# Probabilistic Risk Assessment

- Introduction
- Methodology and Preliminary Results of Level 1 Internal Events PRA
- Methodologies and Status of PRAs for External Events and Low Power Shutdown Modes

# Regulatory Requirement for PRA

- **PRA**
  - 10 CFR 52.47 LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS
    - (a) … must include the following information:
      - (27) A description of the design specific probabilistic risk assessment (PRA) and its results.
  - RG 1.206 COMBINED LICENSE APPLICATIONS FOR NUCLEAR POWER PLANTS
    - C.I.19 PROBABILITY RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION
    - To provide prospective COL applicants with guidance concerning the format and content of the application

# Regulatory Requirement & Guideline

- **PRA Scope and Quality**
  - SRP 19.0
  - SRP 19.1
  - DC/COL-ISG-03
  - RG 1.200 Rev. 2
  - ASME/ANS RA-Sa-2009 PRA Standards
  - RG 1.174 Rev . 1

# Purpose of APR1400 PRA

- **To meet 10 CFR 52.47 (a) (27).**
- **To demonstrate that the risk associated with the design is acceptably low compared to the Commission's goals.**
- **To identify and address potential design and operational vulnerabilities.**

# APR1400 PRA Scope

| Operation Mode | | Level 1 (CDF) | Level 2 (LRF) |
|---|---|---|---|
| At-power | Internal Events | O | O |
| | Internal Fire | O | |
| | Internal Flooding | O | |
| | Seismic* | O | - |
| Low Power and Shutdown | Internal Events | O | O |
| | Internal Fire | △ | △ |
| | Internal Flooding | △ | △ |

* ) PRA-based SMA, △) Qualitative approach

# APR1400 PRA History

• Upgrade PRA method & data

System80+ PRA (1997)

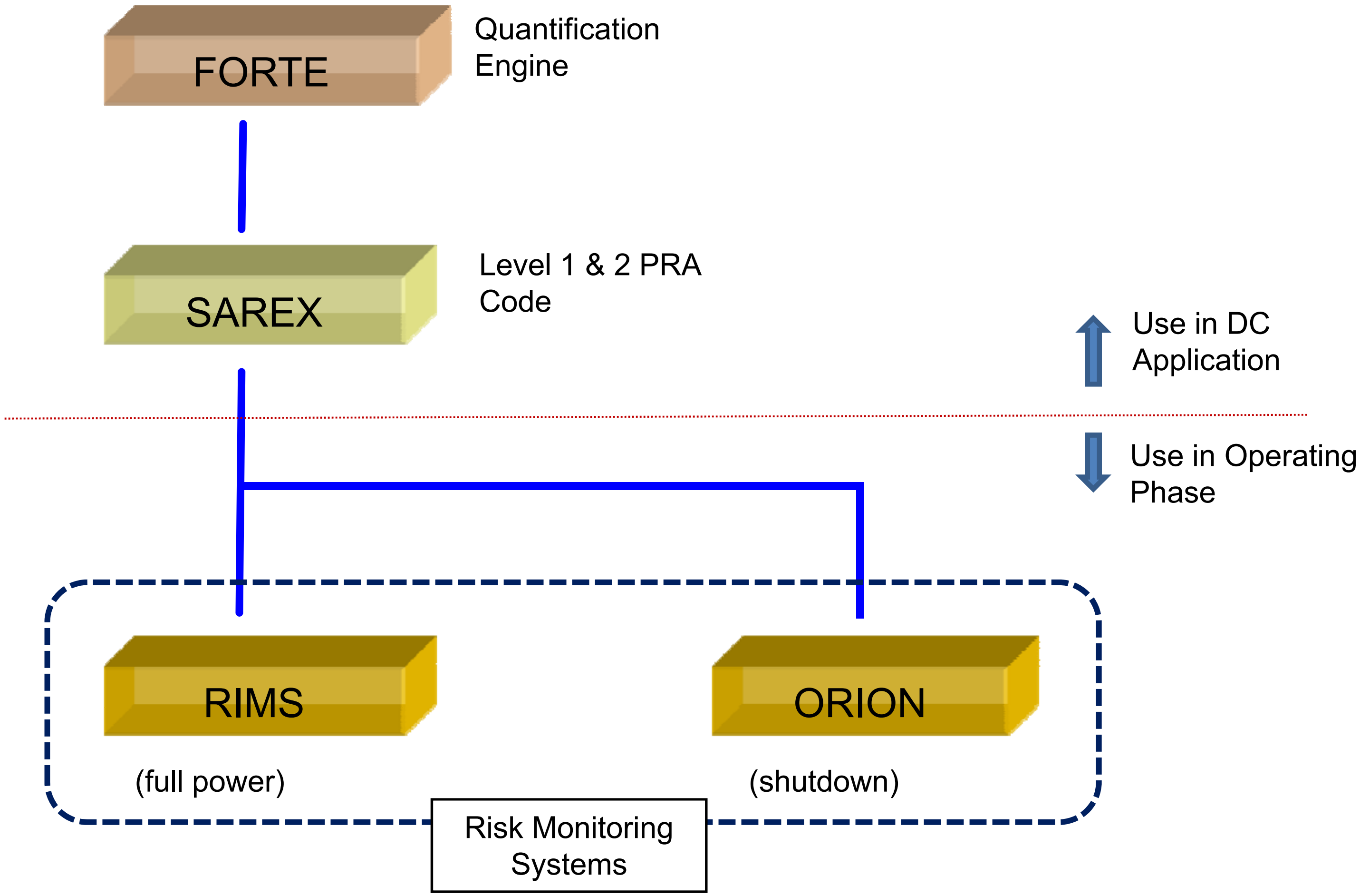
APR1400 PRA (2002)

APR1400 PRA (NRC DC)

SKN3&4 PRA (2011)

• Detail Design Information

# PRA Code Package of KEPCO



# PRA Code Package of KEPCO

- **FORTE**
  - PRA Quantification Engine
  - Support compatible input & output formats
    - Input : FTAP, SET
    - Output : EPRI R&R W/S Std. (RAW), Text format.
- **SAREX**
  - Integrated Level 1/2 PRA Code Package
  - Main Modules
    - Fault Tree Analyzer (FTA)
    - DataBase Analyzer (DBA)
    - Event Tree Analyzer (ETA) : L1 & L2
    - CutSet Analyzer (CSA)
    - ReCovery Analyzer (RCA)
    - Uncertainty Analyzer (UNCA)

# Level 1 Internal Events Analysis

- Methodology
- Preliminary Results

# Methodology

- NUREG/CR-2300, "PRA Procedures Guide"
- NUREG/CR-2815, "Probabilistic Safety Analysis Procedures Guide"
- ASME/ANS RA-Sa-2009 “PRA Standards”

# Initiating Events of APR1400

| Category | Events |
|---|---|
| Loss of Coolant Accident | 1. Large Loss of Coolant Accident (LLOCA) |
| | 2. Medium Loss of Coolant Accident (MLOCA) |
| | 3. Small Loss of Coolant Accident (SLOCA) |
| | 4. Steam Generator Tube Rupture (SGTR) |
| | 5. Interfacing System LOCA (ISLOCA) |
| | 6. Reactor Vessel Rupture (RVR) |
| Transient Accident | 7. Large Secondary Side Breaks (upstream of MSIV) (MSLBU) |
| | 8. Large Secondary Side Breaks (downstream of MSIV) (MSLBD) |
| | 9. Loss of Main Feedwater Transients (LOFW) |
| | 10. Feedwater Line Break (FWLB) |
| | 11. Loss of Instrumentation Air (LOIA) |
| | 12. Loss of Condenser Vacuum (LOCV) |
| | 13. Loss of Class1E 125VDC Vital Bus A (LODCA) |
| | 14. Loss of Class1E 125VDC Vital Bus B (LODCB) |
| | 15. Partial loss of Component Cooling Water (CCW Div. A) (PLOCCW) |
| | 16. Total loss of Component Cooling Water (CCW Div. A & B) (LOCCW) |
| | 17. Partial loss of Essential Service Water (ESW Div. A) (PLOESW) |
| | 18. Total loss of Essential Service Water (ESW Div. A & B) (LOESW) |
| | 19. Loss of Offsite Power (LOOP) |
| | 20. Station Blackout (SBO) |
| | 21. General Transients (TRAN) |
| | 22. Anticipated Transients Without Scram (ATWS) |

# Initiators not included

- **LOHV** **Loss of HVAC**
- **LOKV** **Loss of a 4.16 KV Bus**
- **LOAC** **Loss of a 120V AC Bus**
- **LONSW** **Loss of Non-safety Service Water System**
- **VSLOCA** **Very Small LOCA**
  - A charging pump will make up to 150 gpm.
- **MTRIP** **Manual Trip (Manual Shutdown)**
  - Assumed to be bounded by General Transient

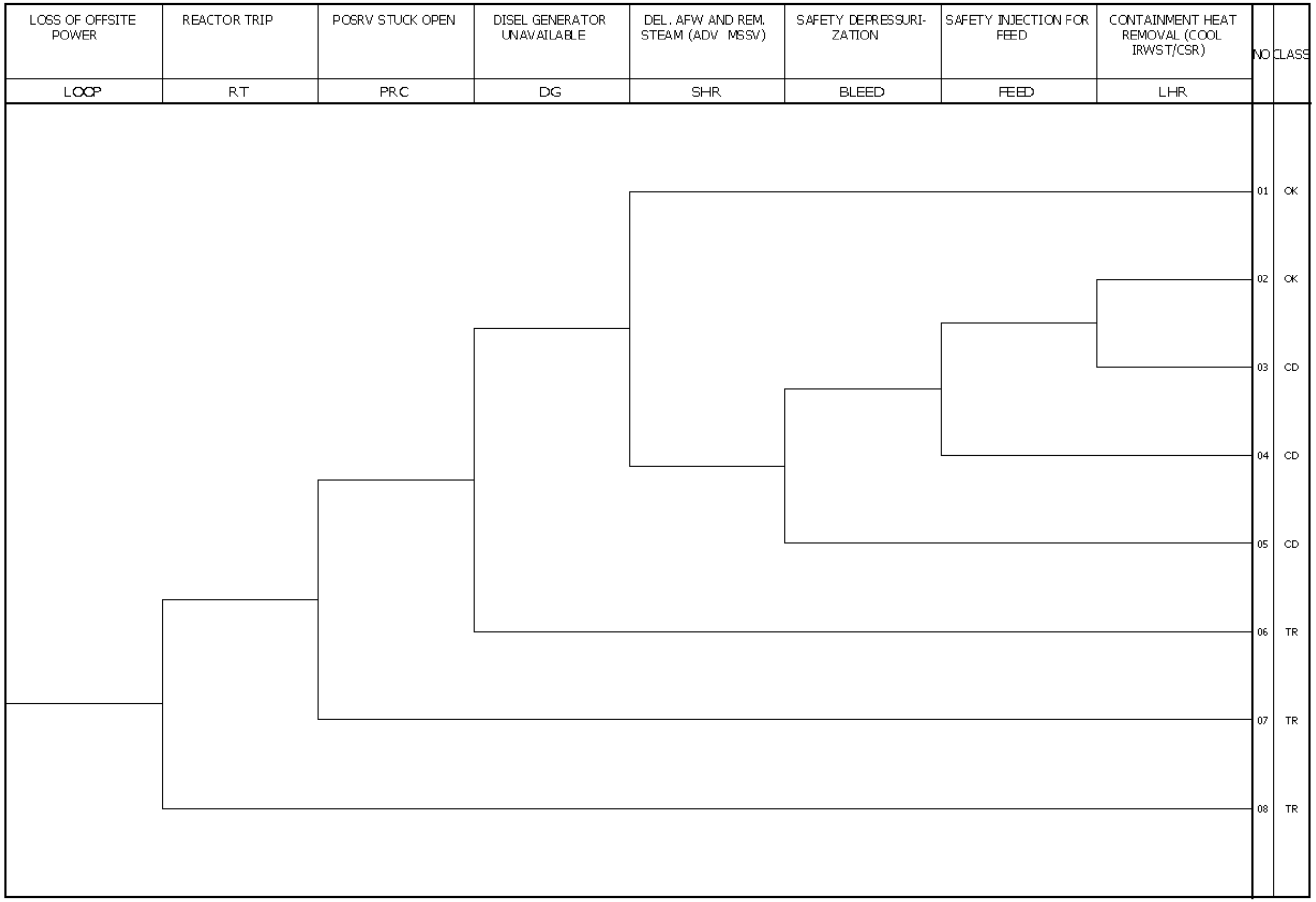
# Accident Sequence Analysis

- **Success Criteria**
  - Definition of Core Damage
    - Consistent with ASME/ANS PRA Standards: Peak node temperature exceeds 1204°C (2200°F)
  - Based on design basis analysis or best estimate analysis
    - Utilize MAAP 4.0.8 , RELAP5 mod3, and safety analysis code
    - Review FSAR Chapter 15

- **Event Tree Development**
  - Small Event Tree/Large Fault Tree
  - Fault Trees are linked to Event Trees

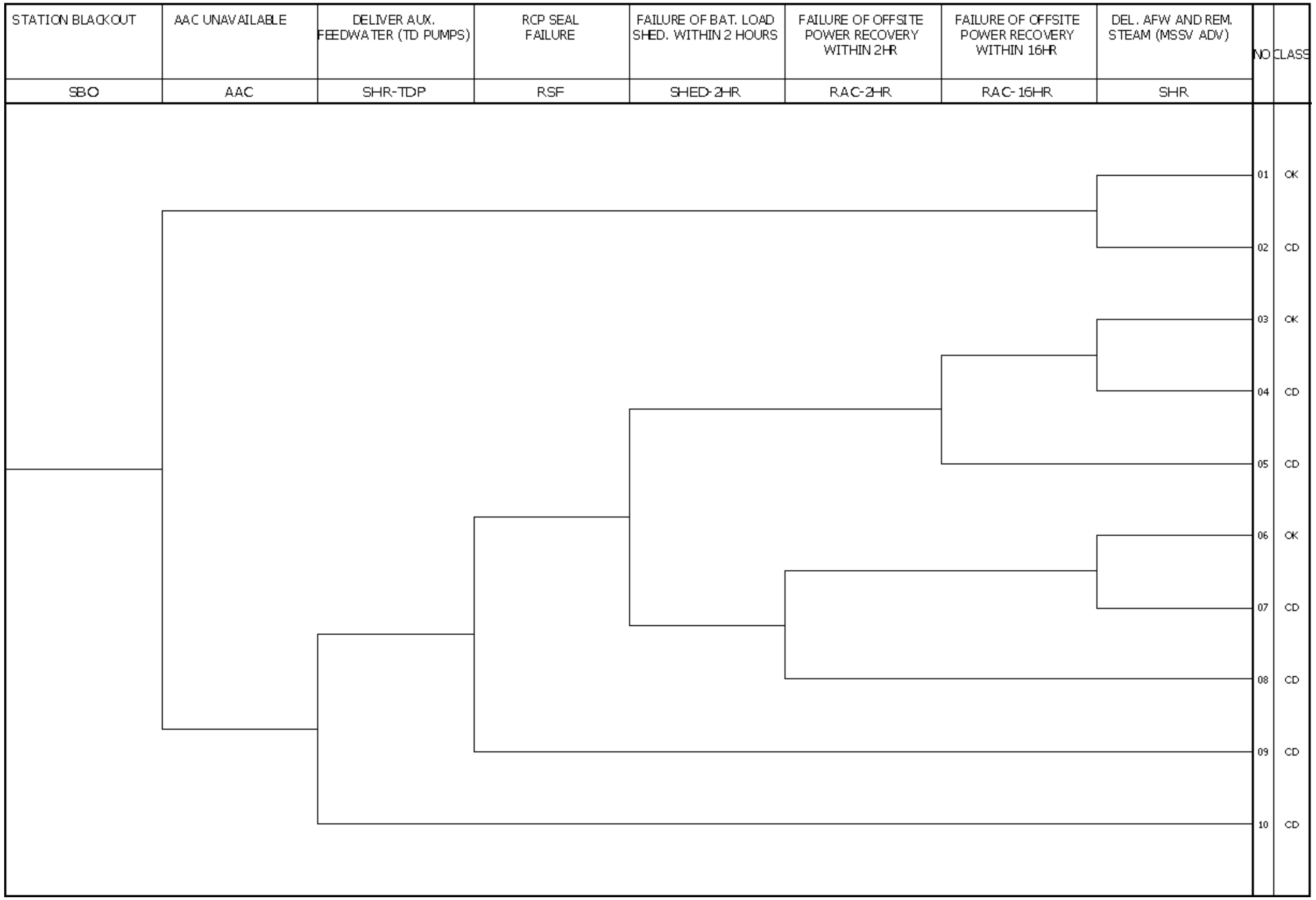
# Event Trees

- **Loss of Offsite Power (LOOP)**



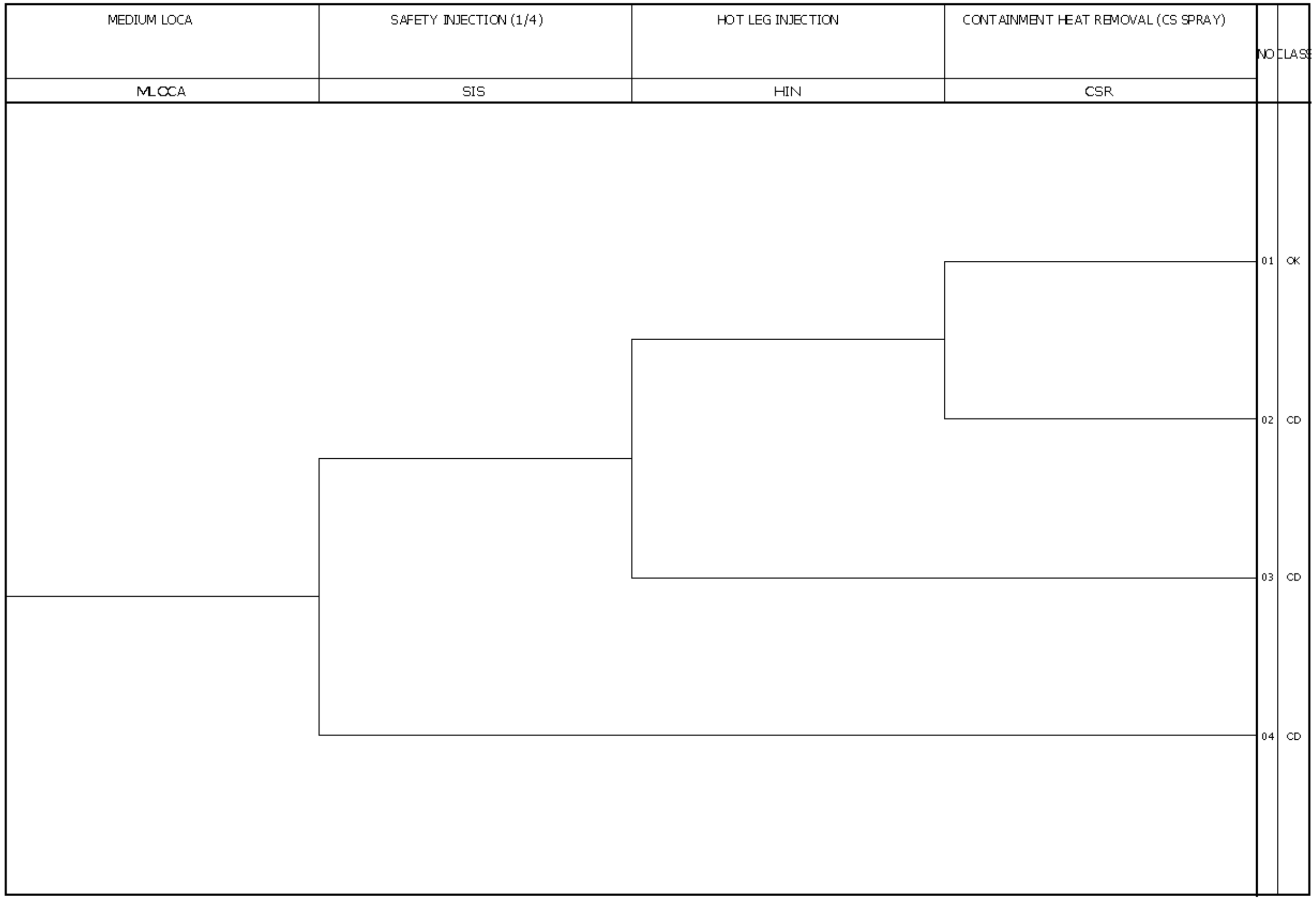
# Event Trees

- **Station Blackout (SBO)**



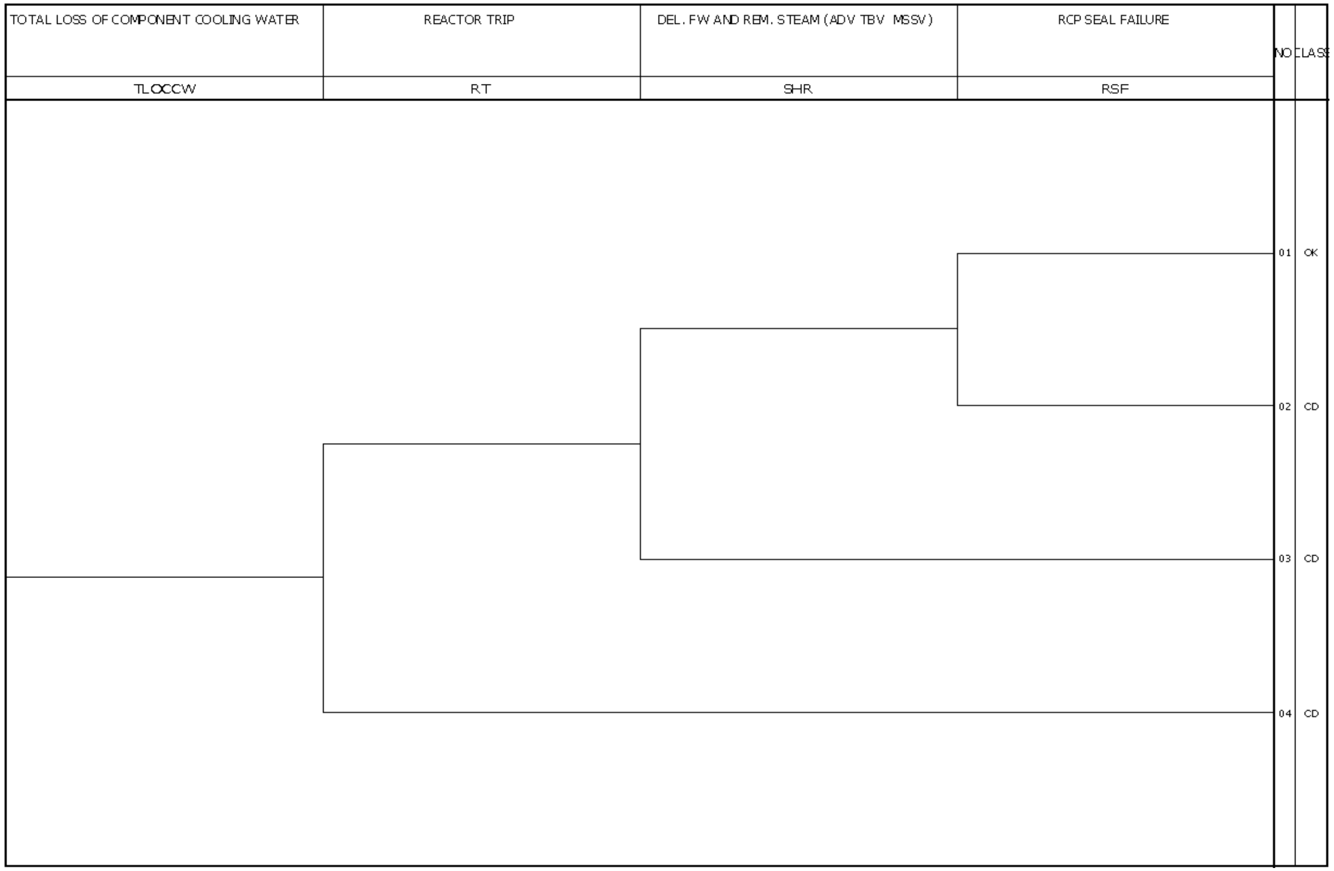
# Event Trees

- **Medium Loss of Coolant Accident (MLOCA)**



# Event Trees

- **Total Loss of Component Cooling Water (TLOCCW)**



# Data

- **Initiating Events**
  - "Initiating Event Datasheet Update 2010" *
    - NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants"
    - NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Plants"
    - NUREG-1829, "Estimating Loss-of-Coolant Accident(LOCA) Frequencies through the Elicitation Process" Component Reliability Data
  - "Component Reliability Data Sheets Update 2010" *
- **Component Unavailability due to Test & Maintenance**
  - "Component Availability Data Sheets Update 2010" *

*http://nrcoe.inel.gov/resultsdb/publicdocs/AvgPerf/ComponentReliabilityDataSheets2010.pdf

# Data

- **Common Cause Failure Data**
  - Multiple Greek Letter (MGL) Method
  - “CCF Parameter Estimations 2010”
    - http://nrcoe.inel.gov/resultsdb/publicdocs/CCF/ccfparamest2010.pdf
  - NUREC/CR-5485, “Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment”
- **Special Data**
  - Non-recovery Probabilities for Off-site Power
    - NUREG/CR-6890, “Reevaluation of Station Blackout Risk at Nuclear Power Plants”

# Human Reliability Analysis (HRA)

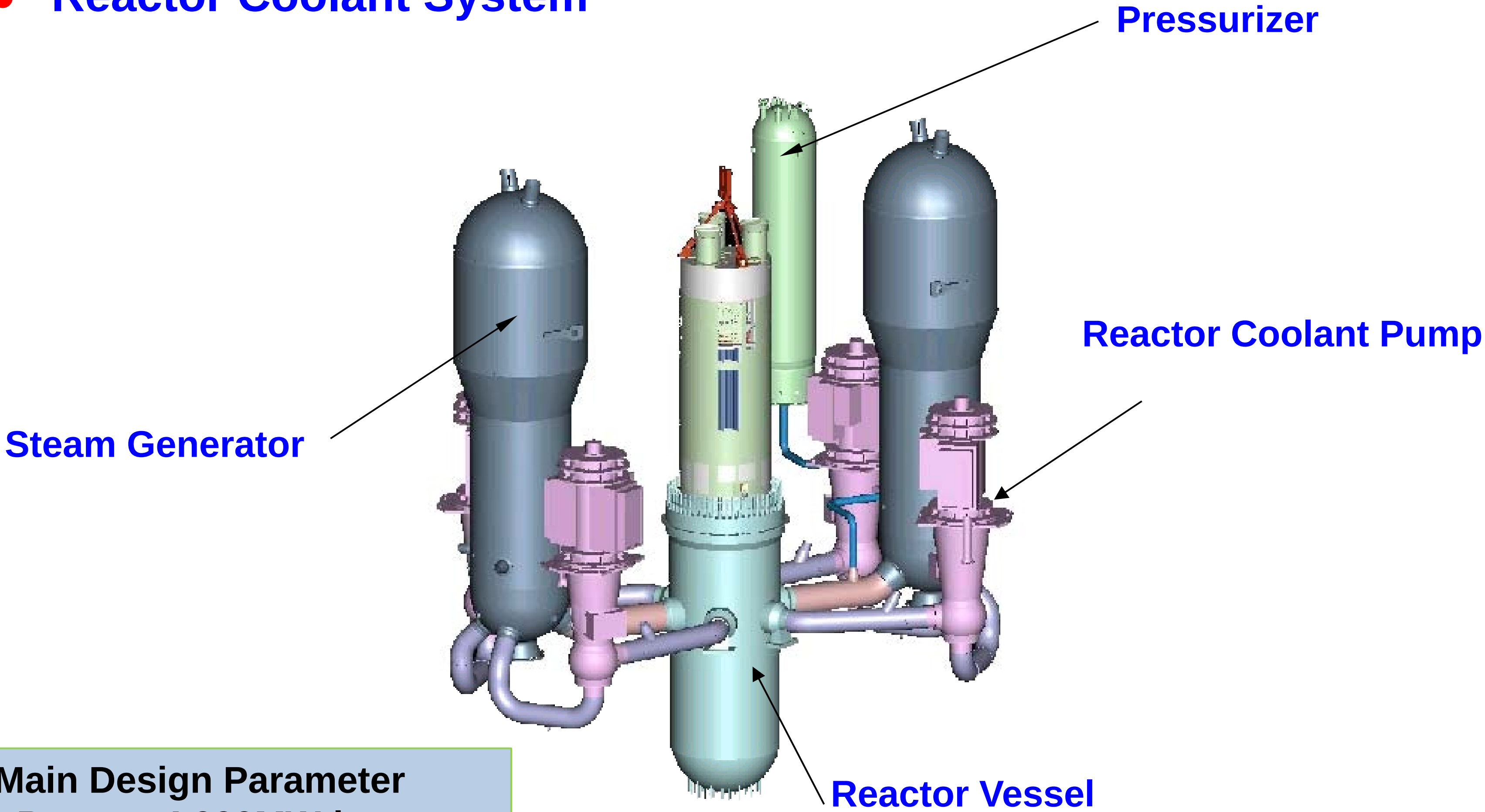
- **Pre-Initiator**
  - ASEP HRA Procedure (NUREG/CR-4772)
- **Post-Initiator**
  - Cause Based Decision Tree Methodology (CBDTM) or Human Cognitive Reliability Operator Response Experiment (HCR/ORE) for Cognitive errors evaluation
  - THERP (NUREG/CR-1278) for execution error evaluation

# Uncertainty Analysis

- **Parametric uncertainty**
  - State of Knowledge Correlation (SOKC) will be addressed.
- **Modeling uncertainty**
  - Plant-specific sources of uncertainty
    - Based on key assumptions
  - Generic sources of uncertainty
    - NUREG-1855, EPRI 1009652
  - Uncertainty characterization
- **Selected sensitivity cases will be performed, e.g., HEPs and CCF factors.**

# Design Features of APR1400

- **Reactor Coolant System**




**Main Design Parameter**
**- Power : 4,000MWth**

# Systems Modeled in PRA

## Front Line Systems

- Safety Injection (SI)
- Shutdown Cooling (SC)
- Containment Spray (CS)
- Chemical & Volume Control (CV)
- Reactor Coolant (RC)
- Auxiliary Feedwater (AF)
- Main Feedwater (MF)
- Steam Generator Blowdown (SD)
- Reactor Protection (RP)

## Support Systems

- Component Cooling (CC)
- Essential Service Water (SX)
- Turbine Generator Building Open Cooling Water (WH)
- Turbine Generator Building Closed Cooling Water (WT)
- Essential Chilled Water (WO)
- Electrical System (AC/DC)
- Instrument & Control
- Instrument Air (IA)
- Heating, Ventilating and Air Cleaning (HVAC)
- Engineered Safety Feature Actuation System (EF)

# Design Features of APR1400

| | System or Component | Configuration |
|---|---|---|
| Primary System | RCS | 2 SGs, 2 Hot Legs, 4 Cold Legs, 4 RCPs |
| | POSRV | 4 |
| Electrical System | Class 1E 4.16kV Switchgears | 4 trains |
| | EDG | 4 |
| | AAC | 1 |
| Safety System | Safety Injection System | 4 trains (Via DVI) |
| | Safety Injection Tank | 4 trains (Via DVI) |
| | Containment Spray System | 2 trains |
| | Shutdown Cooling System | 2 trains (Via DVI) |
| | IRWST | 1 |
| Secondary System | Aux. Feedwater System | 1 MDP and 1 TDP per SG |
| | Main Steam Line | 2 per SG |
| | MSSV | 10 per SG (Total 20) |
| | MSADV | 2 per SG (Total 4) |
| | TBV | 8 |
| Supporting System | ECW, CCW, ESW | 2 divisions (2 trains / division) |

# Design Features of APR1400

- **Safety Injection System (SI)**
  - Inject borated water into the RCS through Direct Vessel Injection (DVI) nozzles.
  - Provide removal of heat from core for extended periods of time.
  - Provide feed flow for feed-and-bleed operation in conjunction with POSRVs.

  - Consist of four mechanically separate trains, four SITs and associated valves, piping and instrumentation.
    - Each train consists of one SI pump and one SIT.
    - Each SI pump takes suction from the IRWST and discharges to DVI nozzle.
    - Two SI pumps can inject the IRWST water to hot legs.

# Design Features of APR1400

- **Containment Spray System (CS)**
  - Remove heat and fission products from containment atmosphere by providing the spray flow from IRWST.
  - Limit the leakage of airborne activity from the containment.

  - Consist of two redundant trains.
  - Each train include a CS pump, a heat exchanger, a main/auxiliary spray header with nozzle, valves, piping and instrumentation per train.

# Design Features of APR1400

- **Safety Depressurization and Vent System (SDVS)**
  - Provide the following functions:
    - Overpressure protection to maintain the RCS Integrity .
    - A safety-grade means of venting non-condensable gases.
    - A means to depressurize RCS for Feed & Bleed operation following LOFW.
    - A means to rapidly depressurize the primary system to low pressure before reactor vessel breach to prevent DCH following a severe accident.
  - Consist of POSRVs and spargers.

# Design Features of APR1400

- **Auxiliary Feedwater System (AF)**
  - Provide an independent means of supplying auxiliary feedwater to SGs when the normal feedwater is unavailable.
  - Provide auxiliary feedwater to each SG by one motor-driven AF pump or one turbine-driven AF pump from AFWST
  - Control auxiliary feedwater flow using a motor-operated AF isolation valve and a solenoid-operated AF modulating valve.

# Design Features of APR1400

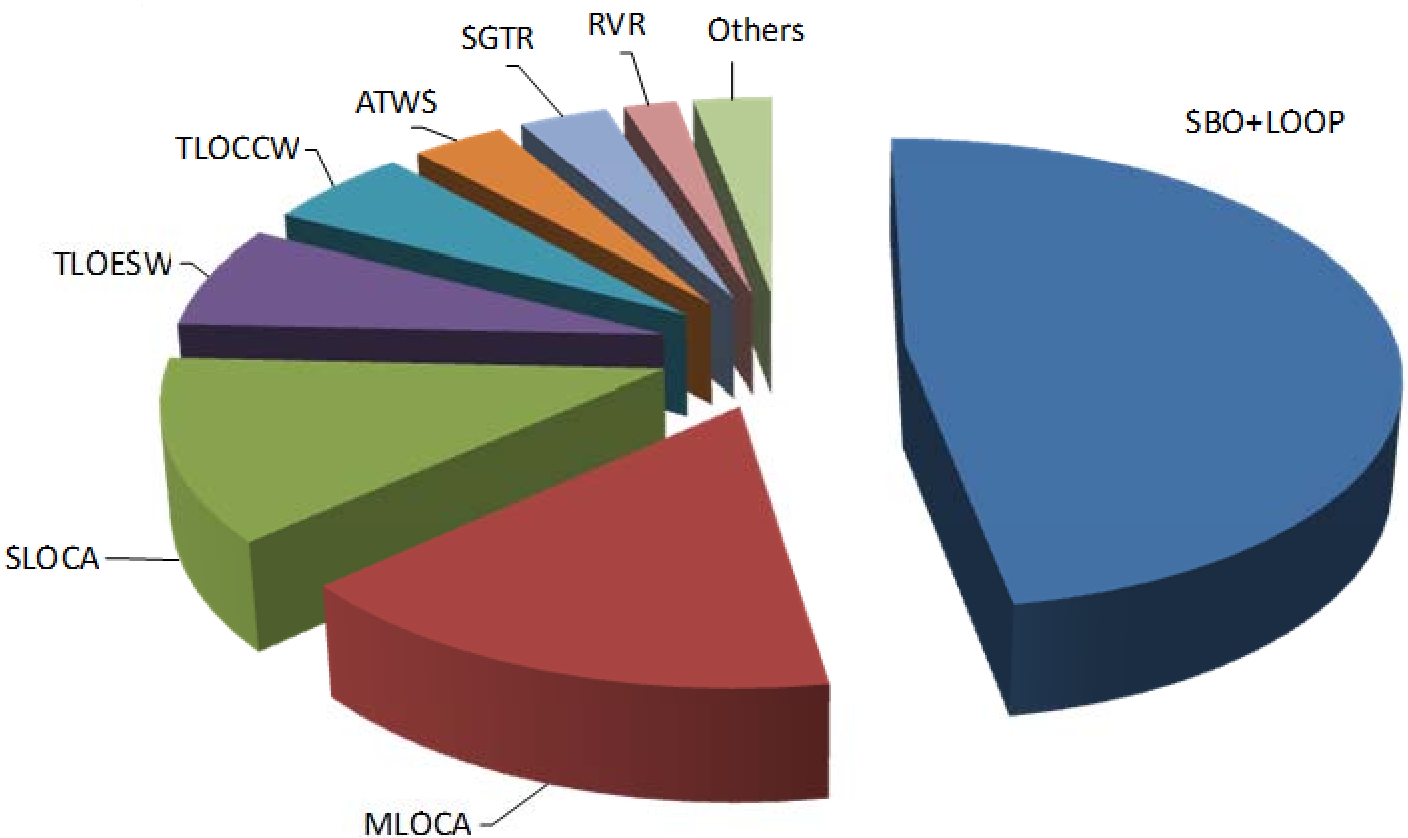
- **Electrical System**
  - It consists of four distribution trains (4 switchgears).
  - Each train has a standby emergency power source supplied by an EDG.
  - 4 EDGs (Emergency Diesel Generators) supply emergency power to each of 1E power distribution trains
  - 1 AAC DG (Alternate AC Diesel Generator) can be used as a common AC source for two Class 1E switchgears.
  - 4 Class 1E Batteries are provided at 125V DC power system.
  - 125V DC power source and 120V AC Instrumentation and control power system consist of four independent and physically separated trains.

# Design Features of APR1400

- **Digital I&C System**
  - APR1400 control room is a highly integrated control room (HICR) that includes:
    - Operator consoles
    - Safety console (Analog)
  - Digital I&C system is modeled on the basis of design concept of Shin-Kori 3&4.

# Preliminary Results of
# Level 1 Internal PRA for At-Power

# L1 PRA Preliminary Results



# L1 PRA Preliminary Results

## ● Top 5 MCSs

(%: Initiator)

| NO | CUTSET | | | Description |
|---|---|---|---|---|
| 1 | **%MLOCA** | SIOPH-S-HLI | MLOCA-PLENTH-RATIO | Hot Leg Injection Failure |
| 2 | **%TLOESW** | CVOPH-S-RCPSEAL | SEAL-AFSUC | RCP Seal Failure |
| 3 | **%GTRN** | I-ATWS-RPMCF | MTC-ATWS | MTC Condition |
| 4 | **%RVR** | | | Reactor Vessel Rupture |
| 5 | **%TLOCCW** | CVOPH-S-RCPSEAL | SEAL-AFSUC | RCP Seal Failure |

# L1 PRA Preliminary Results

## ● Importance Analysis for HRA

| ORDER | EVENT | DESCRIPTION |
|---|---|---|
| 1 | DAOPH-S-AACDG | OPERATOR FAILS TO CONNECT AAC DG WITH C1E 4.16 BUS |
| 2 | CVOPH-S-RCPSEAL | OPERATOR FAILS TO RECOVER RCP SEAL COOLING (CCW CONNTECT. OR AUX. CHG PUMP) |
| 3 | SIOPH-S-HLI | OPERATOR FAILS TO HOT LEG INJECTION |
| 4 | DCOPH-S-SHEDLOAD-MD | OPERATOR FAILS TO SHED LOAD FOR EXTENDING BATTERY TIME (MEDIUM DEPENDENCY WITH DAOPH-S-AACDG) |
| 5 | FWOPH-S-ERY | OPERATOR FAILS TO ALIGN STARTUP FEED WATER PUMP |

# L1 PRA Preliminary Results

- **Importance Analysis for CCF**

| ORDER | EVENT | DESCRIPTION |
|---|---|---|
| 1 | WOCHWQ4-CH03A/4A/3B/4B | DEMAND CCF OF ECW CHILLERS 3A/4A/3B/4B FOR ESW PUMP ROOM |
| 2 | WOCHWQ4-CH01A/2A/1B/2B | DEMAND CCF OF ECW CHILLERS 1A/2A/1B/2B |
| 3 | AFTPKD2-TDP01A/B | 2/2 CCF OF RUNNING AFW TDP PP01/A/B |
| 4 | WOCHKQ4-CH03A/3B/4A/4B | RUNNING CCF OF ECW CHILLERS 3A/4A/3B/4B FOR ESW PUMP ROOM |
| 5 | WOCHKQ4-CH01A/1B/2A/2B | RUNNING CCF OF ECW CHILLERS 1A/2A/1B/2B |

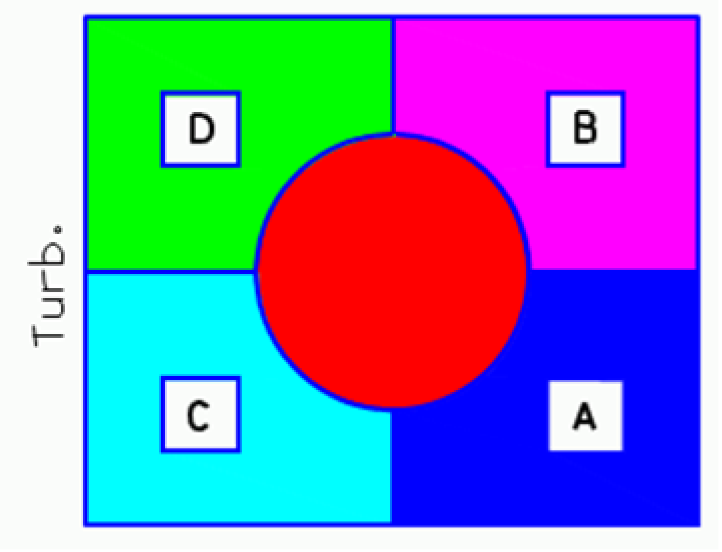
# Level 1 Flooding Analysis

# Internal Flooding PRA

- **Guidance**
  - RG 1.200 Revision 2
  - ASME/ANS RA-Sa-2009 “PRA Standards”
  - SRP 19.1
  - EPRI 1021086 “Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments” Revision 2

# APR1400 Characteristics

- **Auxiliary Building**
  - Quadrant Design
    - Physical quadrant design
    - To strengthen the ability to deal with external accidents such as fire and flooding



# APR1400 Characteristics

- **Auxiliary Building Arrangement**
  - 1st floor
    - Four quadrants separated by watertight boundaries
    - Pumps (SI/CCW/SC/CS/Charging PP)
  - 2nd floor
    - HELB barriers for AFW
    - Class 1E Breakers, AF pumps, Class 1E battery (C/D)
    - Emergency Overflow Lines (EOLs) limit accumulation and direct water to 1st floor.
  - 3rd floor
    - EOLs limit accumulation and direct water to 1st floor via 2nd floor.
    - Class N1E Breakers, HVAC, 2 EDGs (C/D, base floor), Class 1E battery (A/B)
  - 4th floor: MCCs, Spent Fuel pool area
  - 5th floor: RSP, cable spreading room, MG Set, MS valves
  - 6th floor: MCR, I&C equipment room
  - 7th floor: HVAC
  - 8th floor: HVAC

# APR1400 Characteristics

- **EDG Building (A/B) and Compound Building**
  - Limited flood sources
  - Limited potential for propagation
- **Turbine Building**
  - Likely effect is loss of MFW.
  - Limited potential for propagation

# APR1400 Characteristics

- **Flood sources in Aux. Building are closed-loop.**
  - Limited volume
  - Limited propagation
- **Quadrant walls sealed for 9-foot flood elevation on 55-foot elevation.**
- **10-inch EOL limits accumulation on upper elevations.**
- **Ramps/Curbs limit propagation between quadrants on upper elevations.**
- **Flooding alarms in the 1st floor of each quadrant provide rapid indication of flooding to main control room.**

# Flooding Insights

- **Segmentation of rooms limits spray damage.**
  - Damage in one quadrant does not require shutdown.
  - Some electrical bus and distribution failures require TS shutdown.
- **Good instrumentation to detect breaks**
  - Most breaks do not require isolation.
- **Beyond-design-basis fire protection breaks can propagate beyond barriers.**
  - Barriers assumed to fail if challenged.
  - Drainage through emergency overflow lines (EOLs) limits equipment damage due to propagation.
  - Failures not expected to be risk significant.
  - Some failures of ECCS piping can challenge containment integrity.

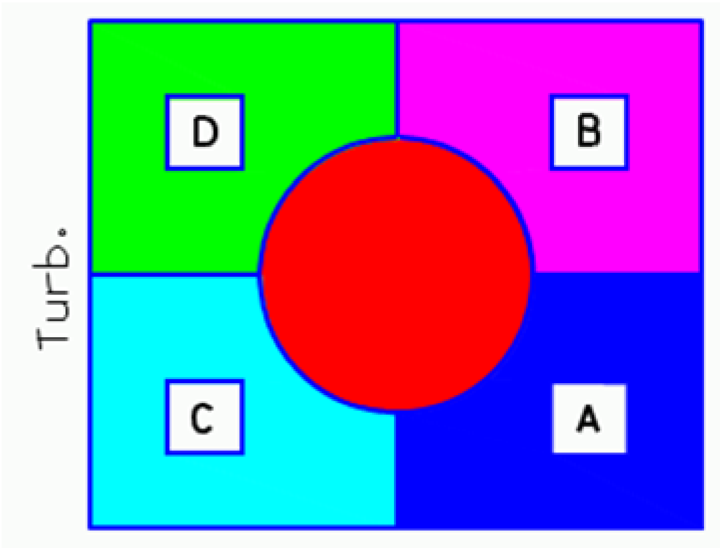
# Level 1 Fire Analysis

# Fire PRA

- **Guidance**
  - NUREG/CR-6850 (EPRI 1011989): Fire PRA
  - ASME/ANS RA-Sa-2009 “PRA Standards”
  - NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidlines“
  - NUREG/CR-6850 (EPRI 1019259) Supplement 1

# APR1400 Characteristics

- **Auxiliary Building Arrangement**
  - Quadrant concept
    - Equipment is arranged according to quadrant concept.
    - Switchgears, Load Centers, ESF pumps, and MCCs are located in separate rooms.



# APR1400 Characteristics

- **Compound Building**
  - No safety-related equipment whose failure affects PRA results.
- **EDG Building**
  - Adjacent to Aux. Building (Quadrants A & B)
  - 2 EDGs A/B (EDGs C/D are located in Aux. Building)
- **Reactor Building**
  - Reactor gas vent related valves
  - Pressurizer related valves
- **Turbine Building**
  - Non Class 1E Switchgears
    - Switchgears M/N installed in separate rooms
- **Yard Transformers**
  - Main transformers (MTR)
  - Unit auxiliary transformers (UAT)
  - Standby auxiliary transformers (SAT)
  - 3-hr fire barrier installed between each transformer
  - Isolated phase bus installed in MTR and UAT

# APR1400 Characteristics

- **3-hr rated fire barrier in all fire areas**
  - 2-hr fire barriers for stairs
- **Main Control Room**
  - Large Display Panel, five Operator Consoles, and Safety Console
  - Safety Console is a backup
  - Independent HVAC system
- **Remote Shutdown Room**
  - Remote Shutdown Panel

# APR1400 Characteristics

- **All cables are IEEE-383 qualified**
- **Control Cabinets are locally distributed.**
  - Group Control Cabinets are located in I&C equipment rooms near MCR.
  - Component Control Cabinets are located near each local room (quadrant design).
- **Automatic Suppression System in Aux. Building**
  - $CO_2$ Suppression system in switchgear/EDG rooms
  - Water sprinkler system in corridors of each floor

# Fire Insights

- The APR1400 has two divisions each consisting of two trains of safety systems. Each division is segregated with physical fire barriers so as to protect the safety function of safety systems from fires impacting the opposite division. Any fire impacting both divisions requires failure of a 3 hour fire barrier.
- The APR1400 Auxiliary Building is divided into four quadrants, two quadrants per division each quadrant is segregated with physical fire barriers. Each quadrant contains the equipment for a safety train.
- The APR1400 has a highly compartmentalized Auxiliary Building comprised of many fire areas with 3 hour fire rating barriers so as to minimize the impact from any single fire in the Auxiliary Building.
- The APR1400 employs fiber optic cables between the Main Control Room, the group controllers and loop controllers thereby minimizing the impact from fire induced spurious hot shorts.
- Fire model consideration of the highly integrated control room (HICR) will be performed using qualitative analysis.

# Level 1 Seismic Assessment

# Methodology

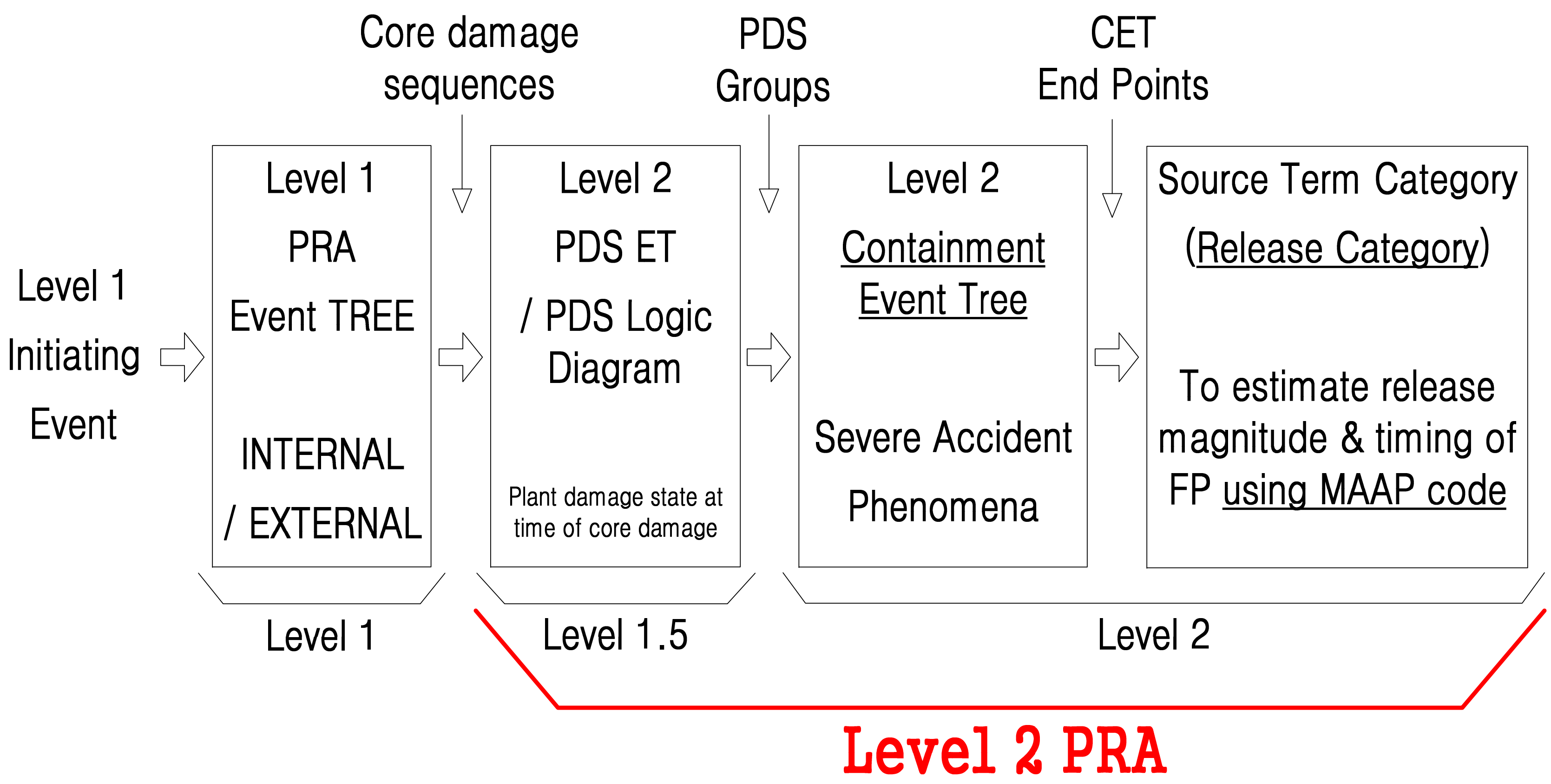
- **DC/COL-ISG-020, Interim Staff Guidance on Implementation of a PRA-Based Seismic Margin Analysis for New Reactors**
  - PRA-based SMA for advanced nuclear power plants
    - Conservative Deterministic Failure Margin (CDFM) Approach for Estimation of Structure and component HCLPF
  - The lowest core damage sequence HCLPF (High Confidence Low Probability of Failure) value should have at least 1.67 times the CSDRS (Certified Seismic Design Response Spectrum)
    - HCLPF > 0.5g
- **Analysis Steps**
  - Select Seismic Equipment List
  - Derive HCLPF values for components and structures (CDFM)
  - Develop Seismic Initiating Event Trees and Fault Tree Models
  - Solve the Seismic Event Trees
  - Calculate HCLPF values for the seismic core damage sequences
  - Determine the plant HCLPF

# Level 2 Analysis

# Level 2 PRA Methodology

- **Level 2 PRA Methodology**
  - NUREG/CR-2300 : PRA Procedures Guide
  - NUREG-1150 : Severe Accident Risks
  - NUREG-1335 : IPE Submittal Guidance
  - CESSAR DC : System 80+ PRA
  - ASME/ANS RA-Sa-2009 “PRA Standards”
- **Containment Event Tree and Decomposition Event Tree (CET/DET) methodology is used.**
- **Computer Codes**
  - MAAP 4.0.8 for analyzing severe accident progression and source term release
  - SAREX 1.2 for developing Level 2 PRA model

# Level 2 PRA Process



# Design Features against Severe Accidents

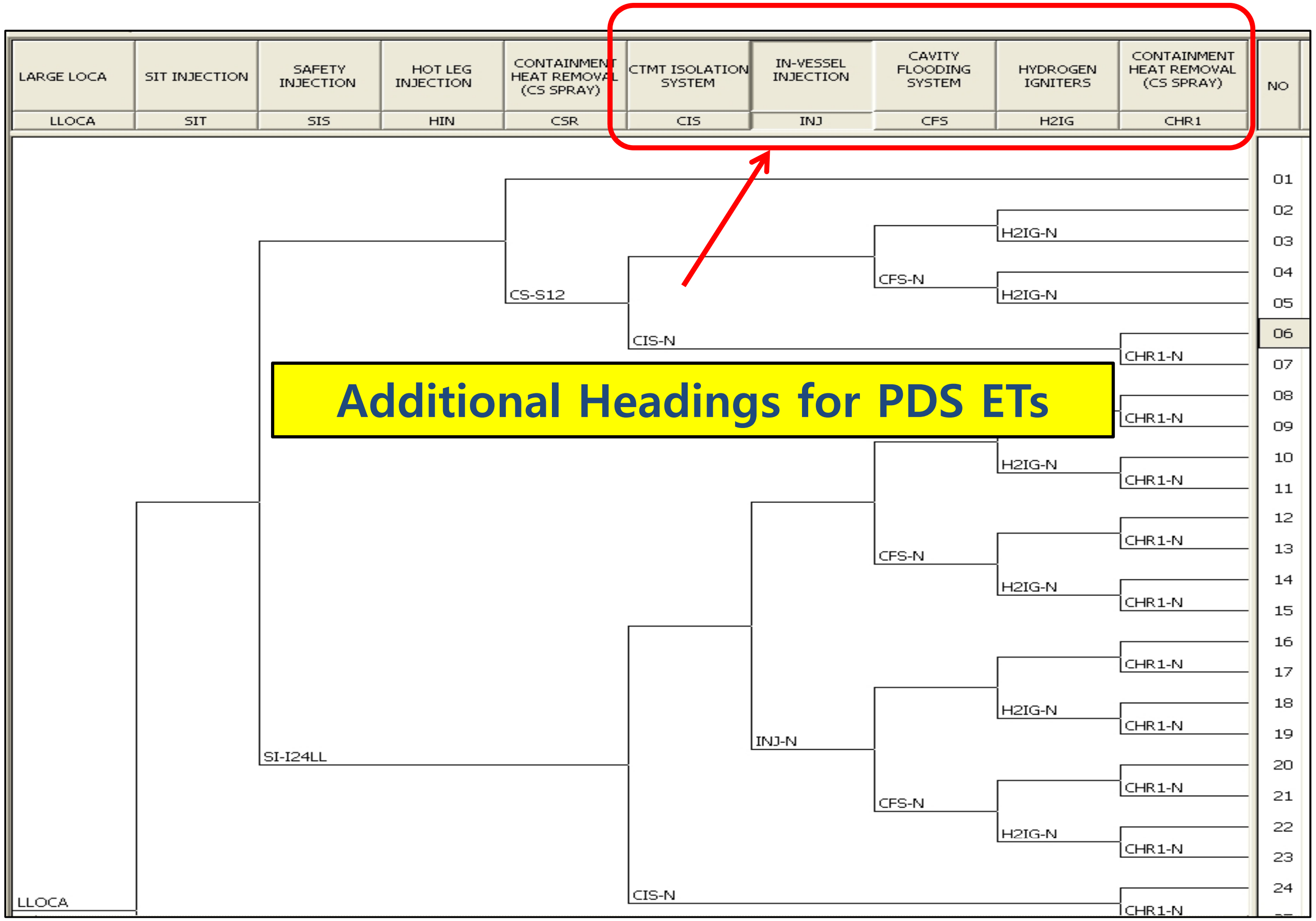
- **Containment Building**
  - Pre-stressed concrete containment with a steel liner plate, Large Dry Containment
- **Reactor Cavity Design**
  - Convoluted Flow Path to decrease the amount of ejected core debris
  - Large cavity floor area (greater than 0.02 $m^2$/MWt)
  - The concrete thickness are 3 feet
- **Cavity Flooding System (CFS)**
  - Minimize or eliminate corium-concrete attack due to MCCI
- **Hydrogen Mitigation System (HMS)**
  - Limit hydrogen concentration in containment, generated from a 100-percent fuel clad-coolant reaction less than 10 v/o
  - 30 PARs and 8 Igniters
- **Safety Depressurization and Vent System (SDVS)**
  - Provide a means to rapidly depressurize the primary system to about 250psia to prevent DCH following severe accidents
- **ECSBS (Emergency Containment Spray Backup System)**
  - An alternate mean of providing containment spray water after 24 hours following severe accidents
  - Deliver water from external water source to the ECSBS containment spray header
  - Use pumping device (i.e., fire engine truck) independent of normal and emergency AC power sources.

# Plant Damage State

- **Plant Damage States (PDSs)**
  - Collections of accident sequence end states according to plant conditions at the onset of severe core damage
  - Represent the interface between the Level 1 and Level 2 analyses
- **Level 1 event trees**
  - Extended as necessary to assess the availability of all systems important to the containment accident progression analysis.
    - Containment isolation system, cavity flooding system, hydrogen mitigation system, containment spray system and so on.
  - Grouped by PDS grouping parameters
- **PDS grouping parameters**
  - Containment Bypass
  - Containment Isolation
  - LOCA or Transients
  - RCS Pressure at the time of core damage
  - Reactor Cavity Condition
  - In-vessel Injection
  - Release Points of RCS Inventory
  - Containment Heat Removal
  - Feedwater Injection into SGs

# Plant Damage State Event Tree

- **PDS Event Tree (Example)**



# Containment Structural Capabilities

- **Considered Containment Failure Locations by Quasi-Static Pressure**
  - Failure due to membrane stresses (including hoop and meridional failure in the cylindrical wall and dome)
  - Radial shear failure of the cylindrical wall at the base
  - Flexural failure of the cylindrical wall at the base
  - Shear failure of the base slab/Flexural failure of the base slab
  - Failure of Equipment Hatch Assembly/Personnel Access Airlock Assembly
  - Failure of Personnel Emergency Exit Assembly/Fuel Transfer Tube
  - Failure at Pipe Penetrations/Electrical Penetrations/Bellows
  - Liner Buckling/Thermal buckling of liner plate

- **The two containment failure modes are determined based on NUREG-1150 and NUREG/CR-6906**
  - Rupture : The containment fails with break size of greater than 1.0 $ft^2$
  - Leak : The containment fails with break size of 0.1 $ft^2$

# Containment Failure Modes

- **NUREG-1335 and NUREG-1150 provide the several containment challenges and the severe accident risks.**
- **The following potential containment challenges are assessed in Level 2 analysis**
  - Direct Bypass (SGTR and ISLOCA)
  - Steam Explosions (In-vessel/Ex-vessel)
  - Hydrogen Combustion Processes (Slow Combustion/Detonation)
  - Steam Over-pressurization
  - Molten Core-Concrete Interaction (Basemat Melt-through)
  - Blowdown Forces (Vessel Thrust Force or Rocket Mode Failure)
  - Liner Melt-Through (Direct Contact of Containment Shell with Fuel Debris)
  - Failure of Containment Building Penetrations

# Containment Event Tree

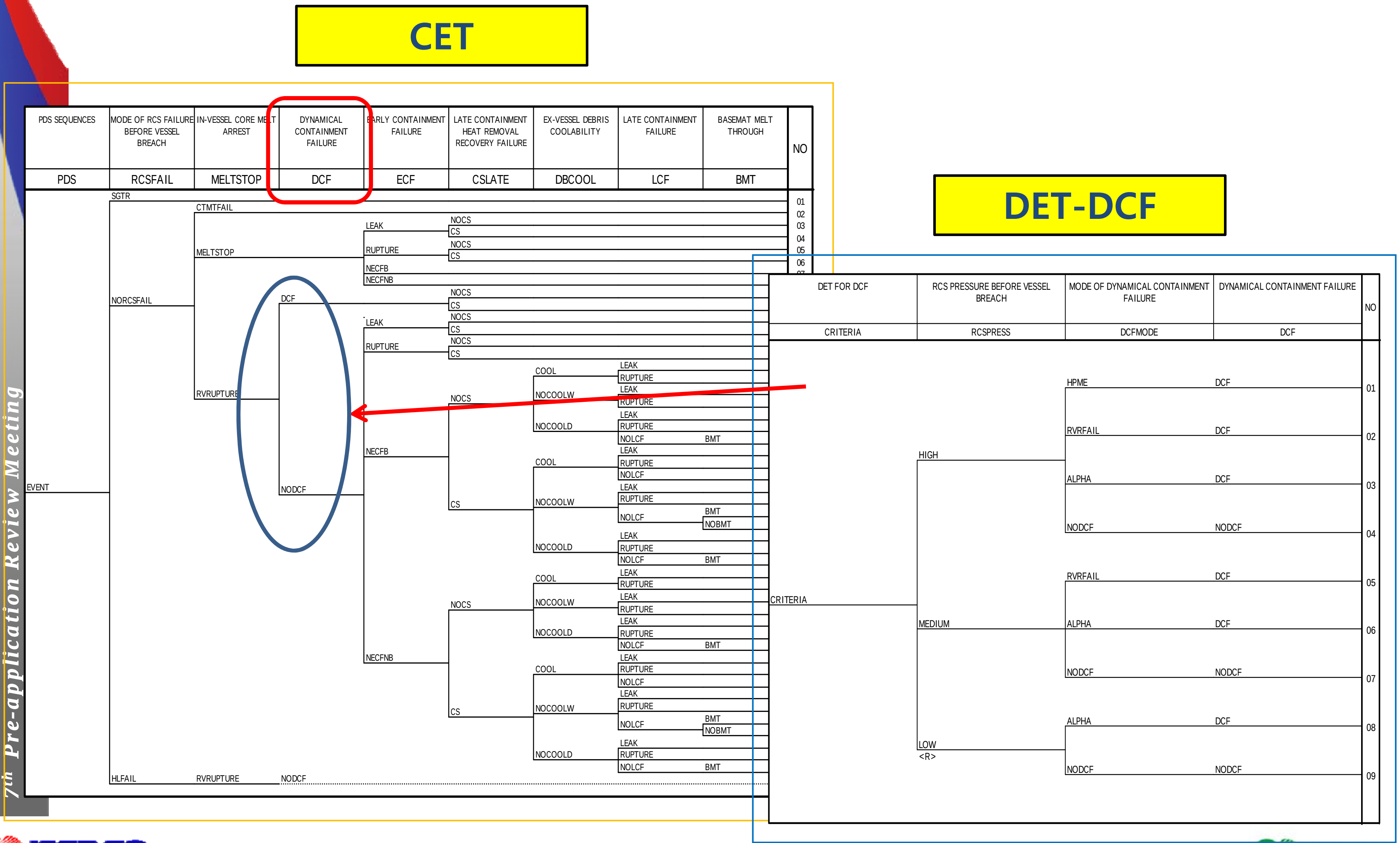
- **The various containment failure mode and the major severe accident phenomena are represented as top events of the CET.**
- **The following headings are considered as top events:**
  - RCS Failure due to Creep (DET-RCSFAIL)
  - In-vessel Retention (DET-MELTSTOP)
  - Dynamical Containment Failure at Vessel Failure (DET-DCF)
  - Early Containment Failure (DET-ECF)
  - Containment Heat Removal after Vessel Failure (DET-CSLATE)
  - Cooling of Core Debris at Ex-vessel (DET-DBCOOL)
  - Late Containment Failure (DET-LCF)
  - Basemat Melt Through (DET-BMT)
- **Detailed evaluation of phenomena for each top event of CET is treated in Decomposition Event Tree (DET)**

# Containment Event Tree

## ● Characteristics of Containment Failure Mode

| Containment Failure Mode | Timing | Severe Accident Phenomena |
|---|---|---|
| Containment Bypass (BYPASS) | Early phase | (Unscrubbed) SGTR and ISLOCA |
| Containment Isolation Failure (NOTISO) | Early phase | Containment isolation failure |
| Early Containment Failure (ECF) | Just or within 2 hrs after reactor vessel failure | DCH, H2 deflagration/detonation<br>In/Ex-vessel steam explosion<br>Blowdown forces(Rocket mode)<br>HPME |
| Containment Failure Before RPV Breach (CFBRB) | Containment failure before reactor vessel failure | Over pressurization |
| Late Containment Failure (LCF) | > 2 hrs after reactor vessel failure | Over pressurization<br>H2 deflagration/detonation |
| Basemat Melt-through (BMT) | > 2 hrs after reactor vessel failure | Molten corium-concrete interaction (MCCI) |

# Containment Event Tree (Example)



# Source Term Category

- **The end points of CETs represent the possible accident sequences that describes complete accident scenarios from initiating events to release into environment.**
  - In general, a detailed source term analysis for all end points is very difficult and seems to be not feasible.
- **It is necessary to group the CET accident sequences that have similar release characteristics into a corresponding release category (or Source Term Category).**
- **Source term release analysis is performed**
  - For a representative sequence of each STC, which is determined based on frequency (or contribution to its STC) and amount of source term released to the environment
  - Using MAAP 4.0.8 code

# Level 2 Results Analysis

- **The Level 2 Results Analysis**
  - Review of significant sequences
  - Identification of Significant contributors to risk
    - To determine if additional recovery actions or equipment redundancy would significantly reduce the risk.
  - Relevant sensitivity analysis
  - Characterization of sources of uncertainty

# Low Power and Shutdown Analysis

# Introduction

- **Scope**
  - Internal Events and External Events (Internal Fire/Flooding)
- **Assumptions**
  - Based on typical Korean nuclear power plant outage experiences and procedures, with modification to consider APR1400 design characteristics
- **Methodology**
  - CESSAR PRA for LPSD operation
  - NUREG/CR-6144
  - PRAs of Korean Standard Nuclear Power Plants PRA with experience from LPSD operation

# Methodology -Analysis Step

- **Internal Events**
  - Plant Operational State Analysis
  - Initiating Events Analysis
  - Accident Sequence Analysis
  - Success Criteria
  - Systems Analysis
  - Human Reliability and Data Analysis
  - Quantification
- **External Events**

# Methodology– Design Features and Mitigating Functions

- **Design Features for LPSD**
  - Instrumentation to monitor reduced inventory operation
  - Procedure to prevent overpressure of RCS during loss of RHR
  - Design change to mitigate Shutdown Cooling pump cavitation caused by entrainment of air in the pump suction
- **Mitigating functions during LPSD can be categorized into two groups**
  - Decay heat removal function
    - RHR using Shutdown Cooling System
    - SG if RHR is unavailable
  - RCS inventory make-up function
    - Safety injection system
    - Chemical and Volume Control System
      - To prevent bulk boiling and maintain the RCS water level

# Methodology– Definition of POS

- **The outage is divided into 17 POSs based on RCS conditions**

| POS | T/S mode | Description |
|---|---|---|
| 1 | 1,2 | Reactor trip and subcritical state |
| 2 | 3 | Hot standby |
| 3 | 4, 5 | Hot and cold shutdown (RCS intact) |
| 4A | 5 | Cold shutdown (Draining RCS) (RCS not intact, RCS vent) |
| 4B | 5 | Cold shutdown (Draining RCS) (RCS not intact, Pressurizer manhole) |
| 5 | 5 | Cold shutdown (Reduced inventory) (RCS not intact) |
| 6 | 5 | Cold shutdown (Filling RCS) (RCS not intact) |
| 7~9 | 6 | Refueling |
| 10 | 5 | Cold shutdown (Filling RCS) (RCS not intact) |
| 11 | 5 | Cold shutdown (Reduced inventory) (RCS not intact) |
| 12A | 5 | Cold shutdown (Filling RCS) (RCS not intact, Pressurizer manhole) |
| 12B | 5 | Cold shutdown (Filling RCS) (RCS not intact, RCS vent) |
| 13 | 4 | Hot and cold shutdown (RCS intact) |
| 14 | 3 | Hot standby |
| 15 | 1,2 | Reactor Startup |

# Methodology– Initiating Events

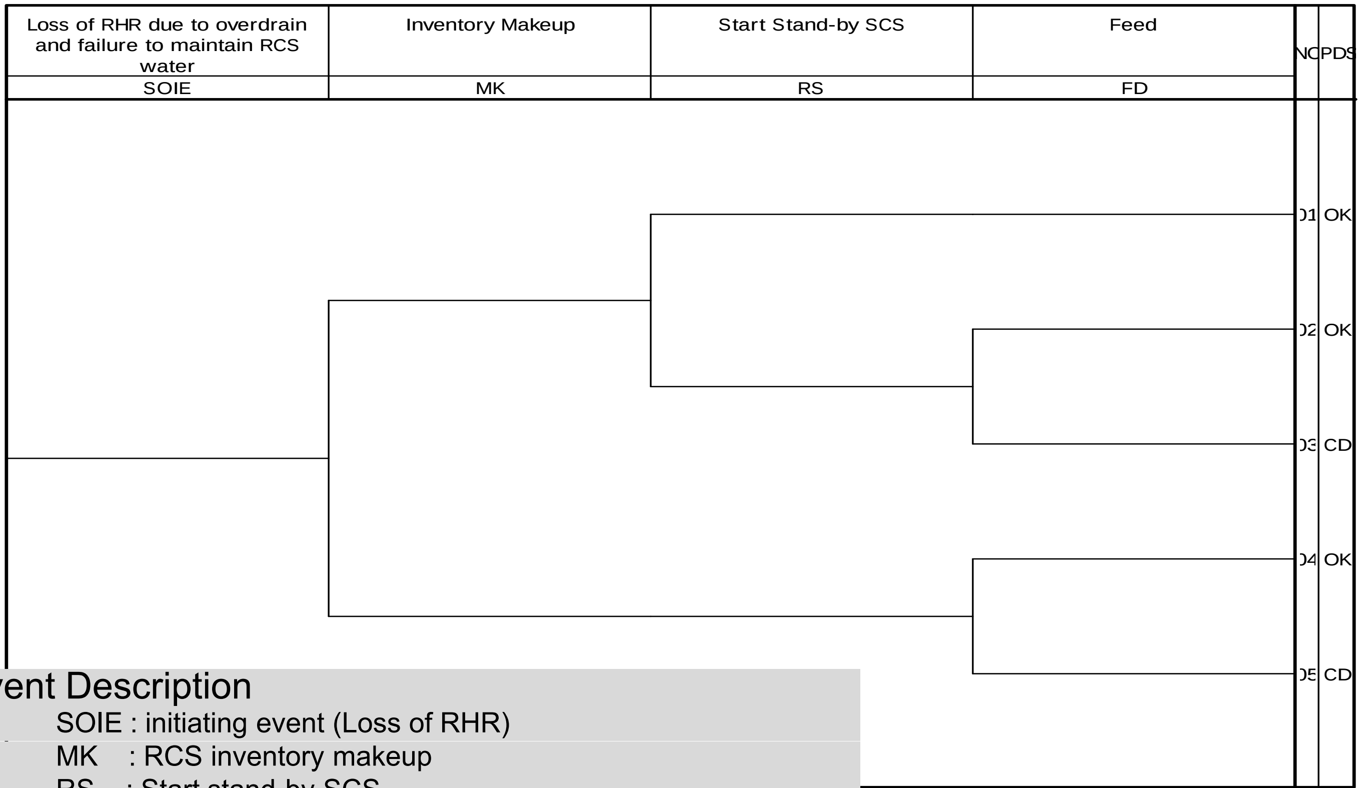
- **Initiating Events for LPSD PSA are listed below**
  - LOCA
    - Large, Medium and Small LOCA, SGTR, Main Steam Line Break during Hot Standby
    - Unrecoverable LOCA occurred by operator error
  - Loss of RHR due to over-drain
    - During transition from the normal condition to reduced inventory operation
  - Loss of RHR caused by failing to maintain water level
    - During reduced inventory operation
  - Loss of RHR caused by other failures
    - Recoverable or Unrecoverable loss of RHR
  - Loss of CCW/ESW
  - Loss of offsite power and SBO
  - Loss of 4.16kV Switchgear Bus
- **The frequency of Initiating Events**
  - NUREG/CR-6144, the latest industry data, and design features for RHR operation

# Methodology– Accident Sequence

- **The Accident Sequence Development**
  - Event Tree development
    - For all initiating events and POSs
  - Procedures and experience of OPR1000
    - Operator procedures, shift supervisor's log book, and so on
  - Success criteria based on thermal hydraulic analysis
    - According to IMC-0609 App G (Inspection manual) of NRC, success criteria of core damage for LPSD is 1,300F of Peak Core Temperature
    - RELAP5 Mod 3 for RCS conditions of each POS
    - MAAP4.0.8 for Containment Heat Removal
  - Design features for RHR operation
    - To meet GL 88-17 requirements during LPSD Operation
    - To prevent RHR operation from loss of shutdown cooling and loss of reactor coolant

# Methodology– Event Tree Analysis

- **Loss of RHR (Example)**
  - Due to overdrain and failure to maintain RCS water during reduced inventory operation




- Event Description
  - SOIE : initiating event (Loss of RHR)
  - MK : RCS inventory makeup
  - RS : Start stand-by SCS
  - FD : Injection water to RCS by using SI and SCS pump

# Methodology– Human Reliability

- **Human Reliability Analysis**
  - LPSD PSA uses the same HRA methodology to that of full power Level 1 internal events PRA

# Closing Remarks

- Use of PRA to Support Design
- Summary

# Use of PRA To Support Design

- **Design Changes from Shin-Kori 3&4**
  - 4 EDGs from 2 EDGs
    - Significant impact on SBO sequences
  - Capacity of Class 1E batteries C/D
    - 16 hours from 8 hours with manual shedding
    - Moderate impact on SBO sequences
  - RTSS (Reactor Trip Switchgear System)
    - Full 2/4 (8 RTBs) from selective 2/4 (4 RTBs)
    - Limited impact on CDF

# Summary

- **The preliminary CDF from internal events for full power is at low $10^{-6}$ per reactor year.**
- **The preliminary results from external events indicate that the quadrant/divisional design is highly effective in risk reduction.**
- **Level 2 and LPSD assessments are still in preliminary stage.**
- **Peer Reviews**
  - Internal Events PRA (including internal flood)