US-APWR

5th Pre-Application Review Meeting
The Methods of Dose Evaluation

March 1, 2007
Mitsubishi Heavy Industries, Ltd.

MITSUBISHI HEAVY INDUSTRIES, LTD. UAP-HF-07014

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.

Yoshinori Takechi (Presenter of Dose Evaluation during normal operation)
Engineering Manager
Radiation Safety Engineering Section
Nuclear Energy Systems Engineering Center
Mitsubishi Heavy Industries, LTD.

Hiromasa Nishino (Presenter of Radiation Shielding Protection)
Engineering Manager
Radiation Safety Engineering Section
Nuclear Energy Systems Engineering Center
Mitsubishi Heavy Industries, LTD.

Terence J. Heames (Presenter of Dose Evaluation under accident conditions)
Technical Consultant

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Objectives of Meeting

- Describe methods and acceptance criteria to be used for US-APWR dose evaluation for normal operation and accident conditions, which are approved by the NRC or used in the U.S.A.
- Describe radiation shielding methodology
- Identify the current acceptance criteria and methodology for demonstrating compliance in this area
- Distinguish between information to be provided in the Design Certification (DC) and the Combined License (COL)
- Obtain the NRC feedback on the methods

Contents

1. General Approach for Dose Evaluation

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1. 1 Scope and Plan

- Our methods of dose evaluation are based on the US requirements and most recent guidance
- We plan to use calculation codes approved by the NRC and used in the USA, therefore no topical report is planned
- We will perform dose evaluations for the DC application using conservative site parameters
- Mitsubishi plans to follow the most up to date guidance provided or referenced in NUREG-0800 and DG-1145

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1.2 Calculation Codes

➤Calculation codes used for US-APWR

	Computer Code	Notes
Dose Evaluation during Normal Operation	ORIGEN2, PWR-GALE, NRCDose	Calculation codes approved by the NRC (XOQDOQ for COL)
Dose Evaluation under Accident Conditions	ORIGEN2, RADTRAD, MicroShield	Calculation codes approved by the NRC (PAVAN and ARCON96 for COL)
Radiation Shielding Protection	ORIGEN2, MicroShield, DORT, ANISN, MCNP, G^3	Calculation codes used in the U.S.A.

2. Dose Evaluation during Normal Operation

2. Dose Evaluation during Normal Operation (1)

US-APWR

➢ **US requirements and most recent guidance**

10CFR20	Standards for protection against radiation
10CFR50	Domestic licensing of production and utilization facilities (Especially 50.34a & Appendix I)
40CFR190	Environmental radiation protection standards for nuclear power operations

R.G.1.109 Rev. 1 (1977)	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I
R.G.1.111 Rev. 1 (1977)	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors
R.G.1.112 Rev. 2 (1997)	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors
R.G.1.113 Rev. 1 (1977)	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I

2. Dose Evaluation during Normal Operation (2)

US-APWR

➢ **Calculation codes**

✓ PWR-GALE (published by the NRC as NUREG-0017)

✓ NRCDose

➢ **Method to Determine χ/Q Values**

✓ Determine atmospheric dispersion factors (χ/Q values) that will bound the atmospheric conditions at most US sites in DC application

✓ In COL application, validate DC conservatism compared to the site specific χ/Q values

2. Dose Evaluation during Normal Operation (3)

US-APWR

➢ Calculation methods of primary and secondary coolant activities

✓Standard concentrations based on ANSI/ANS-18.1 (1999)

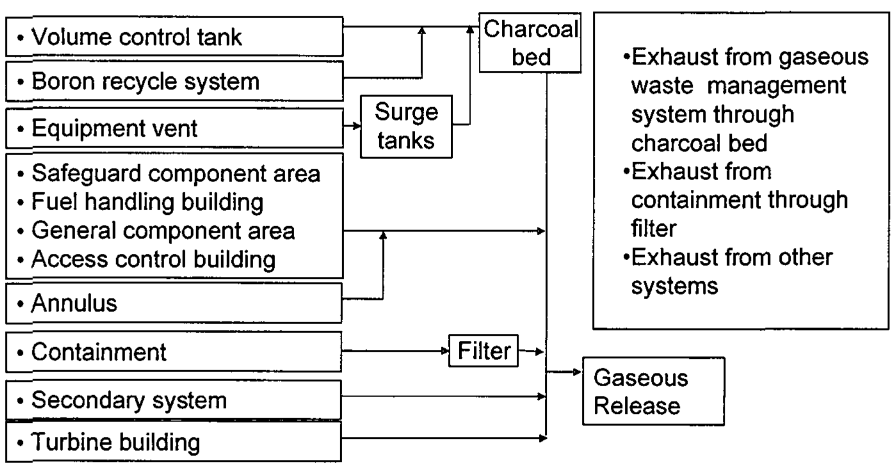
Specific parameters for US-APWR
- Thermal power
- Steam flow rate
- Weight of water in reactor coolant system
- Weight of secondary side water in all SGs
- Reactor coolant let down flow rate
- SG blowdown flow rate, etc.

✓Realistic source term

2. Dose Evaluation during Normal Operation (4)

US-APWR

➢ Pathways for radioactive gaseous release

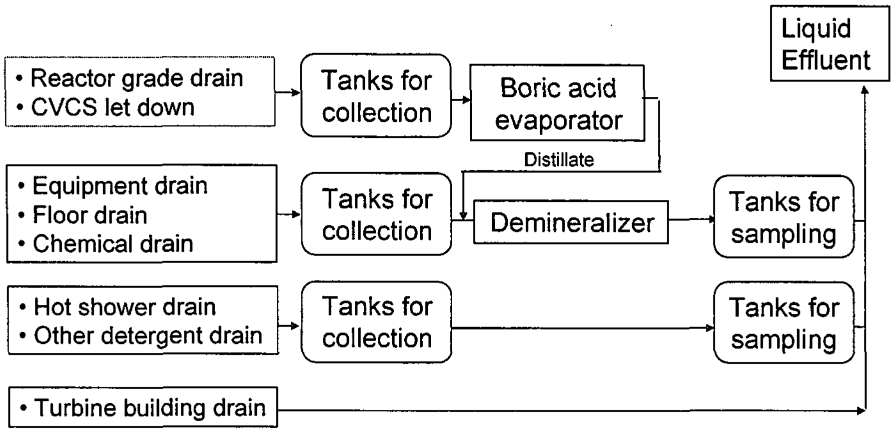


Removal and radioactive decay effect within the system will be considered

2. Dose Evaluation during Normal Operation (5)

US-APWR

➢ Pathways for radioactive liquid effluent



Removal and radioactive decay effect within the system will be considered

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2. Dose Evaluation during Normal Operation (6)

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➢ Summary

✓ Our methods of dose evaluation during normal operation are based on the US requirements and most recent guidance

✓ We plan to use calculation codes already approved by the NRC

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3. Dose Evaluation under Accident Conditions

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3.1 Accidents Evaluated in Design Certification(1)

- The following accidents are quoted from Standard Review Plan
  - ✓ Loss of coolant accident (LOCA)
  - ✓ Steam generator tube rupture (SGTR)
  - ✓ Main steam line break (MSLB)
  - ✓ Fuel handling accident (FHA)

  - ✓ Rod ejection accident (REA)
  - ✓ Locked rotor accident (LRA)
  - ✓ The failure of small lines carrying primary coolant outside containment
  - ✓ Spent fuel cask drop accident

  - ✓ Liquid containing tank failure (evaluated in COL)

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3.1 Accidents Evaluated in Design Certification(2)

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- Evaluate offsite and onsite doses in DC
- Explain about the methods of dose evaluation

	EAB and LPZ Dose	CR Dose
LOCA	O	O
SGTR	O	-
MSLB	O	-
FHA	O	-

EAB = Exclusion Area Boundary
LPZ = Low Population Zone
CR = Control Room

3.2 Requirements Considered

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- The U.S. requirements and most recent guidance for dose evaluation under accident conditions

10CFR100.11	Determination of exclusion area, low population zone, and population center distance
10CFR50.67	Accident source term
10CFR50 App.A	General Design Criteria for Nuclear Power Plants

R.G.1.183	Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors
R.G.1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants
R.G.1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants

NUREG-0737	Clarification of TMI Action Plan Requirements

3.3 Dose Criteria

➢Dose criteria in R. G. 1.183 based on 10 CFR 50.67

Accident		*1EAB and LPZ Dose Criteria	CR Dose Criteria	Analysis Release Duration
LOCA		25 rem TEDE*2	5 rem TEDE	30 days for containment and ECCS leakage
SGTR	Fuel Damage or Pre-incident Iodine Spike	25 rem TEDE	5 rem TEDE	Failed SG: time to isolate; Intact SG(s): until cold shutdown is established
	Coincident Iodine Spike	2.5 rem TEDE	5 rem TEDE	
MSLB	Fuel Damage or Pre-incident Iodine Spike	25 rem TEDE	5 rem TEDE	Until cold shutdown is established
	Coincident Iodine Spike	2.5 rem TEDE	5 rem TEDE	
FHA		6.3 rem TEDE	5 rem TEDE	2 hours

*1EAB; Exclusion Area Boundary, LPZ; Low Population Zone, *2TEDE; Total Effective Dose Equivalent

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3.4 Calculation codes

➢ **Calculation codes used for US-APWR**
- Core Inventory ; ORIGEN2
- Dose ; RADTRAD, MicroShield

➢ **Method to determine χ/Q values in DC**

χ/Q values at the site boundary	Bounding values covering 70 to 80 % of the US sites
χ/Q values at the control room	Calculated to consider locations between release points and receptors

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3.5 Loss-of-coolant Accident (1)

➢ **Assumptions of postulated accidents**

- ✓ Postulated events
  - Large Break LOCA
- ✓ Single failure
  - The worst single active failure
- ✓ Safety features available
  - Safety injection system
  - Containment vessel spray system
  - Annulus air cleanup system
  - Air cleanup system for safeguard component area
- ✓ Other decontamination features
  - Natural deposition in unsprayed zones

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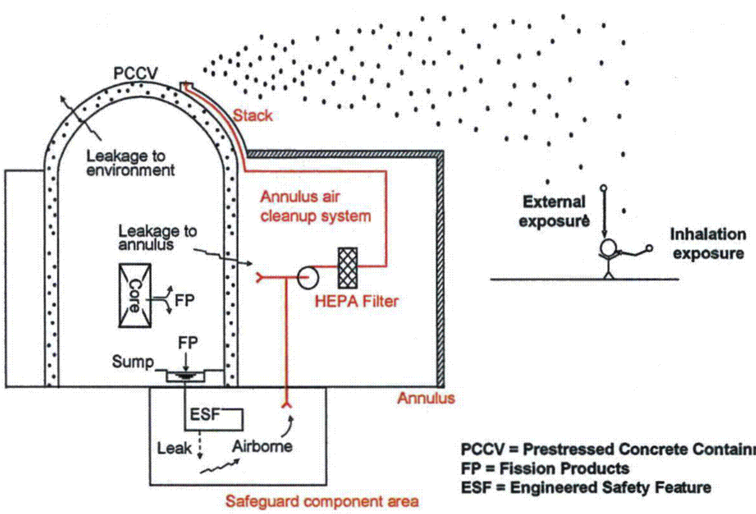
3.5 Loss-of-coolant Accident (2)

➢ **FP pathways (exposure to the public)**



PCCV = Prestressed Concrete Containment
FP = Fission Products
ESF = Engineered Safety Feature

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3.5 Loss-of-coolant Accident (3)

➢ **Source term based on R. G. 1.183**

✓ **Core inventory fraction released into containment**

	Gap Release	Early In Vessel	Total
Noble Gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali	0.05	0.25	0.3
Tellurium Metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble Metals	0.00	0.0025	0.0025
Cerium Group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

✓ **Release phases**

Phase	**Onset**	**Duration**
Gap Release	30 sec	0.5 hr
Early In-Vessel	0.5 hr	1.3 hr

3.5 Loss-of-coolant Accident (4)

➢ **Major conditions for dose evaluation**

Parameters	**Value**
• Source term, release fractions and timing	Source term based on R.G.1.183
• Removal efficiency of the containment spray	Based on SRP 6.5.2
• Containment leakage rate	0 – 24h ; 0.15%/day 24h – 30d ; 0.075%/day
• Removal efficiency of the HEPA filter	Design value
• ESF leakage rate	4 times design leakage rate
• Sump water pH	Larger than 7

3.6 Steam Generator Tube Rupture (1)

➢ Assumptions of postulated accidents

✓ Postulated events

A guillotine break of a single tube in the steam generator.

✓ Single failure

The worst single active failure

✓ Safety features available

Safety injection system

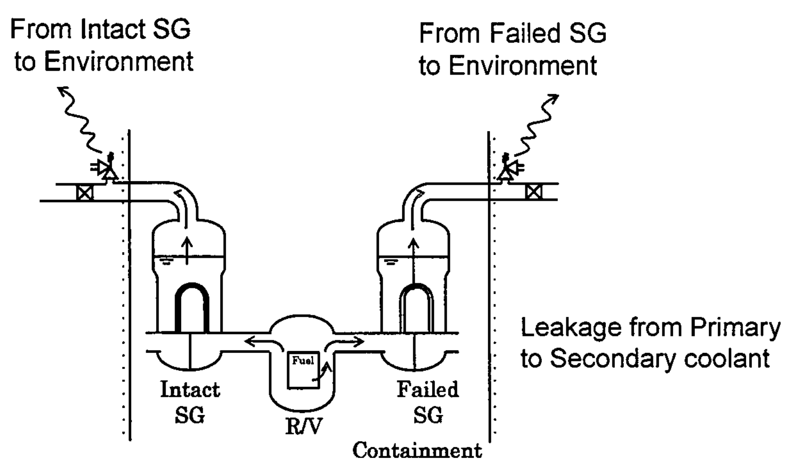
Main steam relief valves

Main steam safety valves

Main steam isolation valves

3.6 Steam Generator Tube Rupture (2)

➢ FP release pathways



3.6 Steam Generator Tube Rupture (3)

➢ Radiological concentration of primary coolant and secondary coolant

Parameters	Value
• Radiological concentration of the primary coolant	Based on the technical specification (1 μ Ci/cc Dose Equivalent I-131)
• Iodine concentration of the primary coolant and iodine spike	
- Pre-incident iodine spike	The maximum value permitted by the technical specification (60 μ Ci/cc Dose Equivalent I-131)
- Coincident iodine spike	Iodine increases to 335 times the equilibrium release rate
• Concentration in secondary coolant	Based on the technical specification (0.1 μ Ci/cc Dose Equivalent I-131)

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3.6 Steam Generator Tube Rupture (4)

➢ Major conditions for dose evaluation

Parameters	Value
• Primary to Secondary leak rate	
- Failed SG:	Based on accident analysis
- Intact SGs:	Based on the technical specification
• Partition Coefficient	
- Iodine	100
- Noble gas	All noble gases released from the primary system are released
• Release Duration	
- Failed SG:	Based on accident analysis
- Intact SGs:	Until shutdown cooling

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3.7 Main Steam Line Break Accident (1)

➢ Assumptions of postulated accidents

✓ Postulated events

A guillotine break of a single main steam pipe

✓ Single failure

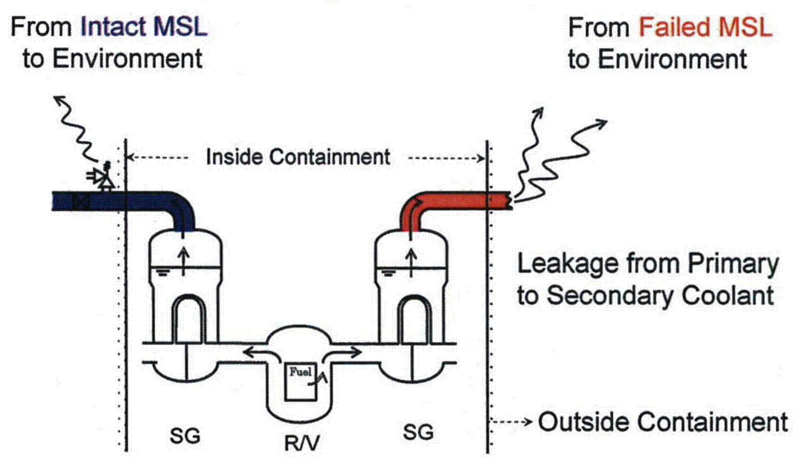
The worst single active failure

✓ Safety features available

Safety injection system
Main steam relief valves
Main steam safety valves
Main steam isolation valves

3.7 Main Steam Line Break Accident (2)

➢ FP release pathways



3.7 Main Steam Line Break Accident (3)

➤ Radiological concentration of primary coolant and secondary system

Parameters	Value
• Radiological concentration in the primary coolant	Based on the technical specification (1 μ Ci/cc Dose Equivalent I-131)
• Iodine concentration in the primary coolant and iodine spike	
- Pre-incident iodine spike	The maximum value permitted by The technical specification (60 μ Ci/cc Dose Equivalent I-131)
- Coincident iodine spike	Iodine increases to 500 times the equilibrium release rate
• Concentration in secondary coolant	Based on the technical specification (0.1 μ Ci/cc Dose Equivalent I-131)

3.7 Main Steam Line Break Accident (4)

➤ Major conditions for dose evaluation

Parameters	Value
• Primary to Secondary leak rate	
- Failed MSL:	Based on technical specifications
- Intact MSLs:	Based on technical specifications
• Partition Coefficient	
- Iodine	100
- Noble gas	All noble gases from the primary system are released
• Release Duration	
- Failed MSL:	Based on accident analysis
- Intact MSLs:	Until shutdown cooling

3.8 Fuel Handling Accident (1)

- **Assumptions of postulated accidents**
  - ✓The following two FHA cases are evaluated
    - FHA in the Fuel Handling Building
    - FHA in the Containment Vessel
  - ✓FPs are released from the gap of the damaged fuel assembly

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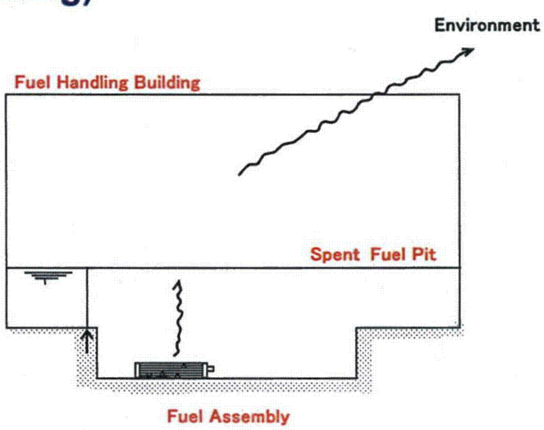
3.8 Fuel Handling Accident (2)

- **FP release pathway (FHA in the Fuel Handling Building)**



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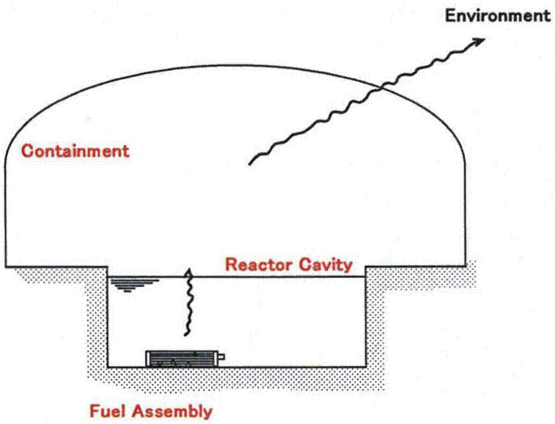
3.8 Fuel Handling Accident (3)

- FP Release pathway (FHA in the Containment Vessel)



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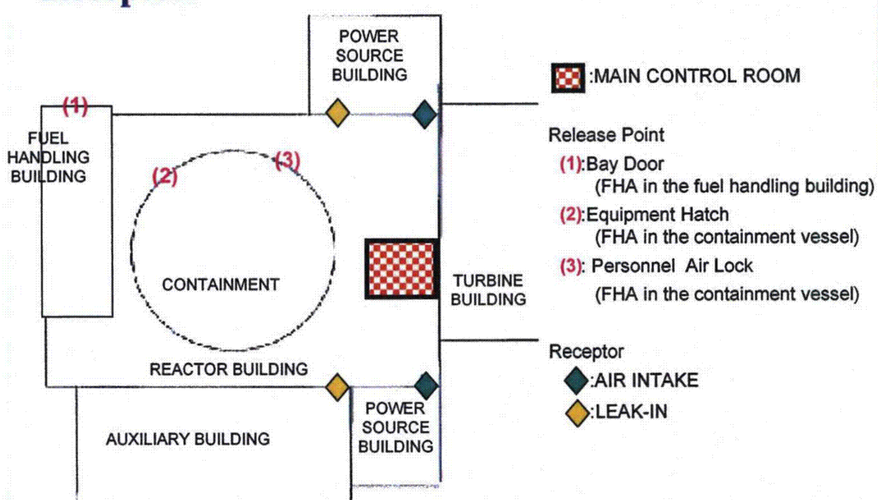
3.8 Fuel Handling Accident (4)

- **FHA Release Points and Control Room Source Receptors**



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3.8 Fuel Handling Accident (5)

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➢ Major conditions for dose evaluation

Parameters	Value	
	Fuel Handling Building	Containment
• Assembly peaking factor	Design value	Design value
• Fraction of degraded fuel	200% (2 assemblies)	200% (2 assemblies)
• Assumed cooling period	1 day	1 day
• Chemical form of iodine		
- Elemental Iodine	99.85 %	99.85 %
- Organic Iodine	0.15%	0.15%
• Minimum Water Depth	23feet ~	23feet ~
• Effective DF		
- Iodine	200	200
- Noble Gases	1	1
• Release Duration	2hr	2hr

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3.9 Main Control Room Doses during LOCA (1)

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➢ **Assumptions of postulated accidents**

✓ Postulated events
  Large Break LOCA

✓ Single failure
  The worst single active failure

✓ Safety features available
  Safety injection system
  Containment vessel spray system
  Annulus air cleanup system
  Air cleanup system for safeguard component area
  Emergency recirculation system for main control room

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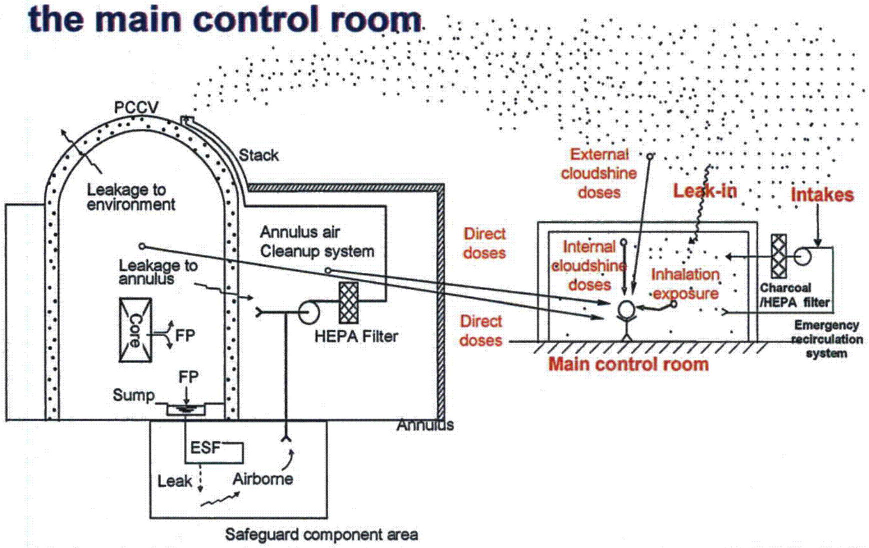
3.9 Main Control Room Doses during LOCA (2)

➢ **Exposure pathways to the operator within the main control room**



3.9 Main Control Room Doses during LOCA (3)

➢ **Dose evaluation conditions within the main control room**

Parameters	Value
FP release rate from CV	Same as offsite dose evaluation at LOCA
MCR emergency air purifier	Pressurized recirculation type
FP pathway of flow into MCR	Fresh-air inlet (through filter) In-leakage (unfiltered leakage)
Capacity of emergency recirculation fan*	1.5 volume changes per hour
Fresh-air (through filter) inlet volume*	0.5 volume changes per hour
Leak-in (unfiltered leakage) volume*	0.05 volume changes per hour
Filtration efficiency of emergency recirculation system	Charcoal filter :95% HEPA filter: 99%

* Values are approximate

3. 10 Summary of Dose Evaluation under Accident Conditions

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➢Summary

✓Our methods of dose evaluation under accident conditions are based on the US requirements and most recent guidance

✓We plan to use calculation codes already approved by the NRC

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4. Methods and Acceptance Criteria for Radiation Shielding Protection

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Contents

4.1 Outline of ALARA

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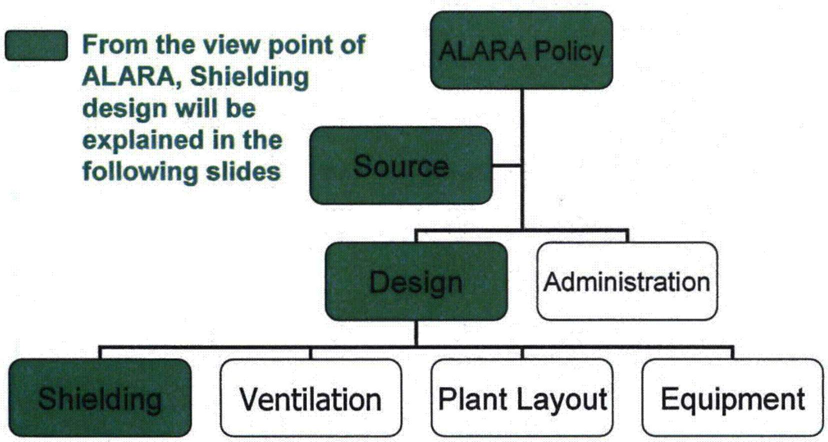
➢**Schematic Diagram to ensure ORE* ALARA**



* ORE:Occupational Radiation Exposure

4.2 Acceptance Criteria

- Design of shielding, HVAC and plant layout will be conducted in full compliance with applicable NRC requirements and regulatory guidance
- NRC Requirements (10 CFR 20, 50)

Item	Limit	Reference
Occupational Dose Limit	5 rem/yr (TEDE)	10CFR20.1201(a)(1)
Dose Limit for Public	0.1 rem/yr (TEDE)	10CFR20.1301(a)(1)
External Dose Limit in any unrestricted area	0.002 rem/hr	10CFR20.1301(a)(2)
Occupational Dose inside the main control room under accident conditions	5 rem (TEDE)	10CFR50 Appendix A 10CFR50.67

HVAC : Heating, Ventilating and Air Conditioning
TEDE : Total Effective Dose Equivalent

4.3 Shielding Design Objectives

- To keep exposure for the public and plant personnel ALARA and within 10 CFR 20 requirements
- Personnel doses under accident conditions should not exceed 5rem in accordance with the limitations prescribed in 10 CFR 50 Appendix A (General Design Criterion 19) and 10 CFR 50.67

4.4 Radiation Sources (1)

(1) Normal Operation Including Shutdown

Location	Radiation Source
Core	Neutron and prompt Gamma-rays (4,451 MWt thermal power)
Reactor Coolant System (RCS) /Steam Generator (SG)	N-16(Gamma-ray) and N-17(Neutron) FP and CP
Chemical and Volume Control System (CVCS)	FP in the Primary Coolant assuming a Fuel Cladding Defect of 1% and CP
	FP and CP in Resin of Demineralizer
Boron Recovery System (BRS)	FP and CP in Boron Acid
	FP and CP in Resin of Demineralizer
Residual Heat Removal System (RHRS)	FP in the Primary Coolant assuming a Fuel Cladding Defect of 1% and CP

FP : Fission Product
CP : Corrosion Product

4.4 Radiation Sources (2)

(1) Normal Operation Including Shutdown (Cont'd)

Location		Radiation Source
Waste Management System	Gaseous	FP in Stripping Gas in Equipment
		FP in Charcoal Bed
	Liquid	FP and CP in Equipment Drain
		FP and CP in Floor Drain
		FP and CP in Condensed Liquid Drain
		FP and CP in Resin of Demineralizer
	Solid	FP and CP in Spent Resin
		Co-60 in Waste Drum

4.4 Radiation Sources (3)

(1) Normal Operation Including Shutdown (Cont'd)

Location	Radiation Source
Secondary Coolant	SG Blow Down Water with Primary-to-Secondary Leakage
Spent Fuel Storage Pit, Reactor Cavity	Spent Fuel Fuel type : 17 x 17, 14 feet, UO_2 U-235 enrichment : based on design value Burn up : based on design value Minimum Decay time before handling : 1 day
Spent Fuel Storage Pit Cooling and purification System	Co-60 in pit water that causes 15 mrem/h at the surface of the water
	FP and CP in resin of demineralizer

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4.4 Radiation Sources (4)

(2) Additional Sources under Accident Conditions

Location	Radiation Source	note
Containment Vessel (CV)	Airborne FP inside CV during loss of coolant accident (LOCA)	Radiological source terms defined in accordance with Regulatory Guide 1.183
Reactor Building(RB)	FP inside Sump Water	
	Airborne FP that leaked into RB	

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4.4 Radiation Shielding Design (1)

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➢Shielding Design is based on these primary references

Reference	Title
Regulatory Guide 1.69	Concrete Radiation Shields for Nuclear Power Plant
ANSI/ANS-N101.6-1972	Concrete Radiation Shields
ANSI/ANS-6.4-1997	Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plant
-	Reactor Shielding Design Manual (Edited by T. Rockwell III)

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4.4 Radiation Shielding Design (2)

US-APWR

➢Calculation Codes used for Shielding Design

Name of code	Notes
ORIGEN2	✓Source term calculation for core inventory and spent fuel
MicroShield	✓For almost all the contained sources besides neutron source (tank, filter, demineralizer, e.t.c.)
DORT ANISN	✓Used inside and around the reactor pressure vessel and equipment of RCS
MCNP	✓For complicated geometries
G^3	✓For scattered gamma-ray dose calculation

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4.4 Radiation Shielding Design (3)

➢ Design Features of US-APWR

(1) Reduction of occupational exposure

(i) pipe trenches or chases to isolate piping containing radioactivity

(ii) valve operating areas to improve maintenance activity

(iii) Cobalt 60 reduction program

(2) Reduction of radiation streaming to prevent unexpected exposure

(i) locate piping or duct penetration high on the shield wall

(ii) minimize the size of penetration and/or fill gap with grout, lead wool or resin

(3) Narrow Gap

Labyrinth structure between the reactor pressure vessel and the primary shielding will be formed to minimize neutron streaming into the upper region of the containment vessel

4. Methods and Acceptance Criteria for Radiation Shielding Protection

➢ Summary

✓ Radiation shielding design of US-APWR is based on the US requirements and most recent guidance

✓ Calculation codes widely known in the USA are used for shielding design

5. Conclusions

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5. Conclusions

- Methods of dose evaluation and shielding design for US-APWR are based on the US requirements and most recent guidance.
- The design and the dose evaluation results will be described in DCD and submitted in December, 2007.

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