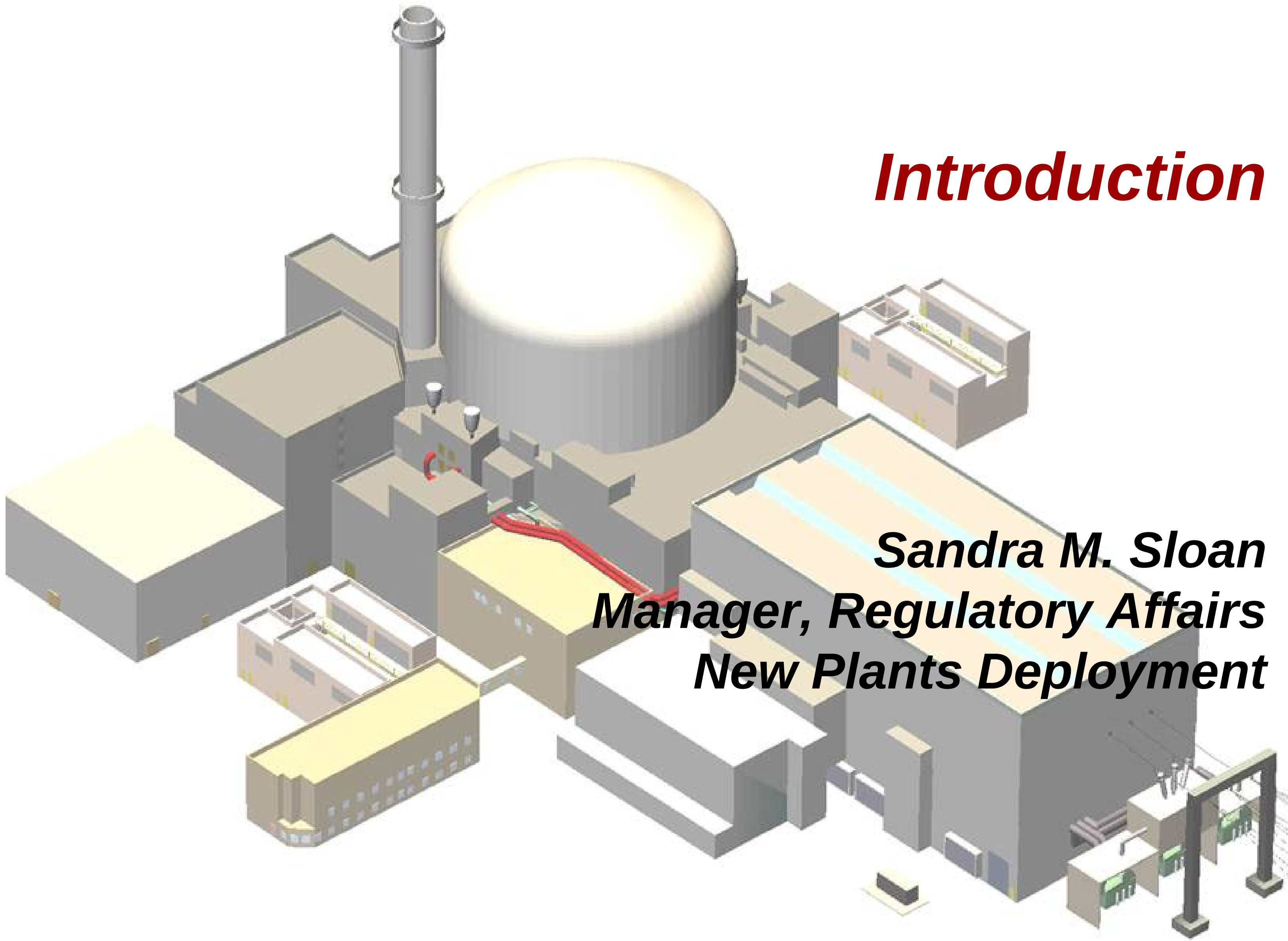
U.S. EPR Pre-Application Review Meeting:

Probabilistic Risk Assessment (PRA) Overview of Approach and Methods

AREVA NP Inc. and the NRC
October 24, 2006



Introduction

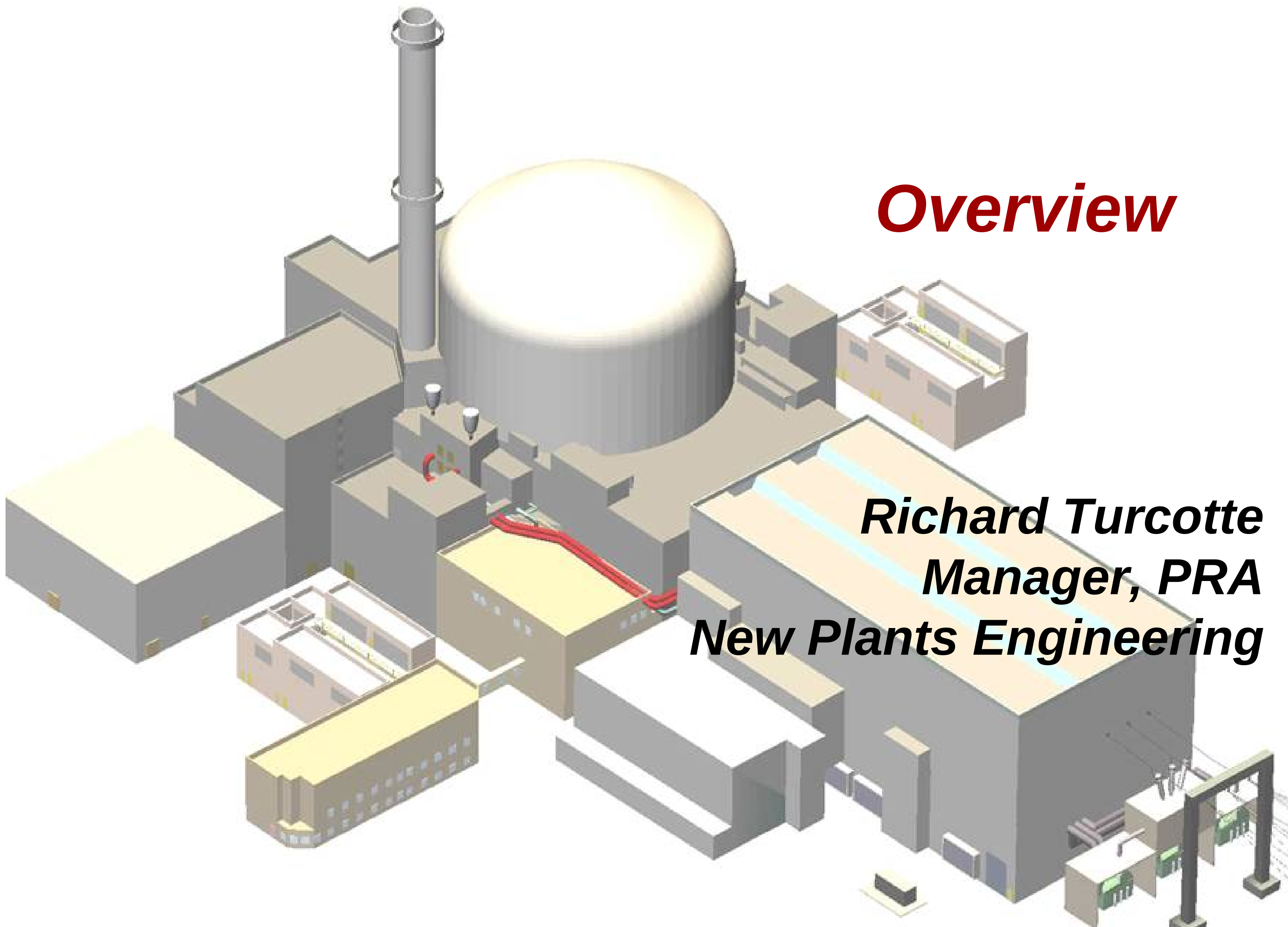
Sandra M. Sloan
Manager, Regulatory Affairs
New Plants Deployment

Meeting Objectives

- Describe the PRA scope, methods and key assumptions used to support the U.S. EPR design certification
- Provide overview of the U.S. EPR design features important from a PRA perspective
- Summarize preliminary PRA results/insights
- Discuss the PRA Methods Report (to be submitted in December)
- Obtain early NRC feedback regarding the PRA approach

Presentation Outline

- > PRA Overview (Richard Turcotte)
- > PRA Methods and Preliminary Results
  - Level 1 Model (Vesna Dimitrijevic)
  - Fire, Flood, Seismic, Shutdown (Vesna Dimitrijevic)
  - Digital I&C (Robert Enzinna)
  - Level 2 and 3 Model (Bob Prior)
- > Summary and Next Steps (Sandra Sloan)



Overview

Richard Turcotte
Manager, PRA
New Plants Engineering

Overview - Topics

> U.S. EPR Design Goals
> PRA Objectives
> PRA Scope
> PRA Quality
> PRA Software
> U.S. EPR Design Features Important to the PRA
> Use of PRA in Design Process

U.S. EPR Design Goals

- Core Damage Frequency $<10^{-5}$ per year
- Large Release Frequency $<10^{-6}$ per year

Including internal and external events (seismic and sabotage excluded) for all operating modes

PRA Approach / Objective

- Use bounding/realistic-type assumptions where detailed design information is not available at Design Certification
- Perform analysis considering Reg. Guide 1.200/ASME PRA Standard
- Follow developments within nuclear industry consensus standards and use of good practices

Objective is to demonstrate robustness of U.S. EPR design features and demonstrate that U.S. EPR design goals are met with margin

Scope

> Level 1 – Core Damage Frequency (CDF)

> Level 2 – Large (and Large Early) Release Frequency (LRF/LERF)

> Level 3 – Offsite Dose Consequence

> Scope of initiating events for design certification
- Internal events (at power and low power/shutdown)
- External events
  - Internal flood and fire events (at power and shutdown)
  - Seismic (PRA-based margins) (at power)
  - Other externals – high level, qualitative

PRA Quality

> **Quality**
- **Qualified personnel**
- **Procedures to control documentation development and revision, independent review/checking of calculations and information used in the PRA**
- **Procedures for documentation and maintenance of records**
- **Corrective action process if information previously used are changed or in error**

> **Technical Adequacy**
- **Meet NRC guidelines on PRA technical adequacy (RG 1.200)**
- **Conduct formal peer review as part of detailed design**
- **Technical review meeting(s) and technical exchange with European counterparts**

PRA Software

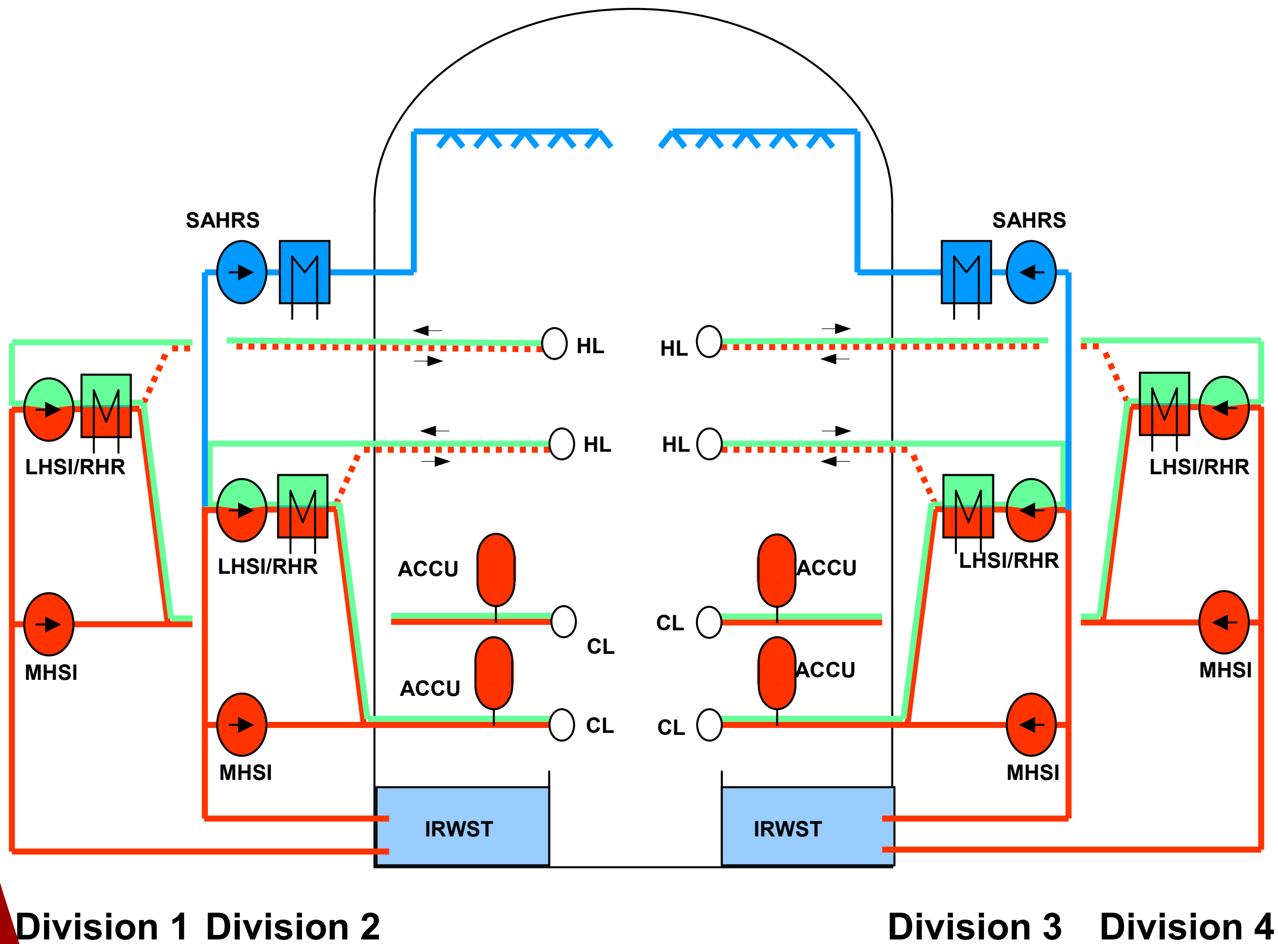
> RiskSpectrum® Professional

- Developed and maintained by Relcon AB
- Used extensively in Europe, Palo Verde in U.S.
- Full license with software/user support
- PRA staff received training
- AREVA NP participates in Users Group
- Performed Relcon V&V to ensure proper installation and operation

Design Features Contributing to Low Risk

- **Increased redundancy and separation**
- **Additional capabilities to mitigate severe accidents**
- **Protection of critical systems from external events**
- **Active components and technology with proven reliability from current operating PWR fleet**

Design Features Contributing to Low Risk



- Four independent trains of SIS
- 2 LHSI pumps powered from SBO diesels
- In-containment refueling water storage pool
- Containment spray for severe accident only

Design Features Contributing to Low Risk

- Four independent safety trains in separate buildings
- Physical separation against internal & external hazards
- Because of large separate areas, internal hazards results in loss of only one division
- Shield building extends airplane crash and external hazard protection to two safeguard buildings and fuel building

Design Features Contributing to Low Risk

- Four Emergency Diesel Generators (one EDG for each safety division)
- Two Station Blackout Diesel Generators
  - Divisions 1 and 4 each contain one SBO diesel
  - Started manually from control room or locally
  - SBO diesels independent/diverse of EDGs - based on consideration of attributes such as: different manufacturer, different controls, batteries
- RCP Stand-Still Seal System: a pneumatic, "metal-to-metal" seal that provides back-up seal capability independent of the normal seal and can prevent RCP shaft leakage

Design Features Contributing to Low Risk

> **Containment & Shield Building**
- **Containment:**
  pre-stressed concrete with steel liner
- **Shield Building:**
  reinforced concrete

> **Protection against aircraft hazard and external explosions**

> **Annulus subatmospheric and filtered to reduce radioisotope release**

Design Features Contributing to Low Risk

- Severe accident mitigation systems
  - Primary Depressurization System prevents high-pressure vessel failure
  - Corium maintained in coolable configuration in spreading area
  - IRWST provides passive cooling of corium and basemat
  - Active SAHRS provides long-term cooling
  - Passive autocatalytic recombiners limit hydrogen buildup

PRA Influence on Design - Examples

> Addition of SBO DGs in divisions 1 and 4 to improve plant response to LOOP events

- Diversity from EDGs based on different manufacturers, controls, batteries

> Permanent alignment of Safety Chilled Water to LHSI pump motors and mechanical seals in Division 1 and 4

> Reduction in support system dependency by realigning power supplies to main steam relief train pilot valves, so that a loss of two electric divisions would not fail the main steam relief

PRA Influence on Design – Examples
(continued)

> **Improvement in reliability of SIS response at mid-loop by adding diverse signals**
  - **Auto start of MHSI on low RCS loop level or low suction pressure to RHR pumps**

> **Diversification of the cooling system for SAHRS by providing a CCW/ESW division dedicated to the cooling of each SAHRS train**

PRA discipline formally tied into ongoing design effort via project design directive and design change process

PRA Methods Report

> **Contents**

Section 1: Introduction and Summary

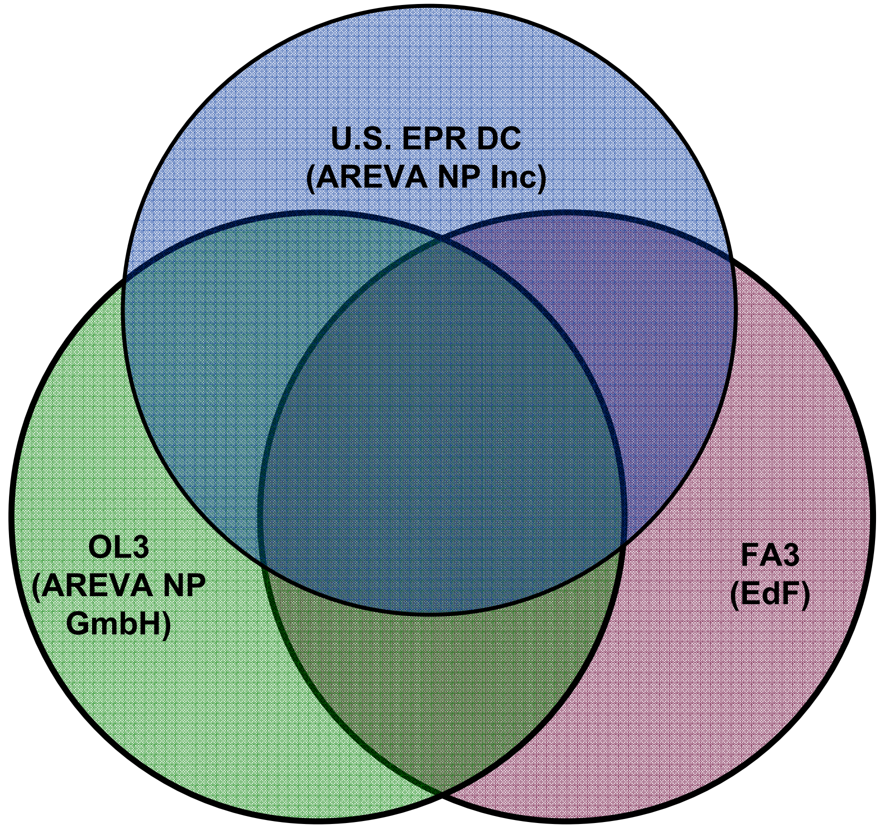
Section 2: PRA Methodology

- **Level 1 Analysis**
- **Internal Flood and Fire Analyses**
- **Seismic Methodology**
- **Other Modes Analysis**
- **Level 2 Analysis**
- **Level 3 Analysis**
- **Uncertainty and Sensitivity Analysis**

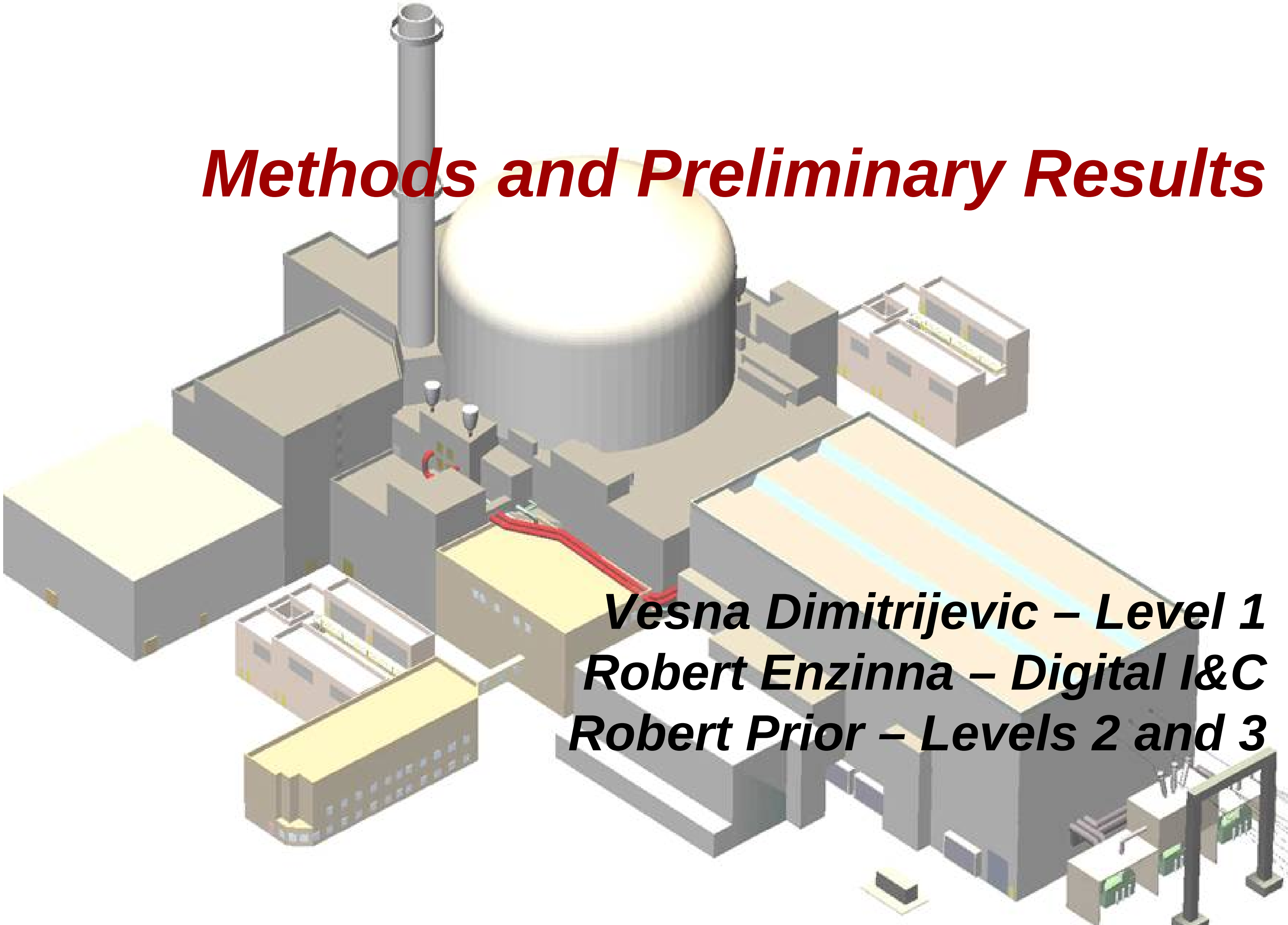
Section 3: Summary and Conclusions

EPR PRA International Cooperation

- Three EPR PRAs:
  - U.S. EPR DC (AREVA NP Inc)
  - OL3 (AREVA NP GmbH)
  - FA3 (EDF)
- PRA objectives similar
- Differences due to:
  - Design variations
  - Phase of design completeness
  - Assumptions
  - Data



Methods and Preliminary Results



Vesna Dimitrijevic – Level 1
Robert Enzinna – Digital I&C
Robert Prior – Levels 2 and 3

AREVA

Methods / Results Overview - Topics

> **Methods and approach selected for Design Certification and preliminary results:**
- **Level 1 PRA**
  - **Initiating Events, Data, Systems**
    Success Criteria, I&C, HRA
    Uncertainty, Sensitivity
- **Internal Flood and Fire Events**
- **External Events – PRA-Based Margins**
- **Low Power Shutdown**
- **Level 2 PRA**
- **Level 3 PRA**
- **Conclusions**

Preliminary Initiating Events Frequency and Basis

Initiating Event	U.S. EPR Freq. [1/yr]	Basis	NUREG 5750 Freq. [1/yr]
General Transients			
TT – Turbine Trip (includes RT – Reactor Trip)	1.2	NUREG/CR 5750	1.2
LOC – Loss of Main Condenser (includes MSIV closure etc.)	1.20E-01	NUREG/CR 5750	1.20E-01
LOMFW – Total Loss of Main Feedwater	8.50E-02	NUREG/CR 5750	8.50E-02
Loss of Coolant Accidents (LOCA)			
SLOCA – Small LOCA (0.6 to 3-inch diameter)	5.00E-04	NUREG/CR 5750	5.00E-04
MLOCA – Medium LOCA (3 to 6-inch diameter)	4.00E-05	NUREG/CR 5750	4.00E-05
LLOCA – Large LOCA (>6-inch diameter)	5.00E-06	NUREG/CR 5750	5.00E-06
SGTR – Steam Generator Tube Rupture	7.00E-03	NUREG/CR 5750	7.00E-03
ISLOCAs	1.00E-09 - 1.20E-13	Fault Tree (FT) Analysis	NA
Secondary Side Breaks			
SLBO – Steam Break Downstream of MSIV	1.00E-02	NUREG/CR 5750	1.00E-02
SLBI – Steam Break Inside Containment	1.00E-03	NUREG/CR 5750	1.00E-03
MSSV – Spurious Opening of Steam Safety Valve	1.00E-03	NUREG/CR 5750	1.00E-03
Support System Failures			
LOOP – Loss of Offsite Power	3.59E-02	NUREG/CR-6890	4.60E-02
LOCCW Trains	1.00E-02 - 5.70E-07	FT Analysis	8.9E-3 (Part. Loss of SW)
LBOP – Loss of Closed Loop Cooling Water or Aux Cooling Water	2.50E-02	FT Analysis	NA
N1BDA – Loss of Divisional Emergency AC (Switchgear N1BDA)	3.49E-02	FT Analysis	1.9E-2 (Vital Med. Volt. AC Bus)

Data Sources

- Current sources of data
  - EGG-SSRE-8875 – Generic Component Failure Database for Light Water and Liquid Sodium Reactors, EG&G Idaho, 1990
  - ZEDB - Centralized Reliability and Events Database of Reliability Data for Nuclear Power Plant Components, Technical Association of Large Power Plant Operators, Germany, Analysis 2000 – includes all German nuclear plants, Dutch Unit Borssele and Swiss unit Goesgen
  - EIREDA95 – European Industry Reliability Data Bank, EIReDA, Volume 2, 1977/1993
- Data compared with U.S. data/EPRI ALWR recommendations

Sample Data Comparison

Group ID	Comp Type	Data Source	Failure Mode Description	Failure Rate [per demand or per hr]
DG-FTR (Emergency Diesel Engine)	Diesel Generator	U.S. EPR PRA	**Emergency Diesel Generator - Fails to run**	**2.40E-03**
		NUREG 5500 Data	Diesel Generator - Fails to run after the 1st hour - No recovery	9.43E-04
		ALWR EPRI Data	Diesel Generator - Fails to run	2.40E-03
DG-FTS (Emergency Diesel Engine)	Diesel Generator	U.S. EPR PRA	**Emergency Diesel Generator - Fails to start**	**4.50E-03**
		NUREG 5500 Data	Diesel Generator - Fails to start and load - No recovery	1.52E-02
		ALWR EPRI Data	Diesel Generator - Fails to start and load	1.40E-02
MDP-FTR A (Medium Head Safety Injection System) (Startup and Shutdown System) (Emergency Feedwater System)	Pump - Motor Driven	U.S. EPR PRA	**Motor-driven Pump - Fails to run**	**5.10E-04**
		ALWR EPRI Data	Motor-driven pump (Emerg. Feed) - Fails to run	1.50E-04
			Motor-driven pump (all other types) - Fails to run	2.50E-05
MDP-FTS A (Medium Head Safety Injection System) (Startup and Shutdown System) (Emergency Feedwater System)	Pump - Motor Driven	U.S. EPR PRA	**Motor-driven Pump - Fails to start**	**1.28E-03**
		NUREG 1715 Data	Motor-driven pump - Fails to start	1.37E-03
		ALWR EPRI Data	Motor-driven pump (Safety Inj.) - Fails to start on demand	1.00E-03
			Motor-driven pump (Emerg. Feed) - Fails to start on demand	3.00E-03
			Motor-driven pump (all other types) - Fails to start on demand	2.00E-03
VLV-MOV-FTC (All Systems)	Valve - Motor Operated	U.S. EPR PRA	**Motor-operated Valve - Fails to Close**	**3.50E-03**
		NUREG 1715 Data	Motor-operated valve - Fails to Close	4.67E-04
		ALWR EPRI Data	Motor-operated valve - Fails to Close	2.00E-03

U.S. EPR data is comparable with NUREG-1715 and ALWR EPRI data

Common Cause Failure Data Comparison

Parameter	European Data	U.S. EPR (NUREG/CR-6819 2003 Update)			
	Generic (4 Components)	Generic	EFW Pump Start	LHSI Pump Run	EDG Start
Beta	0.1	0.0317	0.0374	0.00933	0.0177
Gamma	0.4	0.335	0.679	0.743	0.415
Delta	0.25	0.349	0.347	0.333	0.211
Conditional Four Train Failure Probability	0.010	0.004	0.009	0.002	0.002

U.S. EPR PRA uses NUREG/CR 6819 common cause data

Systems Modeled

> **U.S. EPR frontline systems**

- **Main Feedwater System**
- **Startup Shutdown Feedwater System**
- **Emergency Feedwater System**
- **Main Steam System**
- **Pressurizer Relief System**
- **Medium Head Safety Injection System**
- **Low Head Safety Injection/RHR System**
- **Extra Boration System**
- **Chemical Volume Control System**
- **Reactor Boron and Water Makeup System**
- **Demineralized Water System**
- **Severe Accident Heat Removal System**
- **Fuel Pool Cooling System**
- **Containment Isolation System**
- **Digital I&C (RPS and ESFAS)**

Systems Modeled (continued)

> **U.S. EPR support systems**

- **AC Electrical Distribution System**
- **EDGs and SBO DGs**
- **DC Electrical Distribution System**
- **Essential Service Water System / UHS**
- **Component Cooling Water System**
- **Closed and Auxiliary Cooling Water Systems**
- **Safety Chilled Water System**
- **Operational Chilled Water System**
- **HVAC Systems (safeguard bldg SAC, including switchgear ventilation, diesel room ventilation)**

System Dependencies Example

System/Train Failure	Impacts								
	OCW	SCW	CVCS	RCP	MHSI	LHSI	SAC	EFW	Electrical
CCWS Common Header 1	1 of 2	20	1 of 2	Cooling to 2 pumps					
CCWS Common Header 2	1 of 2	30	1 of 2	Cooling to 2 pumps					
CCWS10					10	10 (Hx)			
CCWS20					20	20			
CCWS30					30	30			
CCWS40					40	40 (Hx)			
Operational Chilled Water (OCW)							50, 80		
Safety Chilled Water 10 (SCW)						10 (Pmp)	10 (61)	10	
Safety Chilled Water 20 (SCW)							20 (62)	20	
Safety Chilled Water 30 (SCW)							30 (63)	30	
Safety Chilled Water 40 (SCW)						40 (Pmp)	40 (64)	40	
SAC10 (SAB1 Ventilation)									10P
SAC20 (SAB2 Ventilation)									20P
SAC30 (SAB3 Ventilation)									30P
SAC40 (SAB4 Ventilation)									40P
SAC50 (Maintenance Train)									(10, 20)P
SAC80 (Maintenance Train)									(30, 40)P

- **10, 20, 30, 40 (50, 80) identifies train divisions**
- **61, 62, 63, 64 identifies SAC divisions for EFW rooms**
- **“Hx” indicates heat exchangers cooling**
- **“Pmp” indicates pump cooling**
- **“P” indicates partial dependency**

Success Criteria

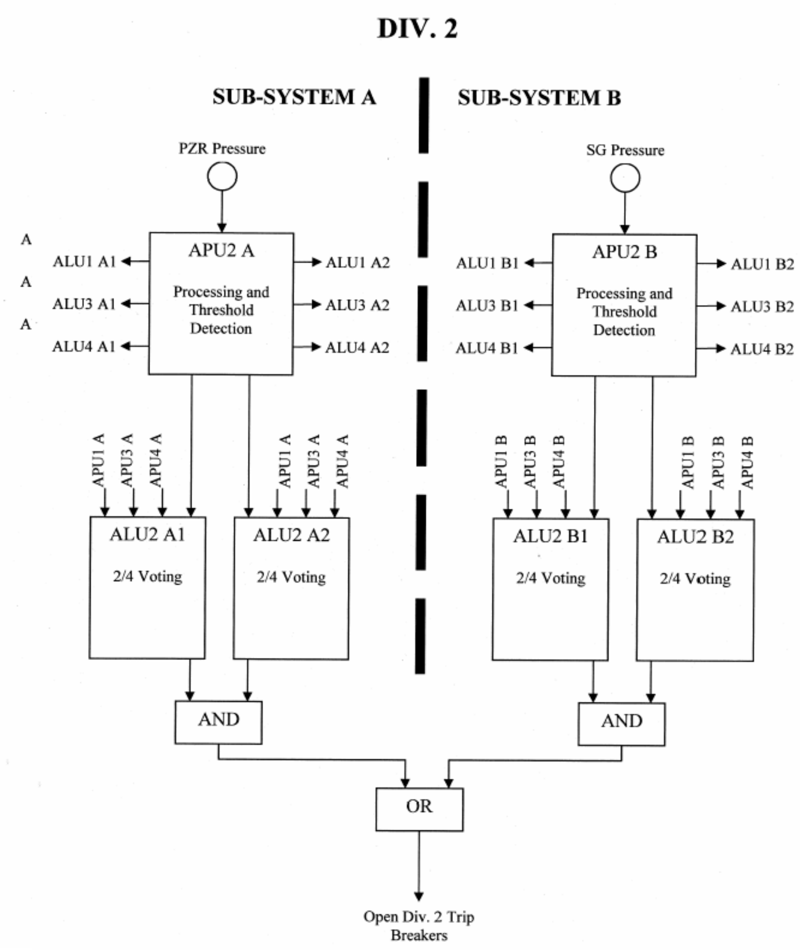
- Postulated events are analyzed to determine whether or not the sequence leads to core damage
- Core damage defined as:
  - Uncovery and heat up of the reactor core to the point at which prolonged oxidation and severe fuel damage involving a large fraction of the core is anticipated
- Acceptance criteria used to avoid core damage
  - For transients and accidents, PCT<2200 ºF
  - ATWS overpressure events, P<130% design pressure
  - For low pressure shutdown events and spent fuel pool, fuel is covered (conservative)
  - Achieve stable state within 24 hours

Success Criteria TH Methodology

- Analysis is conducted with computer models and hand calculations
  - MAAP v4.0.7
    - Integrated accident analysis method
    - Level 1 analysis with understanding of limitations
    - Extensively used in Level 2 analysis
    - Convenient, fast-running
  - S-RELAP5
    - Two-fluid two-phase model
    - Detailed nodalization of system with user flexibility
    - Approved for typical PWR FSAR safety analysis (LOCA, non-LOCA)
    - Can be used for realistic analysis, with realistic input parameters

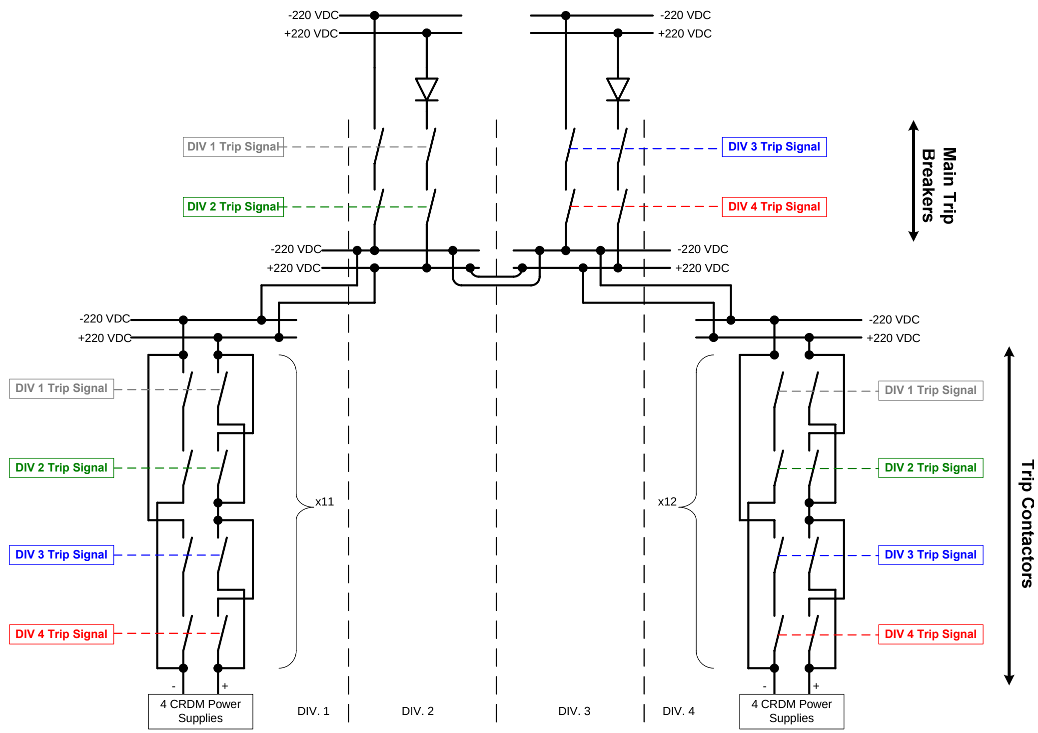
Digital I&C Modeling Approach

- Safety I&C platform is AREVA Teleperm XS
- Detailed fault tree of Protection System (RPS and ESFAS)
  - 4 division redundancy
  - 2 independent subsystems "A/B" functional diversity per division
  - No communication between A/B subsystems
- Backup systems and trips to be modeled later (ATWS, D3)
- Normal plant control systems not modeled in detail



Diverse Reactor Trip Devices

- Four reactor trip breakers
- 23 sets of four contactors
- Fast-acting transistors (not shown) release power to control rod grippers independent of breakers and contactors



I&C Failure Data and Sources

- Manufacturer data for I&C modules from operating history
- Has been operating in European RPS/ESFAS systems for over 10 years
- For example, main computer processor module has over 50 million hours of operating experience
- Conservative Chi-squared (95%) treatment to bound observed failure rate

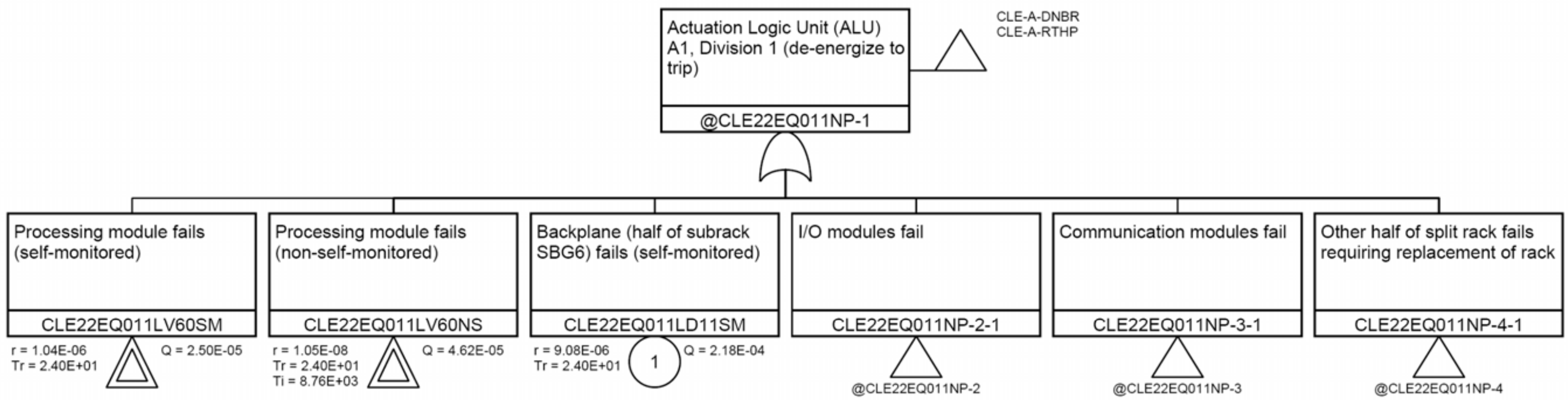
I&C Common Cause Failure Modeling

- Software is robust
  - Software development process “high quality”
  - Software development tools and operating system have been in operation for 10+ years in European RPS/ESFAS systems
  - Application software uses only qualified software modules from function block library (SPACE programming tool)
  - Deterministic operating system improves reliability and predictability
  - Asynchronous operating system reduces CCF potential
  - Functional diversity in independent subsystems
- PRA treats software failure as a CCF mechanism for the computer processors
- CCF applied to computer processors that have the same application software (algorithms) and inputs
- Conservative Beta factor (0.01)
- Model also includes CCF for mechanical components: sensors, reactor trip breakers

I&C Functions Modeled

- Selected representative reactor trip functions: loss of feedwater, turbine trip
- ESFAS functions in detail, for example:
  - EFW actuation
  - Safety injection actuation
  - Main steam relief train open and isolation
  - MSIV
  - MFW isolation
  - Containment isolation

I&C Model and Result Examples



> **Reactor Trip on Loss of Main Feedwater**
- **4 channels of low SG level and 4 channels low DNBR**
- **Backup trips (e.g., ATWS) not yet credited**
- **5E-8/demand**
- **Dominated by CCFs (stuck rods, breakers, computer processors, sensors)**

HRA Methodology

Human Reliability Analysis (HRA) is performed using the following methods:

> **Pre-Accident Human Error Probabilities:**

- **Accident Sequence Evaluation Program (ASEP) method (an abbreviated and slightly modified version of the THERP method) (Reference: NUREG/CR-4772)**

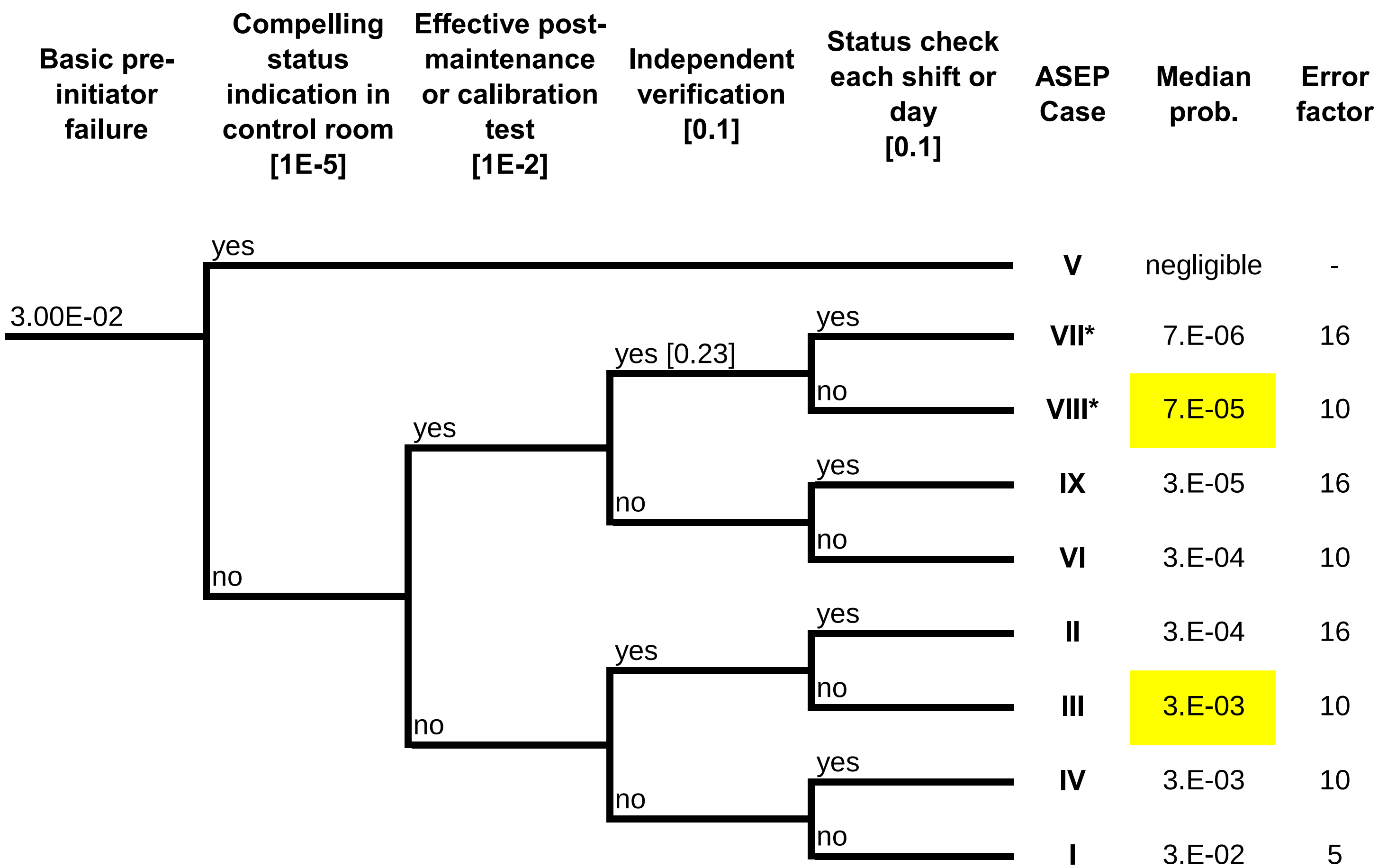
> **Post-Accident Human Error Probabilities:**

- **Standardized Plant Analysis Risk Human Reliability Analysis (SPAR-H) (Reference: NUREG/CR-6883)**

> **Dependencies between Post-Accident Human Actions:**

- **SPAR-H guideline**

Pre-Accident HEP: ASEP Method



Reference: NUREG/CR-4772, Accident Sequence Evaluation Program Human Reliability Analysis Procedure; A.D. Swain; February 1987 (ASEP).

HRA Basic Events Example

Basic Event ID	HEP cognition	HEP action	HEP total	Error Factor	TIME				COGNITION				ACTION			
					Tsw	Tm	Td	Tw	Time	Stress	Complexity	Experience	Time	Stress	Complexity	Experience
OPE-FB-30M (MLOCA)	5.00E-02	2.50E-03	**5.3E-02**	10	30	5	10	15	1	5	2	0.5	1	5	1	0.5
OPE-FB-40M (SLOCA)	1.25E-01	2.50E-03	**1.3E-01**	5	40	5	15	20	1	5	5	0.5	1	5	1	0.5
OPE-FB-90M (LOMFW)	2.50E-03	2.50E-04	**2.8E-03**	10	90	5	15	70	1	5	0.1	0.5	1	5	0.1	0.5
OPE-FCD-30M (MLOCA)	5.00E-02	2.50E-03	**5.3E-02**	10	30	5	10	15	1	5	2	0.5	1	5	1	0.5
OPE-FCD-40M (SLOCA)	1.25E-01	2.50E-03	**1.3E-01**	5	40	5	15	20	1	5	5	0.5	1	5	1	0.5
OPF-EFW-6H	5.00E-05	5.00E-06	**5.5E-05**	10	360	15	60	285	0.1	1	0.1	0.5	0.1	1	0.1	0.5
OPE-RHR-4H	5.00E-04	5.00E-05	**5.5E-04**	10	240	5	60	175	0.1	1	1	0.5	0.1	1	1	0.5

Procedure	Set to 1
Ergonomic	Set to 1
Fitness for Duty	Set to 1
Work Process	Set to 1

Tsw	System Time Window
Tm	Manipulation Time
Td	Time Delay from Transient until Cue is Reached
Tw	Window for Cognitive Response

SPAR-H Dependency Rating System

Crew	Time	Location	Cues	Level of Dependency
Same	Close	Same		**Complete (1)**
		Different		**High (0.5)**
	Not Close	Same	No Additional	**High (0.5)**
			Additional	**Moderate (0.14)**
		Different	No Additional	**Moderate (0.14)**
			Additional	**Low (0.05)**
Different	Close			**Moderate (0.14)**
	Not Close			**Low (0.05)**

Reference: NUREG/CR-6883, The SPAR-H Human Reliability Analysis Method, Idaho National Laboratory, August 2005.

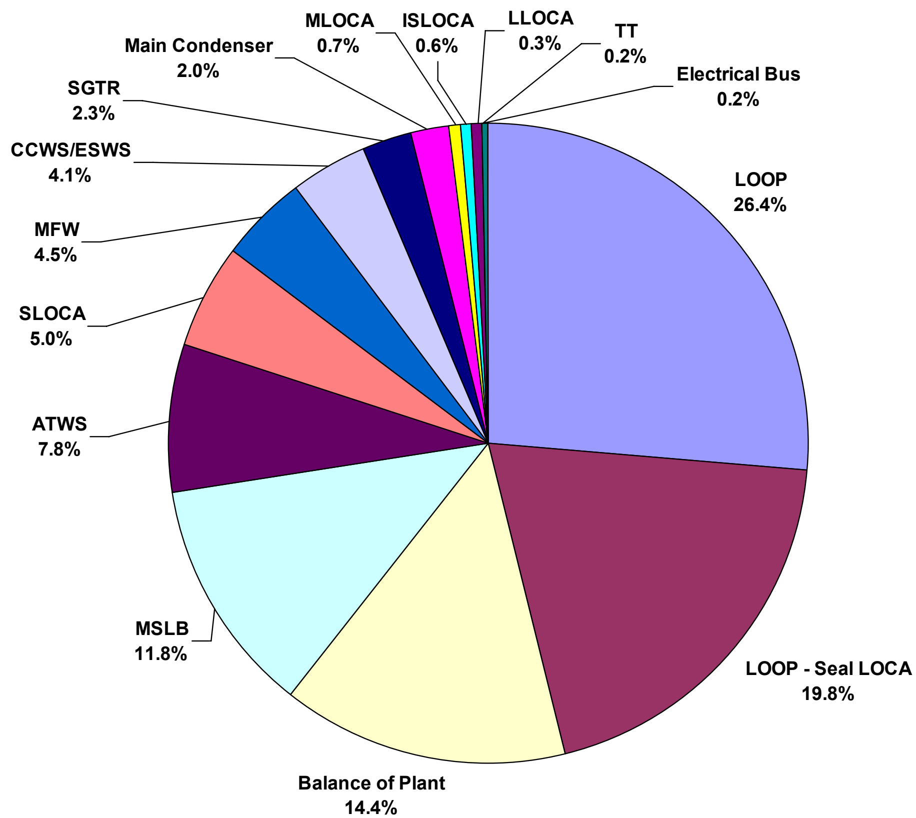
Examples of Human Dependencies in the Current Model

> **Operator action to start feed and bleed after failing to provide EFWT cross-connect/makeup:**
  - **OPE-FB-90M (independent) = 2.8E-3**
  - **OPE-FB-90M (after OPE-EFW-6H) = 0.05 (low dependency)**

> **Operator action to recover room cooling after failing to align HVAC maintenance train**
  - **OPF-SAC-4H (independent) = 1.1E-3**
  - **OPF-SAC-4H (after OPF-SAC-1H) = 0.05 (low dependency)**

> **Operator action to depressurize/start RHR after failing to isolate ISLOCA**
  - **OPE-RHR-4H (independent) = 5.5E-4**
  - **OPE-RHR-4H (after failing to isolate ISLOCA) = 0.14 (medium dependency)**

Preliminary U.S. Results for Level 1 Internal Events (at power)



Total CDF for internal events is 3.5E-07/yr

Preliminary Example of CDF Cutsets

No	Freq.	% Contr	Cumul %	Initiating Event	Event 1	Event 2	Event 3	Event 4
1	4.84E-09	1.36%	1.36%	IE LBOP	DEP-FBL90M-EFW6H	EFWS PM3	OPF-EFW-6H	
					Dependency Between Operator Actions for Long-Term EFW & FB Longer Than 90 Min	EFWS Train 3 Unavailable due to Preventive Maintenance	Operator Fails to Manually Align EFWS Tanks Within 6 Hours	
10	2.83E-09	0.79%	11.51%	IE SLOCA	JNK10AT001SPG_D-ALL			
					CCF of SIS Sump Strainers - Plugged			
11	2.60E-09	0.73%	12.24%	IE SLOCA	JNG13AA005CFO_D-ALL			
					CCF to Open LHSI Check Valves (SIS First Isolation Valves)			
12	2.29E-09	0.64%	12.88%	IE LOOP	OPF-DGSBO-90M	REC OSP 2HR	XKA10_____DFR_D-ALL	
					Operator Fails to Start SBO DGs XJA50/80 or to Close Breakers Within 90 Minutes	Failure to Recover Offsite Power Within 2 Hours	CCF of EDGs to Run	
13	2.23E-09	0.63%	13.51%	IE LOOP	N1BTD01_BATST_D-ALL	OPE-FCD-40M	PROB SEAL LOCA	REC OSP 1HR
					CCF of Batteries to Start	Operator Fails to Initiate Fast Cooldown Within 40 Minutes	Probability of Seal LOCA Occuring Given a Loss of Seal Cooling	Failure to Recover Offsite Power Within 1 Hour
20	2.07E-09	0.58%	17.81%	IE SLOCA	LBA13AA001PFO_D-ALL	OPE-FB-40M		
					CCF to Open Main Steam Relief Isolation Valves	Operator Fails to Initiate Feed & Bleed Within 40 Minutes		

No outliers. Balanced risk.

Approach to Uncertainty and Sensitivity

- Standard Monte-Carlo simulation in RiskSpectrum, including IE, data, CC, human error distributions
- Phenomenological uncertainties (time for human action or success criteria) will be addressed in a sensitivity analysis

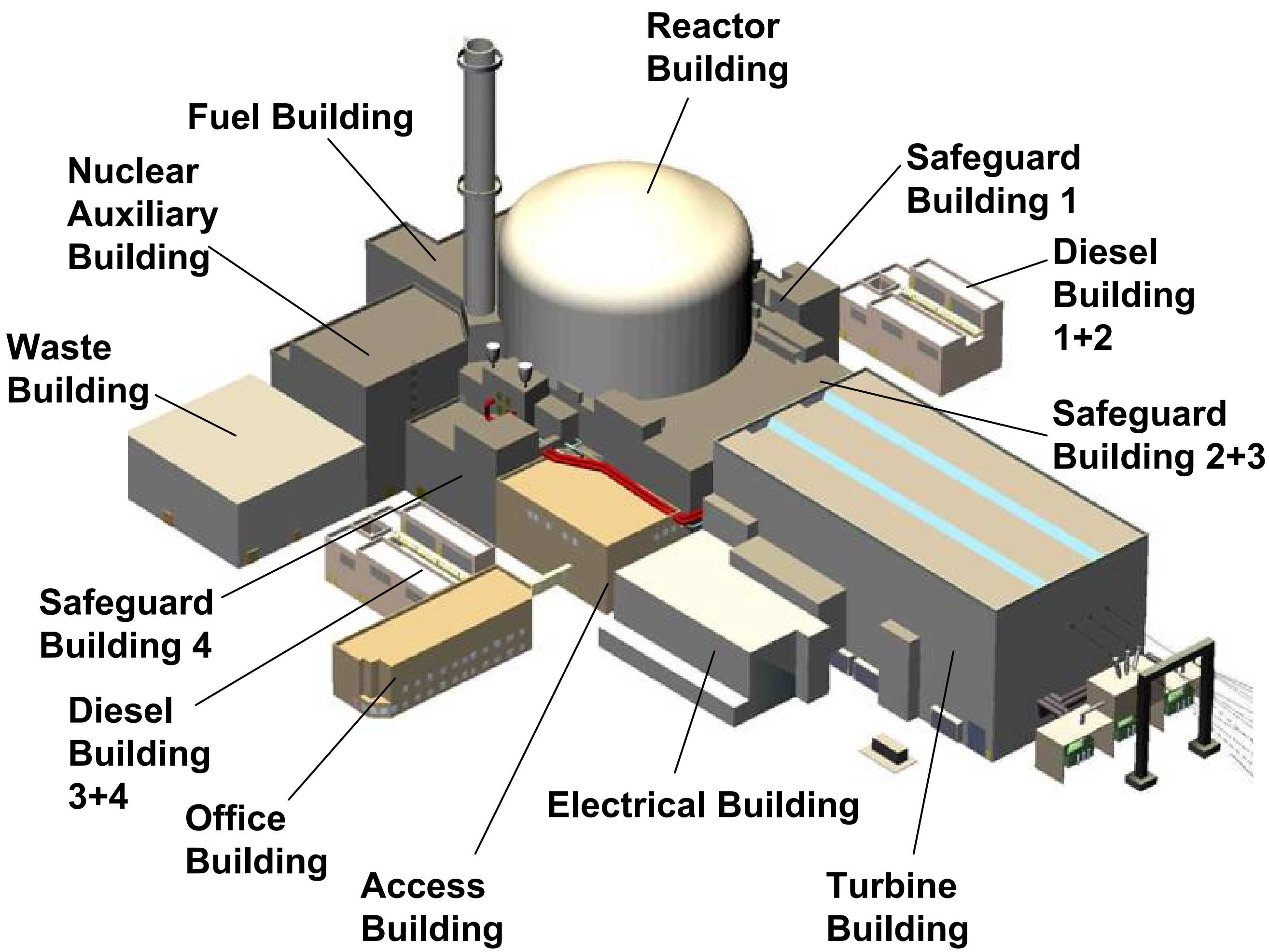
Sensitivity Analysis

> To address
  - Phenomenological uncertainties
  - Various assumptions
> Possible examples (to be determined based on importance)
  - Human error probabilities
  - Selected success criteria
  - Mission time
  - Assumptions on fire/flood IEs
  - Assumptions on the CC grouping
  - Assumptions on factors for RCP seal LOCA/LOOP recoveries
  - Containment performance

Internal Flood & Fire Scope

> Internal flood and fire analysis will be performed for the following buildings/locations
- Containment
- Safeguards Buildings
- Main Control Room/Cable (Spreading) Area
- Turbine Building
- Switchgear Building
- Fuel Building
- Emergency DG Buildings
- ESW Pump Buildings
- Transformer Yard

Spatial Characteristics Important to PRA



Each safety train is independent and located within a physically separate building

Internal Flood & Fire Approach

- Uses a simplified conservative approach based on the currently available information
- Analysis will be updated when final information about pipe routings, cable routings, combustible loadings, and specific equipment locations become available
- Aim of this bounding analysis is to show that the core damage frequency (CDF) due to flood and fire is not going to change the conclusion that the overall CDF meets the U.S. EPR design goal

Internal Flood Approach

> Flooding frequency is calculated for Safeguard Buildings

> Conservative estimates are used for other locations

> Worst scenario is postulated for each location (building) and total area flood frequency is applied

Internal Flood Assumptions

> In calculating SAB internal flood frequencies, the following was considered:
1. Operating systems piping (CCW, ESW, MFW, MS)
2. Standby system piping exposed to tank pressure, up to a pump (SI, EFW)
3. Fire system piping

> Turbine Building flooding frequency is used from NUREG/CR-2300 (Table II.9)

> Breaks in MFW/MS (SAB) are not evaluated because:
- They will not affect other equipment in the building
- Their contribution to LOMFW or SLBO internal event frequencies is negligible

Preliminary Flooding PRA CDF Results

AREA	CONSEQUENCE DESCRIPTION	CCDP	FLOOD FREQ. [1/yr]	FLOOD CDF [1/yr]	FLOOD CDF [%]
SAFEGUARD BULDINGs 1 or 4	Loss of One Safety Division	5.3E-07	4.0E-04	4.2E-10	1.05%
SAFEGUARD BULDINGs 2 or 3	Loss of One Safety Division	5.3E-07	2.5E-04	2.7E-10	0.66%
FUEL BUILDING	Loss of CVCS (TT)	6.5E-10	1.0E-02	6.5E-12	0.02%
TURBINE BUILDING	Loss of BOP	1.3E-06	3.0E-02	3.9E-08	96.69%
SWITHGEAR BUILDING	Loss of BOP	Included in TB			
ESW PUMPS BULDING 1, 2	LUHS1	3.2E-07	1.0E-03	3.2E-10	0.79%
ESW PUMPS BULDING 3, 4	LUHS2	3.2E-07	1.0E-03	3.2E-10	0.79%
EDG BULDING	Loss of 1EDG	no IE			
TRANSFORMER YARD	LOOP (not recoverable)	N/A			
MAIN CONTROL ROOM		N/A			
CABLE (SPREADING) ROOM		N/A			

Total : 4.0E-08

Internal Fire Approach

> Fire frequencies from NUREG/CR-6850 were used when applicable

> Conservative estimates are used for other locations

> Worst scenario is postulated for each location (building) and total area fire frequency is applied

> Fire ignition is considered to grow to a fully developed fire (no severity factors)

> Limited credit is given to fire suppression

> Human recovery actions are credited only for control room fires

Preliminary Fire PRA CDF Results

AREA	CONSEQUENCE DESCRIPTION	CCDP	Supp./ Severity	FIRE FREQ. [1/yr]	FIRE CDF [1/yr]	FIRE CDF [%]
SAFEGUARD BULDINGs 1 or 4	Loss of One Safety Division	8.0E-08	1	1.5E-02	1.2E-09	1.27%
SAFEGUARD BULDINGs 2 or 3	Loss of One Safety Division	8.0E-08	1	1.5E-02	1.2E-09	1.27%
FUEL BUILDING	Loss of CVCS	6.5E-10	1	1.0E-02	6.5E-12	0.01%
TURBINE BUILDING	Loss of BOP	1.3E-06	0.1	5.0E-02	6.5E-09	6.86%
SWITHGEAR BUILDING	Loss of BOP	1.3E-06	1	1.5E-02	2.0E-08	20.59%
ESW PUMPS BULDING 1, 2	LUHS1	3.2E-07	1	1.0E-02	3.2E-09	3.38%
ESW PUMPS BULDING 3, 4	LUHS2	3.2E-07	1	1.0E-02	3.2E-09	3.38%
EDG BULDING	Loss of 1EDG	no IE	1	2.1E-02		
TRANSFORMER YARD	LOOP (not recoverable)	1.3E-05	0.1	2.0E-02	2.5E-08	26.81%
MAIN CONROL ROOM	LBOP w/o F&B	1.2E-04	0.1	2.1E-03	2.5E-08	26.61%
CABLE (SPREADING) ROOM	Loss of Two Safety Division	3.1E-06	1	3.0E-03	9.3E-09	9.82%

Total : 9.5E-08

PRA-Based Seismic Margins Methodology

- Uses internal events PRA model
  - Comprehensive list of SSCs to be evaluated
- Hazard Input: U.S. EPR design based on EUR ground motion spectral shape anchored at 0.3g
- Show margin in design
  - High Confidence Low Probability Failure (HCLPF) capacity above 1.67 times SSE
- Base fragilities primarily on U.S. EPR design and qualification criteria (reasonably achievable)
- Document assumptions to support design development
- Results to be checked against detailed design analysis, when available

LPSD Scope and Approach

Scope:

- **Representative set of plant operating states (POS) conservatively chosen and evaluated**
- **Representative set of initiating events chosen and modeled**
- **Fire and flood evaluated**

Approach:

- **Same methods as power operation PRA except applied to more plant operating states**
- **Several new initiating events (e.g., drain down during mid-loop)**
- **Some new & modified fault trees and operator actions**
- **Preventive maintenance modeling changes**

Preliminary Plant Operating States

POS	DESCRIPTION
A	Full Power to Hot Shutdown (T > 550 F)
B	Steam Generator Heat Removal (T > 248 F)
Ca	RHR Heat Removal with Level in Pressurizer (T ~ 248 to 131 F)
Cb	RHR Heat Removal at Mid-loop with RPV Head On (T ~ 131 F)
D	RHR Heat Removal at Mid-loop with RPV Head Off (T ~ 131 F)
E	Reactor Cavity Flooded (T ~ 131 F)
F	Core off loaded to spent fuel pool

- POS A and B included in power operation PRA
- Remaining POS included in LPSD PRA

LPSD Initiating Events

- Loss of RHR (POS C and D)
  - Includes loss of CCWS, ESWS and AC systems
  - Loss of RHR in POS E not significant
- LOOP (POS C and D)
- Loss of Inventory
  - Level drop (POS Cb and D)
  - Unisolable LOCA (POS C, D and E)
  - RHR LOCA outside containment (POS Ca)
- Loss of Fuel Pool Cooling (all POS)
  - Includes loss of CCWS, ESWS and AC systems

LPSD Initiating Event Insights

- Multiple RHR trains normally operating, only 1 train of RHR required
- Automatic features reduce likelihood of inadvertent "drain-down" and increase likelihood of mitigation (low pressure reducing station letdown auto isolation on level, MHSI auto start on level, RHR protective trip on level)
- Dilution events not considered significant: automatic signals isolate dilution source, reactivity excursion slow and self-regulating

Level 2 Scope

> Full Level 2, with Containment Event Tree including phenomena, systems and human actions

> Frequency of full range of Release Categories covering all release sizes and timing (easily decomposed to LRF / LERF / non-large)

> All plant operating states

Level 2 General Approach

- **RiskSpectrum event tree linking -**
  - Integrates the Level 1 and Level 2 analyses through the definition of Core Damage End States (CDES)
  - Calculates the frequency for each Release Category (RC), combining system and phenomenological analysis in the Containment Event Tree (CET)
- **Supporting studies for -**
  - Accident progression to determine of branch probabilities in the CET
    - Supported by the existing severe accident research and
    - Supplemented by expert judgment
  - Source term for key nuclides released for each RC
  - Containment structural behavior

Level 2 Task Breakdown

- Level 1 / Level 2 Interface
- Severe Accident Phenomenological Evaluation
- Containment Event Tree
- Severe Accident Management / Human Actions
- Supporting MAAP Analysis
- CET Quantification
- Source Term Evaluation
- Analysis of Results
  - Sensitivity
  - Uncertainty Evaluation and
  - Risk Integration
- Documentation

Level 2 Core Damage End States

> Core Damage End States (CDES)

- Set attributes that uniquely define Level 1 core damage sequences
- Group Level 1 sequences sharing similar attributes into a single, uniquely defined bin

> Purpose of the CDES bins

- Transfer groups of sequences to the appropriate Level 2 CET for quantification

Level 2 Systems Analysis

> **Systems addressed in Level 2**
- Dedicated Primary Depressurization System valves
- Passive Autocatalytic Hydrogen Recombiners
- Core Melt Retention System
- Containment Isolation System
- Severe Accident Heat Removal System in the following cooling modes:
  - Gravity fed core melt flooding and
  - Forced core spreading area cooling
- Recovery of RCS injection systems for termination of core damage and prevention of vessel failure

Level 2 Phenomena

- Combines a review of the current state of knowledge (experimental, R&D, analyses) with design specific deterministic analyses
- Phenomenological evaluations developed for:
  - Induced RCS rupture
  - Steam explosion
  - In-vessel recovery
  - Containment loads at vessel failure – including high pressure melt eject
  - Hydrogen generation, mixing, recombination, and combustion modes
  - “Long Term” containment challenges
    - Core-concrete interaction
    - Overpressure / underpressure

Level 2 Phenomenological Evaluation Reports

> Provide a full description of all important phenomena considered in the study

> Demonstrate, by citing test and experimental results, and results of research programs, that the study has been quantified based on state of the art knowledge, and has not solely relied on the prediction of one deterministic analysis tool

> Provide quantified inputs to the CET split fraction process

> Summary of results in terms of direct inputs to split fractions of the CET

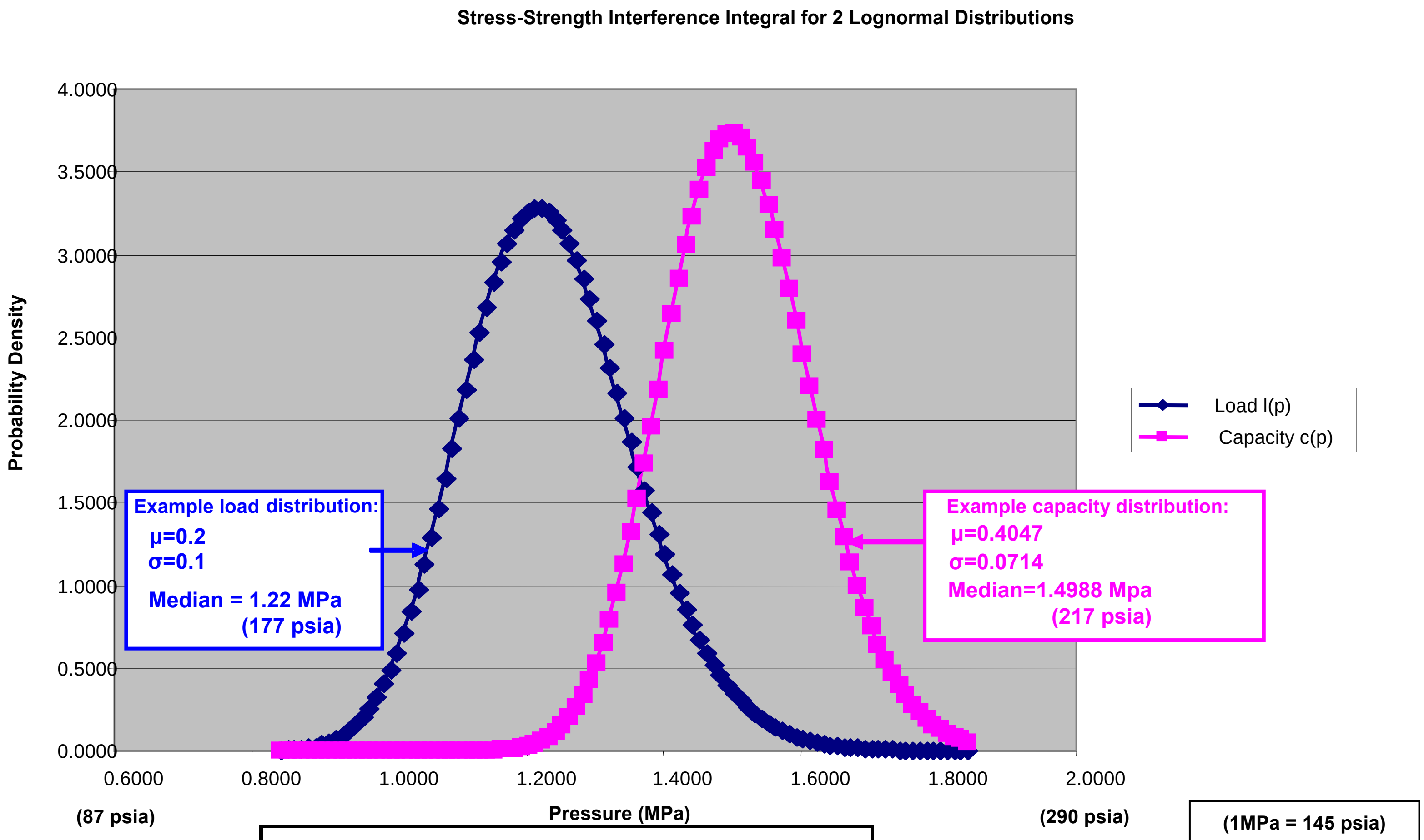
Level 2 Containment Event Tree

- Three time frames
  - “Very early”: from core damage to just before vessel failure
  - “Early”: from just before vessel failure to end of quench
  - “Late”: after end of quench
- Questions
  - Depressurization before vessel failure
  - Induced hot leg failure
  - Induced SG tube failure
  - Containment isolation
  - Very early containment failure
  - SI recovery
  - Vessel breach
  - Ex-vessel steam explosion
  - Early containment failure
  - Melt stabilized
  - Late containment failure
  - SAHRS sprays

Level 2 Human Actions

- Manual actions can influence release risk
  - Key actions will be modeled in Level 2
- SAM is guidance, not procedures – actions subject to TSC evaluation
  - Multiple users/independent verification
- Preliminary list of key actions in Level 2 PRA
  - Depressurize primary system
  - Isolate containment
  - Recover power/safety injection
  - Actuate SAHRS sprays
  - Switch to active SAHRS cooling

Treatment of Uncertainties



$$p_{Fail} = \int_0^\infty l(p) \int_0^p c(p')\,dp'\,dp$$

$$p_{Fail} = 0.051$$

Software and Benchmarking

> RiskSpectrum
- Event Tree / Fault Tree analysis
- Benchmarked against cases supplied by vendor

> MAAP 4.0.7
- Accident progression analysis
- Benchmark
  - Standard EPRI MAAP4 benchmarks – full EPRI QA
  - Specific EPR model benchmarks vs. tests

> Crystal Ball
- Monte Carlo simulation of equipment and containment failure
- Benchmarked by comparisons with cases supplied by vendor

Accident Progression and Phenomenological Analysis

MAAP 4.0.7:

> U.S. EPR design parameter file

> Modeling of U.S. EPR specific systems
  - Debris stabilization system
  - PARs
  - Heavy reflector

> Lumped parameter 27 volume containment model

MAAP analysis:

> Support phenomenological studies and split fraction calculation (batches 1 & 2)

> Event sequence (CET) modeling and quantification (batches 1 & 2)

> Source term characterization (batch 3)

Purpose of Level 3 PRA

- Support design certification
- Support severe accident mitigation design alternatives (SAMDA) analysis for U.S. EPR
- Support comparison to the NRC's quantitative health objectives (QHO), or similar metrics

Codes Used for Level 3

> MACCS2 (Sandia National Laboratories) – an accident consequence code that estimates the potential offsite effects of postulated accident releases

- Radiological doses
- Health effects
- Economic consequences

> RiskIntegrator (originally developed by Duke Energy, modified by AREVA NP) – an Excel spreadsheet that facilitates the linkage between the results of the Level 1 and Level 2 PRAs and the Level 3 PRA

Level 3 Process

- Collect required inputs to develop input files for MACCS2 and RiskIntegrator
  - MACCS2 requires site-specific input (e.g., meteorology, population) – will use Calvert Cliffs as a representative plant site
  - MACCS2 requires input that comes from the output of the MAAP runs
  - MACCS2 requires other plant-specific inputs (e.g., reactor building dimensions, core inventory)
  - RiskIntegrator requires the results of the Level 1 and Level 2 PRAs, as well as the output file of MACCS2
- Execute MACCS2 base case; use RiskIntegrator to evaluate the output

Level 3 Process
(continued)

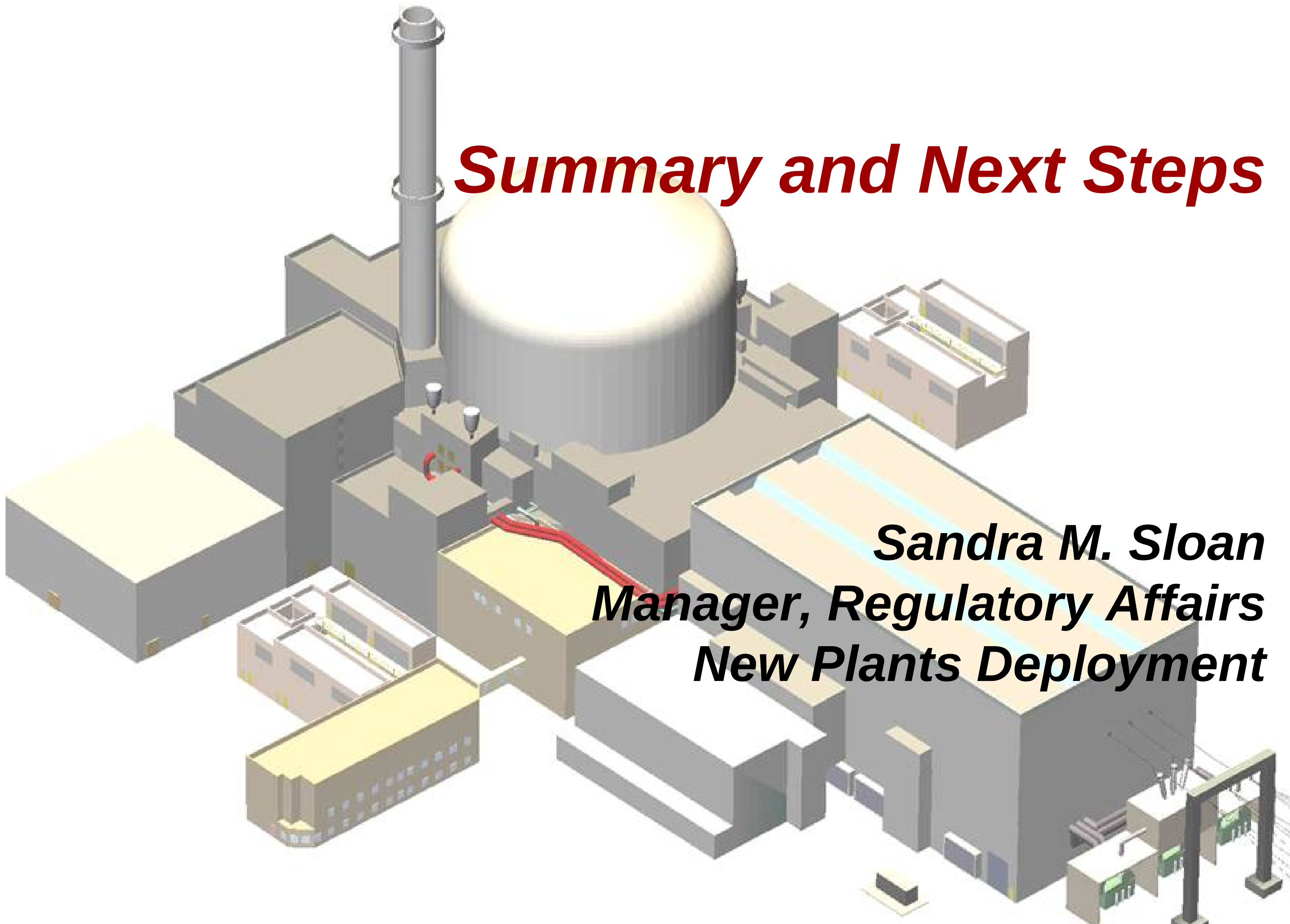
> **Run sensitivity cases**
- Test boundaries of site-specific input data
- Evaluate the sensitivity of some input parameters
- Evaluate the sensitivity of some model assumptions

> **Documentation of the model**
- Development of input parameters
- Interface with Level 1 and 2 PRA results
- Base case results
- Sensitivity studies

Conclusions

- PRA to be done considering guidance from RG 1.200/ASME PRA Standard
- PRA will benefit from international cooperation with European counterparts
- Preliminary stage PRA shows no risk outliers and confirms robustness of the U.S. EPR design
- Preliminary Level 1 results show that the U.S. EPR design goals should be met with margin
- PRA is formally tied into ongoing design effort via project design directive and design change control process
- As detailed design evolves, PRA will be updated/ maintained throughout various project stages up to fuel load

Summary and Next Steps



Sandra M. Sloan
Manager, Regulatory Affairs
New Plants Deployment

Summary

- U.S. EPR PRA uses well-established models and methods to assess plant risk

- The PRA-related reports to be submitted will provide descriptions of PRA methods and preliminary results for Level 1, at-power internal events

- This type of meeting/interaction helps us understand NRC expectations and thus produce a concise submittal and reduce the likelihood of unexpected issues

Next Steps

- Submit PRA Methods Report in December 2006
  - Submit PRA preliminary results for Level 1, at-power internal events in March 2007
- Meetings forthcoming in 2006
  - Unique Design Features and Containment Analysis, October 25
  - Critical Heat Flux Topical Report Pre-Submittal, November 14
  - Equipment Qualification Program Report Pre-Submittal, November 29
  - Human Factors Report Pre-Submittal, December 7

Acronyms

ACCU	Accumulator
ALWR	Advanced Light-Water Reactor
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
CCF	Common Cause Failure
CCW	Component Cooling Water
CDES	Core Damage End States
CDF	Core Damage Frequency
CET	Containment Event Tree
CVCS	Chemical and Volume Control System
DG	Diesel Generator
DNBR	Departure from Nucleate Boiling Ratio
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
EFWT	Emergency Feedwater Tank
ESFAS	Engineered Safety Function Actuation System
ESW	Essential Service Water
F&B	Feed & Bleed
HCLPF	High Confidence Low Probability Failure
HRA	Human Reliability Analysis
HVAC	Heating Ventilation and Air Conditioning
I&C	Instrumentation and Controls
IRWST	In-containment Refueling Water Storage Tank
LERF	Large Early Release Frequency
LHSI	Low Head Safety Injection
LOMFW	Loss of Main Feedwater
LOOP	Loss of Offsite Power
LPSD	Low Power Shutdown
LRF	Large Release Frequency
MHSI	Medium Head Safety Injection

Acronyms *(continued)*

>	MSIV	Main Steam Isolation Valve
>	MSLB	Main Steam Link Break
>	MSSV	Main Steam Safety Valve
>	OCW	Operational Chilled Water
>	POS	Plant Operating State
>	PRA	Probabilistic Risk Assessment
>	PWR	Pressurized Water Reactor
>	QHO	Qualitative Health Objectives
>	RC	Release Category
>	RCP	Reactor Coolant Pump
>	RCS	Reactor Coolant System
>	RHR	Residual Heat Removal
>	SAB	Safeguards Building
>	SAHRS	Severe Accident Heat Removal System
>	SAM	Severe Accident Management
>	SAMDA	Severe Accident Mitigation Design Alternatives
>	SBO	Station Blackout
>	SCW	Safety Chilled Water
>	SGTR	Steam Generator Tube Rupture
>	SI	Safety Injection
>	SLBI	Steam Line Break Inside Containment
>	SLBO	Steam Link Break Outside Containment
>	SLOCA	Small Loss of Coolant Accident
>	SSE	Safe Shutdown Earthquake
>	TB	Turbine Building
>	TH	Thermal Hydraulics
>	TSC	Technical Support Center
>	UHS	Ultimate Heat Sink