US-APWR

8th Pre-Application Review Meeting

Severe Accident Analysis Methodology

July 26, 2007

Mitsubishi Heavy Industries, Ltd.

Meeting Attendees

Makoto Yamagishi[(1)] (Leader of Severe Accident Analysis)

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Hiroshi Goda[(1)] (Representative of Severe Accident Analysis) –Presenter–

Engineer of Safety and Licensing Integration Group

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Objectives of Meeting

- Present MHI's severe accident analysis methodology as proposed at 1 March 2007 meeting with NRC on US-APWR severe accident mitigation features
- Describe MHI's approach to evaluate the effectiveness of US-APWR severe accident mitigation features
  - ✓ Technical approach
  - ✓ Modelling of analysis
- Obtain NRC's feedback
  - ✓ On MHI's approach to evaluate effectiveness of severe accident mitigation features
  - ✓ On MHI's severe accident analysis methodology
- Clarify the necessity for MHI to meet the recently approved requirements by NRC, especially in 10 CFR 52.47

Discussion Outline

1. Definition of severe accident
2. NRC policy and regulations for severe accident mitigation issues on new reactors
3. MHI interpretation of NRC requirements
4. Objectives of severe accident analysis
5. General approach for US-APWR severe accident analysis
6. Severe accident analysis methodology for phenomena addressed in US-APWR design
7. Conclusions

1. Definition of Severe Accident

- Class of accidents beyond the design basis which result in core damage
- May occur if plant conditions significantly exceed the design basis limits, such as:
  - Severe fuel damage
  - RCS pressure boundary stress
  - Containment pressure loads
  - Beyond design basis radiological release

2. NRC SA Policy and Regulations

- **NRC has issued policy statements and regulations regarding severe accident mitigation for new reactors**

  - ✓ 50 FR 32138: Policy statement on severe accidents regarding future designs and existing plants
  - ✓ 10 CFR 52.47: Contents of applications, paragraph (a)(1) (September 2003)
  - ✓ 10 CFR 50.34: Contents of applications; technical information, paragraph (f); "Additional TMI-related requirements"
  - ✓ 10 CFR 50.44: Combustible gas control for nuclear power reactors, paragraph (c); "Requirements for future water-cooled reactor applicants and licensees"

3. MHI Interpretation of NRC Requirements

➢ MHI's interpretation of NRC's severe accident policy statement and regulations:

✓ MHI will demonstrate that US-APWR design will mitigate the consequences of severe accidents through

(1) Demonstration of compliance with current Commission regulations including TMI requirements in 10 CFR 50.34(f)

(2) Demonstration of technical resolution of the applicable unresolved safety issues (USI), and the medium and high-priority generic safety issues (GSI) discussed in NUREG-0933

(3) Development of an appropriate PRA

(4) Submission of DC application for staff review

4. Objectives of SA Analysis

- **Demonstrate compliance with US regulatory requirements**
  - Quantitatively address severe accident challenges to containment integrity
  - Confirm the effectiveness of severe accident mitigation features
  - Establish firm basis for US licensing
- **Obtain adequate input to perform Level 2 PRA**
  - Consider analysis results to determine the branch probabilities of Containment Event Trees (CET) and Decomposition Event Trees (DET)
  - Evaluate the potential for severe accidents

5. General Approach for US-APWR SA Analysis

1. Apply analysis approaches accepted by NRC for former DC applications

- US-APWR's fundamental design concept is very similar to the existing PWR plants
- US-APWR does not introduce new phenomena or configurations, so that current severe accident experimental database is applicable
- Previously employed analytical approaches are applied to US-APWR modelling for severe accident evaluation

5. General Approach for US-APWR SA Analysis

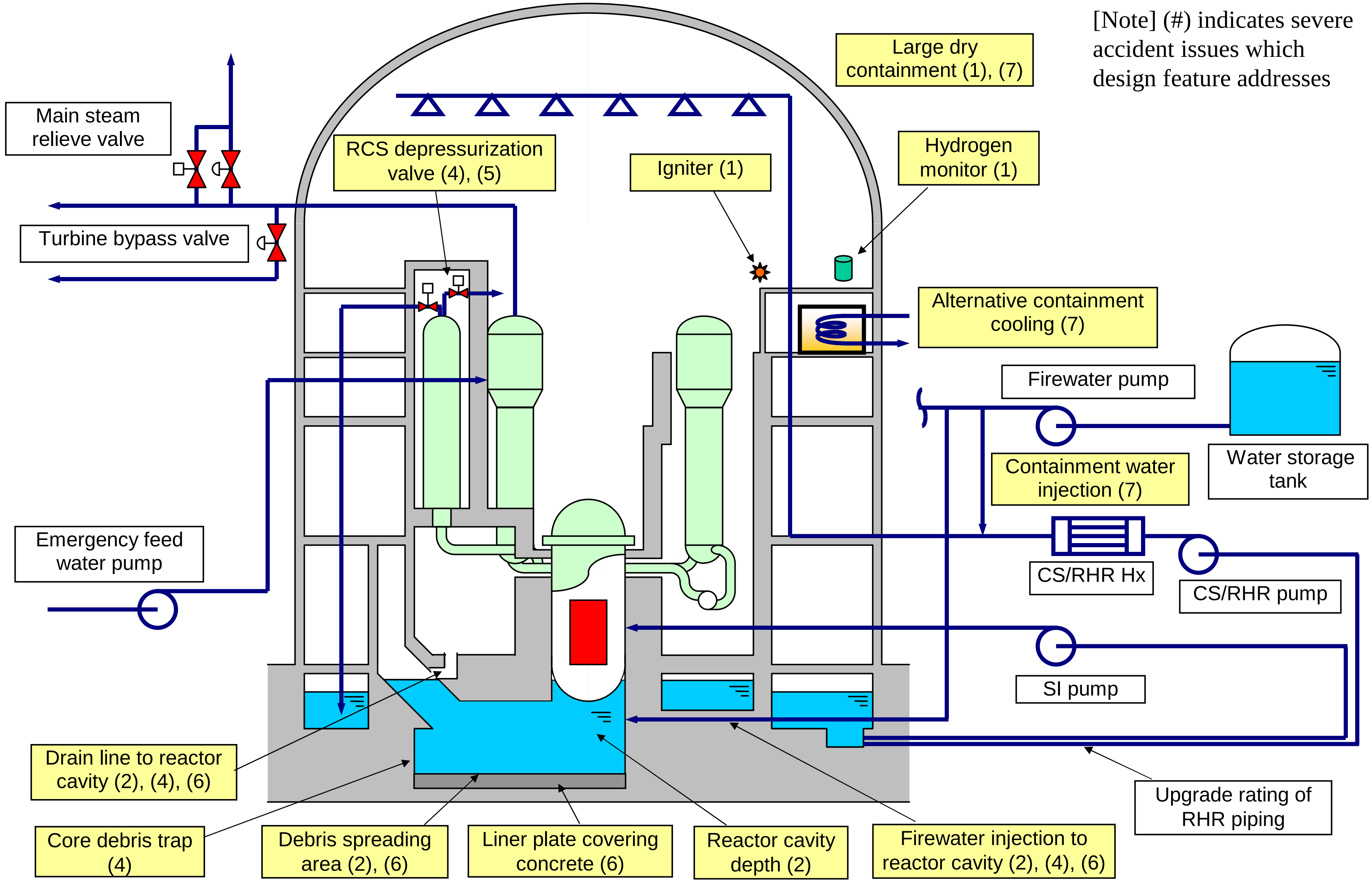
2. **Address inherent uncertainties of severe accident phenomena**
   - Use best-estimate methodologies appropriate for the design stage
   - Perform sensitivity analyses to adequately address the phenomenological uncertainties
   - Review the applicability of understandings obtained from previously performed studies to the US-APWR design
   - Examine analytical outputs to ensure that they are within the expectation of the consequences of existing studies
3. **Employ MAAP 4.0.6 for severe accident progression analysis**
4. **Employ separate effects codes for specific phenomena**

6. SA Analysis Methodology

- **Eight (8) severe accident issues identified for US-APWR**
  - (1) Hydrogen Mixing and Combustion
  - (2) Core Debris Coolability
  - (3) Steam Explosion (In- and Ex-vessel)
  - (4) High Pressure Melt Ejection and Direct Containment Heating
  - (5) Temperature Induced Steam Generator Tube Rupture
  - (6) Molten Core Concrete Interaction
  - (7) Long-term Containment Overpressure
  - (8) Equipment Survivability

Design Features for SA Mitigation

[Note] (#) indicates severe accident issues which design feature addresses



6.1 Hydrogen Mixing and Combustion

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➢ Goals of analysis

1. Demonstrate that containment has capability for ensuring a mixed atmosphere (10 CFR 50.44(c)(1))
2. Demonstrate that uniformly distributed hydrogen concentration is less than 10% by volume when igniters are functional (10 CFR 50.34(f)(2)(ix) and 10 CFR 50.44(c)(2))
3. Demonstrate that containment integrity is maintained when igniters are functional, assuming hydrogen generated from 100% fuel cladding-coolant reaction (10 CFR 50.34(f)(3)(v)(A)(1) and 10 CFR 50.44(c)(5))
4. Demonstrate that containment integrity is maintained against pressure rise assuming Adiabatic Isochoric Complete Combustion (AICC) of hydrogen (10 CFR 50.44(c)(5))

➢ Mitigation features

- Large dry containment
  - ✓ Provide hydrogen mixing and protection against hydrogen burns
- Igniter
  - ✓ Control hydrogen with high reliability

6.1 Hydrogen Mixing and Combustion

US-APWR

➢ Summary of relevant studies and experiments

Paper / Experiment	Findings
NUPEC large scale test (1992) (NUPEC:Nuclear Power Engineering Corporation)	Experiment which modelled Japanese PWR dry containment showed that hydrogen released from SG compartment and annular compartment was well mixed and no local high concentration was observed.
NUPEC large scale hydrogen burn test (1996)	NUPEC reported that no global burn was observed when hydrogen concentration was below 8%. 100% burning efficiency was observed for concentration 10%~15% however pressure rise was less than that calculated by AICC. No Deflagration to Detonation Transition (DDT) was observed for concentration less than 15%.
NUPEC detonation and containment integrity test (2000)	Postulated hydrogen detonation under 13% hydrogen concentration caused approximately 0.6% of maximum plastic strain for PCCV liner plate, which is much lower than fracture strain of 19%. Potential of containment failure due to detonation was confirmed to be very small.
NUREG/CR-4905 (1987)	SNL reported that hydrogen detonation was observed for hydrogen concentration 13.5%~70%.
NUREG/CR-6524 (1999)	BNL reported that DDT occurred at lower hydrogen concentration for higher temperature. However, hydrogen concentration for DDT became higher when either steam or sideward opening existed.

6.1 Hydrogen Mixing and Combustion

US-APWR

- Analysis approach
  1. Evaluate effectiveness of igniters and local concentration of hydrogen
     - ✓ Employ MAAP to evaluate the hydrogen release flow rate to containment atmosphere
     - ✓ Calculate independently the total amount of hydrogen generated from 100% zirconium of active fuel length cladding-coolant reaction
     - ✓ Modify the MAAP hydrogen release flow rate with independently calculated amount of hydrogen generation, and apply as boundary conditions for GOTHIC calculations
     - ✓ Employ GOTHIC with igniter model to evaluate effectiveness of igniters and atmospheric mixing through multi-nodes and sub-divided volumes
     - ✓ Show that local hydrogen concentration during severe accident is less than 10%

6.1 Hydrogen Mixing and Combustion

US-APWR

- **Analysis approach**

2. Evaluate containment structural capability against local hydrogen burn
   - ✓ Investigate structural capability to withstand pressure rise due to hydrogen control by igniters
   - ✓ Evaluate in accordance with the approach specified by ASME standard
   - ✓ Criterion of containment structural capability is based on ultimate capability, not on design capability
3. Evaluate containment structural capability against global hydrogen burn
   - ✓ Evaluate the containment pressure rise assuming Adiabatic Isochoric Complete Combustion (AICC) of hydrogen
   - ✓ Examine containment structural integrity against pressure rise

6.2 Core Debris Coolability

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- ➢ Goals of analysis
  1. Demonstrate that core debris is appropriately cooled when reactor cavity is adequately flooded

     (this goal was established by MHI to ensure termination of severe accident progression for US-APWR)
- ➢ Mitigation features
  - Diverse reactor cavity flooding system
    - ✓ Consists of drain line pathway and firewater injection, to ensure flooding of reactor cavity to meet MHI design goals
  - Reactor cavity geometry
    - ✓ Sufficient reactor cavity floor area and appropriate reactor cavity depth to enhance spreading debris bed for better coolability

6.2 Core Debris Coolability

US-APWR

➢ Summary of relevant studies and experiments

Paper / Experiment	Findings
GL 88-020	NRC staff recommends that assessments be based on available cavity area and an assumed maximum coolable depth of 25 cm.
SWISS (1987)	Debris cooling failed due to formation of stable crust and water pool above melt was kept below boiling point.
MACE (1991&1992)	Debris cooling failed due to formation of stable crust and concrete erosion was not suppressed. Debris coolability cannot be concluded based on this series of experiment programs as observed phenomena are not prototypic to actual plant geometry.
WETCOR (1993)	Influence of sidewall was eliminated by heating. Debris cooling failed due to formation of stable crust. Neither fragmentation of melt nor indication of instability of crust was observed.
COTELS (1999)	Debris was cooled by coolant water and concrete erosion was suppressed. This was caused by water penetration to the porous of debris bed via eroded concrete sidewall clearance.
OECD MCCI (2005)	Debris was cooled by coolant water and concrete erosion was suppressed. Water was able to penetrate the interface between the corium and concrete sidewalls. This cooling mechanism was not observed in MACE M1b, because of inert refractory (MgO) sidewall.

6.2 Core Debris Coolability

- Analysis approach
  1. Confirm adequacy of MAAP analysis model
     - ✓ MAAP assumes crust cracking phenomena and direct water contact with melt
     - ✓ COTELS and OECD MCCI experiments support the MAAP phenomenological assumption of water ingression to melt
  2. Perform severe accident progression analysis
     - ✓ Employ MAAP to investigate core debris coolability
     - ✓ Consider characteristic scenarios for debris cooling
       - Debris drops into water pool
       - Water is injected onto molten debris on cavity floor
  3. Consider inherent phenomenological uncertainties
     - ✓ Examine the effectiveness of debris coolability by heat transfer between debris bed and overlying water pool
     - ✓ Perform sensitivity analysis using MAAP for parameters related to the core debris coolability, such as
       - Water ingression into debris bed

6.3(1) In-vessel Steam Explosion

- Goals of analysis
  - Confirm that in-vessel steam explosion is very unlikely to happen
  - Confirm that existing study conclusions are applicable to US-APWR

    (these goals were established by MHI to adequately address severe accidents for US-APWR)
- Mitigation features
  - None

6.3(1) In-vessel Steam Explosion

US-APWR

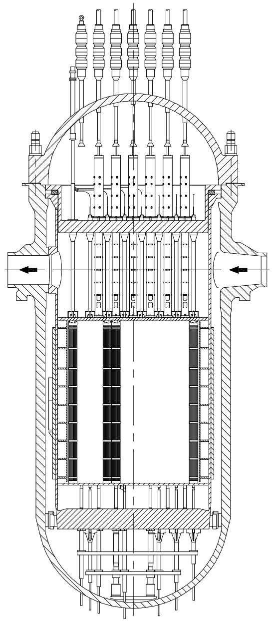
- Summary of relevant studies and experiments

Paper / Experiment	Findings
NUREG-1116 (1985) NUREG-1524 (1996)	It was concluded that the potential for alpha-mode containment failure is negligible and the issue of this failure mode has been resolved from risk point of view.
OECD/CSNI (1997)	No new facts had been identified to question the conclusion of NUREG-1524. It was concluded that alpha-mode failure has no importance with regard to risk.
ALPHA (1995)	ALPHA experiment by JAERI observed no steam explosions for pressure more than 1.0MPa and for saturated water. JAERI concluded that the potential for alpha-mode containment failure may be negligible for medium- and high-pressure accident scenarios.
COTELS (1999) FARO (1997) KROTOS (1997)	No steam explosion was observed when mixture of molten UO_2 and Zr is dropped into water for the experiments of COTELS by NUPEC, and FARO and KROTOS by JRC-Ispra

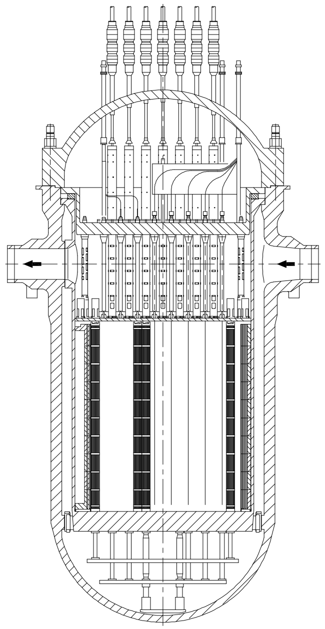
6.3(1) In-vessel Steam Explosion

➢ Analysis approach

1. Examine existing studies
   - ✓ Investigate the likelihood of in-vessel steam explosion in general through existing studies
   - ✓ Examine the applicability of existing studies to US-APWR



US Current 4 Loop Plant



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- Occurrence potential of steam explosion depends on such as system pressure and water temperature in lower plenum
- Challenge to containment depends on RPV upper head missile energy
- No significant differences are identified between US-APWR and existing plants

6.3(2) Ex-vessel Steam Explosion

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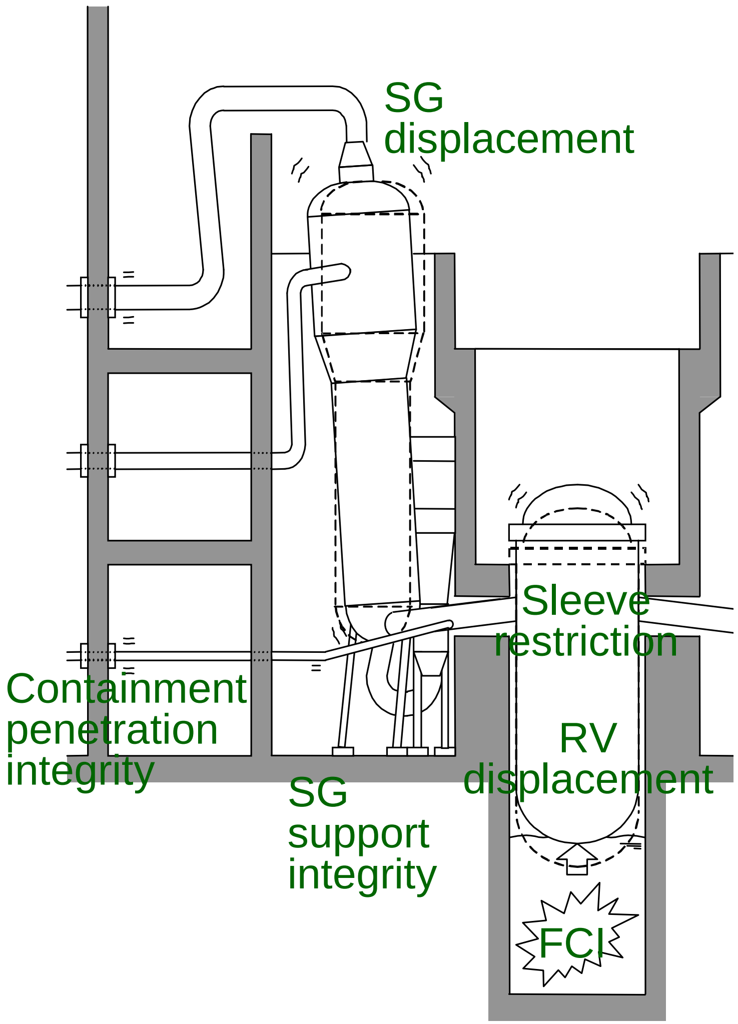
➢ Goals of analysis

- Evaluate the pressure load when ex-vessel steam explosion occurs
- Demonstrate the containment structure has sufficient capability to withstand the pressure load of ex-vessel steam explosion and induced events by the load

  (these goals were established by MHI to adequately address severe accidents for US-APWR)

➢ Mitigation features

- None



6.3(2) Ex-vessel Steam Explosion

US-APWR

- Summary of relevant studies and experiments

Paper / Experiment	Findings
ALPHA, COTELS, etc.	It is commonly understood that steam explosion is unlikely to happen for saturated water. No steam explosion has been observed when water was poured onto molten debris. Potential of steam explosion includes large uncertainty since the occurrence of steam explosion triggering shows statistical behaviour.
ALPHA (1992) NUREG/CR-5372 (1998)	It is considered very limited fraction of corium contributes to steam explosion when large amount of corium drops into water all at once. Fraction of energy conversion from corium to mechanical load is considered to be a few %, or less.
COTELS (1999) FARO (1997) KROTOS (1997)	No steam explosion was observed when mixture of molten UO_2 and Zr is dropped into water for the experiments of COTELS by NUPEC, and FARO and KROTOS by JRC-Ispra

6.3(2) Ex-vessel Steam Explosion

- Analysis approach
  1. Evaluate pressure load
     - ✓ Employ TEXAS-V for pressure load prediction
     - ✓ Utilize MAAP calculation results to set initial conditions for TEXAS-V analysis
     - ✓ Perform sensitivity analyses to address inherent uncertainties
  2. Evaluate containment structural capability
     - ✓ Employ LS-DYNA to evaluate structural capability of reactor cavity to withstand pressure load from steam explosion
     - ✓ Scope of this structural analysis includes
       - Reactor cavity wall
       - Reactor coolant pipes and nozzles
       - Reactor cavity sleeve structure
       - Extent of steam generator displacement
       - Containment penetration integrity

6.4 High Pressure Melt Ejection and Direct Containment Heating

US-APWR

Goals of analysis

- Demonstrate that the capacity of RCS depressurization valve is adequate and accordingly potential of HPME is sufficiently low
- Investigate the ability of debris trap to ensure that very limited amount of core debris is dispersed to containment atmosphere, and accordingly show that challenge by DCH is acceptably low
- Demonstrate that containment structure has sufficient capability to withstand the pressure rise due to DCH

  (these goals were established by MHI to adequately address severe accidents for US-APWR)

Mitigation features

- RCS depressurization valve
  - ✓ Reduce RCS pressure
- Core debris trap
  - ✓ Enhance capturing of ejected molten core in reactor cavity
- Diverse reactor cavity flooding system
  - ✓ Provide reliable flooding of reactor cavity

6.4 High Pressure Melt Ejection and Direct Containment Heating

➢ Summary of relevant studies and experiments

Paper / Experiment	Findings
B.W.Spencer, et al. (1988)	Experiment performed by ANL showed that containment atmosphere temperature rise is very small when reactor cavity was filled with water.
NUREG/CR-6510 (1999) EPRI NP-5127 (1987)	Dispersed debris was captured at traps during flowing within tunnel area and opening of stairs, etc. Influence of DCH was reduced due to this debris capture.
NUREG/CR-6152 (1994)	Scaling experiment by SNL showed that the pressure rise during DCH was as much as 0.5MPa.
NUREG/CR-6075 (1994) NUREG/CR-6109 (1995) NUREG/CR-6338 (1996)	It was concluded from this series of studies that the challenge by DCH have already been resolved for Westinghouse large dry containment. CCFP by DCH for all Westinghouse large dry containments were calculated less than 0.01. It was concluded that DCH issue has been resolved for these plants and no additional studies are required.

6.4 High Pressure Melt Ejection and Direct Containment Heating

➢ Analysis approach

1. Perform severe accident progression analysis for scenarios related to RCS depressurization
   - ✓ Employ MAAP to evaluate the capacity of RCS depressurization valve to prevent HPME
2. Evaluate amount of core debris dispersion
   - ✓ Investigate amount of core debris dispersion anticipated in general through existing studies
   - ✓ Examine the applicability of existing studies to US-APWR
3. Investigate containment structural capability
   - ✓ Assume conservatively the amount of core debris dispersion
   - ✓ Employ two-cell equilibrium model to evaluate pressure rise due to DCH
   - ✓ Examine whether containment structure has sufficient capability to withstand the pressure rise due to DCH

6.5 Temperature Induced Steam Generator Tube Rupture

- Goals of analysis
  - Demonstrate that the capacity of RCS depressurization valve is sufficient to ensure that the potential of TI-SGTR is acceptably low

    (this goal was established by MHI to adequately address severe accidents for US-APWR)
- Mitigation features
  - RCS depressurization valve
    - ✓ Reduce RCS pressure

6.5 Temperature Induced Steam Generator Tube Rupture

US-APWR

➢ Summary of relevant studies and experiments

Paper / Experiment	Findings
NUREG-1150 (1990)	It is considered that TI-SGTR is very unlikely failure mode for high pressure core melt scenarios as long as tubes have no defect
NUREG-1570 (1998)	Analysis result using SCDAP/RELAP5 have shown that surge line break is the most likely failure mode. It has been pointed that TI-SGTR is likely in case of RCP seal LOCA sequences. Although RCP seal leak depressurize RCS, the associated RCS loop seal clearing greatly contributes to the tube failure potential. Secondary system pressure integrity is as important as RCS depressurization.
JAERI-Research 99-067	JAERI performed research focusing on secondary system depressurization during SBO and identified that SG tube integrity was narrowly maintained for the condition. It was however concluded that potential of TI-SGTR could not be ignored taking account of inherent uncertainty of computational calculation.
JNES research (2006) (JNES: Japan Nuclear Energy Safety Organization)	Research focusing on potential of TI-SGTR under condition of secondary system depressurized at core damage, which reported that the mean of probability density for TI-SGTR is 0.50, and that for surge line break is 0.37.

6.5 Temperature Induced Steam Generator Tube Rupture

- Analysis approach
  1. Perform severe accident progression analysis for scenarios related to RCS depressurization
     - ✓ Employ MAAP to analyze RCS high pressure scenarios
     - ✓ Evaluate the capacity of RCS depressurization valve to prevent TI-SGTR
  2. Examine existing studies
     - ✓ Examine the applicability of existing studies to US-APWR

6.6 Molten Core Concrete Interaction

Goals of analysis

- Demonstrate that containment integrity is maintained during pressure rise due to MCCI beyond 24 hours of the onset of core damage
- Demonstrate that basemat melt through does not occur within 24 hours following the onset of core damage

  (these goals were established by MHI to adequately address severe accidents for US-APWR)

Mitigation features

- Overlying concrete on reactor cavity liner plate
  - ✓ Provide protection against challenge to liner plate melt through
- Basemat concrete
  - ✓ Provide protection against fission products release
- Same features as for Core Debris Coolability

6.6 Molten Core Concrete Interaction

➢ Summary of relevant studies and experiments

Paper / Experiment	Findings
BETA (1987)	Experiments performed at Kernforschungszentrum Karlsruhe (KZK). Downward erosion was greater than sideward for high-power experiments. This tendency was more significant for silicate concrete than limestone.
ACE (1988)	Experiments performed at ANL. Melt was thoroughly mixed by gases released from the decomposing concrete and no stratification of oxidized and metallic melts was observed.
TURC NUREG/CR-4521 (1986)	Experiments performed at SNL. Transient heat conduction into concrete was observed in this experiment, resulting in decomposition of concrete. H_2O and CO_2 were reduced to CO and H_2 during decomposition, respectively.
SURC NUREG/CR-4994 (1989) NUREG/CR-5443 (1992) NUREG/CR-5564 (1992)	Experiments performed at SNL to provide information on heat transfer mechanism, gas release chemistry and vaporization release of aerosols. Interaction temperature remained well above the concrete melting point and zirconium chemistry drastically affect the ablation rate and gas composition.

6.6 Molten Core Concrete Interaction

US-APWR

- Analysis approach
  1. Perform severe accident progression analysis
     - ✓ Employ MAAP to investigate the extent of MCCI for characteristic scenarios
       - No water available in reactor cavity
  2. Examine containment structural capability
     - ✓ Investigate whether containment integrity is maintained more than 24 hours following the onset of core damage against
       - Pressure rise by steam and non-condensable gas generation due to MCCI
       - Basemat melt through
  3. Evaluate material properties
     - ✓ Investigate the characteristic difference between basalt and limestone/common sand in terms of MCCI, such as
       - Erosion rate
       - Amount of steam and non-condensable gas generation

6.7 Long-term Containment Overpressure

- **Goals of analysis**
  - Demonstrate the effectiveness of diverse mitigation features against containment overpressure
  - Demonstrate that containment withstands pressurization for more than 24 hours following the onset of core damage

    (these goals were established by MHI to adequately address severe accidents for US-APWR)
- **Mitigation features**
  - Large dry containment
    - ✓ Provide sufficient capability to withstand overpressure
  - Containment spray
    - ✓ Provide primary function to mitigate containment overpressure
  - Alternative containment cooling by recirculation unit
    - ✓ Enhance condensation of surrounding vapor by natural convection
  - Firewater injection to spray header
    - ✓ Delay containment failure (no cooling function)

6.7 Long-term Containment Overpressure

US-APWR

➢ Summary of relevant studies and experiments

Paper / Experiment	Findings
NUREG/CR-6906 (2006)	Containment may generally have pressure capability of a few times design pressure. Global, free-field strains on the order of a few % can be achieved before failure or rupture. However, actual containment may have more complexity than models, thus the capacities of models can be interpreted as an upper bound on actual containment capacity.
NUREG/CR-4119 (1985)	Study on the integrity of containment penetrations under severe accident condition has been summarized. Database to predict leak rate of containment penetrations under severe accident conditions have been established.
NUREG/CR-4149 (1985)	Modelling techniques and analysis procedures to determine ultimate pressure capacity of reinforced and pre-stressed concrete containments have been presented.
NUREG/CR-6809 NUREG/CR-6810 (2003)	Overpressurization test to failure for 1:4 scaled PCCV and the test analysis. Various data were collected, and CV response and failure modes were observed. Post-test analysis predicts liner's strain near weld seams and test itself shows the need for continuous backup bars on all liner seam welds.

6.7 Long-term Containment Overpressure

- Analysis approach
  1. Perform severe accident progression analysis
     - ✓ Employ MAAP to evaluate the effectiveness of mitigation features
       - Containment spray
       - Alternative containment cooling by recirculation unit
       - Firewater injection to spray header
  2. Examine containment structural capability
     - ✓ Investigate whether containment integrity is maintained for more than 24 hours following the onset of core damage

6.8 Equipment Survivability

- Goals of analysis
  - Demonstrate the equipment survivability of systems and components to maintain safe shutdown and containment structural integrity under the environmental conditions created by hydrogen burning (10 CFR 50.44(c)(3))
  - Determine the design specifications to satisfy requirements for equipment survivability

6.8 Equipment Survivability

- Summary of relevant studies and experiments

Paper / Experiment	Findings
NUREG/CR-4763 (1988)	Experiments performed by SNL for pressure transmitter and cables, under conditions of single-burn and multiple-burn have been reported. Equipment survivability for single-burn was confirmed but not for multiple-burn.
NUREG/CR-5334 (1989)	Experimental results on response of 3 types of wire penetrations have been reported. For Westinghouse containment, it was exposed to 400F for 10 days. Electrical capability was maintained for 4 days, and mechanical capability was maintained for 10 days.
EPRI NP-4354 (1985)	Experiments on response of typical safety equipment under hydrogen burn condition were performed. Most of equipment operated normally during and after all tests. It is concluded that the test data may be useful in assessing the survivability of safety equipment.

6.8 Equipment Survivability

- Analysis approach
  1. Determine the scope of analysis
     - ✓ Identify systems and components to be examined during DC stage
     - ✓ Identify time frames necessary to consider in accordance with accident progression
     - ✓ Complete analysis will be given as part of COL
  2. Perform severe accident progression analysis
     - ✓ Employ MAAP to analyze various accident scenarios
     - ✓ Employ GOTHIC to analyze environmental conditions especially for hydrogen combustion
  3. Examine equipment survivability for DC stage
     - ✓ Investigate availability of systems and components under calculated environmental conditions

7. Conclusions

- Severe accident mitigation features for US-APWR were presented at the March 1, 2007 meeting with NRC
- Severe accident analysis methodology for US-APWR has been presented in this meeting
- Strategy and approach for complying with NRC requirements has been presented
- Severe accident analysis results will be reported in DCD