US-APWR

6th Pre-Application Review Meeting
Containment Response Analysis Methodology

March 22, 2007

Mitsubishi Heavy Industries, Ltd.

Meeting Attendees

Makoto Toyama (Responsible for Safety Analysis for US-APWR)
General Manager
Reactor Safety Engineering Department
Nuclear Energy Systems Engineering Center
Mitsubishi Heavy Industries, LTD.

Shigemitsu Umezawa (Responsible for LOCA Methodology Development)
Engineering Manager
Reactor Safety Engineering Department
Nuclear Energy Systems Engineering Center
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Michitaka Kikuta (Responsible for LOCA Analysis for US-APWR)
Engineering Manager
Safeguard System Engineering Section
Nuclear Energy Systems Engineering Center
Mitsubishi Heavy Industries, LTD.

Tom George - Presenter -
Technical Consultant

Objectives of Meeting

- **The purpose of the meeting is to:**
  - ✓ Present information to the NRC on US-APWR Containment Response Analysis Methodology and ensure NRC's expectations are met
  - ✓ Provide an opportunity for the NRC to explain its process, schedule, expectations, and provide feedback to MHI

Presentation Summary (1)

1. **GOTHIC 7.2a Code will be used for US-APWR Containment Response Analysis**
   - **Containment Design Function Evaluation Assuming Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB) for**
     - ✓ Peak pressure
     - ✓ Peak temperature
     - ✓ Pressure at 24 hours

Presentation Summary (2)

2. New Features for US-APWR
   - ➢ Refueling Water Storage Pit (RWSP) inside Containment
     - ✓ No recirculation switchover procedure is required
   - ➢ Improved Safeguard System
     - ✓ 4 advanced accumulators
     - ✓ 4 safety injection pumps
     - ✓ 4 containment spray pumps
3. US-APWR Containment Design Evaluation will be Performed Based on SRP Guidance and Previously Accepted GOTHIC Methodology
4. Methodology and Results will be Reported in Chapter 6 of Design Control Document
   - ➢ Topical Report is not Planned at This Stage

Contents

GOTHIC Code (1)

- GOTHIC 7.2a will be used for US-APWR Containment Response Analysis
- Validation
  - ✓ CVTR
  - ✓ Battelle Model Containment
  - ✓ HDR
  - ✓ Marviken
- Used for U.S. Licensing and Technical Specification Analyses Approved by NRC
  - ✓ Kewaunee, Surry, Fort Calhoun and Others

GOTHIC Code (2)

- GOTHIC Features for US-APWR

Modeling Approach	Lumped Parameter Model
Flow Fields	Vapor/Liquid/Drop (Thermal Non-Equilibrium)
Governing Equations	Conservation of Mass, Energy, Momentum and Drop Surface Area
Heat and Mass Transfer Model	Convection/Conduction/Radiation Diffusion Layer Model (Heat and Mass Transfer Analogy)
Safety Features	Spray, Heat Exchanger, Pump, etc.

US-APWR Plant Parameter Summary (1)

➢ Plant Class of US-APWR

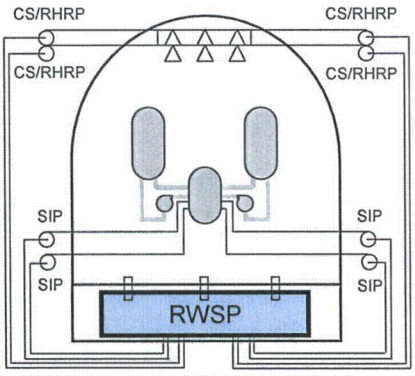
Features	US-APWR	US Current 4 Loop Plant
Core thermal output (MWt)	4,451	3,565
Number of loops, SGs and RCPs	4	4
Containment Vessel Type	Pre-stressed Concrete Containment Vessel With Steel Liner	Pre-stressed Concrete Containment Vessel With Steel Liner
Refueling Water Storage location	Inside CV	Outside CV

➢ Improved Safeguard System

- ✓ 4 advanced accumulators
- ✓ 4 safety injection pumps
- ✓ 4 containment spray pumps

US-APWR Plant Parameter Summary (2)

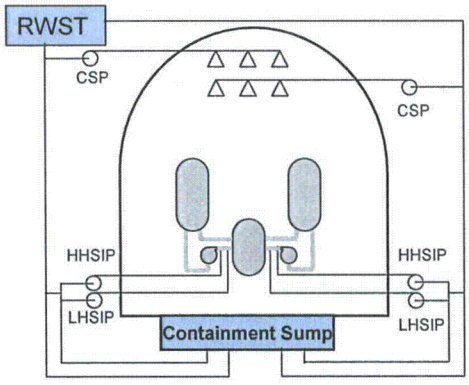
US-APWR



Improved Safety Systems

- ✓ 4 train safeguard system
- ✓ RWSP inside containment
- ✓ No switchover for recirculation

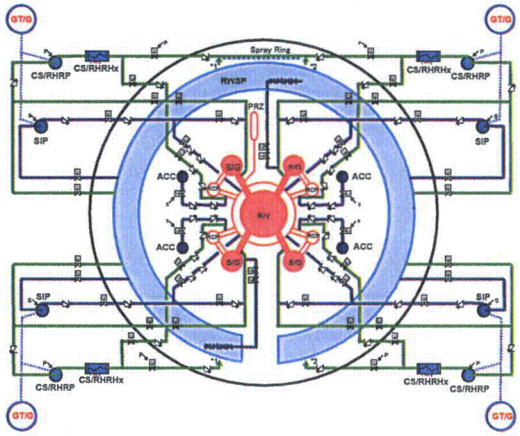
US Current PWR

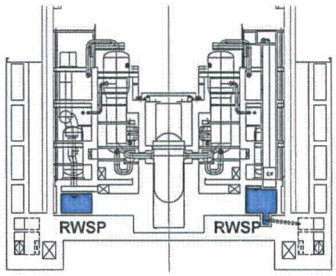


- ✓ 2 train safeguard system
- ✓ RWST outside containment
- ✓ Switch to sump recirculation when the RWST depleted

US-APWR Plant Parameter Summary (3)

- **Dry Containment with Pre-stressed Concrete Containment Vessel**
- **Refueling Water Storage Pit (RWSP) inside Containment**
  - ✓No recirculation switchover procedure is required

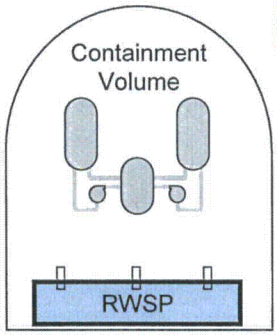




Containment Evaluation Model for LOCA (1)

- **Methodology Guidance**
  - ✓SRP 6.2.1 and 6.2.1.1A
- **Containment Geometric Modeling**
  - ✓Single or two volume system *
  - ✓Minimum Free Volume
  - ✓Minimum Heat Sinks
- **Mass and Energy Release**
  - ✓SATAN
  - ✓WREFLOOD
    - Approved by NRC
    - ANS 5.1-1979 Decay Heat plus uncertainty will be used

*Sensitivity study will be performed

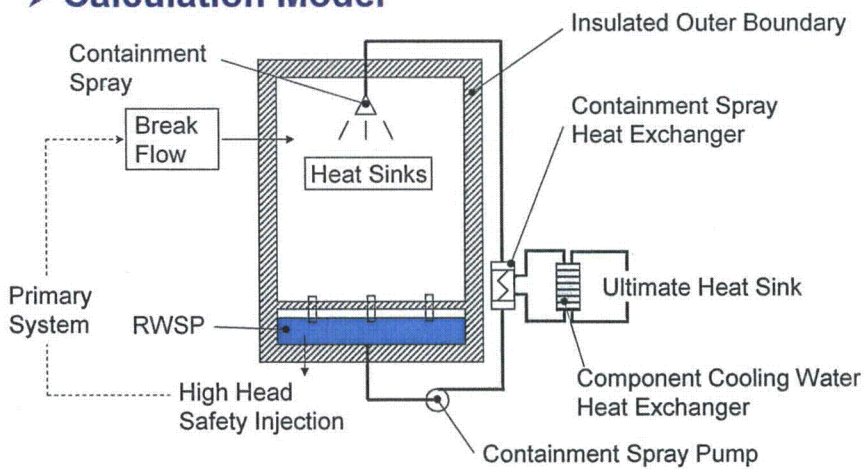


Containment Evaluation Model for LOCA (2)

- **Heat and Mass Transfer Between Containment Atmosphere and Heat Sinks**
  - ✓ **Diffusion Layer Model (DLM) as used in previous NRC approved licensing submittals**
  - ✓ **DLM validated against separate effects test data**
- **Heat Sink Modeling**
  - ✓ **Structures below are modeled as containment internal heat sinks**
    - Containment shell (Liner + Concrete)
    - Internal concrete
    - Heavy components
    - Uninsulated Pipes and supports
    - Other metals
      - Aluminum / Copper

Containment Evaluation Model for LOCA (3)

- **Calculation Model**



Containment Evaluation Model for LOCA (4)

- **Assumptions**
  - ✓ **Low energy removal capability will be assumed with limiting conditions**
    - Available safety features
      - Loss of offsite power will be assumed
      - Limiting single failure will be assumed
      - Online maintenance condition – One train assumed out for maintenance
  - ✓ **Minimum containment spray flow will be used**

Containment Evaluation Model for LOCA (5)

- **Initial Conditions**
  - ✓ **Conservative Initial Conditions for Peak Pressure and Peak Temperature Analysis will be Used**
    - Initial containment pressure
    - Initial atmosphere temperature
    - Relative humidity
    - Initial heat sink temperature
    - RWSP water volume and temperature

Containment Evaluation Model for MSLB

- **Methodology Guidance**
  - ✓ SRP 6.2.1, 6.2.1.1A and NUREG-0588
- **Evaluation Codes**
  - ✓ Mass and energy release - MARVEL
    - Assumes steam phase break flow (100% Quality)
    - Limiting initial power level will be assumed
  - ✓ Containment response - GOTHIC
    - Same as for LOCA
    - Revaporization allowed in superheat conditions
- **Assumptions**
  - ✓ Same as LOCA
  - ✓ Low energy removal capability will be assumed with limiting conditions
  - ✓ Minimum containment spray flow will be used
- **Conservative Initial Conditions will be Used**

Summary

- **GOTHIC will be used for Containment Response Evaluation for US-APWR Analysis**
- **Containment Functional Design Evaluation will be Performed Based on SRP 6.2.1, SRP 6.2.1.1A and NUREG-0588**
- **Containment Design Evaluation Methodology and Analysis Results will be Reported in Design Control Document**