US-APWR

5th Pre-Application Review Meeting
Overview of
Probabilistic Risk Assessment (PRA) Methodology

March 1, 2007

Mitsubishi Heavy Industries, Ltd.

Meeting Attendees

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Leader of Safety and Licensing Group

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Engineering Manager of Safety and Licensing Group

Futoshi Tanaka[1] **(Representative of PRA)** –Presenter–
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Objectives of Meeting

- Overview of the US-APWR PRA methodology for Design Certification application based on applicable U.S. guides and standards
- Receive feedback from NRC concerning planned PRA applications relevant to the US-APWR Design Certification

Outline

1. PRA Related Requirements and Guides
2. Goals and Scope of US-APWR PRA
3. US-APWR Design Features
4. Basic Approach
5. Level 1 PRA Technical Elements
6. Level 2 PRA Technical Elements
7. Level 3 PRA Technical Elements
8. Conclusions
9. Discussion

1. PRA Related Requirements and Guides

- **PRA requirement**
  - ✓10 CFR 52.47 (a)(1)(v)
    "An application for design certification must contain a design-specific probabilistic risk assessment"
- **PRA scope and quality**
  - ✓Reg.Guide 1.200 Rev.1
    - ASME RA-S-2002 and the addenda : ASME RA-Sa-2003, ASME RA-Sb-2005
    - ANSI-ANS 58.21-2003
  - ✓DG-1145 as issued for public comment
    - PRA portion

•PRA results will be documented in **Chapter 19 of design control document (DCD) and the PRA report**
•Topical reports are not anticipated at this stage

2. Goals and Scope of US-APWR PRA

(1) Goals of US-APWR PRA

- Demonstrate that the NRC safety goal and the Mitsubishi Heavy Industries (MHI) target are satisfied
  - ✓MHI safety target
    - Level 1: Core Damage Frequency $<10^{-5}$/ry
    - Level 2: Large Release Frequency $<10^{-6}$/ry
    (MHI Safety Targets announced in 2nd PAR meeting)
- Establish requirements for combined operating license (COL) to assure US-APWR satisfies the MHI safety target
- Identify the dominant risk contributors and consider risk insights in the US-APWR design

2. Goals and Scope of US-APWR PRA

(2) Scope of US-APWR PRA

Initiator	POS*	Level 1	Level 2	Level 3
Internal Event	Full Power	PRA	PRA	PRA
	Low Power and Shutdown	PRA	▲	△
Internal Flood	Full Power	PRA	PRA	△
	Low Power and Shutdown	PRA	▲	△
Internal Fire	Full Power	PRA	PRA	△
	Low Power and Shutdown	PRA	▲	△
Seismic	Full Power	Seismic Margin Method		
	Low Power and Shutdown			
Other External Events	Full Power	Consider screening approach based on frequencies of the events		
	Low Power and Shutdown			

▲ :Conservative or bounding estimation of frequency * POS:Plant Operating State

△ :Conservative estimation of dose

3. US-APWR Design Features

➢ **Basic design concept of US-APWR is similar to current US PWRs. However several new concepts are incorporated**

			Current 4 Loop Plant*	US-APWR
Safety Systems	Trains	Electrical	2 trains	4 trains
		Mechanical	2 trains	4 trains
	Systems	HHSI pump	100% x 2	50% x 4 (Direct Vessel Injection)
		LHSI pump	100% x 2	-
		ACC	4	4 (Advanced ACC)
	RWSP		Outside CV	Inside CV
I & C	Control Room		Conventional	Full Digital
	Safety I&C		Conventional	
	Non-Safety I&C		Full Digital	

*Callaway NPP

3. US-APWR Design Features

➢ **Qualitative effect of US-APWR design features**

✓4 train Core Cooling System

- High reliability due to 4 advanced Accumulators (ACC) and 4 train high head safety injection system

✓In-containment Refueling Water Storage Pit (RWSP)

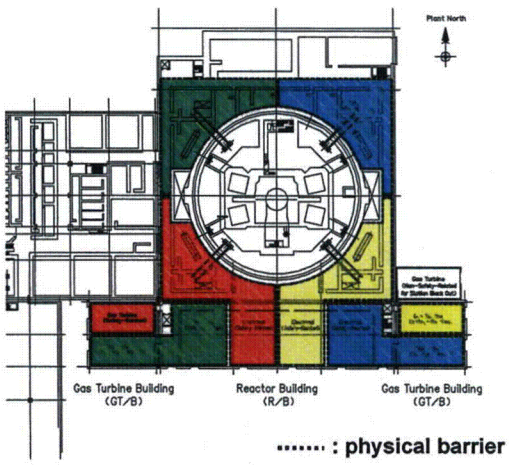
- Elimination of recirculation switchover enhances reliability of long term core cooling after LOCA

✓Full Digital Instrumentations and Controls (I&C)

- Diverse actuation system installed as a counter-measure against common mode failures in software of safety I&C

3. US-APWR Design Features

✓ 4 train safety related systems separated by physical barriers



Robust design against fire and flood events.

Fire and flood events occurring in one train do not affect other trains

4. Basic Approach

- **Basic design concept of US-APWR is similar to current PWRs. Therefore, present guides and standards are considered**
  - ✓ Reg.Guide 1.200 Rev.1
  - ✓ DG-1145
  - ✓ NUREG-0800
  - ✓ Standards endorsed by Reg.Guide 1.200 Rev.1
    - ASME RA-S-2002 and the addenda : ASME RA-Sa-2003, ASME RA-Sb-2005
    - ANSI-ANS 58.21-2003
  - ✓ Areas where no formal standards exists, previous studies or guidance are used

4. Basic Approach

List of documents considered

- Reg.Guide 1.200 Rev.1
- NUREG-0800
- NUREG-1150 (NUREG/CR-4550)
- NUREG-1407
- NUREG-1792
- NUREG-1829
- NUREG-1842
- DG-1145
- NUREG/CR-1278
- NUREG/CR-2300
- NUREG/CR-5497
- NUREG/CR-5750
- NUREG/CR-6850
- NUREG/CR-6890
- NFPA-805
- ASME RA-S-2002
- ASME RA-Sa-2003
- ASME RA-Sb-2005
- ANSI-ANS 58.21-2003

4. Basic Approach

- **Model based on available design information**
  For the areas where specific information will be determined at COL, the PRA will determine limiting boundary conditions and operating assumptions while meeting NRC safety goals.

Assumptions will be used for site specific information such as:

✓Seismic design and anchorage

✓System operating procedures

✓Test and maintenance

5. Level 1 PRA Technical Elements

Contents

5.1 Internal Events at Full Power Operation
- (1) Initiating Event Analysis
- (2) Success Criteria Analysis
- (3) Accident Sequence Analysis
- (4) System Analysis
- (5) Dependencies
- (6) Human Reliability Analysis
- (7) Component Failure Data
- (8) Quantification and Insights

5.2 Internal Events at Low Power and Shutdown

5.3 Internal Flood

5.4 Internal Fire

5.5 Seismic Event

5.6 Other External Events

5.1 Internal Events at Full Power Operation

(1) Initiating Event Analysis

- **Initiating event screening**
  - ✓ Review of U.S. PRAs for relevant events
  - ✓ Review design features of US-APWR that impact the initiators (system screening and Failure Mode and Effects Analysis)
- **Data sources used to quantify initiating event frequencies**
  - ✓ LOCA events (NUREG-1829)
  - ✓ LOOP events (NUREG/CR-6890)
  - ✓ Transients (NUREG/CR-5750)

5.1 Internal Events at Full Power Operation

(2) Success Criteria Analysis

- **Based on design basis analysis or best estimate analysis**
- **Utilize safety analysis codes or MAAP**
- **Definition of core damage**

  Consistent with ASME RA-S-2002 and the addenda, such as peak clad temperature exceed 1200°C (2200°F)

5.1 Internal Events at Full Power Operation

(3) Accident Sequence Analysis

- Event tree analysis
  - ✓Event trees with linked fault trees
  - ✓Event trees are constructed to depict response to initiating event at a level of detail to facilitate analysis and support sequence review and discussion

5.1 Internal Events at Full Power Operation

(4) System Analysis

- The Fault Tree model will include contributions due to following
  - ✓ Random component failures
  - ✓ Outages for maintenance and test
  - ✓ Support systems
  - ✓ Common cause failures
  - ✓ Human errors
    - Failing to restore equipment to its operable state
    - Failing to perform procedural actions

5.1 Internal Events at Full Power Operation

(5) Dependencies

- Components (Common cause failures)
  - ✓ **Methodology: Multiple Greek Letter (MGL)**
  - ✓ **Data source: NUREG/CR-5497**
- Systems
  - ✓ Support systems linked into each system

Component	Failure mode (failure rate)	Common Cause Components Group	MGL Parameter (NUREG/CR-5497)			Failure Probability
			Beta	Gamma	Delta	
High Pressure Safety Injection Pump	Fail to Start (3E-3 /d)*	3	5.53E-2	5.34E-1		Q_1: 2.83E-3 Q_2: 3.87E-5 Q_3: 8.86E-5
		4	5.33E-2	5.40E-1	7.66E-1	Q_1: 2.84E-3 Q_2: 2.45E-5 Q_3: 6.73E-6 Q_4: 6.61E-5

*: NUREG-CR-4550

5.1 Internal Events at Full Power Operation

(6) Human Reliability Analysis

- Consider the technique in NUREG/CR-1278 for human error rate prediction
- Recent information will also be considered, including:
  - ✓ NUREG-1792 "Good Practices for Implementing Human Reliability Analysis"
  - ✓ NUREG-1842 "Evaluation of Human Reliability Analysis Methods Against Good Practices"

5.1 Internal Events at Full Power Operation

(7) Component Failure Data

➢U.S. generic data will be used.

A variety of generic sources will be considered, such as:

✓ **NUREG/CR-4550**, "Analysis of Core Damage Frequency From Internal Events, Methodology Guidelines"

Components	Failure Mode	Failure Rate	Reference
PWR Auxiliary Feedwater Motor-Driven Pumps	Fail to start	3E-3/d	NUREG/CR-4550
	Fail to run	3E-5/hr	
PWR Auxiliary Feedwater Turbine-Driven Pumps	Fail to start	3E-2/d	NUREG/CR-4550
	Fail to run	5E-3/hr	

5.1 Internal Events at Full Power Operation

(8) Quantification and Insights

➢ **Core damage frequency (CDF)**

✓ Total CDF determined along with CDF from initiating event groups

✓ Significant sequences/cutsets which contribute to core damage identified to support generation of insights

✓ Uncertainty of CDF

Monte-Carlo simulation of data parameter uncertainties

➢ **Sensitivity of key assumptions**

✓ Test and Maintenance

✓ Operating procedures

And others

5.1 Internal Events at Full Power Operation

(8) Quantification and Insights

- **PRA Software**

  RiskSpectrum

  - Developed and maintained by Relcon Scandpower AB
  - Capabilities
    - ✓Fault tree linking / Event tree linking
    - ✓Common cause failure (CCF) event modeling
    - ✓Minimal cut set (MCS) analysis
    - ✓Importance analysis
    - ✓Uncertainty analysis

5.2 Internal Events at Low Power and Shutdown

- **Basic methods are same with full power operation PRA**
- **Initiating events**

  Representative set of initiating events chosen and modeled (e.g. overdraining during mid-loop and loss of RHR cooling etc.)
- **Plant operating states (POS)**

  Bounding set of plant operating states conservatively chosen and evaluated

POS	Description	
A	Full Power to Hot Shutdown	Consider as full power PRA (removed from Low power and shutdown)
B	Steam Generator Heat Removal	
C	**RHR Heat Removal with Level in pressurizer**	**Consider in LPSD PRA**
D	**RHR Heat Removal at Mid-loop (RCS closed)**	**Consider in LPSD PRA**
E	**RHR Heat Removal at Mid-loop (RCS opened)**	**Consider in LPSD PRA**
F	Reactor Cavity Flooded	Verify impact on CDF is negligible

5.3 Internal Flood

Basic methodology consistent with ASME RA-S-2002 and the addenda

- **Approach**
  Refer NUREG/CR-2300, NUREG-1150
- **Data**
  Pipe rupture frequency data consistent with most recent U.S. data

Step 1. Qualitative Evaluation	→	Step 2. Quantitative Evaluation
a. Flooding area identification		d. Flooding scenario development
b. Flooding source identification		e. Flooding initiating event evaluation
c. Qualitative screening		f. Quantification of flooding scenario

5.4 Internal Fire

Basic methodology consistent with NFPA-805
"Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants"

- **Approach**
  Refer NUREG/CR-6850 "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities"
- **Data**
  NUREG/CR-6850

Step 1. Qualitative Evaluation	→	Step 2. Quantitative Evaluation
a. Fire compartment identification		d. Fire scenario development
b. Fire component identification		e. Fire frequency calculation
c. Fire compartment screening		f. Quantification of fire scenario

5.5 Seismic Events

➢ **Methodology**

Seismic Margin Analysis (SMA)

Objective is to confirm plant robustness against review level earthquake

✓ Consistent with ANSI-ANS 58.21-2003

✓ SMA Model: Based on internal event PRA Model expanded to account for structural dependencies

✓ Calculation of systems, sequences and plant high confidence of low probability failure(HCLPF)

✓ Review level earthquake:0.5g (SSE 0.3g x 1.67)

5.5 Other External Events

Screening approach will be considered based on the frequency of the events

➢Consistent with NUREG-1407

✓Other external events

- High winds and tornadoes
- External floods
- Transportation and nearby facility accidents

✓Screening criteria such as CDF below 10^{-6}/y will be used

6. Level 2 PRA Technical Elements

Contents

6.1 General Approach for Level 2 PRA

6.2 Level 2 PRA Procedure for Full Power Internal Events

(1) Plant Damage State Analysis

(2) Accident Progression Analysis

(3) Source Term Analysis

(4) Quantification

6.3 Other Events

(1) Low Power and Shutdown Internal Events

(2) Internal Floods and Internal Fire

6.1 General Approach for Level 2 PRA

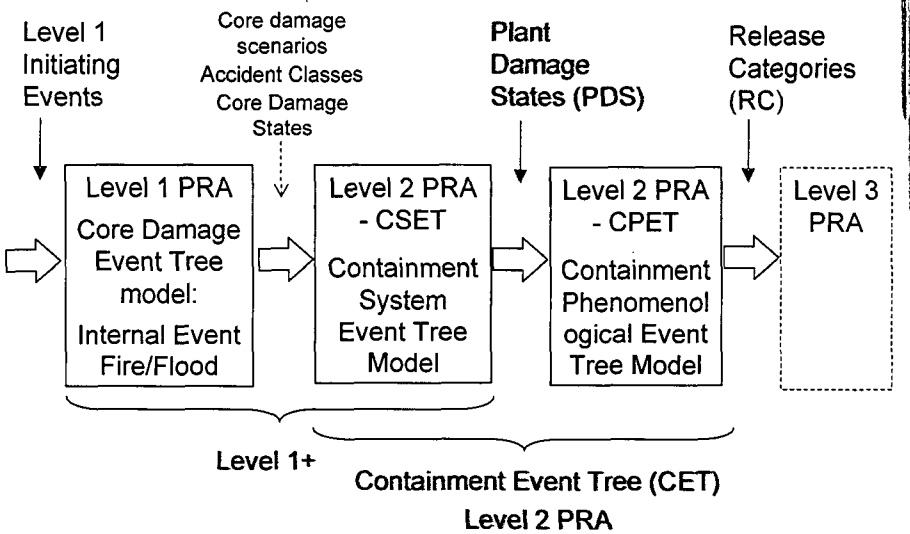
- **Evaluate Large Release Frequency (LRF)** (including early and late containment failure modes)
  - ✓ Reference evaluation approach and guidance
    - NUREG/CR-2300
    - R.G.1.200 Rev.1 (January 2007)
    - ASME RA-S 2002 and addenda (interpreted for LRF)
  - ✓ Demonstrate LRF goal of 10^{-6} /ry is satisfied
- **Full-power internal events are fully analysed, other events (Low Power and Shutdown, Fire, Flood) by bounding analyses**
- **Perform analyses with a method acceptable to NRC**

6.2 Level 2 PRA Procedure for Full-Power Internal Events

- **Technical Elements for Level 2 PRA based on R.G.1.200 Rev.1 (January 2007)**
  - (1) Plant Damage State (PDS) Analysis
  - (2) Accident Progression Analysis
  - (3) Source Term Analysis
  - (4) Quantification
- **Refer to previously performed studies for task level guidance, such as NUREG/CR-2300 or NUREG-1150**

(1) Plant Damage State Analysis

- **PRA sequences and interfaces**



(1) Plant Damage State Analysis

- **Level 1+ model**
  - ✓ Extend Level 1 PRA to include containment systems and actions (CSET)
  - ✓ Account for system dependencies by linking to Level 1 PRA
- **Define PDS by similarity of severe accident progression and containment response in CPET**
  - ✓ RCS pressure
  - ✓ Containment conditions (wet or dry)
  - ✓ Availability of severe accident mitigation features
- **Assign PDS as Level 1+ end states**
- **Quantify PDS frequency**

(2) Accident Progression Analysis

- **Construction of Containment Systems Event Tree (CSET)**
  - ✓ Consider the following containment system features and human actions
    - Containment Isolation
    - Containment Spray
    - Alternative Containment Cooling
    - RCS Depressurization
    - Combustible Gas Control
    - Fire water Cavity Flooding System
    - Recovery of Containment Cooling
- **CSET end states are the Plant Damage States (PDS)**

(2) Accident Progression Analysis

- **Construction of Containment Phenomenological Event Tree (CPET)**
  - ✓ Consider physical phenomena of accident progression such as:
    - Reactor Vessel Failure
    - Hydrogen Mixing and Combustion
    - Core Debris Coolability
    - Steam Explosion (In- and Ex-vessel)
    - Direct Containment Heating
    - Temperature Induced Steam Generator Tube Rupture
    - Molten Core Concrete Interaction
    - Long Term Containment Pressurization
- **Combination of CSET and CPET forms Containment Event Tree (CET)**
- **CET end states are the Release Categories**

(2) Accident Progression Analysis

- **Severe accident phenomenological analysis**
  - ✓ Employ MAAP 4.0.6 to evaluate:
    - Timing of severe accident phenomena
    - Capability of severe accident mitigation features
    - Time margins for activation of severe accident mitigation features
    - Containment loads
- **Containment structural analysis**
  - ✓ Consider specific technique for structural analysis, such as Finite Element Method, to evaluate:
    - Capacity of the containment to withstand severe accident environments
  - ✓ Alternatively substitute minimum pressure capacity specification

(3) Source Term Analysis

- **Define release categories**
  - ✓ Classify CET end states into the defined release categories considering:
    - Timing of containment failure (early or late)
    - Containment bypass
    - Failure of containment isolation
    - Debris coolability
    - Fission product scrubbing/release path
- **Source term evaluation**
  - ✓ Review MAAP 4.0.6 accident progression analysis results for source terms representing each release category
- **Source term and frequencies are transferred to the Level 3 PRA**

(4) Quantification

- **Quantify branch probabilities of CET for every PDS**
  - ✓ for Severe accident mitigation features, including human actions (CSET portion)
    - Employ Fault Tree
  - ✓ for Severe accident phenomena (CPET portion)
    - Employ Decomposition Event Tree (DET)
    - Refer to supporting analyses for severe accident phenomena evaluations or existing PRA
      - Example of Existing PRA: NUREG-1150
    - Stress-strength interference analysis
- **Evaluate LRF uncertainties**
- **Calculate frequency of every Release Category**
  - ✓ Link Level 1 PRA to Level 2 PRA via PDS
- **Evaluate Source Term for every Release Category**
  - ✓ Calculate Source Terms with MAAP 4.0.6

6.3 Other Events

(1) Low Power and Shutdown Internal Events, Internal Flood and Internal Fire

- ✓ Conservatively assume LRF is same as CDF
- ✓ Simple conservative bounding technique will be employed

(2) Internal Flood and Internal Fire

- ✓ Apply the same approach as Full-Power Internal Events for each PDS
- ✓ Modify fault tree models taking into account:
  - Flood/Fire induced failures of severe accident mitigation features
- ✓ Map into internal events via PDS

7. Level 3 PRA Technical Elements

Contents

7.1 General Approach for Level 3 PRA
(1) Targets
(2) Assumptions
7.2 Level 3 PRA Procedure for Full Power Internal Events
(1) Obtain source term of each release category from Level 2 PRA
(2) Calculate meteorological conditions
(3) Estimate atmosphere dispersion and deposition
(4) Estimate the radiation doses
(5) Obtain frequency of each release category from Level 2 PRA
(6) Quantification
7.3 Other Events

7.1 General Approach for Level 3 PRA

- Scope determined by DG-1145 and SAMDA needs
- Targets
  - ✓ Evaluate population whole-body dose (man-rem) out to both 10 miles and 50 miles.
- Use ORIGEN2 for radionuclide inventory and MACCS2 for dose consequence analysis
- Assumptions
  - ✓ The meteorological data for the analysis will bound 70 to 80 percent of current reactor sites in the United States.
  - ✓ The topographical and population data will be based on representative data from a reference site. Large bodies of water are conservatively treated as land in this assessment.

7.2 Level 3 PRA Procedure for Full-Power Internal Events (1)

- Offsite dose evaluation procedure based on NUREG/CR-2300 and NUREG-1150

(1) Obtain source terms from Level 2 PRA
- ✓ The amount of released material
- ✓ Released energy
- ✓ Duration
- ✓ Location
- ✓ Chemical form of iodine

(2) Calculate meteorological conditions
- ✓ Meteorological data (wind direction, wind speed, stability, the amount of rainfall)
- ✓ Categorization of meteorological data and sampling times per meteorological category

7.2 Level 3 PRA Procedure for Full-Power Internal Events (2)

(3) Estimate atmosphere dispersion and deposition
- ✓ Atmosphere dispersion model (Gaussian Plume Model)
- ✓ Deposition processes (dry deposition, wet deposition, etc.)

(4) Estimate the radiation doses
- ✓ Evaluate doses required for SAMDA dose commitment

(5) Obtain release category frequencies from Level 2 PRA

(6) Quantification
- ✓ Multiply the calculated fission product release category frequency by the release category mean dose and sum for each calculation.

7.3 Other Events

- Events other than full power internal event
  - ✓ Full power
    - Internal flood and Internal fire
  - ✓ Low power and shutdown (LPSD)
    - Internal event
    - Internal flood and Internal fire
- Methodology for full power
  - ✓ Internal flood and internal fire
    - Basically same as full power internal events

7.3 Other Events

- Methodologies for LPSD
  - ✓Internal event
    - For dominant release categories
      Whole-body doses are estimated by multiplying consequences of each release category by frequency
    - For other release categories
      Whole-body doses are conservatively estimated by multiplying the representative release category consequence by frequencies of the release categories
  - ✓Internal flood and internal fire
    - Basically same as LPSD internal event

8. Conclusions

- Overview of PRA methodologies planned for the US-APWR have been provided to the NRC
- Comments and suggestions received from NRC will be considered in the US-APWR PRA
- PRA results will be documented in DCD and the PRA report

9. Discussion

- Are there any NRC comments or questions on MHI PRA methodology?
- Are there any planned changes to requirements and guidance?
- Are changes planned to DG-1145 relative to PRA?