US-APWR

4th Pre-Application Review Meeting

Non-LOCA Methodology

February 1, 2007

Mitsubishi Heavy Industries, Ltd.

Meeting Attendees

Makoto Toyama (Responsible for Safety Analysis for US-APWR)
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Hisanaga Takahashi (Responsible for Non-LOCA Safety Analysis)
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Junto Ogawa (Responsible for Non-LOCA Code & Methodology)
Acting Manager
Reactor Control & Protection System Engineering Section
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Objectives of Meeting

- Identify computer codes and analysis methods used to evaluate Non-LOCA events for the US-APWR
- Confirm that the Topical Report (TR) contents for Non-LOCA methodology meet the NRC's requirements
- TR of Non-LOCA codes and methodology will be submitted by the end of July, 2007

Outline of Presentation

1. Introduction
2. Contents of the Topical Report
3. Non-LOCA Computer Codes
4. Scope of Non-LOCA Transients
5. Event-Specific Methodology
6. Summary

1. Introduction

- ➢ **US-APWR design methods are the same as current PWRs in the U.S. and Japan**
  - ✓ Primary and secondary system configuration
  - ✓ Thermal hydraulics characteristics of the coolant
  - ✓ Fuel properties
  - ✓ Core kinetics
  - ✓ Reactor control and protection system functional design
  - ✓ Active safety systems

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1. Introduction (Cont'd)

- ➢ **US-APWR Plant Parameter Summary**
  - ✓ Large core thermal output
  - ✓ Large thermal margins due to the lower average linear heat rate

Features	US-APWR	US Current 4 Loop Plant
Core thermal output (MWt)	4,451	3,565
Number of loops, SGs and RCPs	4	4
Number of fuel assemblies	257	193
Fuel rod lattice	17 x 17	17 x 17
Active fuel length (ft)	14	12
Average linear heat rate (kW/ft)	4.6	5.7
Reactor coolant pump type	Centrifugal	Centrifugal
Steam generator type	U-Tube	U-Tube

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1. Introduction (Cont'd)

➢ **US-APWR Design Features**

✓ Very similar to current U.S. PWRs

Features	Effects on Non-LOCA Analysis
Neutron Reflector	Negligible change on point kinetics parameters
Simplified core lower plenum	Core inlet mixing between loops approximately the same
Pressurizer	Larger steam space moderates pressure transients
Steam generator	Smaller U-tube diameter improves the SGTR*1 transient
EFWS*2	4 separate trains with 1 pump in each loop
Digital reactor protection system	Similar to newer U.S. operating plants
Diverse actuation system	Addresses design requirements for ATWS*3
Advanced Accumulator	Not expected to actuate in Non-LOCA analysis

*1 Steam Generator Tube Rupture
*2 Emergency Feed Water System
*3 Anticipated Transient Without Scram

1. Introduction (Cont'd)

➢ **Conventional Non-LOCA codes and methodologies can be applied to US-APWR analysis**

➢ **Non-LOCA analysis follows regulatory guidance in Standard Review Plan (SRP) Chapter 15**

1. Introduction (Cont'd)

➢ Non-LOCA Computer Codes

MARVEL (WCAP-8843-P)

✓Plant system transient analysis code

TWINKLE (WCAP-7979-P-A)

✓Multi-dimensional neutron kinetics code

VIPRE-01 (NP-2511-CCM-A)

✓Subchannel thermal hydraulics analysis and fuel transient code

- DNB correlation : WRB-2 (WCAP-10444-P-A)
- Thermal design method : RTDP* (WCAP-11397-P-A)

* Revised Thermal Design Procedure

2. Contents of Topical Report

➢ Non-LOCA Methodology

✓ Computer Codes used for US-APWR
- MARVEL
  - Overview
  - Theoretical models
- TWINKLE, VIPRE-01
  - Overview

✓ Verification of MARVEL Code
- Comparison with approved code for typical events

✓ Scope of Non-LOCA Transients
- SRP Chapter 15 events

✓ Event-Specific Methodologies
- Selected typical events

✓ Sample Transient Analysis

3. Non-LOCA Computer Codes

US-APWR

➢ **MARVEL**

Plant system transient analysis code

- ✓ Original version developed by Westinghouse and used for licensing analysis in the U.S.
- ✓ NRC approved; WCAP-8843-P (1983)
- ✓ MHI implemented modifications;
  - Extended from 2-loop to 4-loop simulation
  - Built in Reactor Coolant Pump (RCP) model
- ✓ Verification for modifications to be provided in TR
  - Comparison with 4-loop LOFTRAN code (WCAP-7907-P-A) for typical events
  - Comparison with PHOENIX code (WCAP-7551) for RCP coast down transient

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3. Non-LOCA Computer Codes

US-APWR

➢ **MARVEL (Cont'd)**

- ✓ Point neutron kinetics with various reactivity effects
- ✓ Core section has 4 channels by 4 axial nodes
- ✓ SG primary side contains 4 nodes and the secondary side 1 node with saturated mixture model
- ✓ Reactor vessel mixing in the inlet and outlet plenum is simulated
- ✓ Pressurizer uses the non-equilibrium model
- ✓ Safety injection system and CVCS* are simulated
- ✓ Reactor control and protection systems are simulated

* Chemical and Volume Control System

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3. Non-LOCA Computer Codes

US-APWR

- TWINKLE

Multi-dimensional neutron kinetics code

- ✓ Developed by Westinghouse and used for licensing analysis in the U.S.
- ✓ NRC approved; WCAP-7979-P-A (1975)
- ✓ Axial one dimensional simulation with Doppler weighting factor
- ✓ Used for Reactivity Initiated Events (RIE) as follows
  - RCCA* withdrawal from subcritical
  - RCCA ejection

* Rod Cluster Control Assembly

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3. Non-LOCA Computer Codes

US-APWR

- VIPRE-01

Subchannel T/H analysis code

- ✓ Developed from COBRA code by Battelle for EPRI and used for licensing analyses in the U.S.
- ✓ NRC approved; NP-2511-CCM-A Rev.3 (1993)
- ✓ Used for the minimum DNBR* and fuel temperature transient analyses
- ✓ MHI implemented modifications concerning DNB correlation and fuel thermal conductivity degradation
- ✓ TR will be submitted in May, 2007 (T/H group)

* Departure from Nucleate Boiling Ratio

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4. Scope of Non-LOCA Transients

- Non-LOCA NSSS accidents in NRC Standard Review Plan (SRP) Chapter 15 are included
- ANSI 18.2 Classification will be utilized for Conditions II, III and IV events
- SRP Acceptance Criteria will be followed

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4. Scope of Non-LOCA Transients

- SRP Chapter 15 Events, Classification, Computer Codes

1. Increase in Heat Removal from the Primary System

Faults	Class.	Computer codes
(1) Feedwater system malfunctions causing a reduction in feedwater temperature	Cond.II	MARVEL
(2) Feedwater system malfunctions causing an increase in feedwater flow	Cond.II	MARVEL
(3) Excessive increase in secondary steam flow	Cond.II	MARVEL
(4) Inadvertent opening of a steam generator relief or safety valve	Cond.II	MARVEL, VIPRE-01*1
(5) Steam system piping failure	Cond.III*2 Cond.IV*3	MARVEL, VIPRE-01*1

*1 Steady state analysis
*2 Minor breaks
*3 Major breaks

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4. Scope of Non-LOCA Transients

- **SRP Chapter 15 Events, Classification, Computer Codes**

  2. Decrease in Heat Removal by the Secondary System

Faults	Class.	Computer codes
(1) Loss of external electrical load and/or turbine trip	Cond.II	MARVEL
(2) Inadvertent closure of main steam isolation valves	Cond.II	MARVEL
(3) Loss of condenser vacuum and other events resulting in turbine trip	Cond.II	MARVEL
(4) Loss of non-emergency ac power to the station auxiliaries	Cond.II	MARVEL
(5) Loss of normal feedwater flow	Cond.II	MARVEL
(6) Feedwater system pipe break	Cond.IV	MARVEL

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4. Scope of Non-LOCA Transients

- **SRP Chapter 15 Events, Classification, Computer Codes**

  3. Decrease in Reactor Coolant System Flow Rate

Faults	Class.	Computer codes
(1) Partial loss of forced reactor coolant flow	Cond.II	MARVEL, VIPRE-01
(2) Complete loss of forced reactor coolant flow	Cond.III	MARVEL, VIPRE-01
(3) Reactor coolant pump shaft seizure (locked rotor)	Cond.IV	MARVEL, VIPRE-01
(4) Reactor coolant pump shaft break	Cond.IV	MARVEL, VIPRE-01

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4. Scope of Non-LOCA Transients

➢ **SRP Chapter 15 Events, Classification, Computer Codes**

4. Reactivity and Power Distribution Anomalies

Faults	Class.	Computer codes
(1) Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Cond.II	TWINKLE, VIPRE-01
(2) Uncontrolled RCCA bank withdrawal at power	Cond.II	MARVEL
(3) RCCA misalignment (Dropped RCCA, Statically misalignment and Withdrawal of a single RCCA)	Cond.II Cond.III	MARVEL VIPRE-01*1
(4) Startup of an inactive reactor coolant pump at an incorrect temperature	Cond.II	N-1 loop operation not allowed
(5) CVCS*2 malfunction that results in a decrease in the boron concentration in the reactor coolant	Cond.II	Evaluation without computer code
(6) Spectrum of RCCA ejection accidents	Cond.IV	TWINKLE, VIPRE-01

*1 Steady state analysis
*2 Chemical and Volume Control System

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4. Scope of Non-LOCA Transients

➢ **SRP Chapter 15 Events, Classification, Computer Codes**

5. Increase in Reactor Coolant Inventory

Faults	Class.	Computer codes
(1) Inadvertent operation of the emergency core cooling system during power operation	Cond.II	N/A*1
(2) CVCS*2 malfunction that increases reactor coolant inventory	Cond.II	MARVEL

*1 Not Applicable - Safety Injection pump shut off head below normal operation pressure
*2 Chemical and Volume Control System

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4. Scope of Non-LOCA Transients

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➢ **SRP Chapter 15 Events, Classification, Computer Codes**

6. Decrease in Reactor Coolant Inventory

Faults	Class.	Computer codes
(1) Inadvertent opening of a pressurizer safety valve	Cond.II	MARVEL
(2) Steam generator tube rupture	Cond.IV	MARVEL

5. Event-Specific Methodology

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➢ **Typical Events Described in TR**

✓ Transients analyzed by MARVEL only

✓ Transients analyzed by MARVEL / VIPRE-01

✓ Transients analyzed by TWINKLE / VIPRE-01

✓ Transients requiring special treatment

5. Event-Specific Methodology

➤ Typical Events Described in TR

✓ Transients analyzed by MARVEL only
- Uncontrolled RCCA* Bank Withdrawal at Power

✓ Transients analyzed by MARVEL / VIPRE-01
- Complete Loss of Forced Reactor Coolant Flow

✓ Transients analyzed by TWINKLE / VIPRE-01
- Spectrum of RCCA* Ejection Accidents

✓ Transients requiring special treatment
- Steam System Piping Failure
- Feedwater System Pipe Break
- Steam Generator Tube Rupture

* Rod Cluster Control Assembly

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5. Event-Specific Methodology

➤ Typical Events Described in TR

✓ Uncontrolled RCCA* Bank Withdrawal at Power
✓ Complete Loss of Forced Reactor Coolant Flow
✓ Spectrum of RCCA* Ejection Accidents
✓ Steam System Piping Failure
✓ Feedwater System Pipe Break
✓ Steam Generator Tube Rupture

* Rod Cluster Control Assembly

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5. Event-Specific Methodology

US-APWR

➢ Uncontrolled RCCA Bank Withdrawal at Power

✓ Analyses Codes
- Plant response and DNBR : MARVEL

✓ Reactor Protection
- High neutron flux
- Overtemperature ΔT

✓ Calculation Case
- Beginning of cycle (BOC), end of cycle (EOC)
- Wide range reactivity insertion

✓ Calculation Conditions
- Moderator density coefficient
  - Least positive (BOC), Most positive (EOC)
- Doppler power coefficient
  - Least negative (BOC), Most negative (EOC)
- Conservative trip time and reactivity insertion curve

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5. Event-Specific Methodology

US-APWR

➢ Complete Loss of Forced Reactor Coolant Flow

✓ Analyses Codes
- Plant response : MARVEL
- Minimum DNBR : VIPRE-01

✓ Reactor Protection
- RCP underspeed

✓ Calculation Conditions
- Conservative inertia momentum of the RCP flywheel
- Moderator density coefficient
  - Least positive
- Doppler power coefficient
  - Most negative
- Conservative trip time and reactivity insertion curve

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5. Event-Specific Methodology

➢ Spectrum of RCCA Ejection Accidents

✓ Analyses Codes
- Core kinetics : TWINKLE
- Fuel temperature transient : VIPRE-01

✓ Reactor Protection
- High neutron flux (high and low setting)

✓ Calculation Case
- BOC and EOC hot full power
- BOC and EOC hot zero power

✓ Calculation Conditions
- Axial one dimensional simulation with Doppler weighting factor
- Conservative Doppler power defect
- Large reactivity insertion and power peaking factor
- Minimum delayed neutron fraction
- Conservative trip time and RCCA insertion curve

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5. Event-Specific Methodology

➢ Steam System Piping Failure (DNBR evaluation)

✓ Analyses Codes
- Plant response : MARVEL
- Minimum DNBR : VIPRE-01

✓ Reactor Protection
- Low steam line pressure signal in any loop
- Overpower ΔT

✓ Calculation Case
- Post-scram : Double-ended rupture, EOC hot zero power
- Pre-scram : Spectrum of break sizes, EOC hot full power

✓ Calculation Conditions
- Moderator density coefficient and Doppler power coefficient considering a stuck rod effect
- Minimum shutdown margin (hot zero power)
- Power distribution is calculated by detailed core analysis considering a stuck rod at the coldest core sector
- Core inlet mixing modeled to reflect non-uniform effects
- Limiting single failure is assumed

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5. Event-Specific Methodology

➢ Feedwater System Pipe Break

- ✓ Analyses Codes
  - Plant response : MARVEL
- ✓ Reactor Protection
  - Low SG water level in either SG
- ✓ Calculation Conditions
  - Double-ended rupture at BOC hot full power
  - Conservative scenario
    - Loss of normal feedwater is assumed at 0 seconds
    - Reactor trip by low SG water level in either SG
    - Feedwater pipe break is assumed to occur after trip
  - Discharge quality assumption
    - Quality is assumed 0 during transient
  - Limiting single failure is assumed

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5. Event-Specific Methodology

➢ Steam Generator Tube Rupture

- ✓ Analyses Codes
  - Plant response : MARVEL
- ✓ Reactor Protection
  - Low pressurizer pressure
  - Overtemperature ΔT
- ✓ Calculation Case
  - Fluid discharge for the dose evaluation
  - Margin to SG overfill analysis
- ✓ Calculation Conditions
  - Operator action time sequence established and justified
  - Operator actions are the same as current U.S. PWRs
  - Simple leak flow model used
    - Based on the primary-secondary differential pressure
    - Initial break flow rate determined by Zaloudeck correlation
  - Limiting single failure is assumed

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6. Summary

- Topical Report (TR) will identify and discuss the MHI methodology for analysis of US-APWR Non-LOCA events
- In the TR, MHI will identify computer codes, initial conditions, and methods familiar to the NRC
- TR will discuss modifications made to the analysis methods for evaluation of typical Non-LOCA events
- TR for codes and Non-LOCA methodology will be submitted by the end of July, 2007
- Analyses for each accident using these methodologies will be provided in DCD

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