference therein.
    - Will close DAC (Design ITAAC) for representative examples of ASME class CS, ASME class 1/2 piping and components, Fuel assemblies/RCC, and HFE design
  - *NRC Audit* [during DCD review phase]
    - Review of activities and documents related to detailed design
    - Will close DAC (Design ITAAC) for remaining ASME class 1 piping and components and new design of ASME class 2 components.
- Remaining detailed design results will be verified and/or reconciled during plant construction phase
  - Close ITAAC for ASME class 2 & 3 piping and components

US-APWR Standard Design Completion (cont'd)



US-APWR Approach for DAC-ITAAC

Areas	Use of DAC			Supplier: MHI	Supplier: Purchased	DCD at submittal: DAC	DCD at submittal: ITAAC Design	DCD at submittal: ITAAC Construction	During DCD Review: ITAAC met Design	During DCD Review: ITAAC met Construction	COL / Construction: ITAAC met Design	COL / Construction: ITAAC met Construction	T/R or Audit Available Schedule
Components and pipng	ASME Class CS	Reactor Internals	Stress Analyses	X	-	X	X	X	X (T/R)	-	-	Reconciliation	T/R 6/2009
	ASME Class 1	Reactor Vessel		X	-	X	X	X	X (T/R)	-	-	Reconciliation	T/R 6/2009
		Steam Generator Pressurizer Reactor Coolant Pump CRDM		X	-	X	X	X	X (Audit)	-	-	Reconciliation	Audit 9/2009
		Reactor Coolant Loop Piping and Branch Piping		X	-	X	X	X	X (Audit)	-	-	Reconciliation	Audit 9/2009
		Pressurizer Surge Line Piping		-	X	X	X	X	X (T/R)	-	-	Reconciliation	T/R 6/2009
		Valves		-	X	X	X	X	-	-	X	Reconciliation	N/A
	ASME Class 2	Accumulator		X	-	X	X	X	X (Audit)	-	-	Reconciliation	Audit 9/2009
		Other Components		-	X	X	X	X	-	-	X	Reconciliation	N/A
		MS Piping		-	X	X	X	X	X (T/R)	-	-	Reconciliation	T/R 6/2009
		Other Piping		-	X	X	X	X	-	-	X	Reconciliation	N/A
	ASME Class 3	Components		-	X	X	X	X	-	-	X	Reconciliation	N/A
		Piping		-	X	X	X	X	-	-	X	Reconciliation	N/A
	Others	New and spent fuel storage racks		-	X	X	X	X	X (T/R)	-	-	Reconciliation	T/R 3/2009
Fuel system	Fuel assemblies and RCC		structure response analysis under seismic and LOCA	X	-	X	X	-	X (T/R)	-	-	-	T/R 6/2009
I & C	HSI Design			-		X	X	X	X (T/R)	-	-	X	T/R 6/2009
	US Operator V & V					X	X	X	X (T/R)	-	-	X	T/R 12/2008

Note) T/R: Technical Report; HSI: Human Interface System; V&V: Verification and Validation.

3. US-APWR Report Submittal Plan

Topical Report and Technical Report

- Prior to DCD, MHI submitted 12 topical reports and 1 technical report to the NRC.
  - Reports were part of DCD pre-application activities
  - Reports address specific technical topics to assure that the methodologies included in the DCD are acceptable and are in accordance with the NRC's guidance.
  - Reports referred to in the DCD
- MHI submitted 15 additional technical reports with DCD on Dec.31, 2007.
- Additional technical reports will be submitted in support of the Design certification application during DCD review phase.

Number of Reports

Chapter	Pre-application Review Phase	DCD Acceptance Review Phase		DCD Review Phase
		With DCD	After DCD	
1	-	1	-	1
2	-	-	-	-
3	-	1	1	6
4	2	5	1	2
5	-	1	-	1
6	2	-	1	-
7	3	1	1	-
8	1	-	-	-
9	-	-	1	1
10	-	2	-	-
11	-	-	-	-
12	-	-	-	-
13	-	-	1	2
14	-	-	-	-
15	3	1	-	-
16	-	2	-	-
17	1	-	-	-
18	1	-	-	2
19	-	1	-	1
Total	13	15	6	16

Structure, Components and Piping
(Ch.3 and Ch.5)

Pre-application Review Phase

-

DCD Acceptance Review Phase

- Comprehensive Vibration Assessment Program for US-APWR Reactor Internals (12/2007)
- Structure Analysis for US-APWR RCP Motor Flywheel (12/2007)
- Enhanced information for PS/B design (2/2008)

DCD Review Phase

- Dynamic analysis of the coupled RCL-R/B-PCCV-CIS Lumped Mass Stick Model (4/2008)
- Environmental Qualification Program (12/2008)
- Summary of design transient (1/2009)
- Summary of stress analysis results (Reactor Internals, Reactor Vessel, Pzr Surge line, M/S line; 6/2009)

Fuel System
(Ch.4)

US-APWR

Pre-application Review Phase

- Mitsubishi Fuel Design Criteria and Methodology (5/2007)

DCD Acceptance Review Phase

- US-APWR Fuel System Design Parameters List (12/2007)
- US-APWR Fuel System Design Evaluation (2/2008)

DCD Review Phase

- FINDS: Mitsubishi Fuel Assemblies Seismic Analysis Code (3/2008)
- Evaluation Results of Structural Response Analysis of US-APWR Fuel System under Seismic and LOCA (6/2009)

Nuclear, Thermal Hydraulics, Reactor Internals
(Ch.1 and Ch.4)

US-APWR

Pre-application Review Phase

- Thermal Design Methodology (5/2007)

DCD Acceptance Review Phase

- APWR Reactor Internals 1/5 Scale Model Flow Test Report(12/2007)
- Qualification of Nuclear Design Methodology using PARAGON/ANC (12/2007)
- Validation of Criticality Safety Methodology (12/2007)
- US-APWR In-Core Power Distribution Evaluation Methodology (12/2007)
- FMEA for Control Rod Drive Mechanism Control System (12/2007)

DCD Review Phase

- US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Report (7/2008)

ESF and Safety Analyses
(Ch.6 and Ch.15)

US-APWR

Pre-application Review Phase

- Advanced accumulator (3/2007)
- Large Break LOCA Code Applicability Report for US-APWR (7/2007)
- LOCA Mass and Energy Release Analysis Code Applicability Report for US-APWR (7/2007)
- Small Break LOCA Methodology for US-APWR (7/2007)
- Non-LOCA Methodology (7/2007)

DCD Acceptance Review Phase

- Small Break LOCA Sensitivity Analyses for US-APWR (12/2007)
- Subcompartment analysis for US-APWR Design Confirmation(2/2008)

DCD Review Phase

-

I & C and HFE
(Ch.7 and CH.18)

US-APWR

Pre-application Review Phase

- Safety I & C system design process and description (3/2007)
- Safety system digital platform - MELTAC - (3/2007)
- Defense-in-Depth and Diversity (4/2007)
- HSI system description and HFE process (4/2007)

DCD Acceptance Review Phase

- Defense in Depth and Diversity Coping Analysis (12/2007)
- Software Program Manual (2/2008)

DCD Review Phase

- US operator static V & V results (including HFE analysis results) (12/2008)
- HSI design (6/2009)

Miscellaneous
(Ch.8, Ch.9, and Ch.10)

US-APWR

Pre-application Review Phase

- Qualification and test plan for gas turbine generator (11/2007) : Ch.8

DCD Acceptance Review Phase

- Probability of Missile Generation from Low Pressure Turbines (12/2007) : Ch.10
- Probabilistic Evaluation of Turbine Valve Test Frequency (12/2007) : Ch.10
- Criticality analysis for US-APWR new and spent fuel racks (2/2008) : Ch.9

DCD Review Phase

- Mechanical analyses for US-APWR new and spent fuel racks (3/2009) : Ch.9

Miscellaneous
(Ch.16, Ch.17, and Ch.19)

US-APWR

Pre-application Review Phase

- Quality assurance program description for design certification of the US-APWR (1/2007) : Ch.17

DCD Acceptance Review Phase

- Justification for deviations between NUREG-1431 Rev.3.1 and US-APWR Technical Specification (12/2007) : Ch.16
- Mitsubishi Reload Evaluation Methodology (12/2007) : Ch.16
- Probabilistic Risk Assessment (Detailed design-specific Level 1 and Level 2 PRA and severe accident evaluation) (12/2007) : Ch.19

DCD Review Phase

- Probabilistic Risk Assessment (Level 3) (3/2008) : Ch.19

Physical Security
(Ch.13)

US-APWR

Pre-application Review Phase

-

DCD Acceptance Review Phase

- Design Certification Physical Security Elements Review (2 weeks after MHI receives clearance for SGI)
  [Identification of Vital equipment and Vital areas and Other required information]

DCD Review Phase

- Security assessment report (7/2008)
  - High assurance evaluation
  - Mitigative measures evaluation
  - Cyber assurance evaluation
- Evaluation of mitigation of Beyond DBT aircraft crash (To be determined later)

Summary

US-APWR

- US-APWR DCD scope complies with regulatory position.
- Completion of standard design
  - ✓ Most of the standard design is presented in the DCD and the Topical reports/Technical reports that have been submitted.
  - ✓ Additional technical reports will be submitted after the filing of the DCD and will become part of the DCD.
  - ✓ DAC is applied to the areas where there are constraints to completing design such as stress analyses of components/piping and HFE design.
- MHI will closely communicate with the NRC staffs to optimize the review process.

US-APWR

Design Certification Application Orientation

Detail of FSAR
Tier 1

January 15,16, 2008
Mitsubishi Heavy Industries, Ltd.

Presenter

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Contents

1. Overview

- **Tier 1 contents**
  - ✓ Based on SRP 14.3.
  - ✓ Contains information to be certified by the NRC.
    - Tier 2 contains more detailed information that is approved but not certified by the NRC.
  - ✓ The type of information and the level of detail are based on a graded approach commensurate with the safety significance of the SSCs.
    - The Tier 1 material is derived from the Tier 2 material.
  - ✓ The top-level information includes the principal performance characteristics and safety functions of the SSCs, which are to be verified appropriately by ITAAC.
    - The successful completion of all ITAAC is a prerequisite for fuel load and a condition of the license.
    - The Tier 1 provides ITAAC for structures and systems within the scope of the DCD.

1. Overview

➢ Organization

✓ Chapter 1: Introduction
- Discussion of Tier 1 contents, definitions, format etc.

✓ Chapter 2: Design Description & ITAAC
- Top-level design description and commitment
- Tables and figures
- ITAAC

✓ Chapter 3: Interface Requirements
- Safety significant Interfaces between the standard design and COL applicantion.

1. Overview

Section	Subject	Section	Subject
1.0	Introduction	2.8	Radiation Protection
2.0	Design Descriptions and ITAAC	2.9	Human Factors Engineering
2.1	Site Parameters	2.10	Emergency Planning
2.2	Structural and Systems Engineering	2.11	Containment Systems
2.3	Piping Systems and Components	2.12	Physical Security Hardware
2.4	Reactor Systems	2.13	Design Reliability Assurance Program
2.5	Instrumentation and Controls	2.14	Initial Test Program
2.6	Electrical Systems	3.0	Interface Requirements
2.7	Plant Systems		

1. Overview

- **General approach**
  - ✓ Derived from Tier 2
  - ✓ Address the most safety-significant aspects, top-level design features, performance characteristics most significant to safety
  - ✓ Level of detail governed by graded approach (more detail in safety-related systems)
  - ✓ Figures supplement narrative in most cases
  - ✓ Regulations, codes, and standards not cited in general
  - ✓ Tables and figures identify SSCs to be verified by ITAAC

2. Design Description

- **System design descriptions are to address, as applicable**
  - ✓ Significant performance characteristics and safety functions
  - ✓ Whether the system is safety related or not
  - ✓ System location
  - ✓ Key design features
  - ✓ Seismic and ASME Code classifications
  - ✓ Important alarms, displays and controls
  - ✓ Logic circuits (for direct safety functions)
  - ✓ Interlocks (for direct safety functions)
  - ✓ Interface requirements

3. ITAAC

US-APWR

➢ Format

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	1.	1.
2.	2.	2.

- ✓ 1^{st} column – proposed design requirement and/or commitment to be verified
- ✓ 2^{nd} column – proposed method by which the licensee will verify the requirement or commitment
- ✓ 3^{rd} column – proposed acceptance criteria to be met

3. ITAAC

US-APWR

➢ System based ITAAC

- ✓ Reactor Systems, Plant Systems, Containment Systems
- ✓ Instrumentation and Controls
- ✓ Electrical Systems

➢ Non-system based ITAAC

- ✓ Structural and Systems Engineering
- ✓ Piping Systems and Components
- ✓ Radiation Protection, Human Factor Engineering
- ✓ Emergency Planning
- ✓ Physical Security Hardware
- ✓ Design Reliability Assurance Program

➢ No ITAAC

- ✓ Site Parameters
- ✓ Initial Test Program

3. ITAAC

US-APWR

➢ **Contents Example for Fluid Systems (system based ITAAC)**
- ✓ System functional arrangement
- ✓ Pressure boundary integrity
- ✓ Pumps having sufficient net positive suction head
- ✓ System divisions being powered from respective Class 1E division
- ✓ Mechanical division separation
- ✓ Control room alarms, displays, and controls

3. ITAAC

US-APWR

➢ **Contents Example for Structural and Systems Engineering (non-system based ITAAC)**
- ✓ Pressure boundary integrity
- ✓ Pipe rupture
- ✓ Normal loads
- ✓ Seismic loads
- ✓ Floods, wind, and tornados
- ✓ Rain and snow
- ✓ Codes and standards
- ✓ Containment integrity
- ✓ As-built reconciliation

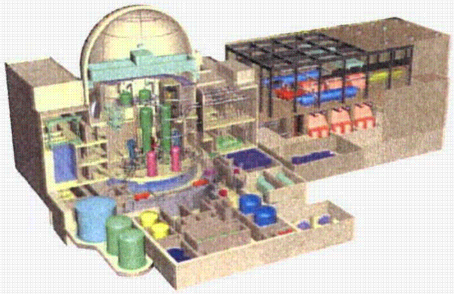
4. Summary

US-APWR

- ➢ Tier 1 contents are based on SRP 14.3.
- ➢ Tier 1 material is derived from the Tier 2, based on a graded approach.
- ➢ The top-level information are to be verified by ITAAC.

MITSUBISHI HEAVY INDUSTRIES, LTD.

Design Feature of US-APWR



MITSUBISHI US-APWR

January 15, 16, 2008

Atsushi Kumaki
Engineering Manager
APWR Promoting Department
Nuclear Energy Systems
Headquarters

Contents

1.What is US-APWR

US-APWR **satisfies U.S. customer's requirements for** ***Safety, Economy, Operation,*** **and** ***Maintenance!***

1. What is US-APWR (cont'd)

Evolutionary (not "Revolutionary") Design

- **US-APWR design is based on Japanese APWR**
- **New technologies of APWR are fully tested, well-verified and established**

1. What is US-APWR (cont'd)

US-APWR is slightly modified

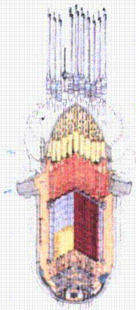
- ✓ **to increase electric output**
- ✓ **to meet the U.S. utilities requirements**

2. Key Plant Parameters

	APWR	US-APWR
Electric Output	1,538 MWe	**1,700 MWe Class**
Core Thermal Output	4,451 MWt	4,451 MWt
Core	12 ft Fuel 257Assem.	**14 ft Fuel** 257 Assem.
SG Heat Transfer Area per SG	70,000 ft^2	**91,500 ft^2**
Thermal Design Flow rate per loop	113,600 GPM	112,000 GPM
Turbine	54 inch blades	**70 inch class blades**
Containment Vessel	PCCV	PCCV
Safety Systems	Electrical 2 trains Mechanical 4 trains	**Electrical 4 trains** Mechanical 4 trains
	HHSI x 4 Advanced Accumulator x 4 Elimination of LHSI	HHSI x 4 Advanced Accumulator x 4 Elimination of LHSI
I&C	Full Digital	Full Digital

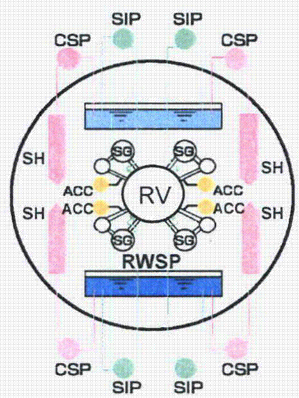
3. APWR's Advanced Technology

Reactor



- 1500 MWe class large capacity
- Neutron reflector

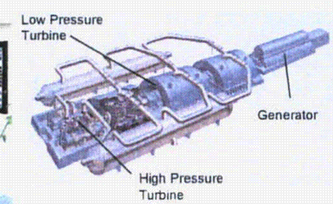
Engineering Safety Features



- Simplified configuration with 4 mechanical sub-systems
- In-containment RWSP
- Advanced accumulator

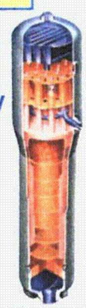
Turbine

- 54 inch-length blades in LP turbine
- Fully integrated LP turbine rotor



Steam Generator

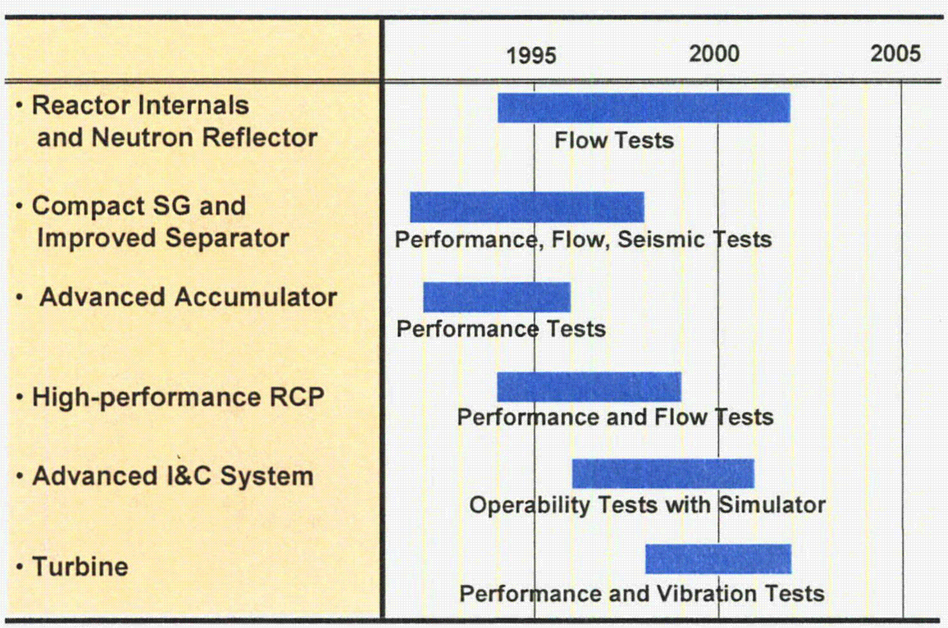
- High performance separator
- Increased capacity with compact sizing

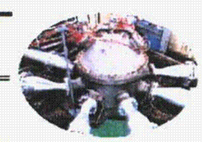


I & C

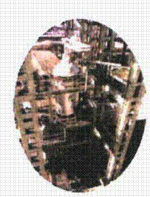
- Digital control & protection systems
- Compact console

4. Verification for Advanced Designs





Reactor Flow Test



SG Separator Test

LP Turbine Test

5. Features of US-APWR

Enhanced Safety

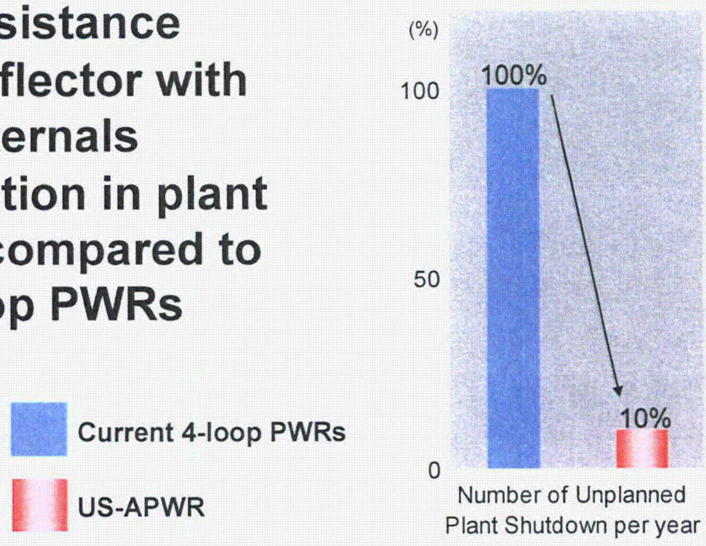
- Four-train safety systems for enhanced redundancy
- advanced accumulator
- In-containment refueling water storage pit
- Fully digital I&C

➢ CDF of US-APWR is reduced to 1/20 of operating PWRs in U.S.

5. Features of US-APWR

Enhanced Reliability

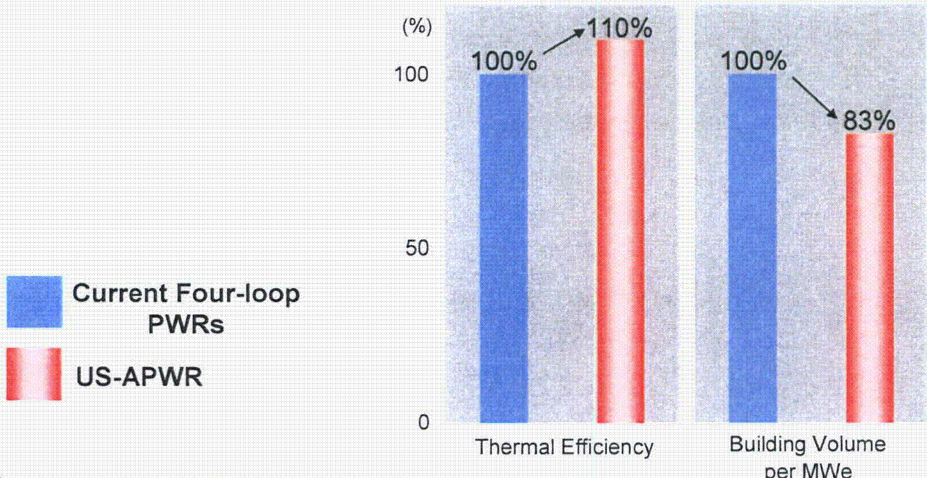
- A steam generator with high corrosion resistance
- A neutron reflector with improved internals
- A 90% reduction in plant shutdowns compared to current 4-loop PWRs



5. Features of US-APWR

Attractive Economics

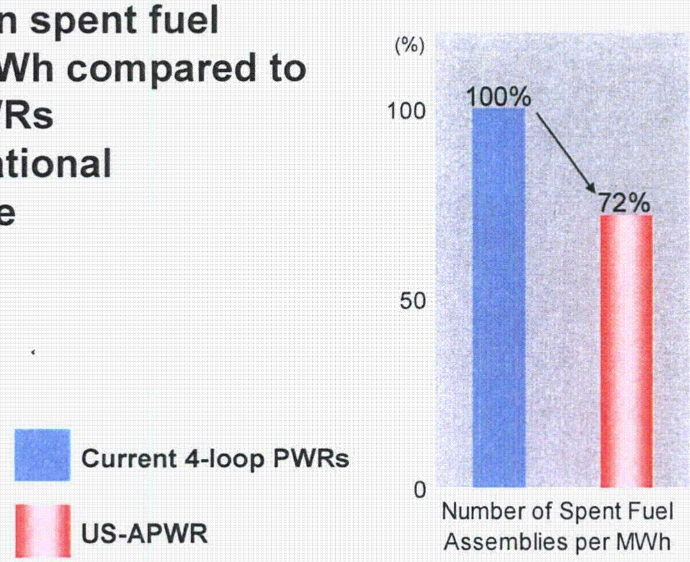
- A large core improving fuel efficiency
  Building volume per MWe that is four –fifths that of other 4-loop PWRs



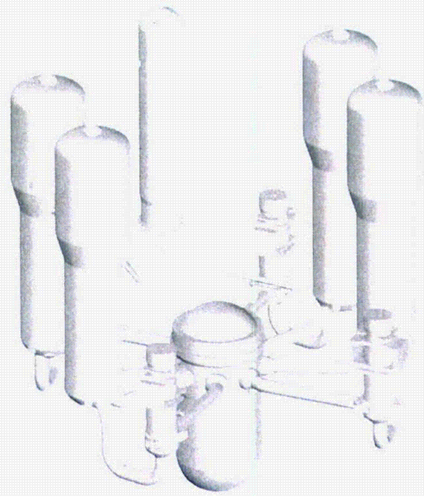
5. Features of US-APWR

More Environmentally Friendly

- A 28% reduction in spent fuel assemblies per MWh compared to current 4-loop PWRs
- Reduction occupational radiation exposure



6. Key Design Features



Comparison of Fuel, Core & Internals

		U.S. Current 4 Loop	APWR	US-APWR
Core Thermal Output		3,411MWt	4,451 MWt	4,451 MWt
Core and Fuel	NO. of Fuel Assem.	193	257	257
	Fuel Latice	17 x 17	17 x 17	17 x 17
	Active Fuel Length	12ft	12ft	14 ft
Reactor internals		Baffle/former structure	Neutron Reflector	Neutron Reflector
In-core Instrumentation		Bottom mounted	Bottom mounted	Top mounted

- APWR
  - ✓ Large capacity core by increasing number of fuel assemblies
  - ✓ Installation of neutron reflector to enhance reliability and fuel economy
- US-APWR
  - ✓ Low power density core using 14ft. fuel assemblies with the same reactor vessel as APWR to enhance fuel economy for 24 months operation
  - ✓ Enhanced reliability and maintainability of reactor vessel by top mounted ICIS

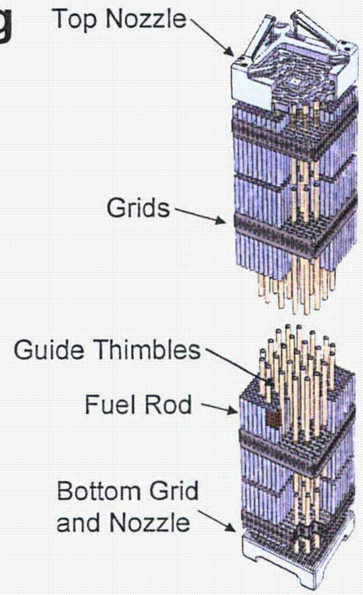
Fuel

US-APWR

> Low power density core using 14ft. FAs for 24 months operation

> Higher Density Pellet (97%T.D.)

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



Comparison of Output & Main Components

US-APWR

		U.S. Current 4 Loop	APWR	US-APWR
Electric Output		1,180 MWe	1,538 MWe	1,700 MWe Class
Core Thermal Output		3,411MWt	4,451 MWt	4,451 MWt
Steam Generator	Model	54F	70F-1	91TT-1
	Tube size	7/8"	3/4"	3/4"
Reactor Coolant Pump	Model	93A-1	100A	100A
Turbine	LP last-stage blade	44 inch	54 inch	70 inch class

> APWR

✓1538MWe output is achieved by large capacity core and large capacity main components such as SG, RCP, turbine, etc.

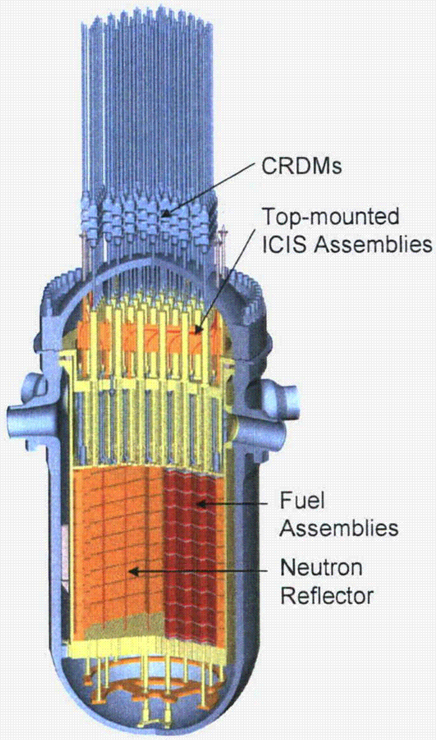
> US-APWR

✓1700MWe class output is achieved from a 10% higher efficiency than APWR.

- Same core thermal output with APWR
- High-performance, large capacity steam generator
- High-performance turbine

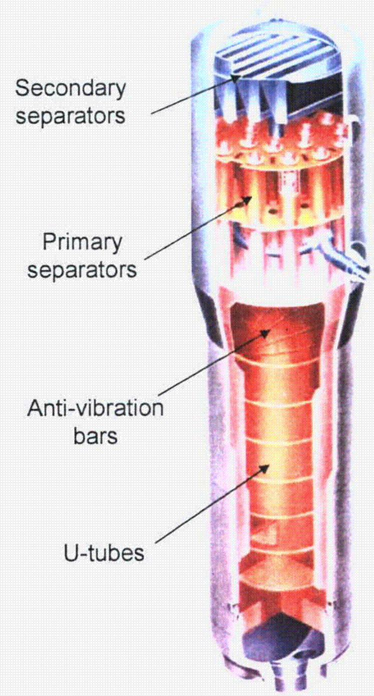
Reactor Vessel

- Core thermal output: 4,451MWt
- 14 feet fuel length
- RV size is same as APWR
- Eliminate the bottom mounted ICIS



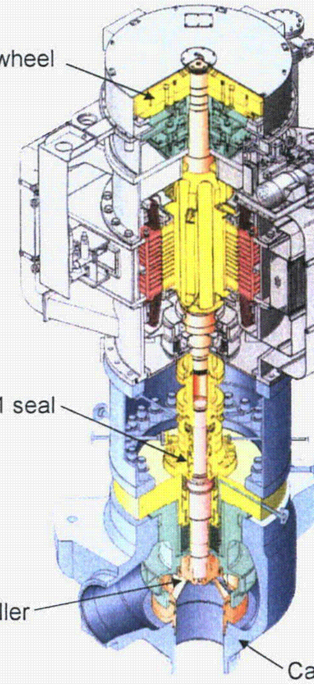
Steam Generator

- High Performance Separator
- Increased Capacity with Compact Sizing
- High Corrosion Resistance Tubes



Reactor Coolant Pump

- Improved Hydraulic performance
- Advanced Seal
  -Improved Seal Characteristic and Durability



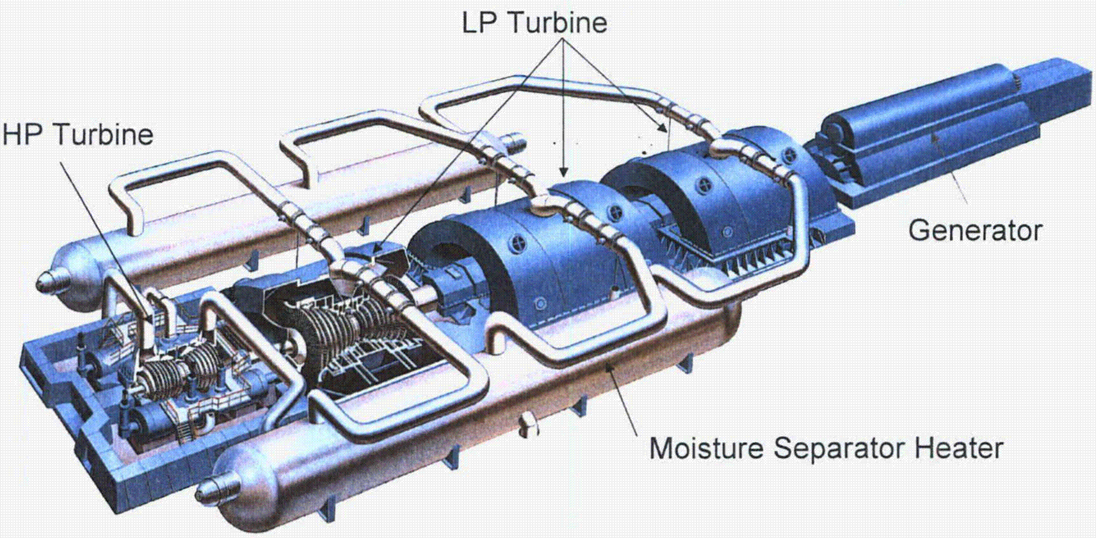
Turbine Generator

HIGHER EFFICIENCY

- TWO STAGE REHEAT MSR
- HIGH EFFICIENCY REACTION BLADES

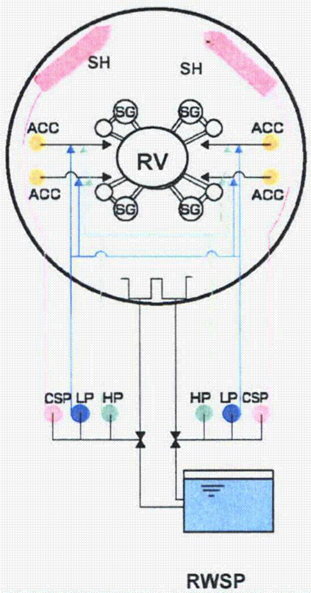
HIGHER RELIABILITY

- INTEGRAL SHROUD LP END BLADE - ISB
- MONOBLOCK LP ROTOR



ECCS and CSS/RHRS

Current 4 Loop PWR (2 train)

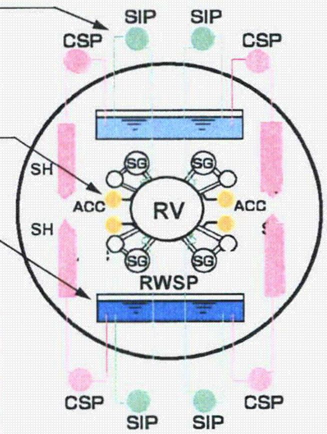


US-APWR (4 train)

◆4 train (DVI)
→ Higher Reliability
Simplified Pipe Routing

◆Advanced Accumulator
→ Elimination of LP

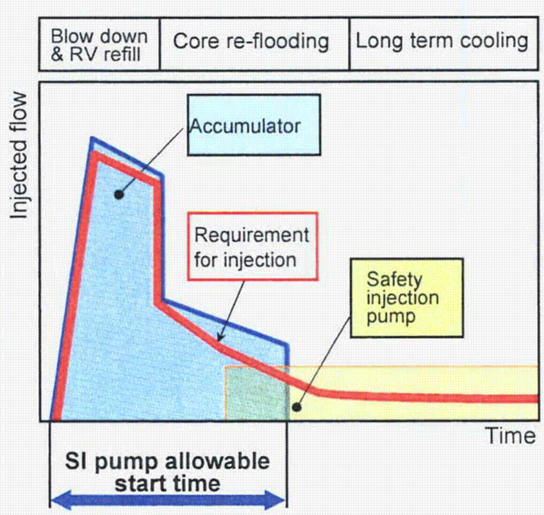
◆In-containment RWSP
→ Higher Reliability

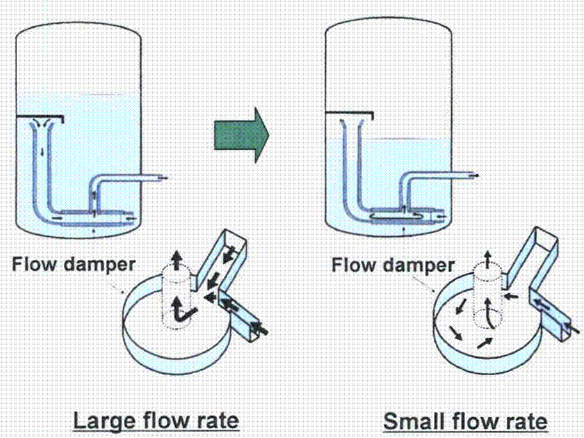


ACC	: Accumulator
HP	: High Head SIP
LP	: Low Head SIP
SIP	: Safety Injection Pump
CSP	: Containment Spray Pump
SH	: Spray Header
RV	: Reactor Vessel
RWSP	: Refueling Water Storage Pit

Advanced Accumulator

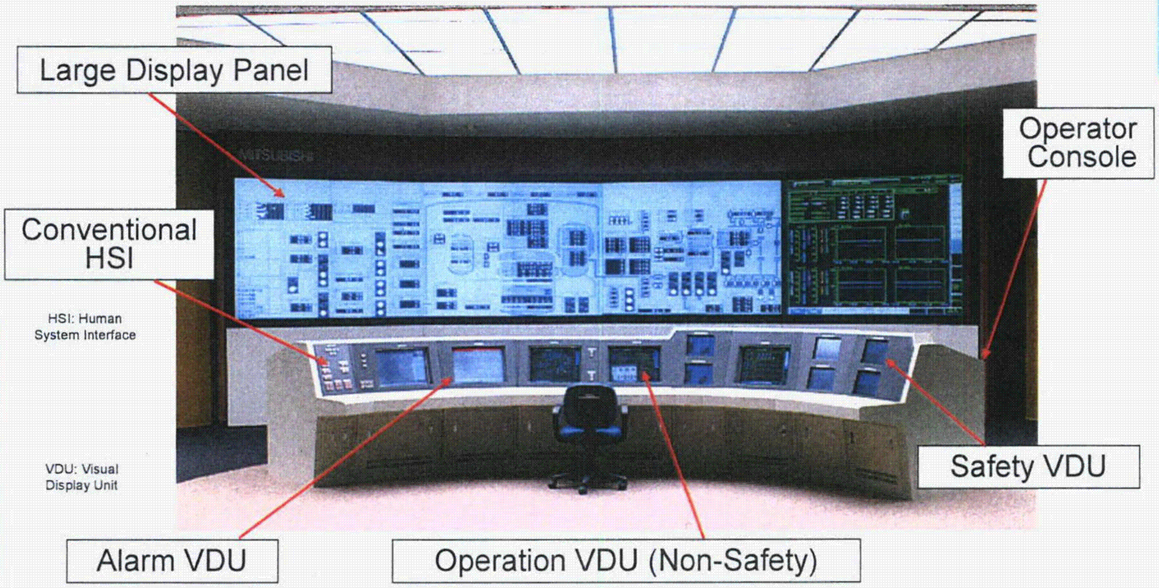
- Automatic switching of injection flow rate by flow damper
- Integrated function of low head injection system
- Long accumulator injection time allows more time for safety injection pump to start





Improved Operability

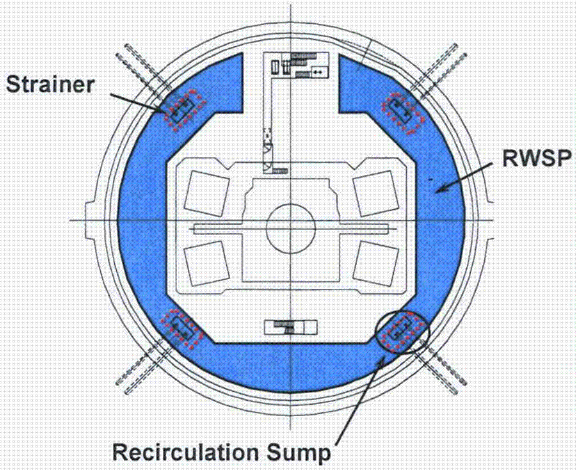
- **Fully digital control and protection systems**

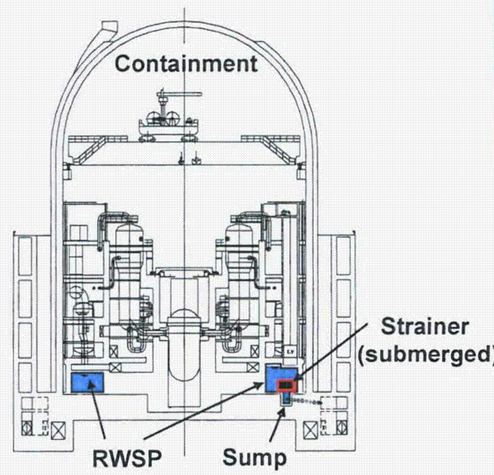


Refueling Water Storage Pit

- RWSP is installed inside containment vessel
- Easy to meet the GSI-191 because the surface area of strainer can be increased easily

GSI-191: Generic Safety Issue of PWR Sump Performance

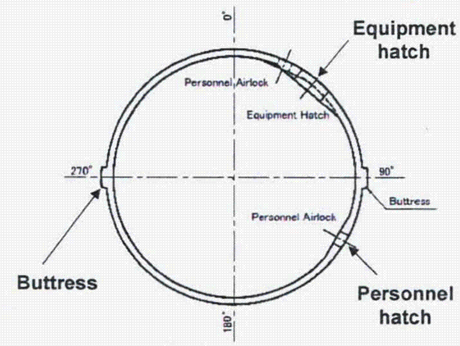


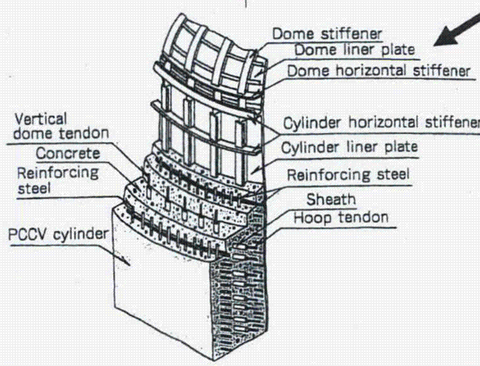


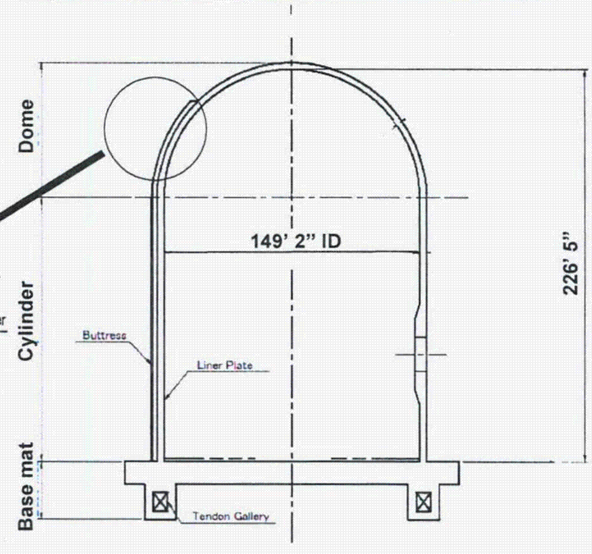
PCCV

➤Robust and reliable pressure vessel with steel liner

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

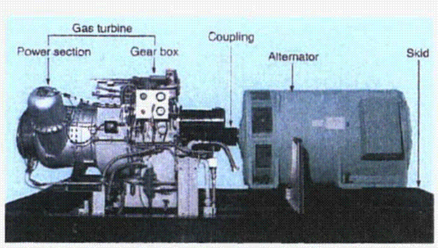






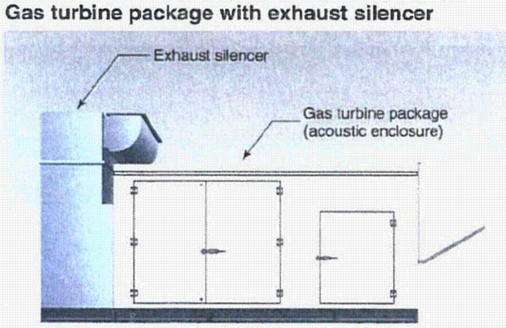
Gas Turbine Generator for EPS

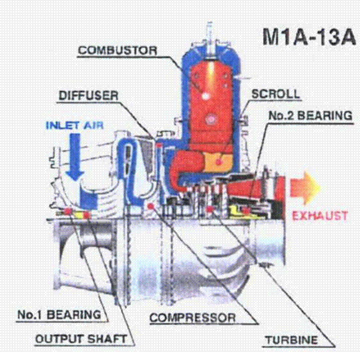
➤Gas-Turbine Generators are applied to the Emergency Power Source



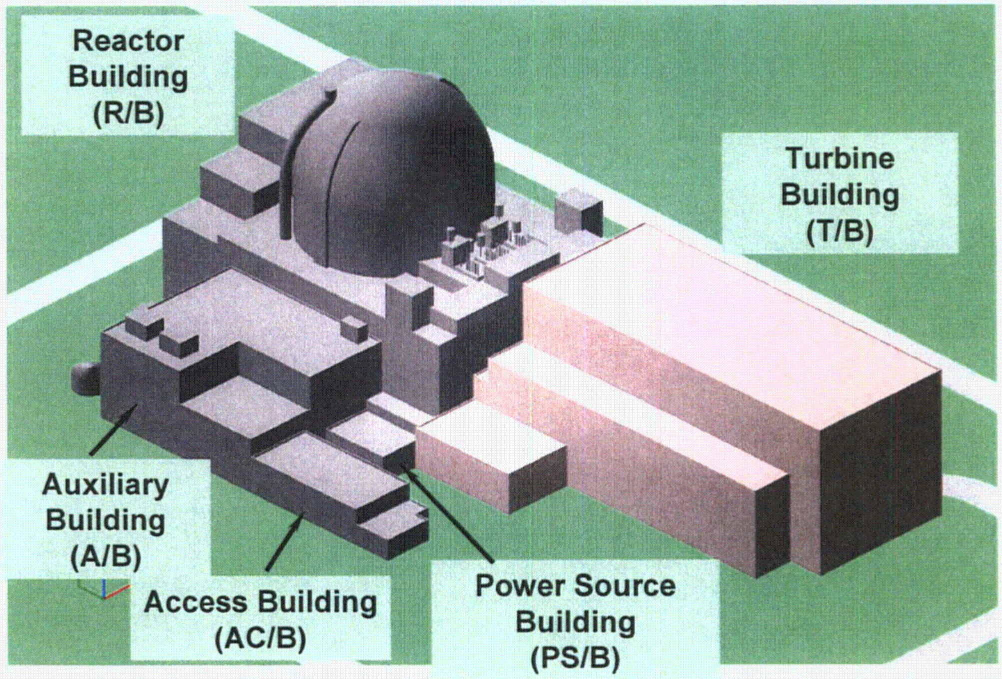
- **The Gas Turbine is a very simple rotating engine with few components**
- **A water cooling system is not required**

Gas turbine package with exhaust silencer



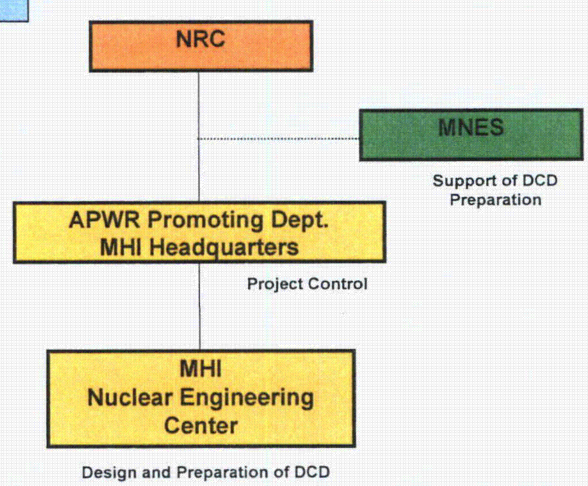


Arrangement of Main Power Block



7. Deployment Organization

At the DC Stage



MNES: Mitsubishi Nuclear Energy Systems, Inc.
MHI: Mitsubishi Heavy Industries, Ltd.

8. Conclusions

- US-APWR design is based on Japanese APWR and is modified to meet the U.S. utility's requirements
- US-APWR is 1700MWe class large NPP and high performance efficiency
- US-APWR is a well balanced Nuclear Power Plant where new and evolutional technologies are adapted

US-APWR

Design Certification Application Orientation

Detail of FSAR

Tier 1

January 15,16, 2008

Mitsubishi Heavy Industries, Ltd.

Presenter

US-APWR

Atsushi Kumaki

Engineering Manager
APWR Promoting Department
Nuclear Energy Systems Headquarters
Mitsubishi Heavy Industries, LTD.

Contents

1. Overview
   - Tier 1 Contents
   - Organization
   - General Approach
2. Design Description
3. ITAAC
4. Summary

1. Overview

- **Tier 1 contents**
  - ✓ Based on SRP 14.3.
  - ✓ Contains information to be certified by the NRC.
    - Tier 2 contains more detailed information that is approved but not certified by the NRC.
  - ✓ The type of information and the level of detail are based on a graded approach commensurate with the safety significance of the SSCs.
    - The Tier 1 material is derived from the Tier 2 material.
  - ✓ The top-level information includes the principal performance characteristics and safety functions of the SSCs, which are to be verified appropriately by ITAAC.
    - The successful completion of all ITAAC is a prerequisite for fuel load and a condition of the license.
    - The Tier 1 provides ITAAC for structures and systems within the scope of the DCD.

1. Overview

US-APWR

➢ Organization

✓ Chapter 1: Introduction

- Discussion of Tier 1 contents, definitions, format etc.

✓ Chapter 2: Design Description & ITAAC

- Top-level design description and commitment
- Tables and figures
- ITAAC

✓ Chapter 3: Interface Requirements

- Safety significant Interfaces between the standard design and COL applicantion.

1. Overview

US-APWR

Section	Subject	Section	Subject
1.0	Introduction	2.8	Radiation Protection
2.0	Design Descriptions and ITAAC	2.9	Human Factors Engineering
2.1	Site Parameters	2.10	Emergency Planning
2.2	Structural and Systems Engineering	2.11	Containment Systems
2.3	Piping Systems and Components	2.12	Physical Security Hardware
2.4	Reactor Systems	2.13	Design Reliability Assurance Program
2.5	Instrumentation and Controls	2.14	Initial Test Program
2.6	Electrical Systems	3.0	Interface Requirements
2.7	Plant Systems		

1. Overview

- **General approach**
  - ✓ Derived from Tier 2
  - ✓ Address the most safety-significant aspects, top-level design features, performance characteristics most significant to safety
  - ✓ Level of detail governed by graded approach (more detail in safety-related systems)
  - ✓ Figures supplement narrative in most cases
  - ✓ Regulations, codes, and standards not cited in general
  - ✓ Tables and figures identify SSCs to be verified by ITAAC

2. Design Description

- **System design descriptions are to address, as applicable**
  - ✓ Significant performance characteristics and safety functions
  - ✓ Whether the system is safety related or not
  - ✓ System location
  - ✓ Key design features
  - ✓ Seismic and ASME Code classifications
  - ✓ Important alarms, displays and controls
  - ✓ Logic circuits (for direct safety functions)
  - ✓ Interlocks (for direct safety functions)
  - ✓ Interface requirements

3. ITAAC

US-APWR

➢ Format

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	1.	1.
2.	2.	2.

- ✓ 1st column – proposed design requirement and/or commitment to be verified
- ✓ 2nd column – proposed method by which the licensee will verify the requirement or commitment
- ✓ 3rd column – proposed acceptance criteria to be met

3. ITAAC

US-APWR

➢ System based ITAAC

- ✓ Reactor Systems, Plant Systems, Containment Systems
- ✓ Instrumentation and Controls
- ✓ Electrical Systems

➢ Non-system based ITAAC

- ✓ Structural and Systems Engineering
- ✓ Piping Systems and Components
- ✓ Radiation Protection, Human Factor Engineering
- ✓ Emergency Planning
- ✓ Physical Security Hardware
- ✓ Design Reliability Assurance Program

➢ No ITAAC

- ✓ Site Parameters
- ✓ Initial Test Program

3. ITAAC

US-APWR

- **Contents Example for Fluid Systems (system based ITAAC)**
  - ✓ System functional arrangement
  - ✓ Pressure boundary integrity
  - ✓ Pumps having sufficient net positive suction head
  - ✓ System divisions being powered from respective Class 1E division
  - ✓ Mechanical division separation
  - ✓ Control room alarms, displays, and controls

3. ITAAC

US-APWR

- **Contents Example for Structural and Systems Engineering (non-system based ITAAC)**
  - ✓ Pressure boundary integrity
  - ✓ Pipe rupture
  - ✓ Normal loads
  - ✓ Seismic loads
  - ✓ Floods, wind, and tornados
  - ✓ Rain and snow
  - ✓ Codes and standards
  - ✓ Containment integrity
  - ✓ As-built reconciliation

4. Summary

US-APWR

- Tier 1 contents are based on SRP 14.3.
- Tier 1 material is derived from the Tier 2, based on a graded approach.
- The top-level information are to be verified by ITAAC.

US-APWR
Application of the Design Certification Overview

January 15, 16, 2008

Mitsubishi Heavy Industries, Ltd.

Contents

US-APWR

1. Contents of DC Application
2. Overview of DCD
3. Overview of ER

1. Contents of DC Application

- **Applicant:** Mitsubishi Heavy Industries, Ltd.
- **Submittal date:** US-APWR Design Certification (DC) Application was submitted to the NRC on December 31, 2007
- **Contents: US-APWR DC Application consists of the following two documents,**

DCD(FSAR)
Tier1 and Tier2

Applicant's Environmental Report

2. Overview of DCD

- **Design control Document (DCD) constitutes the final safety analysis report (FSAR) for the US-APWR standard plant design required by 10 CFR 52.47(a)**
- **DCD consists of the followings;**
  - ✓ Tier 1 contains standard design information and ITAAC information to be certified by the NRC.
  - ✓ Tier 2 consists of Chapter 1 - 19 contains more detailed information to be approved but not certified by the NRC.
  - ✓ COL information is addressed in each Chapter, also compiled in Chapter 1
- **DCD meets the NRC requirements such as;**
  - ✓ NUREG-0800 (Standard Review Plan)
  - ✓ Regulatory Guide 1.206

2. Overview of DCD (cont'd)

- The US-APWR DCD was submitted electronically on compact discs "CDs" in two versions
- The first version includes certain information, designated pursuant to commission guidance as sensitive unclassified non-safeguards information, reffed to as security-related information ("SRI"), that is to be withheld from public disclosure under 10 CFR 2.390
- The second version omits the SRI and is suitable for public disclosure

3. Ovrview of ER

- Environmental Report was submitted in accordance with 10 CFR 52.47(b)(2)
- The title of Environmental Report is "Applicant's Environmental Report-Standard Design Certification"
- This document provides an evaluation of Severe Accident Mitigation Design Alternatives (SAMDA)

US-APWR

Design Certification Application Orientation

Detail of FSAR

Tier2: Chapter 1

January 15,16, 2008

Mitsubishi Heavy Industries, Ltd.

Presenter

US-APWR

Atsushi Kumaki

Engineering Manager
APWR Promoting Department
Nuclear Energy Systems Headquarters
Mitsubishi Heavy Industries, LTD.

Contents

1. Overview of Chapter

- Title of Chapter

Chapter 1: INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

- Scope of Chapter

Provides a basic understanding of the overall facility without needing to refer to subsequent chapters

2. Chapter 1 Contents

- 1.1 Introduction

  License Requested: Approval and certification for US-APWR standard design

  Containment Type , Reactor Type and Power Output

  Format and Content (for all chapters)

  Acronyms and abbreviations (for all chapters)

2. Chapter 1 Contents (cont'd)

- 1.2 General Plant Description

  - The principal design criteria, safety considerations and operating characteristics

    (engineered safety features (ESF) , emergency systems, I&C, electrical systems, power conversion system, etc.)

  - The general arrangement of major structures and equipments

2. Chapter 1 Contents (cont'd)

- 1.3 Comparison with Other Facilities

  Comparison with the J-APWR and the Standardized Nuclear Unit Power Plant System (SNUPPS)

- 1.4 Identification of Agents and Contractors

  Applicant ; Mitsubishi Heavy Industries, Ltd.
  Participant ; Obayashi Corporation,
  Engineering Development Co. ,Ltd. ,
  Washington Division of URS Corporation

2. Chapter 1 Contents (cont'd)

- 1.5 Requirement for Additional Technical Information

  Verification and confirmation tests of certain unique features of US-APWR

  Advanced Accumulator, Reactor Internals, Digital I&C system, Gas Turbine Generator

- 1.6 Material Referenced

  A list of topical reports incorporated by reference as part of the DC application

2. Chapter 1 Contents (cont'd)

- **1.7 Drawing and Other Detailed Information**
  - Figure lists of Tier-2 Chapters :

    I&C functional diagrams, electrical one-line diagrams and system drawings
  - Legend descriptions

- **1.8 Interfaces for Standard Design**

  Table of COL Items in US-APWR DCD

2. Chapter 1 Contents (cont'd)

Example: Table of COL Items

Table 1.8-2 Compilation of All Combined License Applicant Items for Chapters 1-19 (sheet 1 of 44)

COL ITEM NO.	COL ITEM
COL 1.1(1)	*The COL Applicant is to provide scheduled completion date and estimated commercial operation date of nuclear power plants referencing the US-APWR design certification.*
COL 1.2(1)	*The COL Applicant is to develop a complete and detailed site plan in the site specific licensing process.*
COL 1.8(1)	*The COL Applicant is to demonstrate that the interface requirements established for the design have been met.*
COL 2.1(1)	*The COL Applicant is to describe the site geography and demography including the specified site parameters.*
COL 2.2(1)	*The COL Applicant is to describe nearby industrial, transportation, and military facilities in the vicinity of the site of the US-APWR standard plant design. The COL Applicant is to establish the presence of potential hazards and effects of potential accidents in the vicinity of the site and determine whether these accidents are to be considered as DBEs.*

2. Chapter 1 Contents (cont'd)

US-APWR

➢ 1.9 Conformance with Regulatory Criteria

Regulatory Guides, Standard Review Plan, Generic Issues, SECY, etc.

Conformance Evaluation Tables:

Number, Title, Status and Corresponding Chapter/section/subsection

2. Chapter 1 Contents (cont'd)

US-APWR

Example: Conformance with Regulatory Guides

Table 1.9.1-1 US-APWR Conformance with Division 1 Regulatory Guides (sheet 5 of 15)

Reg Guide Number	Title	Status	Corresponding Chapter/Section /Subsection
1.68.3	Preoperational Testing of Instrument and Control Air Systems (Rev. 0, April 1982)	Conformance with exceptions. C.7, C.11: This criterion applies to instrument and control air system important safety. US-APWR does not have the instrument and control air system. C.8.b: US-APWR does not perform the gradual reduction pressure test because suddenly air pressure shutoff test can be verified that the affected components respond properly.	9.3.1.4, 14.2.7
1.69	Concrete Radiation Shields for Nuclear Power Plants (Rev. 0, December 1973)	Conformance with exceptions. Criterion 5 is not applicable to US-APWR design certification. There is no plan which uses metal for the aggregate of concrete shielding.	12.3.2
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) (Rev. 3, November 1978)	Not applicable. Format and content for new reactors established in RG 1.206.	N/A
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 1, March 2007)	Conformance with no exceptions identified.	5.2.3.3.2, 5.2.3.4.3
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin (Rev. 2, November 1978)	Not applicable. US-APWR design does not use Spray Pond. The spray water of US-APWR is supplied from RWSP in containment.	N/A
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	Conformance with no exceptions identified.	8.1.5.3
1.75	Physical Independence of Electric Systems (Rev. 3, February 2005)	Conformance with no exceptions identified.	7.1.3, 8.1.5.3
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, March 2007)	Conformance with no exceptions identified. Note: COL Applicant will verify site-specific data is bounded by data used in DCD analyses.	2.3, table 2.0-1, 3.3.2, 3.5.1.4

3. Summary

US-APWR

- Chapter 1 gives an introduction to the DCD and a general description of the US-APWR.
- This chapter helps the review of the subsequent detailed chapters by facilitating a basic understanding of the overall facility.

US-APWR

Design Certification Application Orientation

OVERVIEW OF ENVIRONMENTAL REPORT

January 15,16, 2008

Mitsubishi Heavy Industries, Ltd.

Presenter

Katsunori Kawai

Engineering Manager

APWR Promoting Department

Nuclear Energy Systems Headquarters

Mitsubishi Heavy Industries, LTD.

Contents

1. Overview of ER
   - Title of ER
   - Scope of ER
2. Contents of ER

1. Overview of ER

➢Title of ER

US-APWR

Applicant's Environmental Report- Standard Design Certification

➢Scope of ER

This report includes the following contents:

- **An Evaluation of severe accident mitigation design alternatives (SAMDA)**

2. Contents of ER

- Contents of ER
  1. INTRODUCTION
  2. SUMMARY
  3. SELECTION AND DESCRIPTION OF SAMDAS
     - 3.1 SCREENING METHOD
     - 3.2 SCREENING CRITERIA
  4. METHODOLOGY
     - 4.1 Risk Reduction Potential of Design Improvements
     - 4.2 Algorithms for Calculating Value of Risk Avoided
     - 4.3 Estimate of Risk for Design
  5. SUMMARY OF RISK SIGNIFICANT ENHANCEMENTS
  6. REFERENCE SITE CHARACTERISTICS
  7. VALUE OF ELIMINATING RISK
  8. EVALUATION OF POTENTIAL IMPROVEMENTS
  9. RESULTS
  10. REFERENCES

  APPENDIX-A: OFFSITE POPULATION DOSE AND PROPERTY DAMAGE RISK QUANTIFICATION METHODOLOGY

2. Contents of ER

- 2.1 Introduction
  - Regulatory Requirements:
    - NEPA, Section 102.(C)(iii)
    - 10CFR52.47(b)(2)
    - 10CFR51.55
      - ✓ Requires each applicant for a standard design certification to submit an ER evaluating SAMDAs
    - 10CFR50.34(f)(1)(i)
      - ✓ Requires a plant/site specific PRA
      - ✓ Seek improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant
  - The information provided in this report complies with applicable parts of NUREG-1555

2. Contents of ER

- 2.2 Summary
  - Cost-benefit methodology follows:
    - NUREG/BR-0184
    - NUREG/BR-0058
    - SECY-99-169
  - Identification and screening of SAMDAs:
    - NEI 05-01
  - The total maximum averted cost: $289k
    - Replacement power cost (65%)
    - Cleanup and Decontamination costs (24%)
    - Offsite Exposure cost (10%)
    - Onsite Exposure cost (1%)
    - Offsite Property Damage (<1%)

2. Contents of ER

- 2.3 Selection and Description of SAMDAs
  - Sources of SAMDAs:156 candidates
    - NEI 05-01 for PWRs
    - US-APWR specific candidates
  - Two phases of screenings
    - Phase I Qualitative: 156 -> 10 candidates
      - ✓ NEI 05-01 screening criteria
        - » Not applicable (20)
        - » Already implemented (22)
        - » Not a design alternatives (29)
        - » Combines (3)
        - » Excessive implementation cost (3)
        - » Very low benefit (69)
    - Phase II Quantitative: 10 -> 0 candidates
      - ✓ Comparison to the Maximum Averted Costs

2. Contents of ER

➢ 2.4 Methodology (1/2)

- Risk Reduction Potential of Design Improvements
  - Five principal component costs:
    - ✓ Offsite Exposure Cost
    - ✓ Onsite Exposure Cost
    - ✓ Offsite Property Damage
    - ✓ Cleanup and Decontamination Cost
    - ✓ Replacement Power Cost
  - 50-mile radius from the plant site
  - Monetary conversion factor: 2000 ($/person-rem)
  - Real discount rate: 0.07 (/year)

2. Contents of ER

➢ 2.4 Methodology (2/2)

- Estimate of Risk for Design
  - Risk assessment covers four categories of events:
    - ✓ Internal events
    - ✓ Internal fire
    - ✓ Internal flood
    - ✓ Low-power and shutdown (LPSD)
  - Level-3 PRA
    - ✓ MACCS2 code
    - ✓ Done for Internal events at power
    - ✓ CDF scaling factor is applied to the other events

2. Contents of ER

US-APWR

- 2.5 Summary of Risk Significant Enhancement
  - Design enhancements already incorporated into the US-APWR, due to probabilistic risk assessment insights and results
    - Redundancy
      - ✓ Four train ECCS, EFWS, etc.
    - Diversity
      - ✓ AAC power supply, etc.
    - Simplicity
      - ✓ In-containment RWSP, etc.
    - Prevention
      - ✓ Higher rated piping of RHRS, etc.
    - Inherent margin of safety
      - ✓ Robust structure against steam explosion, etc.

2. Contents of ER

US-APWR

- 2.6 Reference Site Characteristics
  - Meteorological data of the Surry site has been used as "typical"
  - The 50-mile population data of the Surry site in the MACCS2 code sample input file has been adjusted to be representative of about 80% of the U.S. nuclear plant sites in NUREG/CR-2239

2. Contents of ER

US-APWR

➤ 2.7 Value of Eliminating Risk

- Cost estimate based on NEI 05-01

	Design Alternative	Cost
1	Provide additional dc battery capacity	$2,000k
2	Provide an additional gas turbine generator	$10,000k
3	Install an additional, buried off-site power source	$10,000k
4	Provide an additional high pressure injection pump with independent diesel	$1,000k
5	Add a service water pump	$5,900k
6	Install an independent reactor coolant pump seal injection system, with dedicated diesel	$3,800k
7	Install an additional component cooling water pump	$1,500k
8	Add a motor-driven feed-water pump	$2,000k
9	Install a filtered containment vent to remove decay heat	$3,000k
10	Install a redundant containment spray system	$870k

2. Contents of ER

US-APWR

➤ 2.8 Evaluation of Potential Improvements

- Conservatism was retained in the cost estimates
  - Older studies were used without attempting to adjust to present-day dollars
  - Costs were maintained without adjustment by power scaling factor
    - ✓ In only one case, cost was scaled from a lower-power to the current larger power (SAMDA #10)
- Potential improvements are substantially more costly than the calculated maximum averted cost.

2. Contents of ER

- 2.9 Conclusion
  - There are no additional design alternatives that are shown to be cost-beneficial in severe accident mitigation design.

US-APWR

Design Certification Application Orientation

Detail of FSAR

Tier2: Chapter 2

January 15,16, 2008

Mitsubishi Heavy Industries, Ltd.

Presenter

Hiroyuki Fuyama, Ph.D.

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Contents

1. Overview of Chapter

➢ Title of Chapter

Chapter 2: SITE CHARACTERISTICS

➢ Scope of Chapter

This Chapter includes geological, seismological, hydrological and meteorological characteristics.

COLA confirms site characteristics are bounded, or provides site-specific qualification.

2. Site Characteristics

➢ Key Site Parameters

Bounds estimated 75% to 80% of US Landmass

Site is defined as contiguous real estate with legal right to control access by individuals, and to restrict land use.

Table 2.0-1 is a summary identifying specific site parameters for the US-APWR.

2. Site Characteristics (cont'd)

➢ Chapter 2 consists of 5 sections.

Section	Title	Description
2.1	Geography and Demography	Site Specific Characteristics
2.2	Nearby Industrial, Transportation, and Military Facilities	Site Specific Characteristics
2.3	Meteorology	1. 50 psf Maximum snow and precipitation weight 2. 230 mph Maximum tornado wind speed 3. 1.2 psi Maximum tornado pressure drop 4. Tornado-generated missile characteristics in accordance with RG 1.76 Rev. 1 5. Bounding limits of atmospheric dispersion factors and deposition factors presented in Table 2.0-1 6. 155 mph extreme wind speed is for 3-second gusts at 33 ft above ground level

2. Site Characteristics (cont'd)

➢ Outline of Subsections (cont'd)

Section	Title	Description
2.4	Hydrologic Engineering	1. Groundwater elevation minimum of 1 ft. below plant grade 2. Maximum level for flood or tsunami of 1 ft. below plant grade 3. Maximum local intense precipitation of 19.4 in./hr
2.5	Geology, Seismology, and Geotechnical Engineering	1. Peak ground acceleration = 0.3g 2. Uses modified high frequency approach per RG 1.60 (further detail in Subsection 3.7.1.1) 3. SSE is based on Certified Seismic Design Response Spectra

3. Summary

➢ Chapter 2 defines site parameters of the US-APWR standard plant as well as identifies important site parameters.

➢ COLA is to confirm that DCD site parameters envelope site-specific parameters.

➢ Table 2.0-1 is a summary identifying specific site parameters for the US-APWR.

US-APWR

Design Certification Application Orientation

Detail of FSAR

Tier2: Chapter 3

January 15,16, 2008

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Presenter

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Mitsubishi Heavy Industries, LTD.

Contents

1. Overview of Chapter

- Title of Chapter

  Chapter 3: DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS, AND EQUIPMENT

- Scope of Chapter

  This chapter includes the design criteria and design description for buildings, structures, components, and piping.

Table of Contents for Chapter 3

Appendices

2. Design Features

- **Application of LBB Criteria (Section 3.6.3)**
  LBB Criteria apply for Reactor Coolant System Pipe and Main Steam Pipe
- **Seismic Design (Section 3.7)**
  Peak Grand Acceleration : 0.3G
  Response Spectrum Curve: Modified R.G 1.60
- **Component Design (Section 3.9)**
  Reactor Internals with neutron reflector

3. DCD Chapter 3 contents

3.1 Conformance with NRC GDC

GDC: General Design Criteria

- Each Criterion is first quoted and then discussed in sufficient detail to demonstrate compliance of the US-APWR.
- The Design Criteria establish the necessary design, fabrication, construction, testing, and performance requirements for safety-related SSCs.
- Section 3.1 briefly discusses extent which Design Criteria for the safety-related SSCs comply with 10 CFR, Part 50, appendix A, General Design Criteria for Nuclear Power Plants.
- Total 64 criteria are quoted and conformance discussions are made.

3.1 Conformance with NRC GDC (cont'd)

Example GDC and Associated Conformance Discussion:

➢Criterion 16 – Containment Design

"The reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

➢Discussion (Partial Extraction)

The prestressed concrete containment vessel (PCCV) is composed of a pre-stressed post-tensioned concrete vessel, featuring a vertical cylinder, a hemispherical dome, and a flat reinforced concrete foundation. The PCCV is surrounded by the concrete reactor building (R/B). The steel-lined PCCV completely encloses the reactor, RCS, and other related systems. The lines that penetrate the containment vessel are provided with containment isolation valves according the provisions of GDCs 54, 55, 56, and 57. The steel-lined PCCV provides an essentially leak-tight barrier and provides environmental radiation protection under all postulated accident conditions.

3.2 Classification

Subsection	Title	Description
3.2.1	Seismic Classification	Safety-related SSCs are designed to withstand the effects of earthquake without loss of capability to perform safety function. Three Seismic Categories: Seismic Category I Seismic Category II Non Seismic
3.2.2	System Quality Group Classification	Subsection describes the following contents: ✓Acceptable deterministic approach to classify safety-related systems ✓Applicable construction codes & standards ✓Safety-related SSCs designed, fabricated, erected and tested to quality standards commensurate with their safety function.
Key Tables	Table 3.2-2	Classification of Mechanical and Fluid Systems, Components, and Equipment
	Table 3.2-4	Seismic Classification of Buildings & Structures

Classification of Mechanical and Fluid Systems, Components, and Equipment (example)

System and Components	Equip. Class	Location	Quality Group	10 CFR 50 Appendix B (Ref 3.2-8)	Code and Standards	Seismic Category
Primary System						
1. Reactor Systems						
Fuel assemblies	1	PCCV	A	YES	5	I
Rod control cluster	1	PCCV	A	YES	5	I
Burnable poison	1	PCCV	A	YES	5	I
Neutron source assemblies	1	PCCV	A	YES	5	I
Upper core support	1	PCCV	A	YES	ASME III, CS	I
Lower core support	1	PCCV	A	YES	ASME III, CS	I
Guide tube assemblies	1	PCCV	A	YES	5	I
Control rod drive mechanism latch housing	1	PCCV	A	YES	1	I
Control rod drive mechanism rod travel housing	1	PCCV	A	YES	1	I

Seismic Classification of Buildings and Structures

US-APWR

Structure	Acronym	Seismic Category
Reactor Building	R/B	I
Prestressed Concrete Containment Vessel	PCCV	I
Containment Internal Structure	CIS	I
Power Source Building (East and West)	PS/B	I
Power Source Fuel Storage Vault	PSFSV	I
Essential Service Water Pipe Tunnel	ESWPT	I
UHS Related Structures	UHSRS	I
Auxiliary Building	A/B	II
Turbine Building	T/B	II
Access Building	AC/B	NS
Outside Building	O/B	NS
Turbine Generator Pedestal	T/G Pedestal	NS

3.3 Wind and Tornado

US-APWR

- This Section provides wind and tornado loads used in the design of Seismic Category I and Category II structures.

Subsection	Title	Description
3.3.1	Wind Loadings	Design basis wind load in accordance with ASCE/SEI 7-05 Load combinations defined in applicable codes and as modified by the relevant RGs and SRPs 155 mph wind speed for 3 second gusts at 33 ft above ground for exposure category C 1.15 Importance Factor for essential facilities

3.3 Wind and Tornado (cont'd)

Subsection	Title	Description
3.3.2	Tornado Loadings	Parameters based on RG 1.76 Rev. 1 Region 1 tornado which envelopes all other regions 230 mph tornado wind speed including rotational and translational speeds Radius of maximum rotational wind from center of tornado, R_m = 150 ft Atmospheric pressure drop = 1.2 psi Rate of pressure change = 0.5 psi/second Seismic category II structures are required to be designed for same tornado wind loads as seismic category I structures

3.4 Water Level (Flood) Design

➤ The US-APWR is designed to protect safety-related SSCs for maximum water levels caused by flooding sources that are both external and internal to the plant

The external water source events

- Probable maximum precipitation (PMP)
- Probable maximum flood (PMF) of streams and rivers
- Probable maximum tsunami hazards
- Probable maximum surge, seiche flooding, and wave action ,etc...

The internal water source events

- Earthquakes
- Pipe breaks and cracks
- Fire fighting operations
- Pump mechanical seal failures

3.5 Missile Protection

- **The US-APWR is designed to protect safety-related SSCs for missiles**

  The missiles are generated by pressurized components, rotating machinery, explosions within the plant, falling objects, and by tornados or transportation accidents eternal to the plant.

- **Missile protection for SSCs important to safety is adequate if provided by one or more of the following methods:**
  - Locating the system or component in a missile-proof structure
  - Separating redundant systems or components for the missile path or range
  - Providing local shields and barriers for systems and components
  - Designing the equipment to withstand the impact of the most damaging missile
  - Providing design features to prevent the generation of missiles
  - Orienting missile sources to prevent missiles from striking equipment important to safety

3.6 Postulated Piping Rupture

- **The US-APWR is designed to protect safety-related SSCs for the dynamic effects of postulated pipe break accidents.**

  It is specifically in the localized regions of the pipe break, including pipe whip, jet impingement, subcompartment pressurization, fluid system decompression in the ruptured pipe, the environmental effects, spray wetting, and flooding.

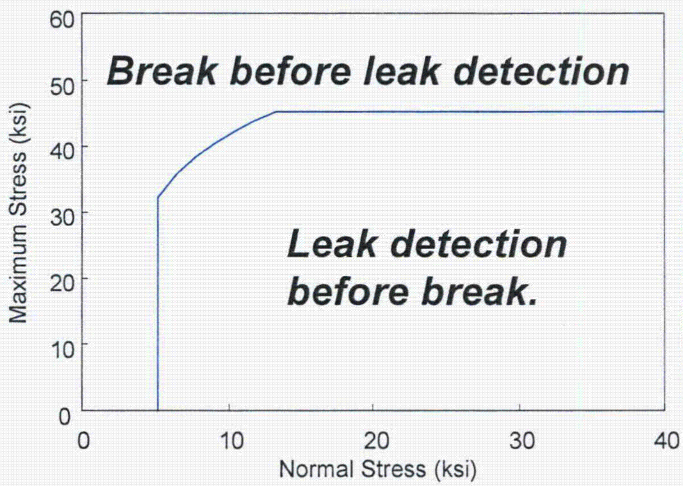
- **LBB criteria are applied to the following high energy piping systems.**
  - RCL pipes
  - RCL branch pipes with nominal diameter of 6 inches or larger, except for steam within the piping for the pressurizer safety valve and power operated relief valve
  - Main steam pipe in PCCV

Bounding Analysis Curve (BAC)

US-APWR

➤ BACs are employed to assure successful application of LBB.

For Primary Loop Hot Leg



Item	Value
System	RCS
Nominal Diameter	31"ID
Outside Diameter	37.120 inch
Normal Thickness	3.060 inch
Pipe Material	SA-182 F316 (t<=5)
Normal Operating Pressure	2235.0 psig
Normal Operating Temperature	617.0 F

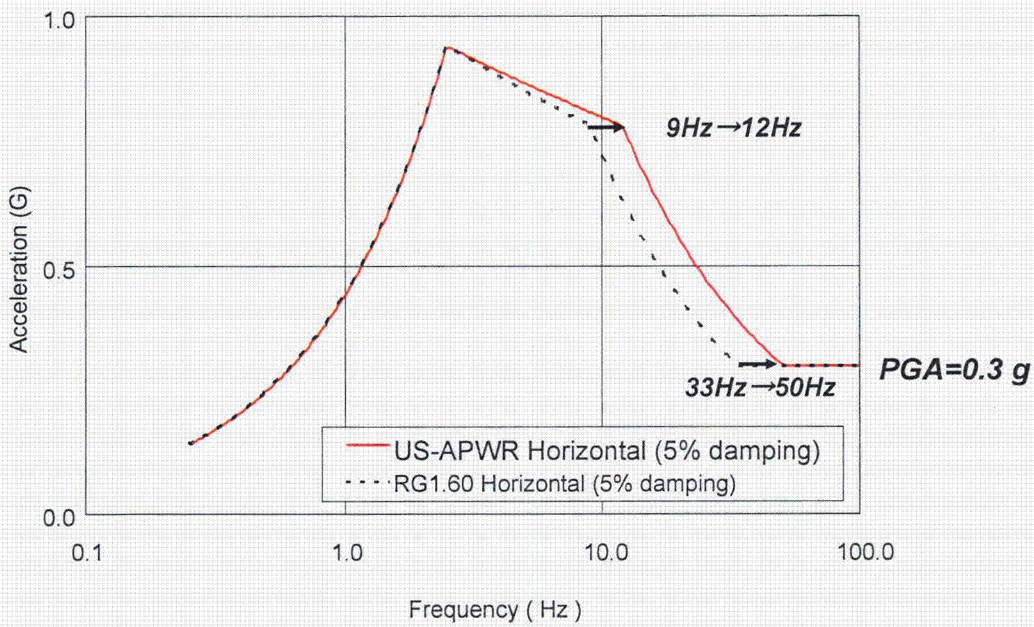
3.7 Seismic Design

US-APWR

Subsection	Title	Description
3.7.1	Seismic Design Parameters	PGA of CSDRS is 0.3g. OBE=1/3SSE ✓OBE is not considered in design. ✓In the fatigue evaluation, earthquake cycles are considered equivalent to five OBE events followed by one SSE event.
3.7.2	Seismic System Analysis	R/B, PCCV, CIR & PS/B ✓Seismic category I ✓3D Lumped Mass Stick Model A/B & T/B ✓Seismic category II ✓Designed to protect collapse during SSE.
3.7.3	Seismic Subsystem Analysis	In accordance with SRP 3.7.3 and related RGs.
3.7.4	Seismic Instrumentation	In accordance with RG 1.12 and RG 1.166.

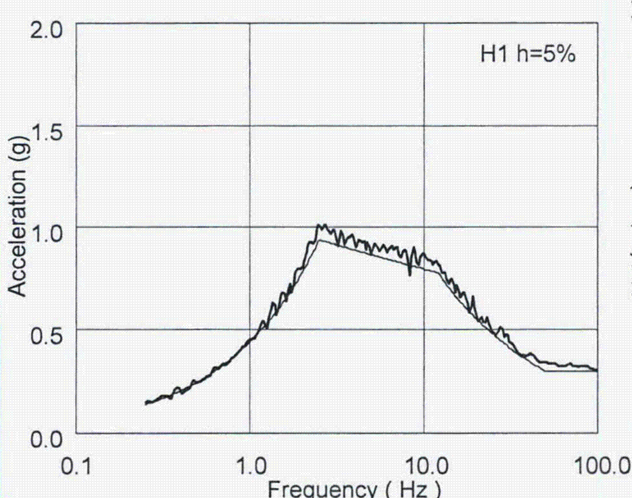
CSDRS

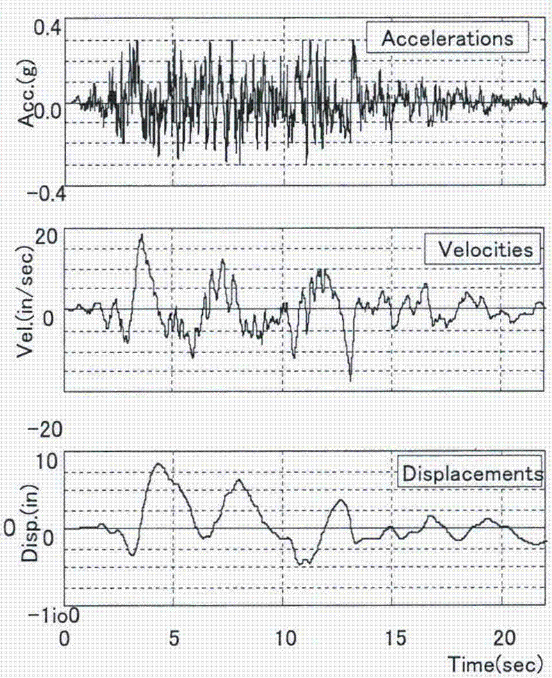
CSDRS: Certified Seismic Design Response Spectra



Design Ground Motion Time History

Option 1, Approach 2 is adopted as described Section II of SRP3.7.1

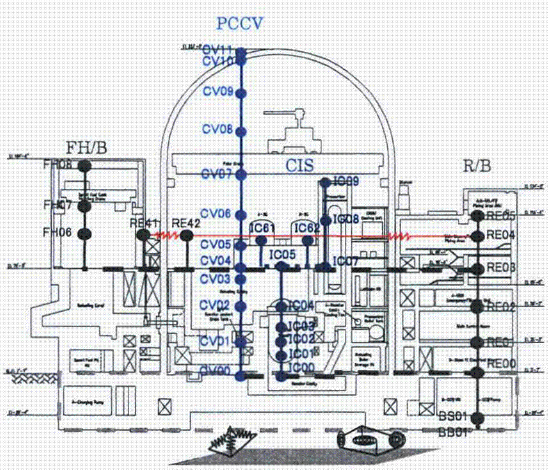


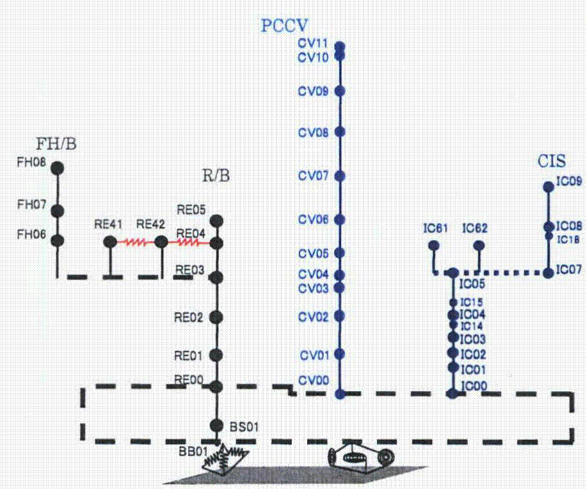


Summary of Seismic Analysis

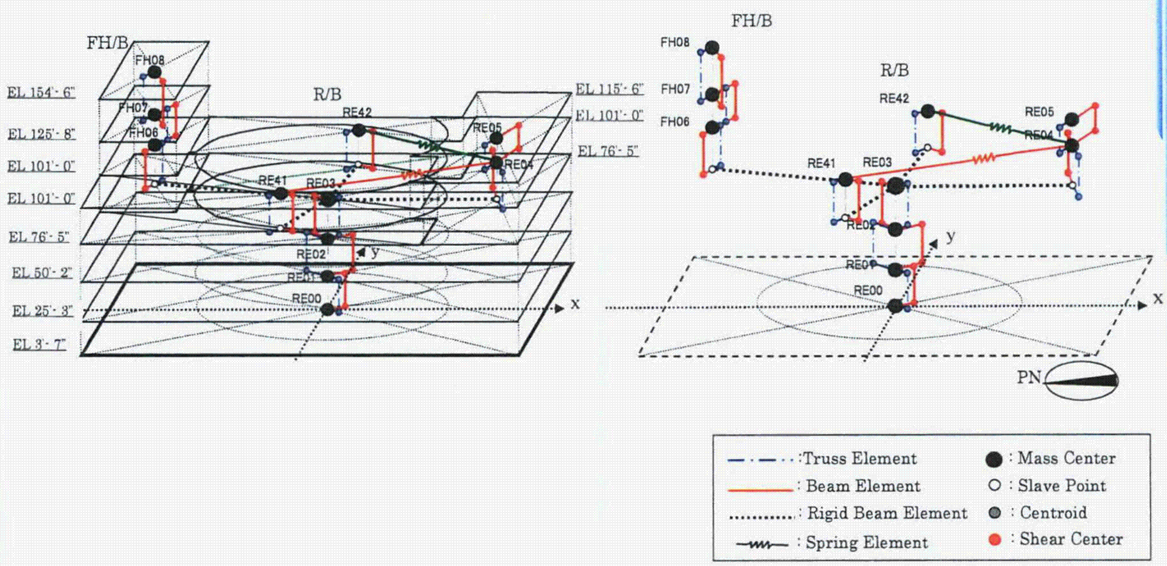
Model	Analysis Method	Program	Three Components Combination	Modal Combination
3D-R/B-PCCV-CIS Lumped Mass Stick Model	Direct Integration Time History Analysis	ANSYS	SRSS	N/A
3D-R/B-PCCV-CIS FE Model (For Validation of Lumped Mass Stick Model)	Time History Analysis in Frequency Domain	NASTRAN	N/A	N/A
3D RCL Piping Lumped Mass Stick Model	Direct Integration Time History Analysis	ANSYS	SRSS	N/A
3D PS/Bs Lumped Mass Stick Models	Direct Integration Time History Analysis	ANSYS	SRSS	N/A
3D RCL-R/B-PCCV/CIS Lumped Mass Stick Model	Direct Integration Time History Analysis	ANSYS	SRSS	N/A

Lumped Mass Stick Model





Modeling the Torsional Effect



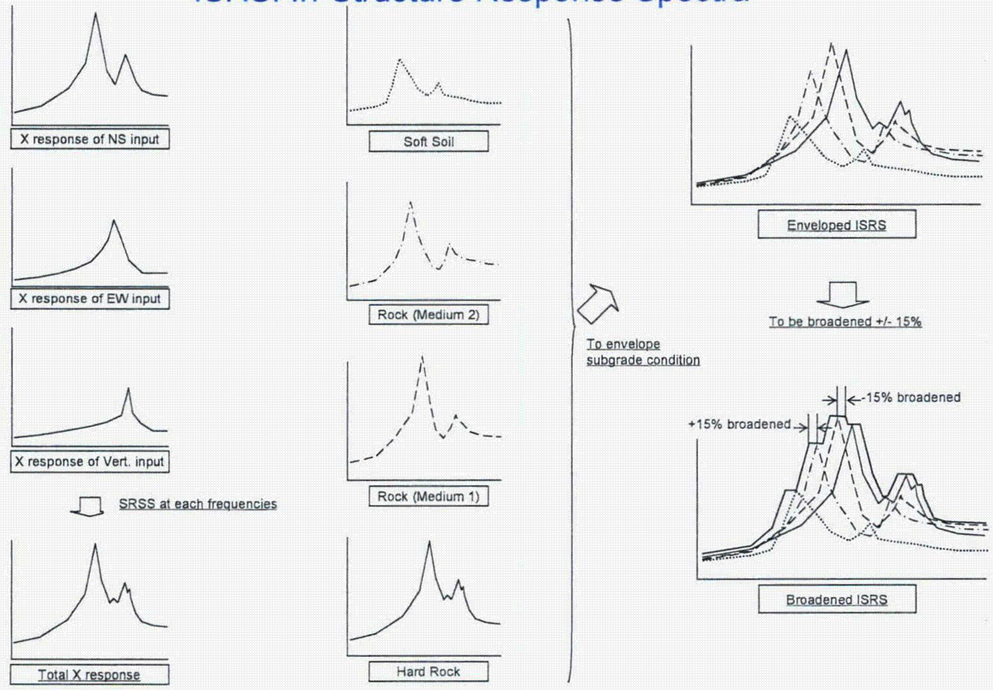
Soil-Structure Interaction (SSI)

Item		Soft Soil Site	Rock Site (Medium1)	Rock Site (Medium2)	Hard Rock Site
Vs	(fps)	1,000	3,500	6,500	8,000
	(m/s)	300	1,100	1,980	2,500
Density (pcf)		110	130	140	160
Poisson's Ratio		0.40	0.35	0.35	0.30

- Lumped parameter approach is used for SSI model.
  - ✓Reducing damping values to 60% of theoretical for translational terms.
  - ✓Fixed base model is used for Hard Rock Site.

ISRS for Subsystem Analysis

ISRS: In-Structure Response Spectra

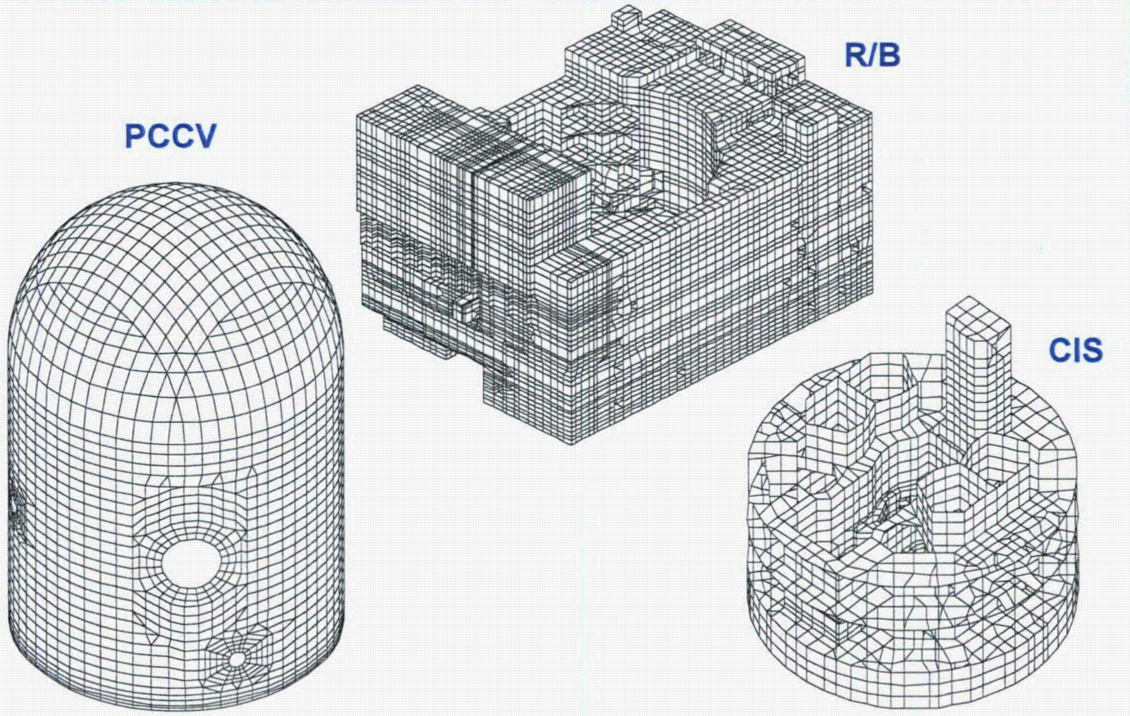


3.8 Category I Structures

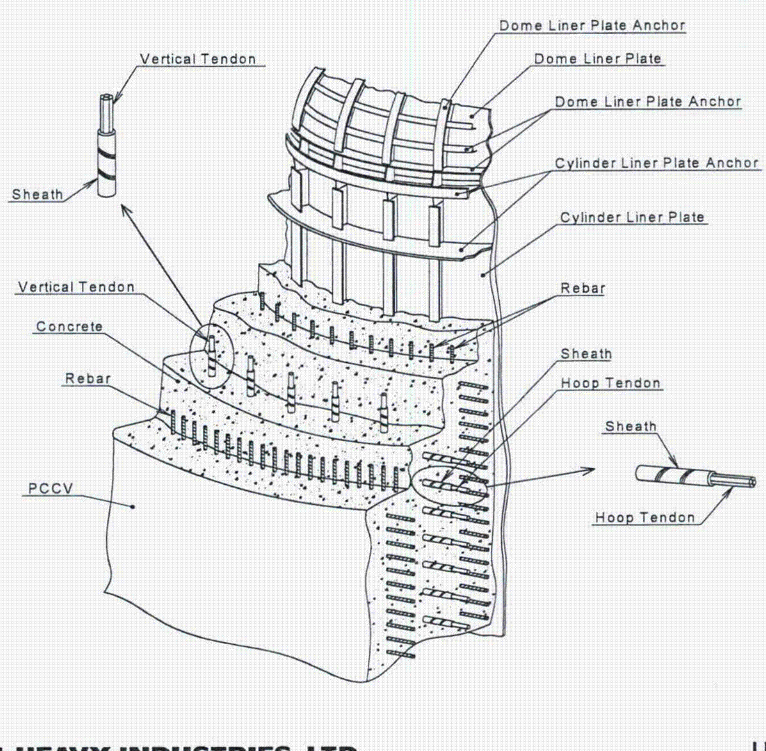
Subsection	Title	Description
3.8.1	Concrete Containment	Details of structural analysis and design as follows: ✓ General Description of Structures ✓ Applicable Codes , Standards, and Specification ✓ Loads and Load Combinations ✓ Design and Analysis Procedures ✓ Structural Acceptance Criteria ✓ Material, Quality Control, and Special Construction Techniques ✓Structural Design Results ✓ Testing and Inservice Inspection Requirements
3.8.3	Internal Structures	
3.8.4	Other Seismic Category I	
3.8.5	Foundation	
3.8.2	Steel Containment	N/A

(In the following pages, some parts of the design details of the PCCV as an example of the structures are indicated.)

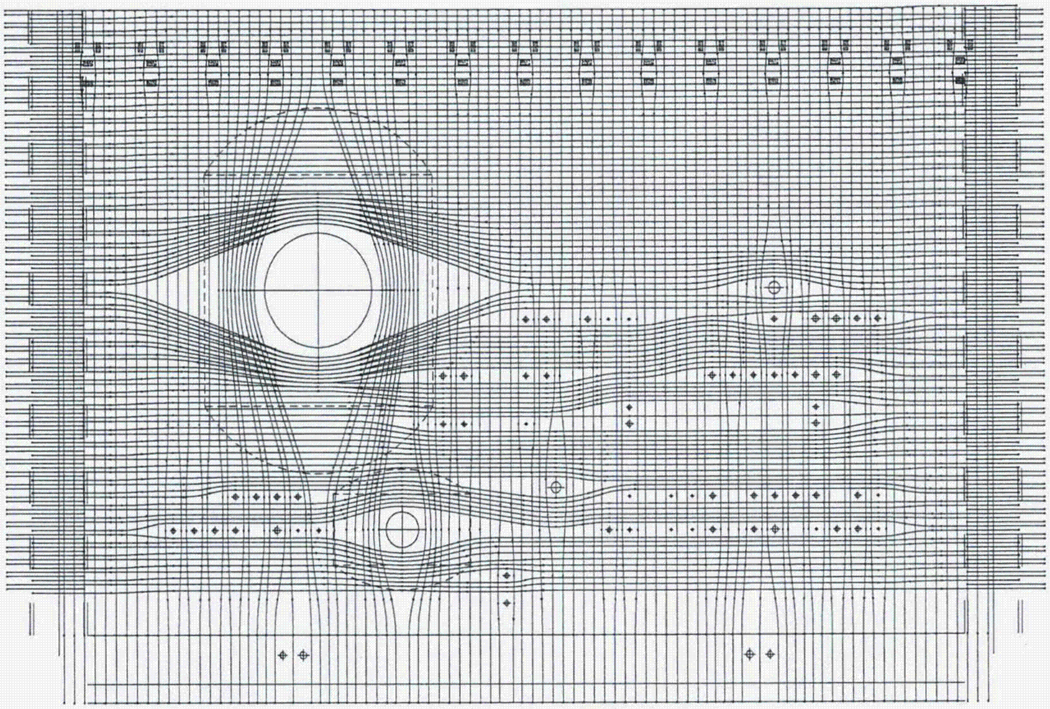
FE Model



PCCV Schematic Detail

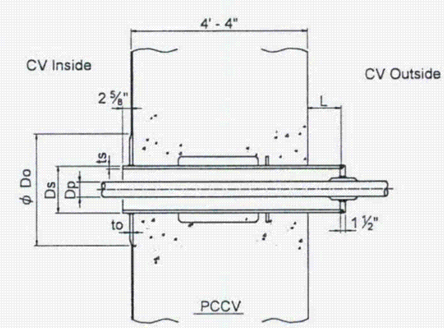


PCCV Tendons Layout



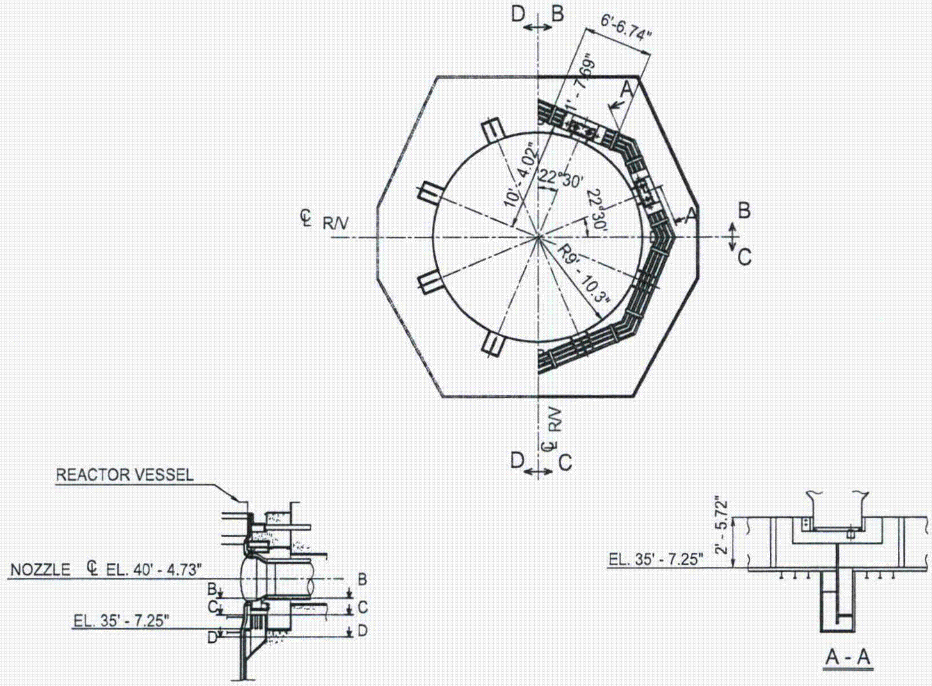
Containment Penetrations

TYPE-1 (Pipe Size=follow 4B)

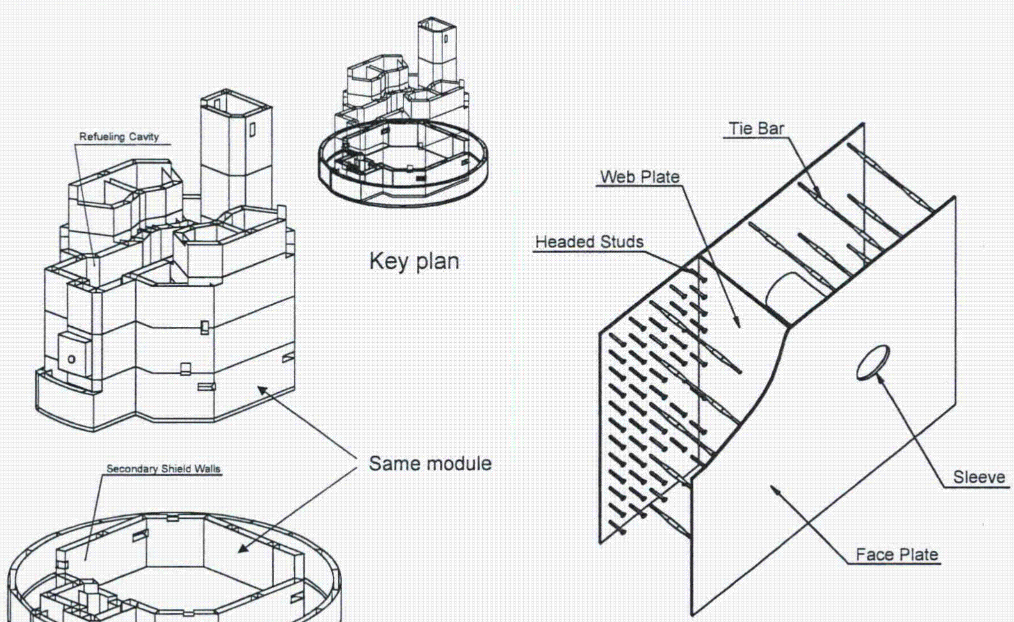


Ds	ts	Dp	L	Do	to	SLEEVE NO.
6B	1/2"	1-1/2B	7"	11 5/8"	1/2"	P279,P280,P281,P282
		3/4B	7"			P220,P222,P231,P270,P416,P417
		1B	7"			P236,P247,P265,P266
		2B	7"			P207,P230,P245,P253,P284
10B	3/8"	3B	7"	1'-5 1/4"	1/2"	P205,P248,P260,P283
14B	5/8"	4B	7"	1'-8 3/4"	1/2"	P162,P210,P227,P233,P235,P258
						P274,P278
			15"			P277

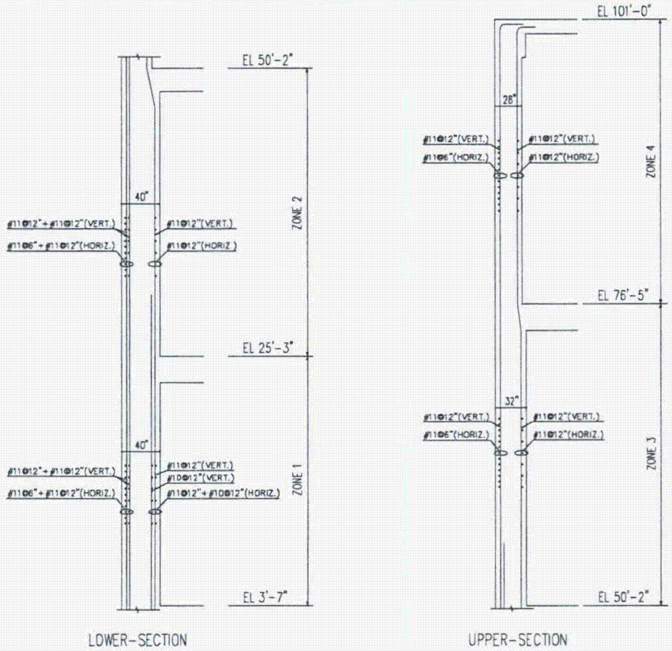
RV Support System



Containment Internal Structures



Typical Reinforcement of R/B

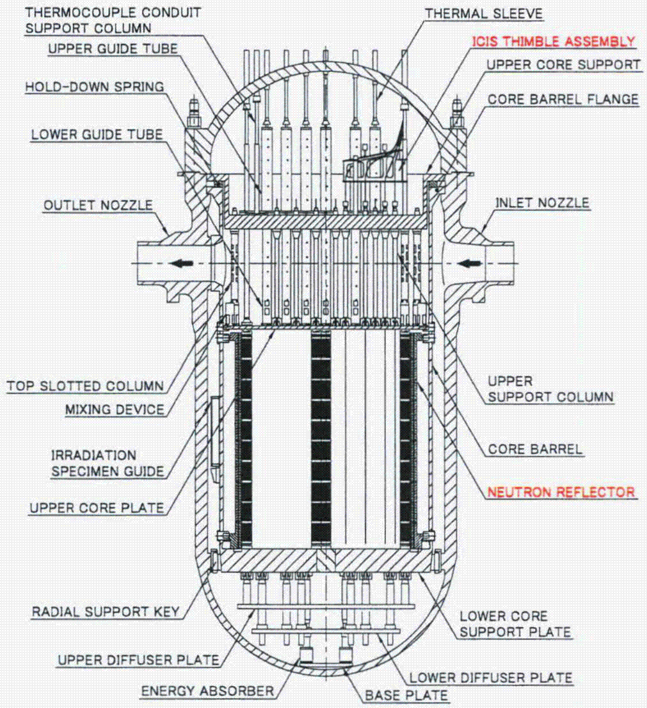


Exterior Wall of West Side

3. 9 Mechanical System and Components

Subsection	Title	Description
3.9.1	Special Topics	➢ Information on design transients for ASME Code Sec. III, class 1 Component, Component Support, and Core Support Structures
3.9.2	Dynamic Testing and Analysis	➢ Dynamic testing and analysis method of piping, components and equipment ➢ FIV analysis and testing of Reactor Internal
3.9.3	ASME Code Class 1,2,and 3	➢ Loading combination and stress limits of ASME Code class 1, 2, and 3 Component, Component Support, and CSSs
3.9.4	Control Rod Drive System	➢ Descriptive information ➢ Functional requirements ➢ Design loads, stress limits and allowable deformations
3.9.5	Reactor Pressure Vessel Internals	➢ Design arrangements ➢ Design basis for reactor internals
3.9.6	Functional design Qualification, and Inservice Testing Program	➢ Functional design and Qualification of Pump, and Dynamic Restraints ➢ IST Program for Pump, Valves, and Dynamic Restraints

Reactor Internals



ICIS THIMBLE ASSEMBLY

- Offer the neutron detector's path
- Support and bundle the detector guide thimbles

NEUTRON REFLECTOR

- Consists of multiple stocked ring blocks
- Form the core cavity instead of the baffle plates
- Reduction in the number of threaded fasteners

3.10 Seismic and Dynamic Qualification

Subsection	Title	Description
3.10.1	Seismic Qualification Criteria	➢ Decision criteria for selecting a particular test or method of analysis. ➢ Considerations defining the seismic and other relevant dynamic load input motion. ➢ Process to demonstrate the adequacy of the seismic qualification program.
3.10.2	Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation	➢ Qualification on Seismic category I mechanical and electrical equipment in accordance with IEEE Std 344-1987 and RG 1.100. ➢ Qualification by test, analysis, or combination of test and analysis. ➢ Discussion on high frequency issue in the CEUS.

3.10 Seismic and Dynamic Qualification (cont'd)

Subsection	Title	Description
3.10.3	Methods and Procedures of Analysis or Testing of Supports of Mechanical and Electrical Equipment and Instrumentation	Qualification of supports by either tests or analyses to assure structural capability, including anchorage.
3.10.4	Test and Analyses Results and Experience Database	Establishment and maintenance of complete and auditable records in the equipment qualification file.

3.11 Environmental Qualification

Subsection	Title	Description
3.11.1	Equipment Location and Environmental Conditions	➢ Identification of the equipment and selection basis ➢ Anticipated operational occurrences ➢ Accident & Post-accident ➢ Test environmental conditions.
3.11.2	Qualification Tests and Analyses	The testing and analyses are described to verify conformance to the EQ Program objectives. The environmental conditions in Appendix 3D reflect the worst-case scenario.
3.11.3	Qualification Test Results	The testing is performed to verify that safety-related SSCs, as well as those important to safety, The test results are documented and evaluated to assure that EQ Requirements have been satisfied.

3.11 Environmental Qualification (cont'd)

Subsection	Title	Description
3.11.4	Loss of Ventilation	Equipment which may be Impacted by Inadequate Ventilation or a Loss of Environmental Control is Identified During the Design Process.
3.11.5	Estimated Chemical and Radiation Environment	The Impact of the various chemicals used in the plant is factored into the design and EQ process. Equipment subject to chemical exposure will be qualified Electrical and mechanical equipment subject to radiation exposure is qualified. Equipment, that is only located in areas considered harsh by the potential present of radiation, will be qualified by analysis and partial test data.

3.11 Environmental Qualification (cont'd)

Subsection	Title	Description
3.11.6	Qualification of Mechanical Equipment	Active and passive mechanical equipment is qualified as part of the US-APWR EQ program. The EQ program provides for qualification of non-metallic components such as gaskets, O-rings, seals, lubricants for safety-related and important to safety mechanical equipment. Non-active mechanical equipment, that is equipment whose primary safety function is structural integrity, is qualified pursuant to the requirements of ASME Code, Section III.

3.12 Piping Design Review

US-APWR

- This section covers the design of piping system and piping supports which comprises seismic category I, seismic category II and non-seismic piping systems.
- Piping analysis uses the 1992 Edition including 1992 Addenda of the ASME Code, Section III, Division 1, Subsections NB, NC, and ND.
- Seismic category I pipe supports are designed in accordance with subsection NF of the ASME Code, Section III, 2001 Edition for Level A, B, and C service conditions. For Level D service condition, Nonmandatory Appendix F of Section III of the ASME Code, 2001 Edition is utilized.

3.13 Threaded Fasteners

US-APWR

Subsection	Title	Description
3.13.1	Design Considerations	Guidance for the selection of threaded fasteners for ASME Code, Section III, Class 1, 2, and 3 Systems. The materials used for all threaded fasteners must be suitable for, and compatible with, the plant design temperatures, pressures, loads, stresses, and operating service conditions. Table 3.13-1 Lists the applicable criteria in ASME Code.
3.13.2	Inservice Inspection Requirements	The preservice inspection and ISI of threaded fasteners shall comply with requirements of 10 CFR 50.55a, and ASME Code. Table 3.13-2 lists the ASME Section XI examination categories for ISI of mechanical joints in ASME Code, Class 1 and 2 systems that are secured by threaded fasteners.

4. Summary

US-APWR

- Mechanical and Electrical Components are identified in System Quality Group Classification (Section 3.2.2), IST Program (Subsection 3.9.6), and EQ program (Section 3.11).
- Design Criteria are presented for Missile Protection, Flood Design, Component and Piping Design.
- CSDRS is developed considering high frequency issue for CEUS.
- Designs of all seismic category I buildings/structures are completed.

US-APWR

Design Certification Application Orientation

Detail of FSAR

Tier 2: Chapter 4

January 15,16, 2008

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Mitsubishi Heavy Industries, LTD.

Contents

1. Overview of Chapter

Title of Chapter

Chapter 4: Reactor

Scope of Chapter

This chapter includes the design description of "Reactor", such as,

- 4.1 Summary
- 4.2 Fuel System design
- 4.3 Nuclear design
- 4.4 Thermal-Hydraulic design
- 4.5 Reactor material
- 4.6 Functional design of reactivity control

2. Format and Contents

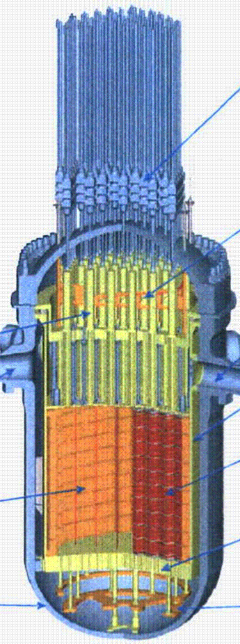
- Format and content conform to with RG 1.206.
- Content includes the information required by NUREG-0800: Standard Review Plan.
- Comprehensive descriptions for design bases, methodology, evaluation and testing/verification for each design area
- Detailed information for methodology and applications are issued in topical/technical reports.

3. Design Features

Reactor Design Features

US-APWR

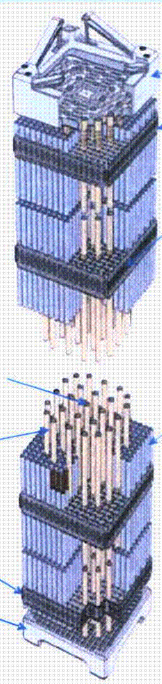
- Flexible Operation
- Enhanced Economy
- Improved Reliability



Fuel Assembly Design Features

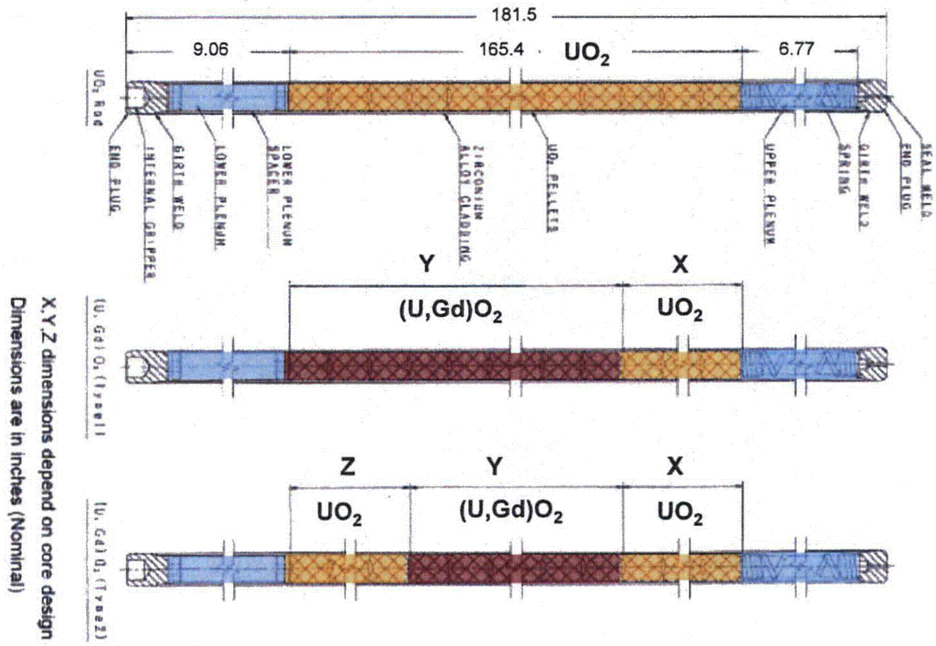
US-APWR

- Flexible Operation
- Enhanced Economy
- Improved Reliability



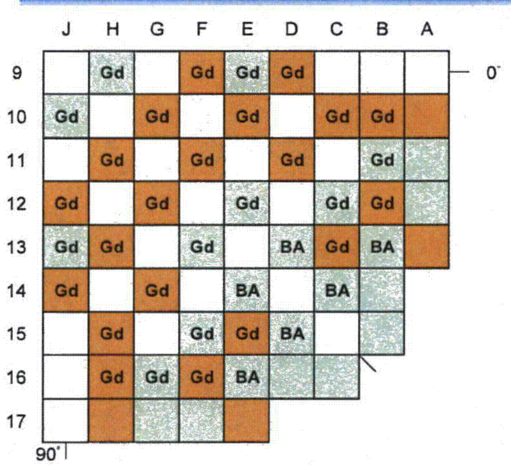
Fuel Rods

US-APWR



Core Loading Pattern (Initial core, 24 Months)

US-APWR



Region 1 – 2.05wt% U.-Enrichment

Region 2 – 3.55wt% U.-Enrichment

Region 3 – 4.15wt% U.-Enrichment

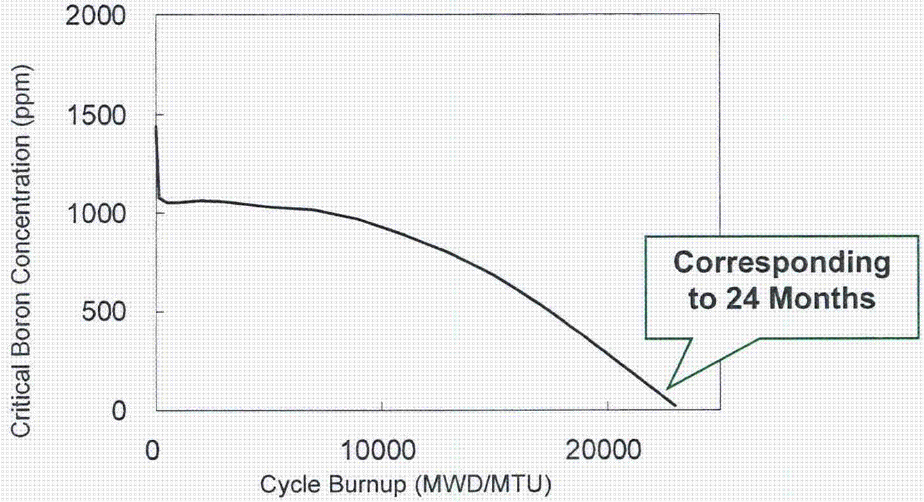
Gd Gadolinia Integrated Fuel

BA Burnable Absorber Rods (Discrete)

- **3 Regions, Checker-board Loading**
- **Variety of Burnable Poisons:**

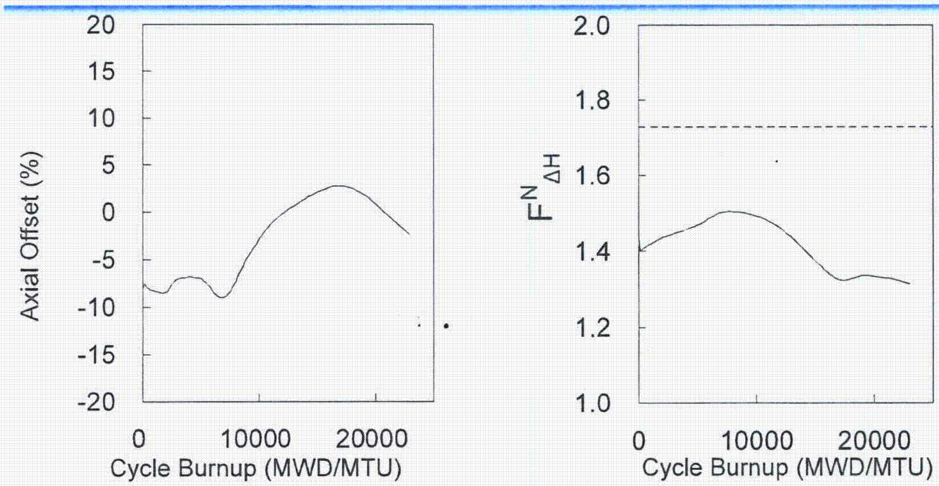
  Content of Gadolinia, and the number of BA rods in an assembly are selectable.

Boron Concentration vs. Core Depletion



- Boron concentration at BOC is properly suppressed.
  → Negative moderator temperature coefficient at HZP

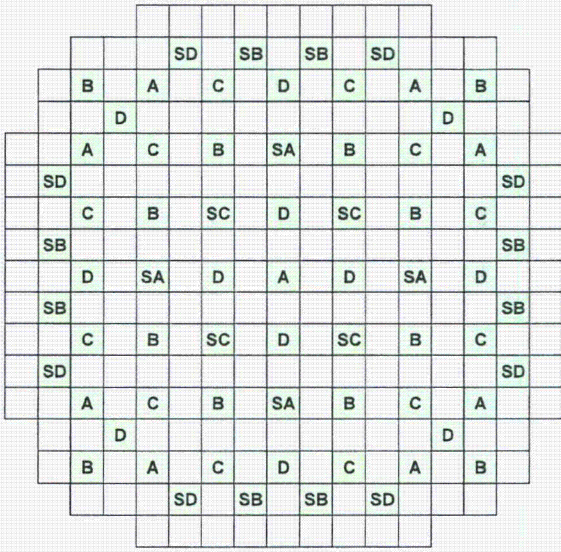
Power Peaking Factors for Initial Core



	Design Limits	Evaluated Values
F_Q	2.60	2.05 (2.35)
$F^N_{\Delta H}$	1.73	1.50 (1.66)

( ): Including Uncertainties

Shutdown Margin



RCCA Core Configuration

Shutdown Margin (Initial Core)

	BOC	EOC
Calculated SDM	2.95	2.16
Required SDM	1.60	1.60

(%Δρ)

A to D: Control group Bank
SA to SD: Shutdown group Bank

4. Analytical Techniques

Analytical Techniques

The US-APWR design

- Based on proven technologies applied in conventional PWRs
- Existing methods are essentially adequate

Methodologies and codes selection

- Maximum use of methods & codes already approved by the NRC

Analytical Techniques - Fuel Design

Design Category	Analysis Techniques/Approach	Primary Code
Rod key parameters (rod internal pressure, fuel temperatures, etc)	Fuel performance models (Thermal model, fission gas release model, etc)	FINE
FA key parameters (loads, stress, deflection)	Static and dynamic analyses	FEM codes (ANSYS, ABAQUS)

Analytical Techniques - Nuclear Design

Design Category	Analysis Techniques/Approach	Primary Code
Few-group microscopic and macroscopic cross-sections	2D current coupling collision probability methods (CCCP)	PARAGON
3D power distributions, boron concentrations, and other nuclear parameters	3D 2-group diffusion theory applied with a nodal expansion method (NEM)	ANC

Analytical Techniques - Vessel Irradiation / Criticality

Design Category	Analysis Techniques/Approach	Primary Code
Fast neutron flux	Discrete ordinates Sn transport methodology	DORT
Criticality of reactor, fuel assemblies, new and spent fuel racks, fuel handling	Monte-Carlo methodology	MCNP

Analytical Techniques - Thermal-Hydraulic Design

US-APWR

Design Category	Analysis Techniques/Approach	Primary Code
Steady-state and transient conditions	Subchannel analysis of local fluid conditions in the core, solving mass, momentum and energy conservation equations for steady state/transient conditions	VIPRE-01M

5. Summary

US-APWR

- The DCD format and content comply with RG 1.206 and SRP requirements.
- The reactor provides flexible operation with enhanced economy and improved reliability based on proven technologies.
- Methodologies and codes already approved by the NRC are selected to a maximum extent. Codes requiring NRC approval are described in topical reports. Supplemental information is provided in technical reports.

US-APWR

Design Certification Application Orientation

Detail of FSAR

Tier2: Chapter 5

January 15,16, 2008

Mitsubishi Heavy Industries, Ltd.

Presenter

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Principal Engineer

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Contents

1. Overview of Chapter
   - ✓ Title of Chapter
   - ✓ Scope of Chapter
2. Design Features of Reactor Coolant System
3. Design Features of RCPB Components
   - 3-1. Design Features of Reactor Vessel
   - 3-2. Design Features of Reactor Coolant Pump
   - 3-3. Design Features of Steam Generator

1. Overview of Chapter

➢ Title of Chapter

Chapter 5:REACTOR COOLANT AND CONNECTING SYSTEMS

➢ Scope of Chapter

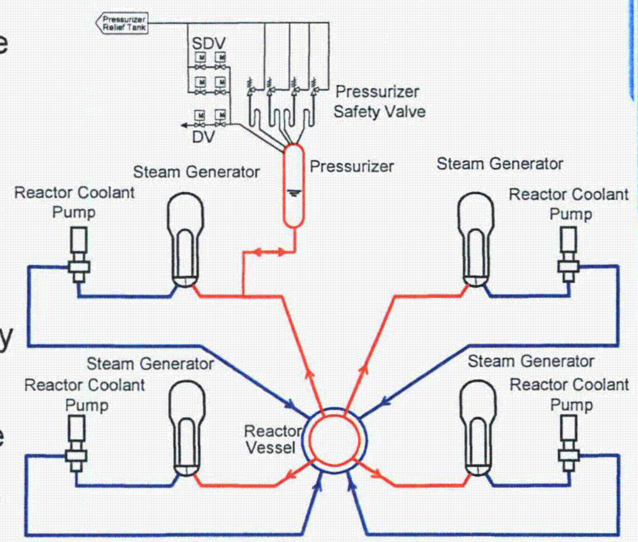
This chapter includes the following Contents;

5.1 Summary description
5.2 Integrity of Reactor Coolant Pressure Boundary
5.3 Reactor Vessel
5.4 Reactor Coolant System Component and Subsystem design

2. Design Features of 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
- ✓ Safety depressurization valves (SDVs), manual motor operated valves, are used to depress RCS in the case safety shutdown, SGTR.
- ✓ Another depressurization valve is installed for severe accident.



2. Design Features of Reactor Coolant System (Cont'd)

➤ Material Specifications of RCPB

- ✓ The materials used in the RCPB satisfy the applicable material requirements of ASME Code Section III and conform to the applicable ASME Code Section II material specifications.
- ✓ ASME SA-508 materials are used for the Class 1 ferritic pressure boundary forgings and SA-533 materials are used for the Class 1 ferritic pressure boundary parts formed from plates. These materials are covered with stainless steel or nickel-chromium-iron cladding for corrosion resistance.
- ✓ Austenitic stainless steel base materials for RCPB applications are solution heat treated to prevent stress corrosion cracking.
- ✓ Nickel-chromium-iron alloy materials for RCPB applications are thermally treated to enhance their resistance to PWSCC.

2. Design Features of Reactor Coolant System (Cont'd)

US-APWR

- **Overpressure Protection**

Overpressure protection system has the following design features:

- Pressurizer safety valves are installed on separate relief lines at the top of the pressurizer.
- Low temperature overpressure protection (LTOP) is provided by the CS/RHR pump suction relief valves installed in the each train of residual heat removal (RHR) system.

3. Design Features of RCPB Components

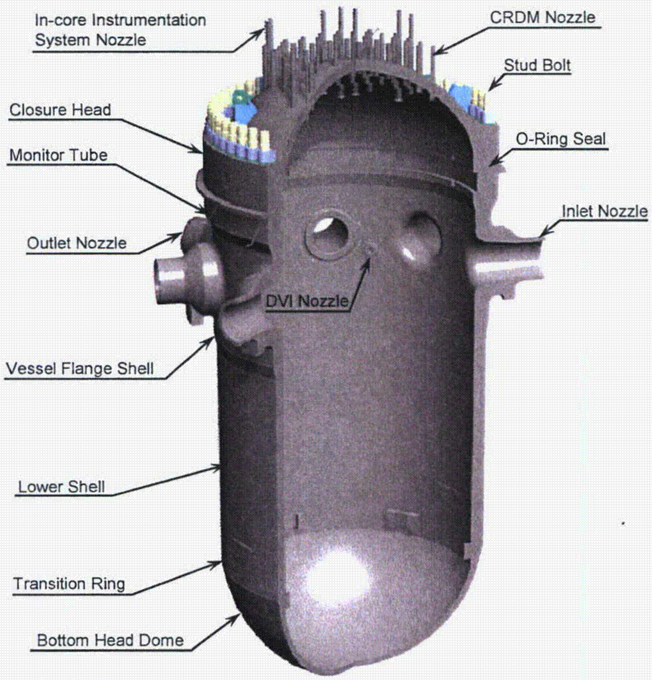
US-APWR

Overall Design Concept

- RCS components are based on conventional and established technologies
- Improvements from past experience have been incorporated

3-1. Design Features of Reactor Vessel

Reactor Vessel Configuration:



Structural Design Feature Improvements

Key improvements to the structural design features include:

- No penetrations on Bottom Head Dome
- Reduced number of instrumentation nozzles.
- Reduced number of weld lines by using integrated forging for Vessel Flange Shell.

3-1. Design Features of Reactor Vessel (Cont'd)

Reactor Vessel Materials

1. SA-508 Gr. 3 Cl. 1 with inner stainless steel cladding is used for the main pressure boundary forgings.
2. Fracture toughness requirements of ASME Code Sec. III and 10 CFR 50 Appendix G are satisfied.
3. Thermally treated Alloy 690 material is used to the Closure Head nozzles.

Manufacturing

1. Requirements of ASME Code Sec. III NB-4000 and applicable Regulatory Guides are applied.
2. Applicable welding processes include GTAW, GMAW, SMAW, PAW and SAW. Electroslag welding is applied for inner cladding.

Inspection

1. During manufacturing, ASME Code Sec. III requirements are applied.
2. PSI and ISI plans are established in accordance with the applicable requirements of ASME Code Sec. III and XI.

3-1. Design Features of Reactor Vessel (Cont'd)

Surveillance Program

1. **The reactor vessel material surveillance program is established in accordance with 10 CFR 50 Appendix H and ASTM E-185. and evaluates radiation damage of the beltline region materials.**
2. **Six (6) surveillance capsules containing test specimens for the beltline region weld metal, base metal and HAZ metal are provided.**

Pressure-Temperature Limits

1. **P-T limits are established in accordance with 10 CFR 50 Appendix G and ASME Code Sec. XI Appendix G to protect against non-ductile failure.**

Pressurized Thermal Shock

1. **RT_{PTS} values at the end of plant life are evaluated and satisfy the screening criteria of 10 CFR 50.61.**

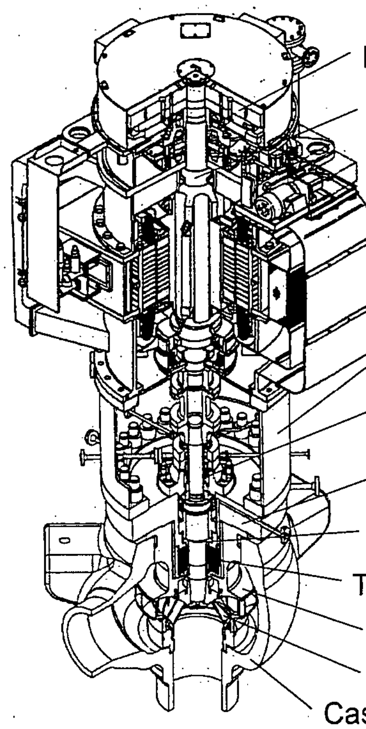
3-2. Design Features of Reactor Coolant Pump

- **Pump Performance**
  - ✓RCP assures adequate reactor coolant flow rate
  - ✓Shaft seals employ a well-established seal system that has been used in many operating plants.
- **Pump flywheel Design**
  - ✓The results of the following analyses satisfy the flywheel design requirements of RG 1.14:
    - Ductile failure analysis
    - Nonductile failure analysis
    - Fatigue Crack growth analysis
    - Excessive deformation analysis

3-2. Design Features of Reactor Coolant Pump (Cont'd)

US-APWR

Flow Rate	112,000 gpm
Head	306.9 ft
Rotating speed, synchronous	1,200 rpm
Unit design pressure	2,485 psig
Unit design temperature	650 °F
Unit overall height	28 ft
Power	8,200 hp
Voltage	6,600 V



3-3. Design Features of Steam Generator

US-APWR

Basic design
- SGs are of vertical inverted U-tube recirculation type based on conventional technologies
- SGs are capable of supporting a power of 1116.5 MWth/SG

Tube material
- Alloy 690 Thermally Treated tube material for improved corrosion resistance

Tube supports
- 405 stainless steel broached tube support plates to minimize corrosion and denting
- Anti-vibration bar design in U-bends to minimize vibration wear

Feedwater ring
- Feedwater ring design to mitigate water hammer, thermal stratification, and trapping loose parts incoming feedwater

Moisture separation system
- Centrifugal primary separators and single tier secondary separators

3-3. Design Features of Steam Generator (cont'd)

US-APWR

SG Configuration

Tube Design Data

Heat transfer area	91,500 ft^2
Number of tubes	6747
Tube outside diameter	0.75 in.
Tube pitch (triangular)	1.00 in.