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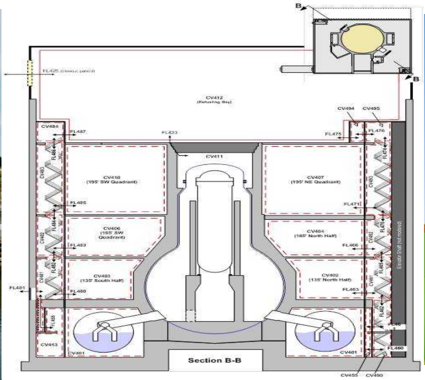

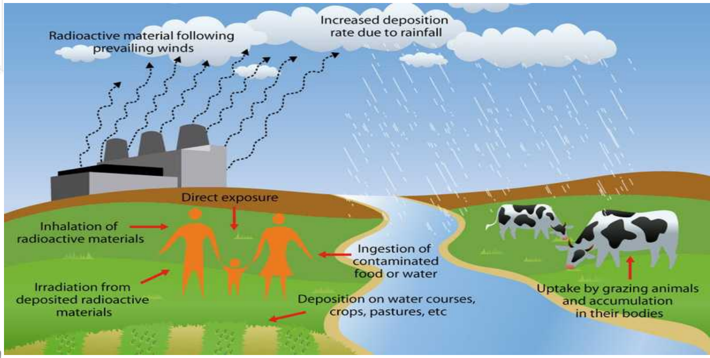

U.S. Nuclear Regulatory Commission's State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

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Sandia National Laboratories

4th Arab Forum on the Prospects of Nuclear Power – October 11, 2017

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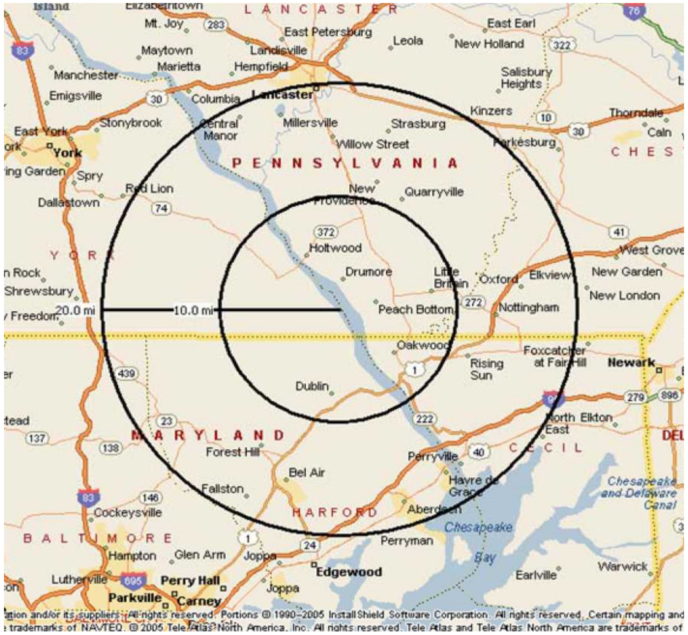
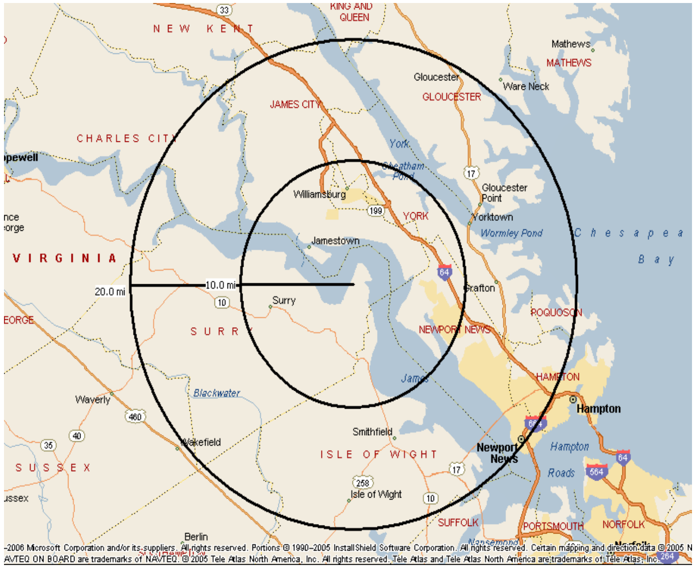
Overview

- Background
- Deterministic Methodology
- Uncertainty Methodology
- Uncertainty Conclusions



Peach Bottom Atomic Power Station
10 and 20 mile analysis areas



Surry Power Station
10 and 20 mile analysis areas

Background

Peach Bottom Atomic Power Station

Surry Power Station

Sequoyah Nuclear Generating Station

Objectives of SOARCA

- The U.S. Nuclear Regulatory Commission sponsored Sandia National Laboratories to conduct the SOARCA Project

- The primary objective of the SOARCA project is to develop a body of knowledge of the realistic outcomes of severe reactor accidents in the U.S. civilian nuclear reactor sites
  - The central focus of the SOARCA project was to introduce the use of a detailed, best-estimate, integrated quantification of sequences based on current scientific knowledge and plant capabilities.

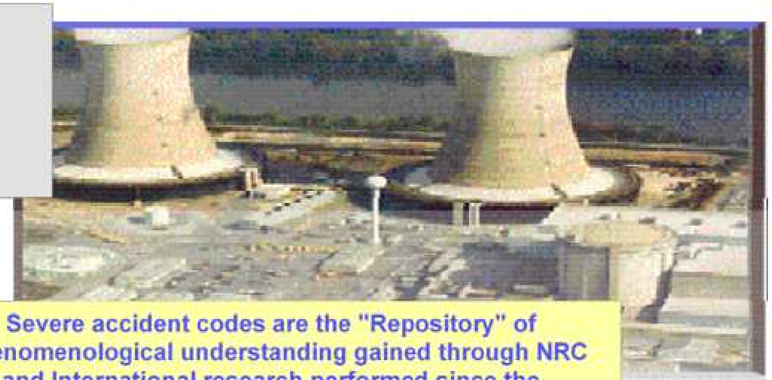
- The SOARCA project is demonstrating new approaches to evaluating consequences of severe accidents that are not equivalent to current full scope PRA concepts

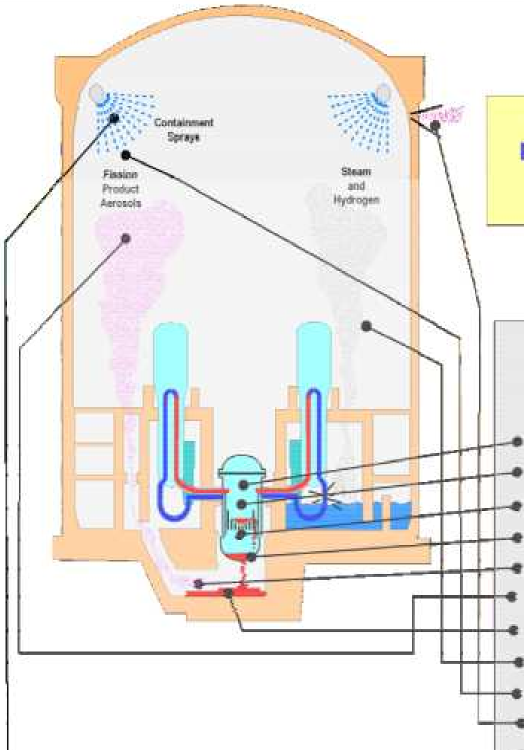
- NUREG-1935; State-of-the-Art Reactor Consequence Analyses (SOARCA) Report
- NUREG/CR-7110 Volume 1; SOARCA Project Volume 1: Peach Bottom Integrated Analysis
- NUREG/CR-7110 Volume 2; SOARCA Project Volume 2: Surry Integrated Analysis
- NUREG/BR-0359; Modeling Potential Reactor Accident Consequences

MELCOR

Severe Reactor Accident System Code

Modeling and Analysis of Severe Accidents in Nuclear Power Plants





Severe accident codes are the "Repository" of phenomenological understanding gained through NRC and International research performed since the TMI-2 accident in 1979

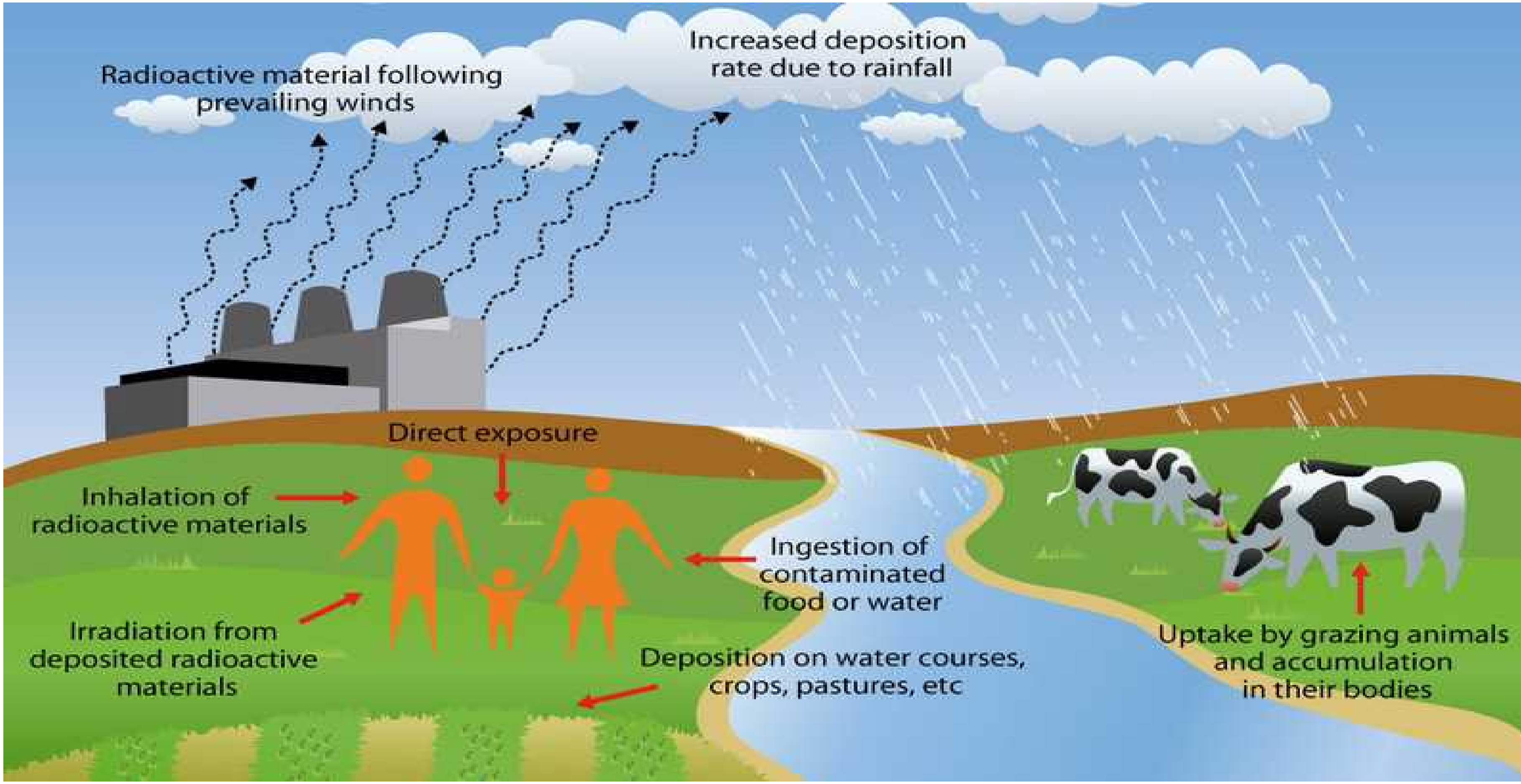
Integrated models required for self consistent analysis

Important Severe Accident Phenomena (MELCOR, CONTAIN, VICTORIA, SCDAP RELAP5)

- Accident initiation
- Reactor coolant thermal hydraulics
- Loss of core coolant
- Core meltdown and fission product release
- Reactor vessel failure
- Transport of fission products in RCS and Containment
- Fission product aerosol dynamics
- Molten core/basemat interactions
- Containment thermal hydraulics
- Fission product removal processes
- Release of fission products to environment
- Engineered safety systems - sprays, fan coolers, etc
- Iodine chemistry, and more

MACCS

MELCOR Accident Consequence Code System



SOARCA Uncertainty Analysis Objectives

1. Identify the uncertainty in the input and parameters used in the SOARCA deterministic, and

2. Develop insight into the overall sensitivity of the SOARCA results to uncertainty in key modeling inputs
   - Assess key MELCOR and MACCS modeling uncertainties in an integrated fashion to quantify of the relative importance of each uncertain input on the potential consequences

- NUREG/CR-7155; SOARCA Project Uncertainty Analysis of the Unmitigated Long-term Station Blackout of the Peach Bottom Atomic Power Station

- ML15224A001; SOARCA Project Uncertainty Analysis of the Unmitigated Short-term Station Blackout of the Surry Power Station – *DRAFT* – *being updated*

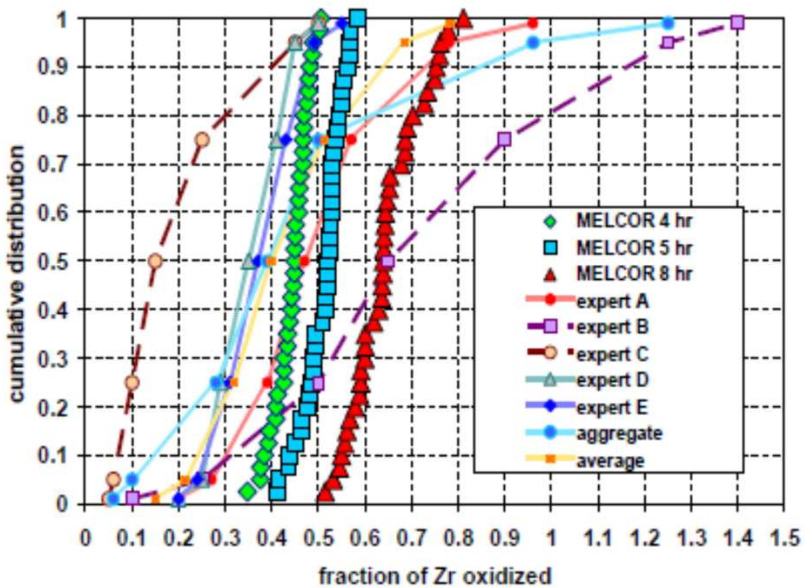
- MLXXXXX; SOARCA Project Sequoyah Integrated Deterministic and Uncertainty Analyses – Final *DRAFT*

SOARCA Uncertainty Analysis

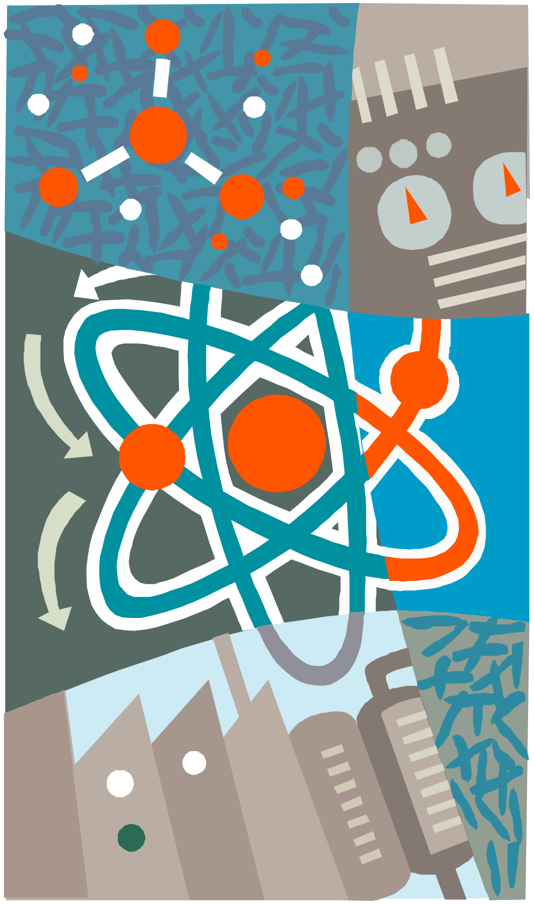
- Focus is on epistemic (state-of-knowledge) uncertainty in input parameter values
  - Model uncertainty addressed to the extent that some parameters represent or capture alternate model effects or in separate sensitivity analyses
  - Aleatory (random) uncertainty due to weather is handled in the same way as the SOARCA study
- Scenario definition has not changed as a result of Fukushima
  - Daiichi - station blackout with a loss of ultimate heat sink
  - Daini – consider it as 'success story' for 'lessons learned'
  - TurboFRMAC – IAEA working group for radiological assessment methods?
- Looking at uncertainty in key model inputs
  - MELCOR parameters
  - MACCS parameters

Deterministic versus Probabilistic

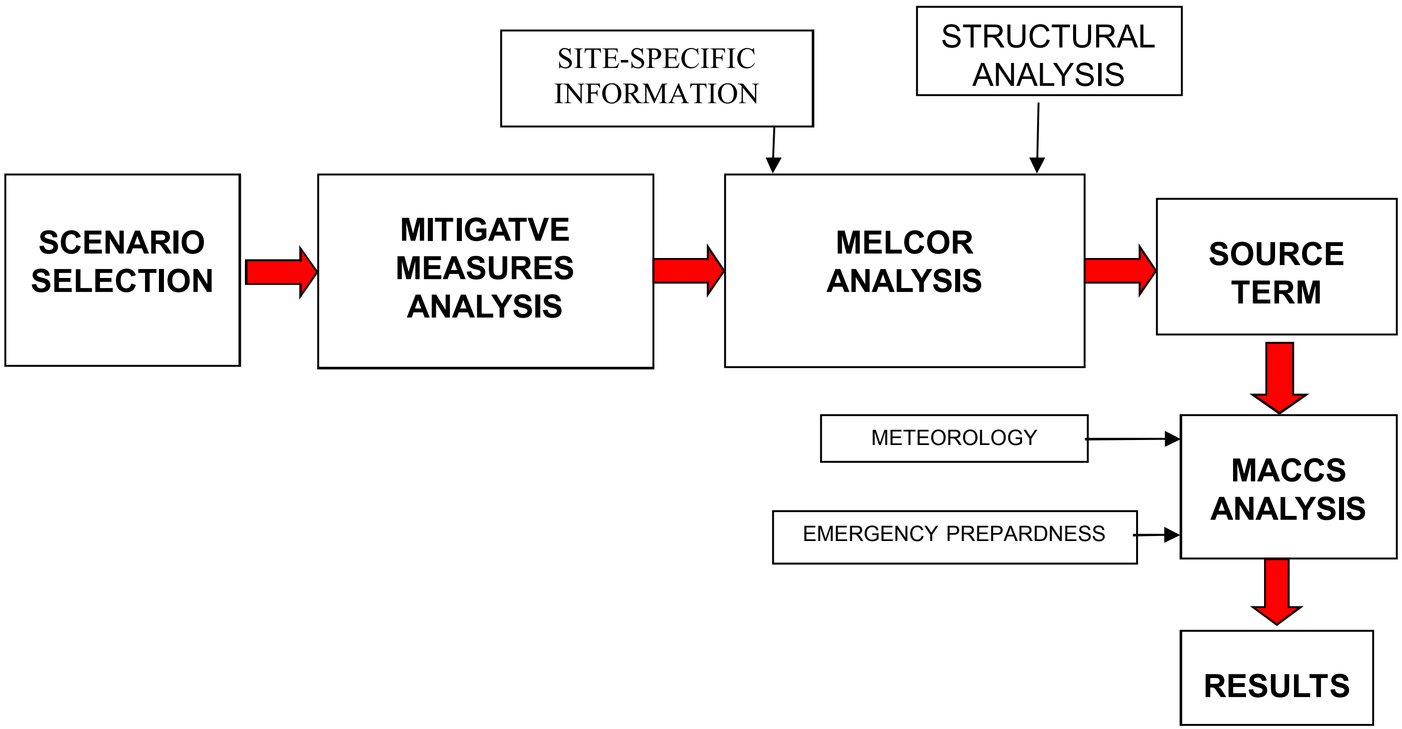
- Traditional safety analyses
  - Deterministic methods
  - ‘Conservative’ input assumptions
    - Safety relief valve setpoint drift
  - Produce defensible bounding analyses
  - Can be overly conservative
    - Excessive regulatory burden
- Expert elicitation
  - Only as good as experience of experts
- Objective Uncertainty Analysis
  - Quantification of uncertainty
  - Doesn’t combine unrealistically all worst case parameters
  - Characterizes safety margins
    - What is likely and expected vs. regulatory boundaries
    - Simple set of sensitivity cases or ‘one-offs’



Deterministic & Uncertainty Methodologies

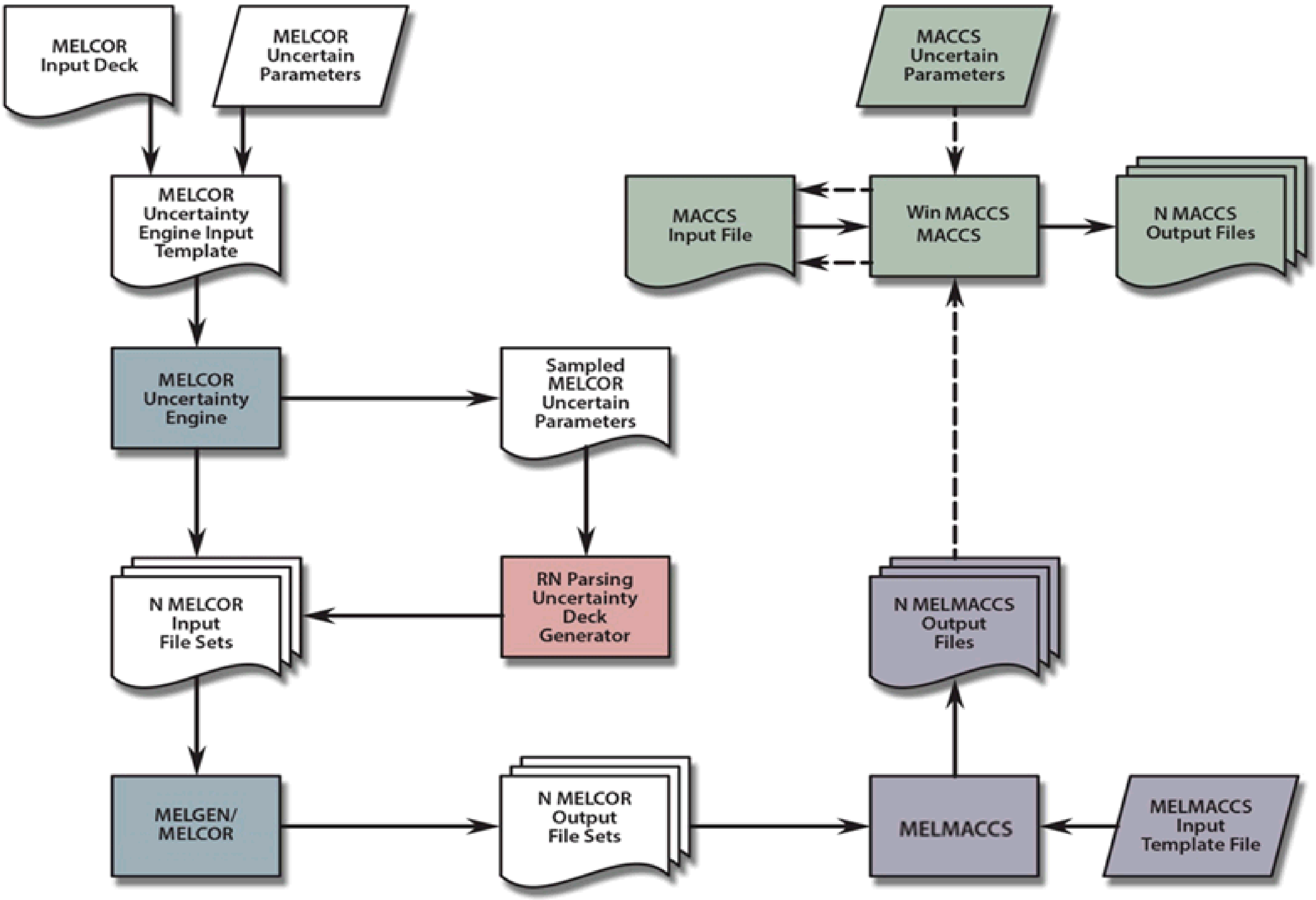


Deterministic SOARCA Approach

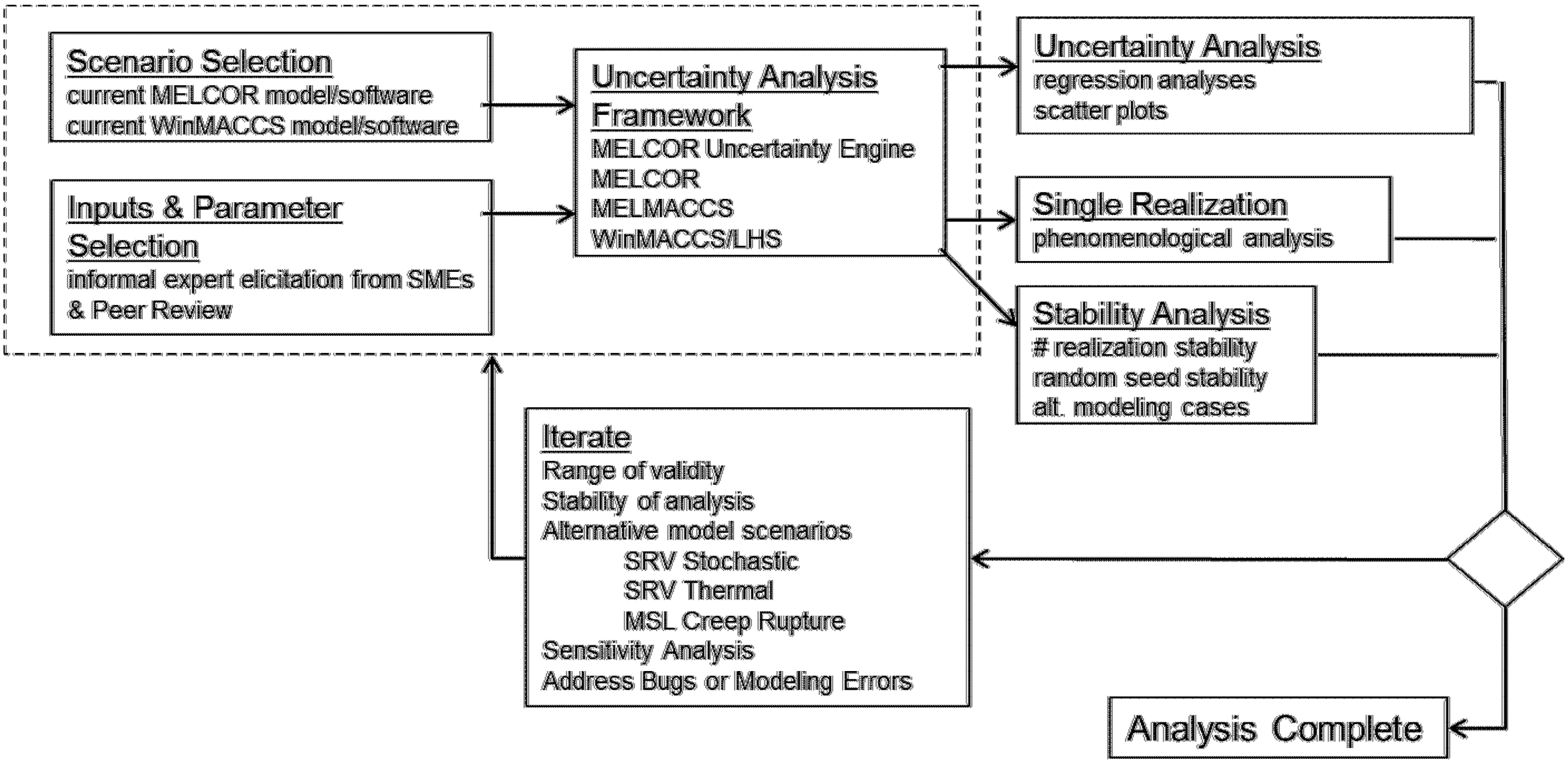


SOARCA Uncertainty Analysis

Information Flow



Uncertainty Analysis is an Iterative Process

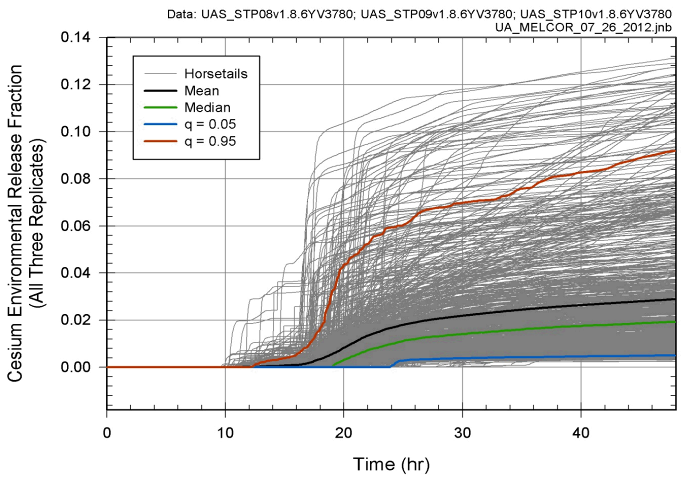


Uncertainty Analysis Post Processing

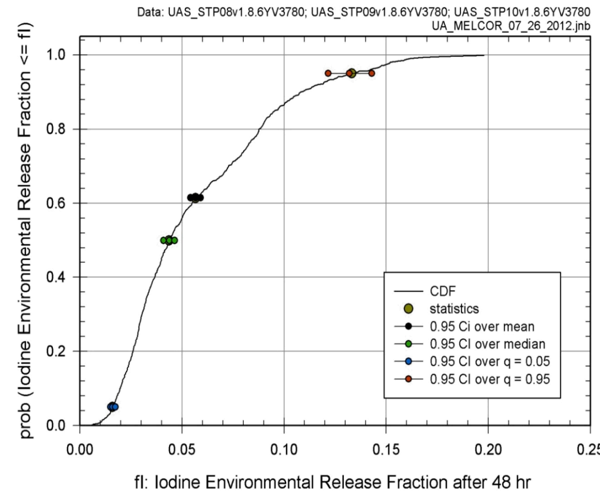
- Determination of Metrics (SOARCA)
  - Analysis of source term releases including Cesium and Iodine release over time
  - Latent cancer fatality risk and early fatality risk using a liner-no-threshold dose-response model
  - Description of most influential uncertain parameters in study
- Analysis of Output Metrics:
  - Statistical regression
  - Plot of all realizations vs. time (horse-tail plot) including median, mean, 5th and 95th percentiles
  - CDFs (at selected times) and CCDFs with confidence bounds
  - Phenomenological investigation of individual realizations of interest

Uncertainty Analysis Post Processing

(continued)



horse-tail plot with percentiles



CDF at a given time with confidence bounds

Iodine Release at 48 hours	Rank Regression			Quadratic			Recursive Partitioning			MARS		
Final R^2	0.69			0.76			0.93			0.80		
Input name*	**R^2 inc.**	**R^2 cont.**	**SRRC**	**S_i**	**T_i**	**p-val**	**S_i**	**T_i**	**p-val**	**S_i**	**T_i**	**p-val**
SRVLAM	0.49	0.49	-0.72	0.46	0.68	0.00	0.55	0.78	0.00	0.64	0.70	0.00
CHEMFORM	0.58	0.09	0.30	0.10	0.16	0.00	0.07	0.22	0.00	0.09	0.12	0.00
FL904A	0.64	0.06	0.26	0.05	0.06	0.22	0.02	0.12	0.00	0.05	0.08	0.00
RRDOOR	0.67	0.03	0.28	0.01	0.06	0.03	0.04	0.07	0.00	---	---	---
SRVOAFRAC	0.69	0.02	-0.12	0.06	0.13	0.00	0.05	0.20	0.00	0.06	0.16	0.00
FFC	0.69	0.00	0.06	0.03	0.03	0.17	---	---	---	0.02	0.00	1.00

regression coefficient table

Uncertainty Conclusions

Uncertainty Analyses corroborate SOARCA study conclusions:

- Public health consequences from severe nuclear accident scenarios modeled are smaller than 1982 SNL Siting Study (NUREG/CR-2239)
- The delay in releases calculated provide more time for emergency response actions such as evacuating or sheltering

Overall, Uncertainty Analyses provide the following insights:

- A major determinant of source term magnitude is the timing of the sticking open of the safety relief valve
- Health-effect risks vary sublinearly with source term because people are not allowed to return to their homes until dose is below habitability criterion
- The use of multiple regression techniques, most of which include nonlinear interactions between input variables, to post-process Monte Carlo results provides better explanatory power of which input parameters are most important to uncertainty in results

International SOARCA Efforts

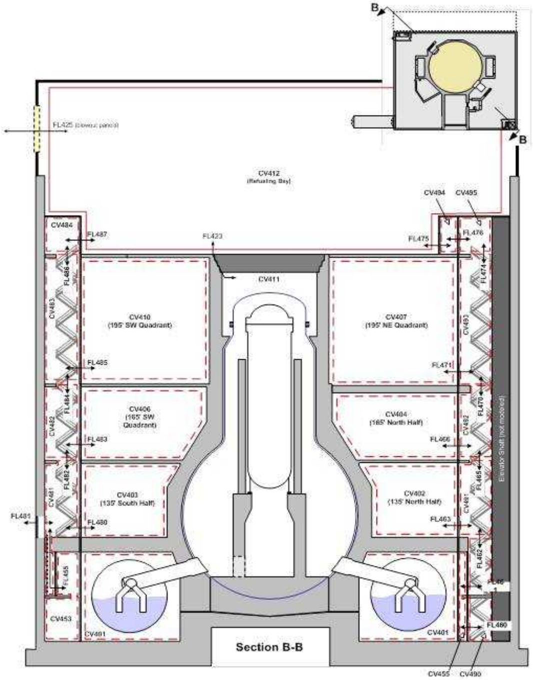
- Korea Hydro & Nuclear Power Company
- APR-1400 as the SOARCA reactor type (PWR)
- Shin Kori as site for MACCS analysis

http://www.ansnuclearcafe.org/wp-content/uploads/2014/02/KEPCOECAPR1400units.png

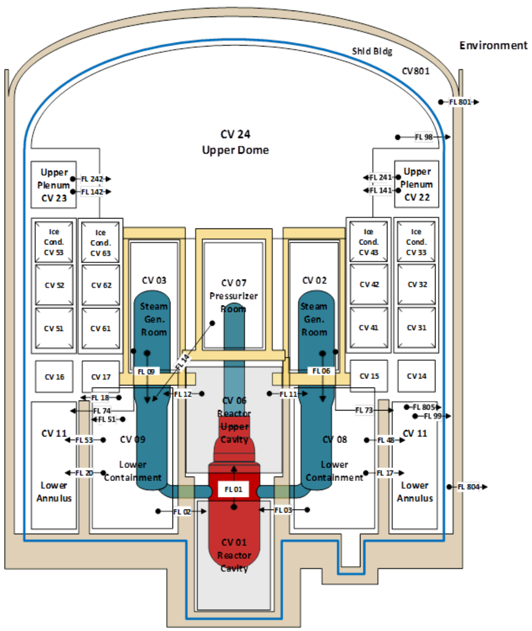
Precluding Severe Accidents

- PWR Owners Group
  - Westinghouse, Areva, Combustion Engineering, & KHNP
- BWR Owners' Group
  - General Electric - Hitachi
- Post-Fukushima generic Emergency Procedures / Severe Accident Guidance
  - BWROG – to be released in early 2018
  - PWROG – released in February 2017
- Nuclear Energy Institute – FLEX Guidance

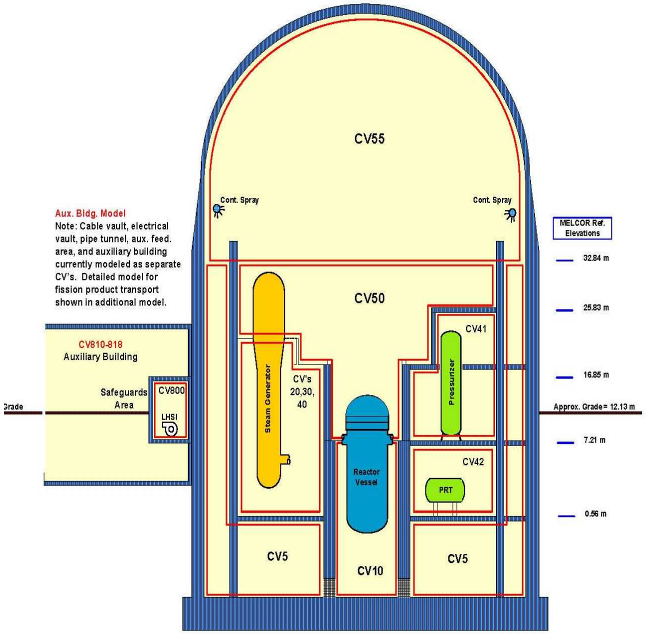
Questions



Peach Bottom Atomic Power Station
Reactor Building MELCOR Nodalization



Sequoyah Nuclear Generating Station
Containment MELCOR Nodalization



Surry Power Station
Containment MELCOR Nodalization

Backup Slides

SOARCA & Fukushima

NRC State-of-the-Art Reactor Consequence Analyses included BWR station blackout scenarios (SBO) performed *before* Fukushima accidents

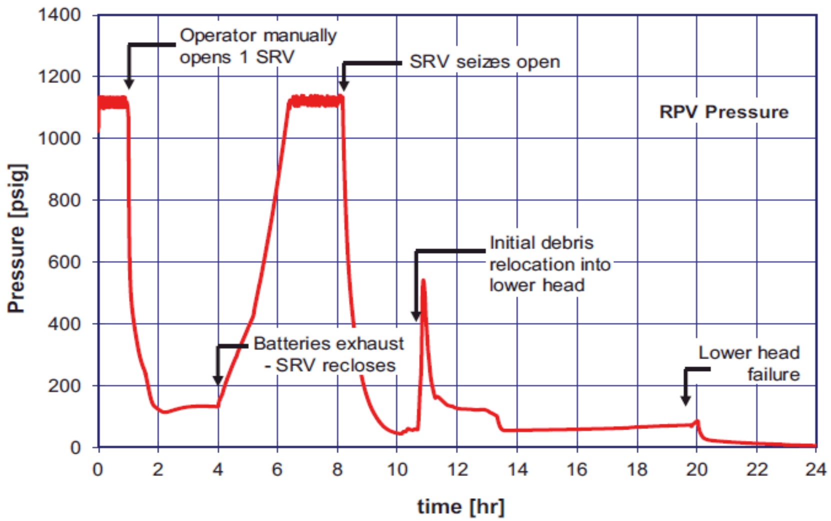
- Sequences observed at Fukushima
  - *Striking similar trends*
- Accidents are classic and 'usual suspects' for analysis

Fukushima critical equipment performance brought new insights

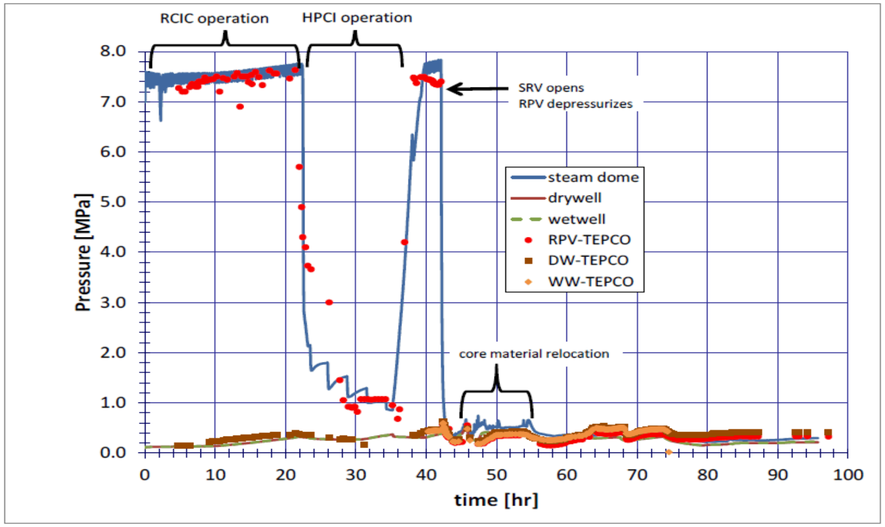
- Understanding of real-world operations can delay or prevent severe accidents

More information will come from decommissioning activities

- Main steam line failure, safety relief valve seizure, and containment liner failure

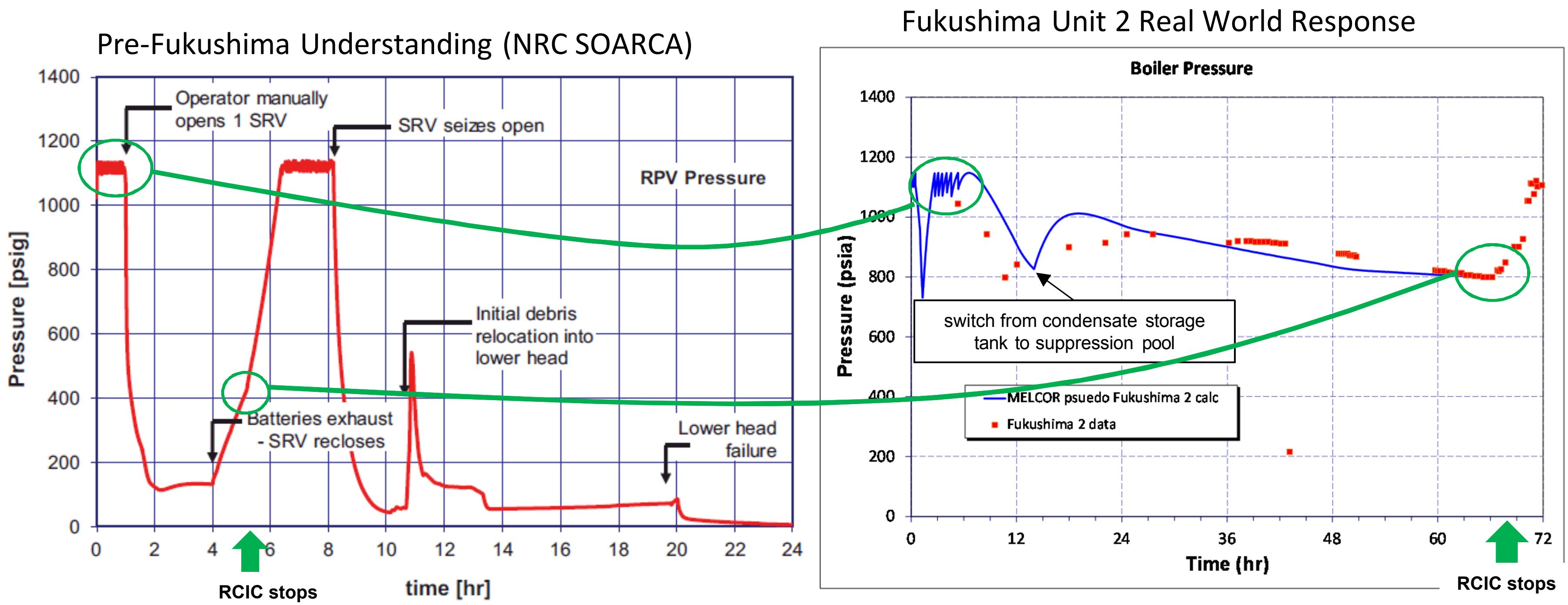


SOARCA Peach Bottom Long-Term Station Blackout



Fukushima Unit 3

Modeling of Station Blackout Accident before and after Fukushima (MELCOR Analyses and Fukushima Data)

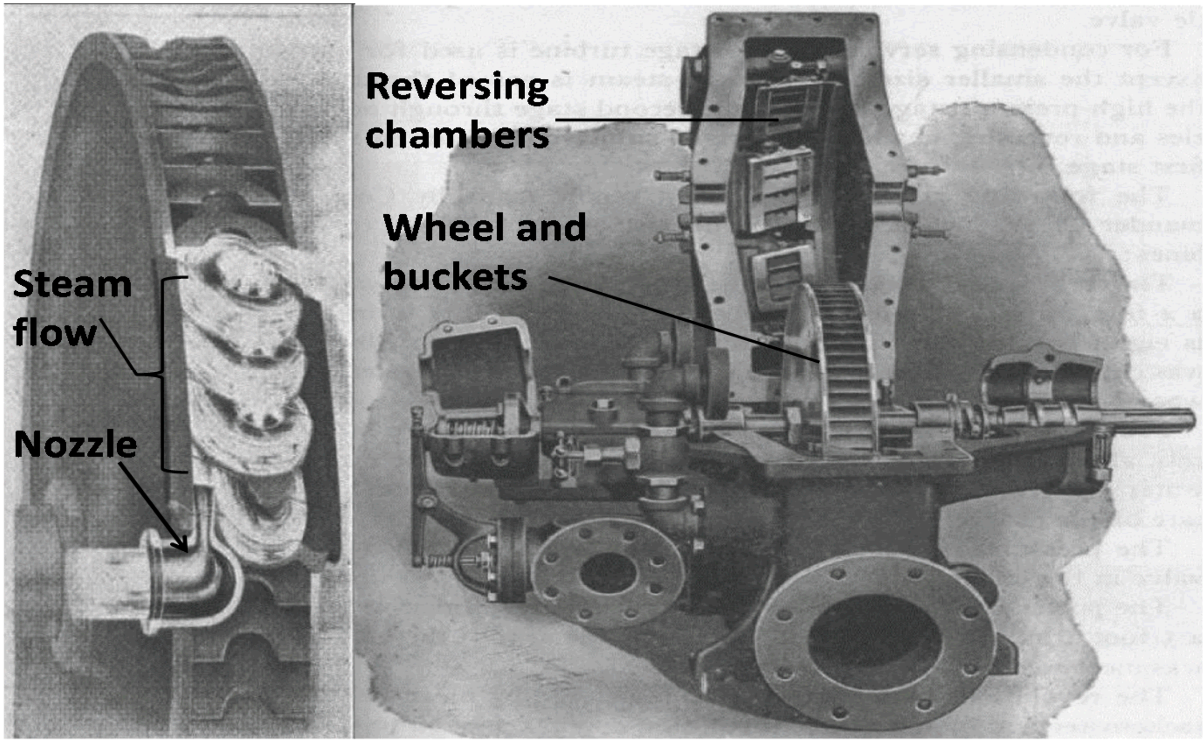


- Turbine-driven RCIC injection maintains desired water level in reactor pressure vessel (RPV)
- Battery depleted @ 4 hours
  - *SRV closes and RCIC runs full on*
  - *RPV overfills, MSL floods, water enters RCIC turbine, and RCIC assumed to fail @ ~5.2 hrs*
- Core meltdown at 10 hours

- Turbine-driven RCIC injection maintains desired water level in RPV at start of event
- Batteries fail @ 45 minutes from tsunami flooding
  - *RPV overfills, MSL floods, water enters RCIC turbine, but RCIC turbine does not fail*
  - *RCIC self-regulates RPV water level in cyclic mode*
- Core damage avoided for nearly 3 days

Reactor Core Isolation Cooling (RCIC)

- RCIC pump is driven by Terry Turbine developed circa 1900
  - *Pure impulse turbine with robust design that tolerates 'wet' steam (i.e. water/steam)*
  - *Very inefficient (~2-5% of steam energy converted to pumping power)*
- Prior assumptions held that steam line flooding would fail RCIC
  - *SOARCA predicted ~1.2 hrs till RCIC failed*
- Fukushima Unit 2 experience shows otherwise
  - *Operated in a 'self regulating' mode for days*



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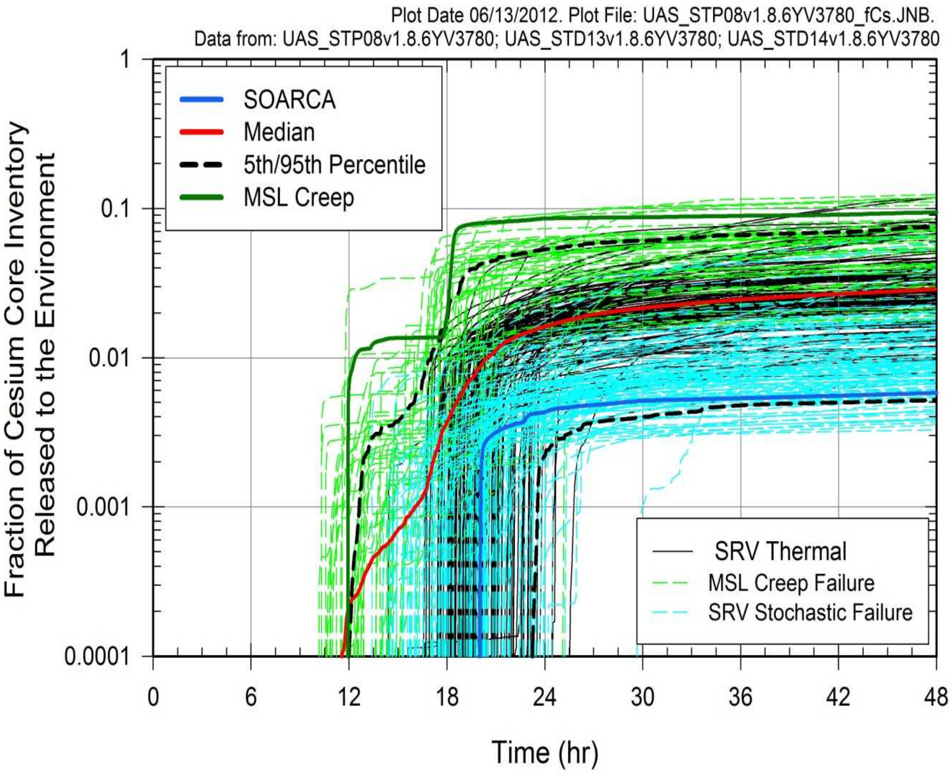
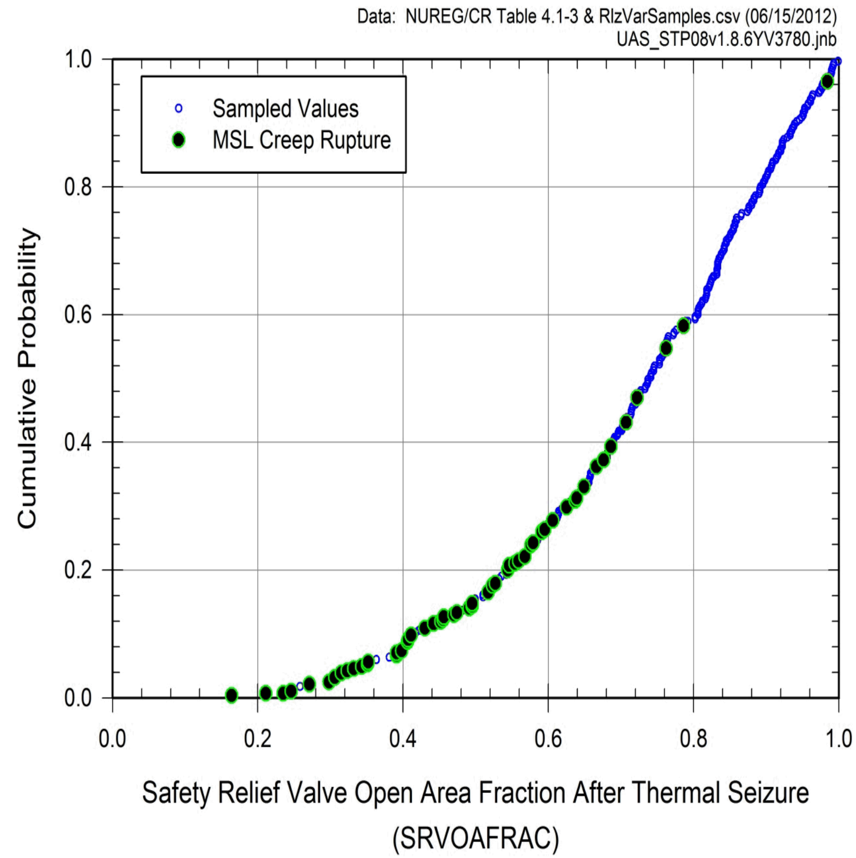
Peach Bottom SOARCA Uncertainty Inputs

MELCOR	MACCS
Epistemic Uncertainty (21 variables)	**Epistemic Uncertainty (350 variables)**
Sequence Issues	**Deposition**
SRV stochastic failure to reclose (SRVLAM)	Wet deposition model (CWASH1)
Battery Duration (BATTDUR)	Dry deposition velocities (VEDPOS)
In-Vessel Accident Progression Parameters	**Shielding Factors**
Zircaloy melt breakout temperature (SC1131(2))	Shielding factors (CSFACT, GSFAC, PROTIN)
Molten clad drainage rate (SC1141(2))	**Early Health Effects**
SRV thermal seizure criterion (SRVFAILT)	Early health effects (EFFACA, EFFACB, EFFTHR)
SRV open area fraction (SRVOAFRAC)	**Latent health effects**
Main Steam line creep rupture area fraction (SLCRFRAC)	Groundshine (GSHFAC)
Fuel failure criterion (FFC)	Dose and dose rate effectiveness factor (DDREFA)
Radial debris relocation time constants (RDMTC, RDSTC)	Mortality risk coefficient (CFRISK)
Ex-Vessel Accident Progression Parameters	Inhalation dose coefficients (radionuclide specific)
Debris lateral relocation – cavity spillover and spreading rate (DHEADSOL, DHEADLIQ)	**Dispersion Parameters**
Containment Behavior Parameters	Crosswind dispersion coefficients (CYSIGA)
Drywell liner failure flow area (FL904A)	Vertical dispersion coefficients (CZSIGA)
Hydrogen ignition criteria (H2IGNC)	**Relocation Parameters**
Railroad door open fraction (RRIDRFAC, RRODRFAC)	Hotspot relocation (DOSHOT, TIMHOT)
Drywell head flange leakage (K, E, δ)	Normal relocation (DOSNRM, TIMNRM)
Chemical Forms of Iodine and Cesium	**Evacuation Parameters**
Iodine and Cesium fraction (CHEMFORM)	Evacuation delay (DLTEVA)
Aerosol Deposition	Evacuation speed (ESPEED)
Particle Density (RHONOM)	**Aleatory Uncertainty (984 weather trials)**
865 MELCOR source terms developed	Weather Trials

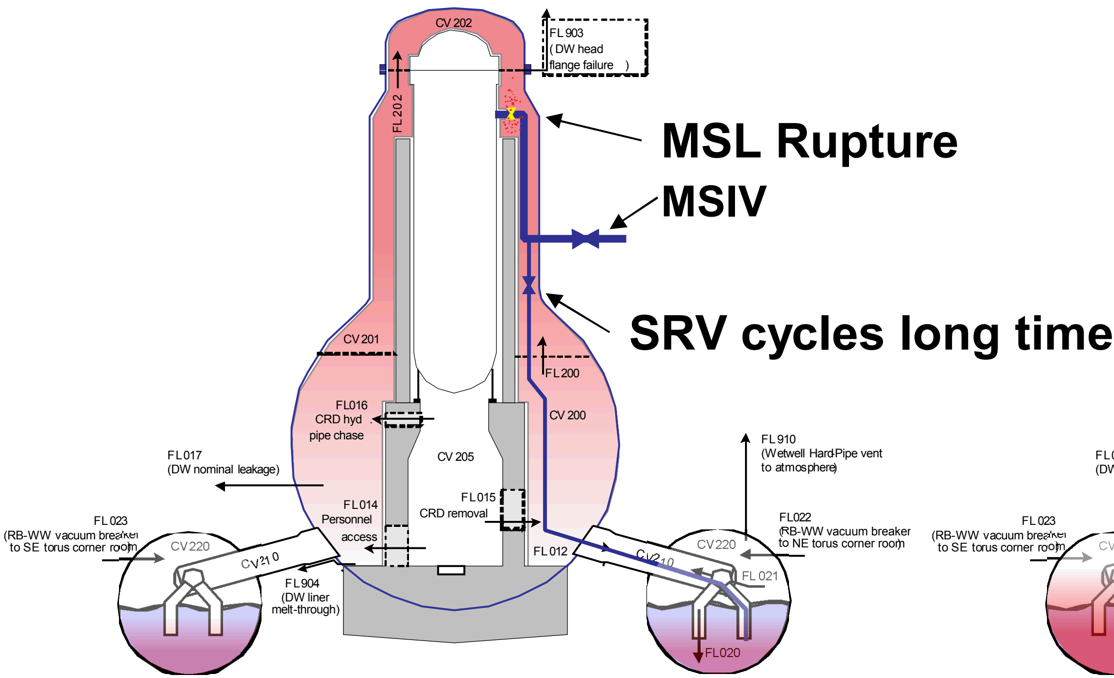
Influence of safety valve thermal thermal failure on Cesium release for Peach Bottom

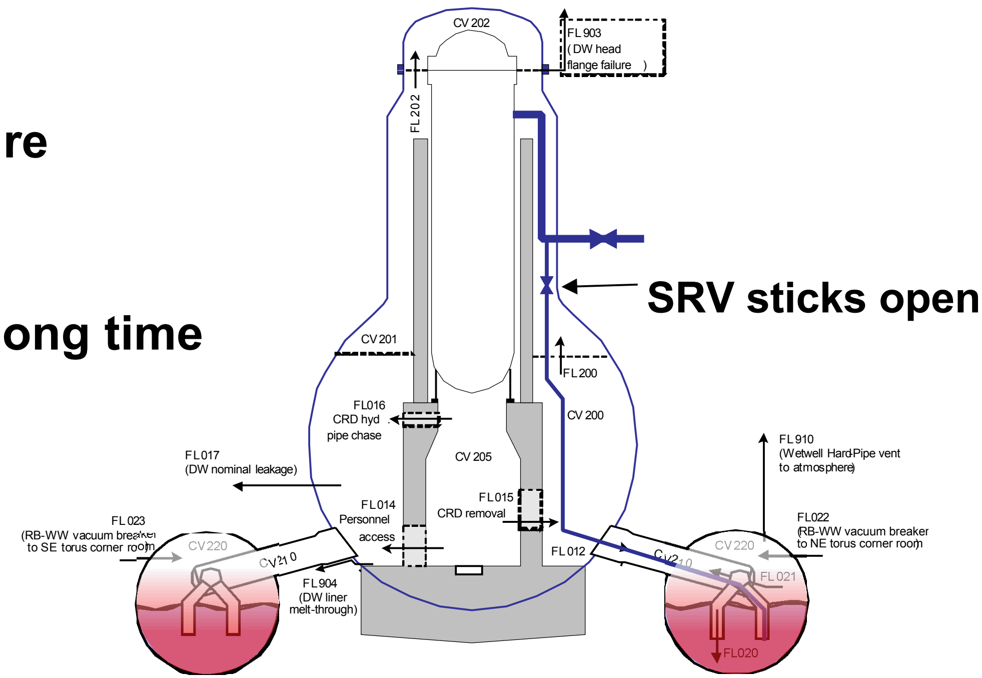




- Potential for main steam line creep failure controlled by SRVOAFRAC
- SRVOAFRAC is a log uniform [0.05,1] distribution
- Timing of safety relief valve failure is important (Stochastic vs. Thermal)
- SRV will fail thermally first in all cases for main steam line (MSL) creep failure

SRV failure vs. MSL failure





Main steam line failure vents fission products to drywell

Release to environment via drywell head flange or drywell liner melt through

SRV failure vents fission products into wetwell

Wetwell scrubbing minimizes release to the environment