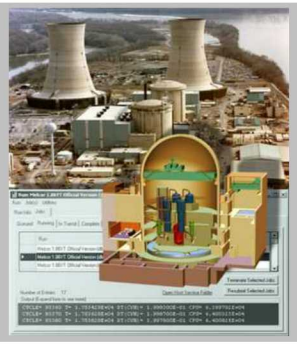
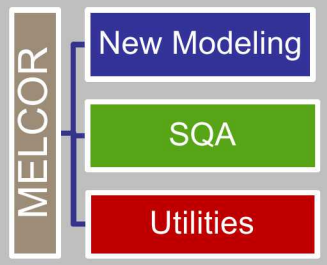
SAND2019-13216C



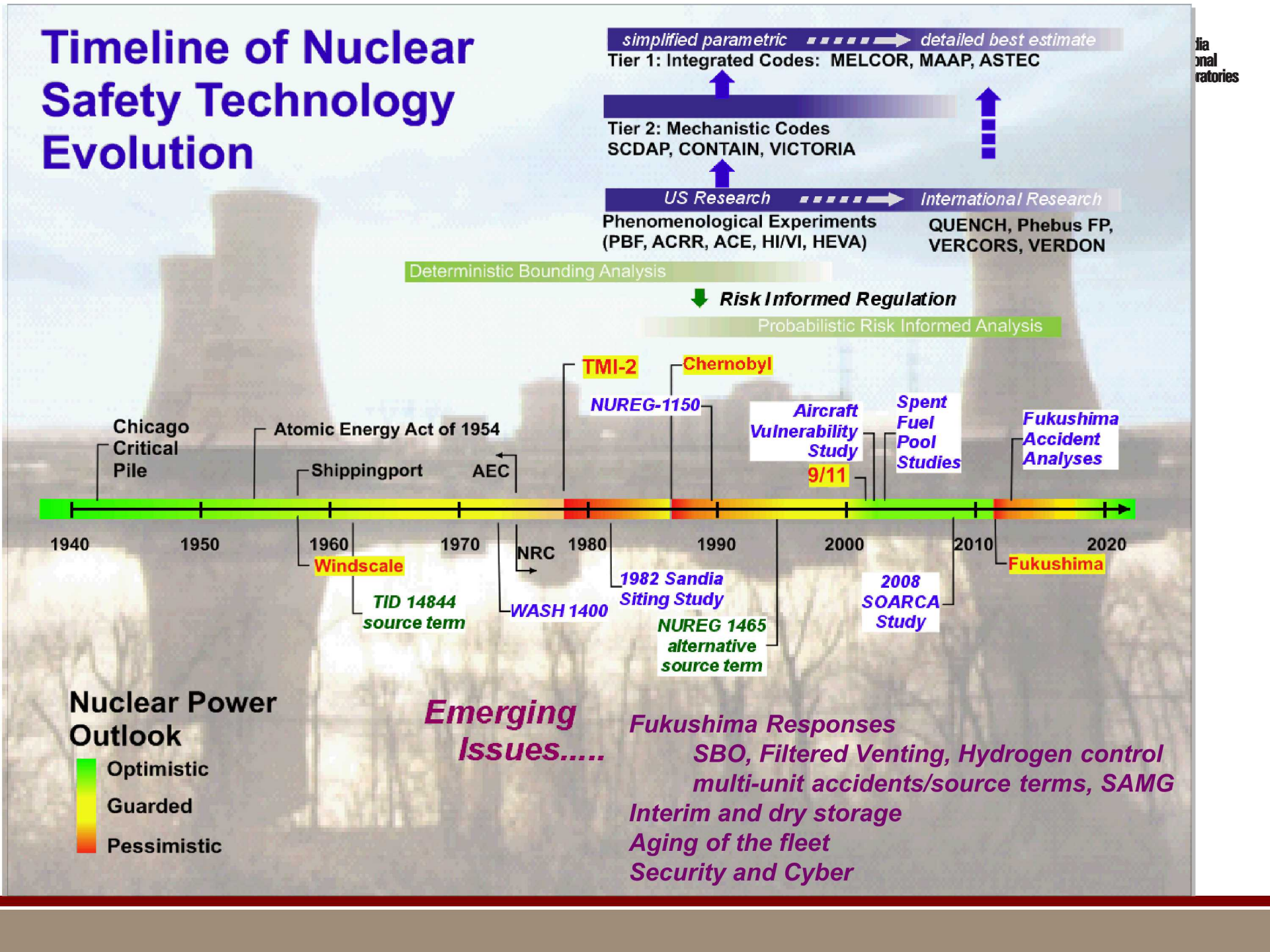


Status of MELCOR 2.2 and Plans For MELCOR 3.0

R.O. Gauntt
Severe Accident Analysis Department

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Timeline of Nuclear Safety Technology Evolution

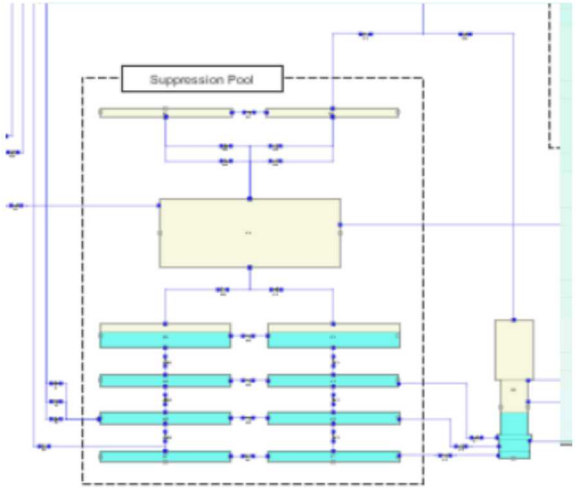


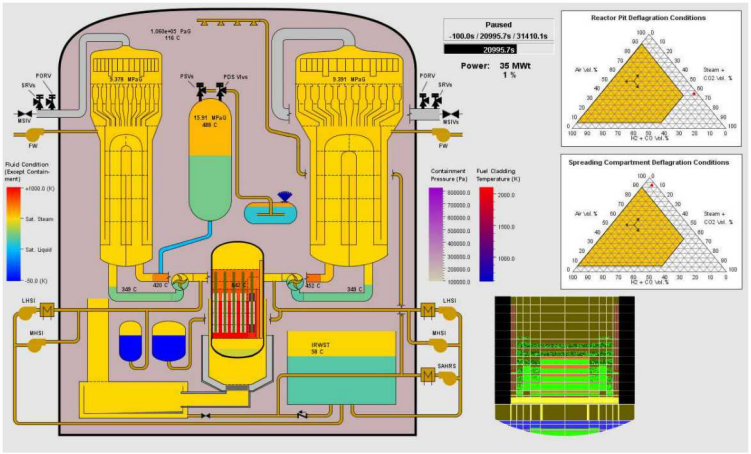
Emerging Issues.....

- Fukushima Responses
  - SBO, Filtered Venting, Hydrogen control
  - multi-unit accidents/source terms, SAMG
- Interim and dry storage
- Aging of the fleet
- Security and Cyber

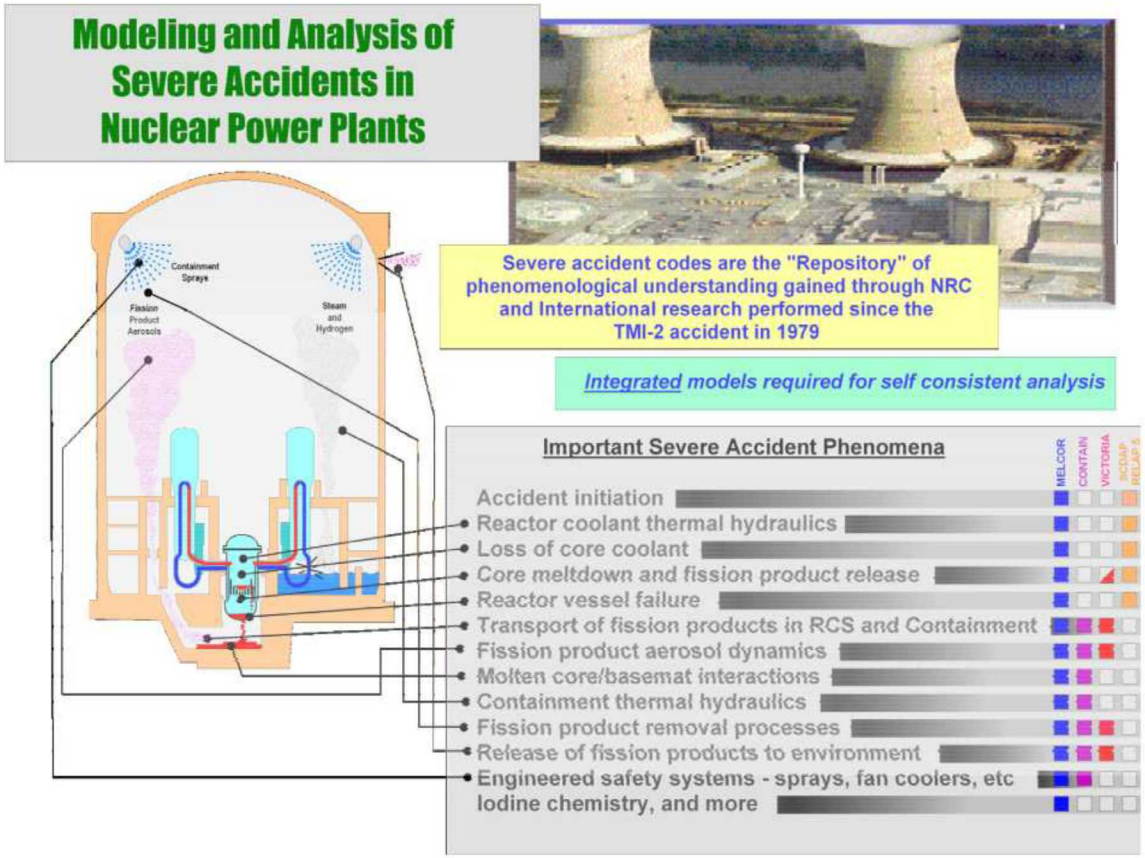
What is Required of a Severe Accident Code

- Fully Integrated, engineering-level code
  - Thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings;
  - Core heat-up, degradation, and relocation;
  - Core-concrete attack;
  - Hydrogen production, transport, and combustion;
  - Fission product release and transport behavior
- Application
  - User constructs models from basic constructs
    - Control volumes, flow paths, heat structures,
  - Multiple 'CORE' designs
    - PWR, BWR, HTGR (Pebble Bed & PMR), PWR-SFP, BWR-SFP, SMR, Sodium (Containment)
  - Adaptability to new reactor designs
- Validated physical models
  - ISPs, benchmarks, experiments, accidents
- Uncertainty Analysis
  - Relatively fast-running
  - Characterized numerical variance
- User Convenience
  - Windows/Linux versions
  - Utilities for constructing input decks (GUI)
  - Capabilities for post-processing, visualization
  - Extensive documentation





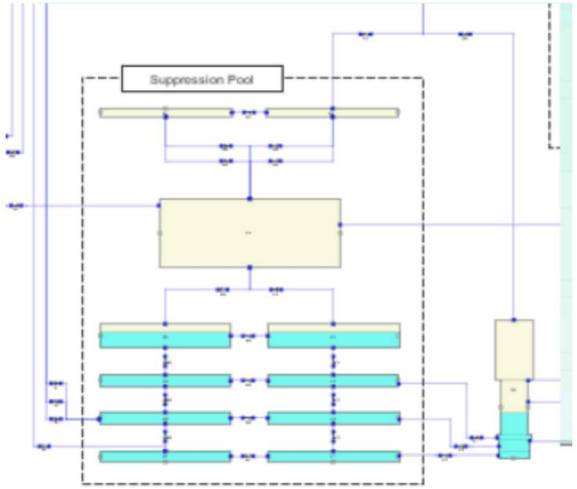
MELCOR – Severe Accident Analysis Code

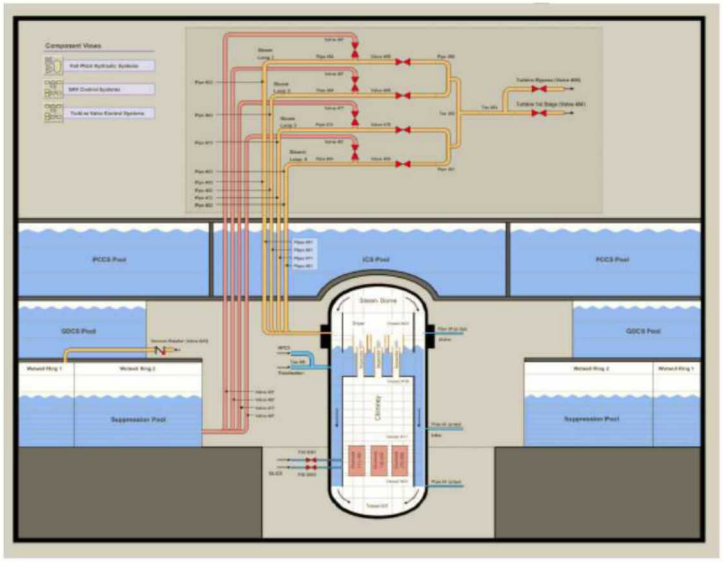


- Started in 1982 as integrated replacement for Source Term Code Package
- System level plant model of Nuclear Power Plant
- Emphasis on “best estimate”
- Repository of knowledge for future generations
- Global standard used internationally in over 31 nations
- Evolves to meet emerging regulatory issues

What is the MELCOR Code

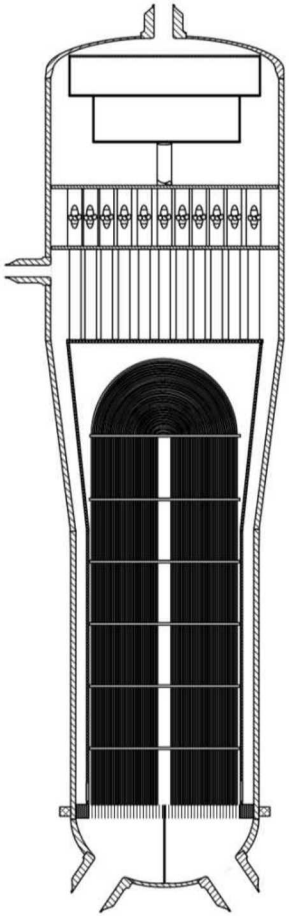
- NRC sponsored simulation code for analysis of accidents in nuclear power plants
  - Also applied to containment DBA simulation
  - PWR, BWR, HTGR, PWR-SFP, BWR-SFP
- Fully Integrated, engineering-level code
  - Thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings;
  - Core heat-up, degradation, and relocation;
  - Core-concrete attack;
  - Hydrogen production, transport, and combustion;
  - Fission product release and transport behavior
- Desk-top application
  - Windows/Linux versions
  - Relatively fast-running
    - One or two days common
    - One or two weeks possible
    - Project to improve code performance
  - SNAP for post-processing, visualization, and GUI

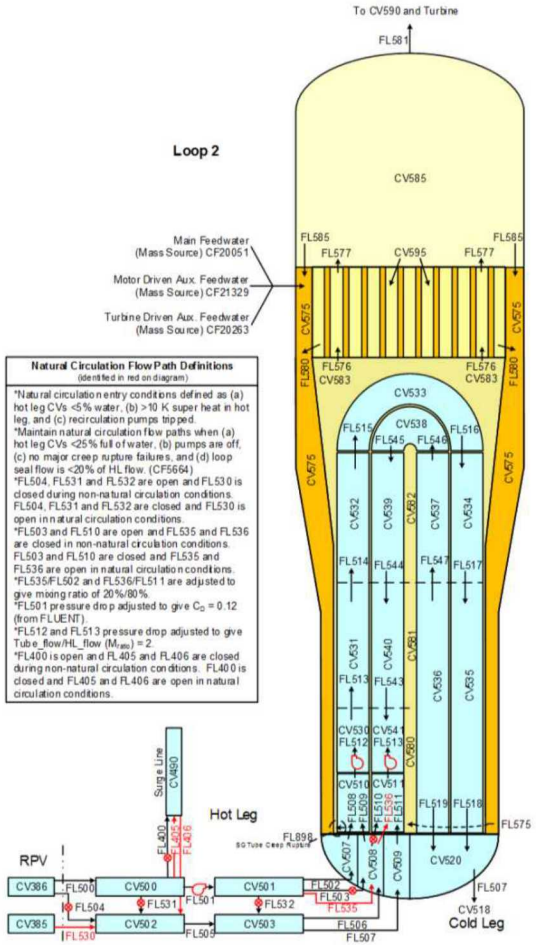


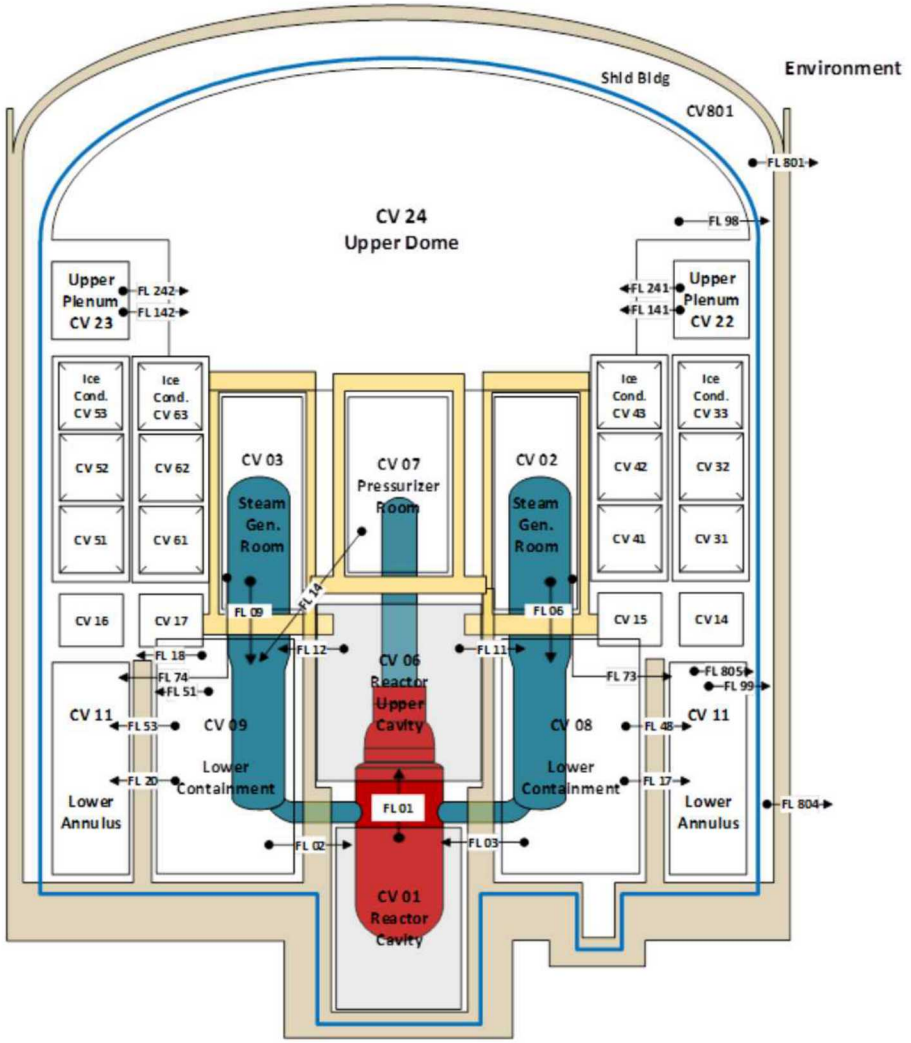


Modeling Nuclear Power Plant

Sandia National Laboratories







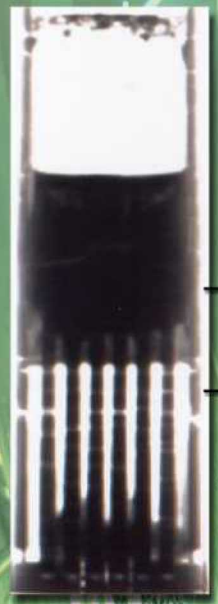
History of Core Melt Progression Research at Sandia

DF-1 & 2
Fuel Rod Heatup, Oxidation & Initial Damage

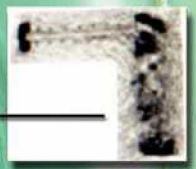
DF-3 & 4
PWR & BWR Control Rod Behavior in Fueled Bundle

ST-1 & 2
Irradiated Fuel & Fission Product Effects

MP-1 & 2
Late Phase Melt Progression Behavior

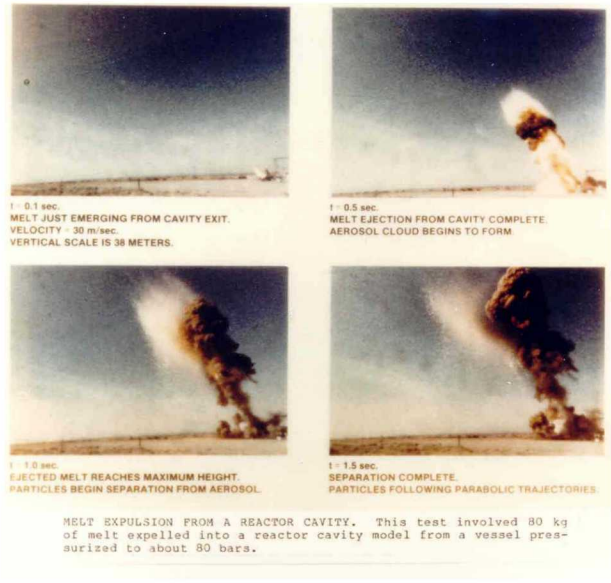


XR-1 & 2
Metallic Melt Relocation Behavior in BWRs



LWR Research into Severe Accident Phenomena

Molten Core/Concrete Interactions China Syndrome



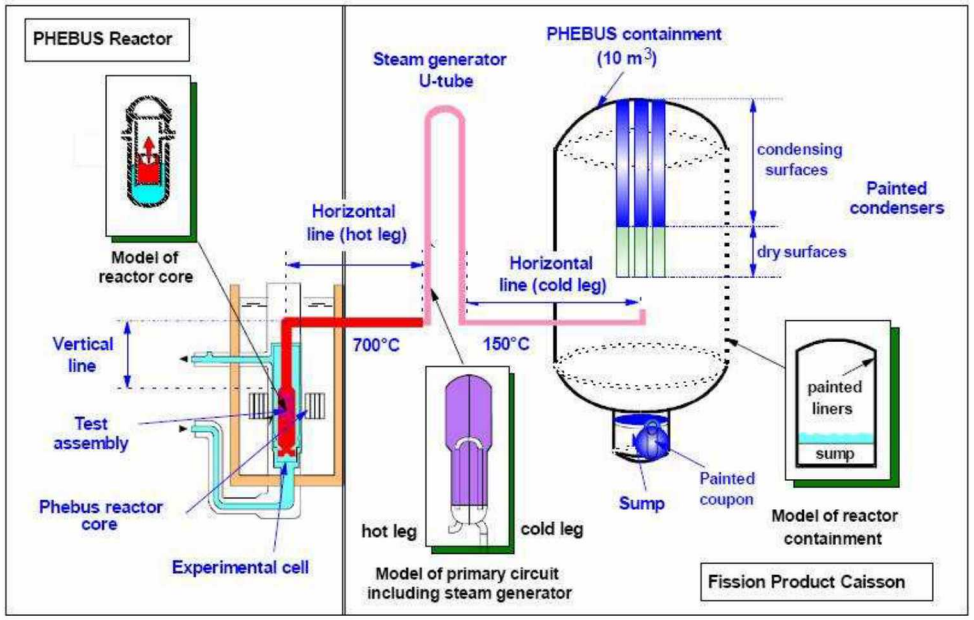
Direct Containment Heating Experiments

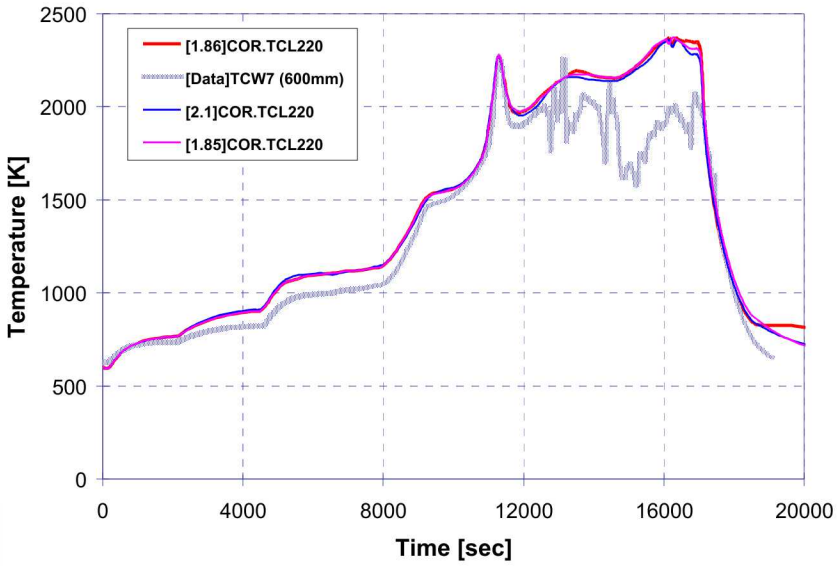
Steam explosion research

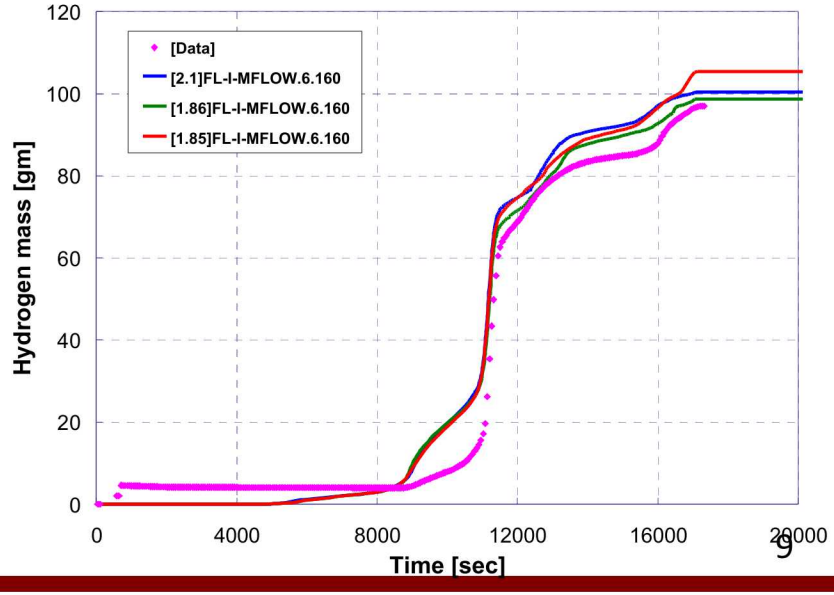
Containment response to severe accident loads

FPT-1 (ISP-46)

- ***General Description***
  - FPT-1 experiment was an in-pile, irradiated fuel experiment conducted in the PHEBUS Fission Product Facility by the Nuclear Safety and Protection Institute (IPSN) at Cadarache, France, on July 26, 1996. The objective of the fuel bundle assembly was to assess fuel degradation and fission product release
- ***Important Physics***
  - Oxidation
  - Material relocation
  - Fission product release, transport, and deposition



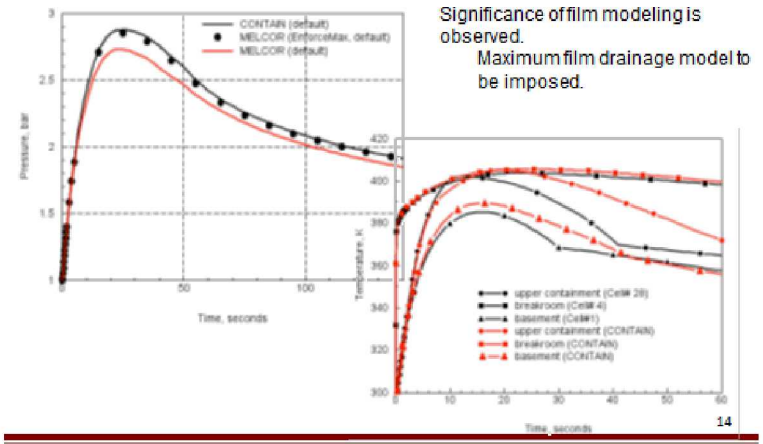




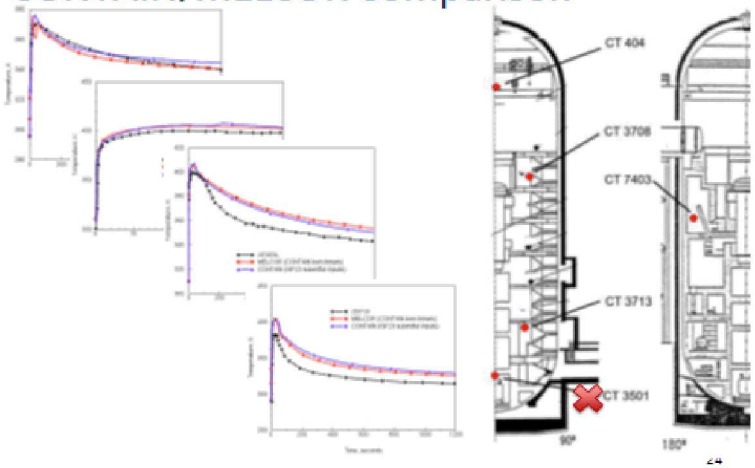
MELCOR Containment Thermal Hydraulics

- Test provides an indication of the effect of forced convective condensation during a blowdown event.
- Significance of film modeling is observed.
  - Code enhancement to permit a maximum film drainage model to be imposed (like CONTAIN).
  - New model permits investigation of the relevance of film depth, the corresponding heat transfer, and impact to peak pressure

V44 (ISP 16) CONTAIN/MELCOR Comparison



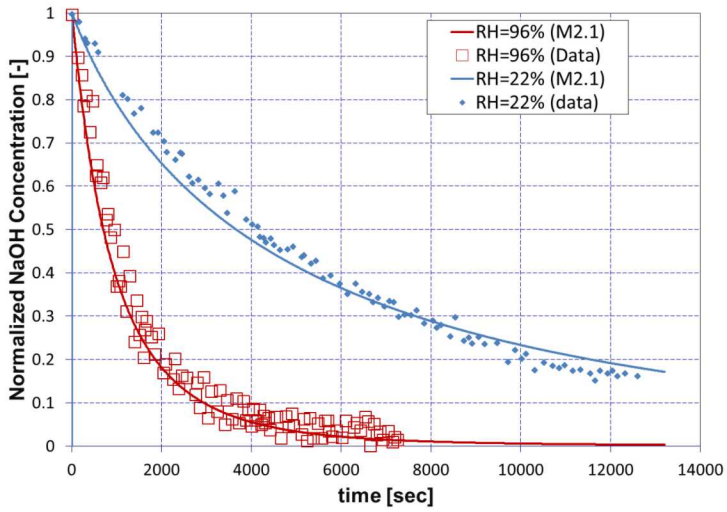
T31.5 (ISP-23) CONTAIN/MELCOR Comparison



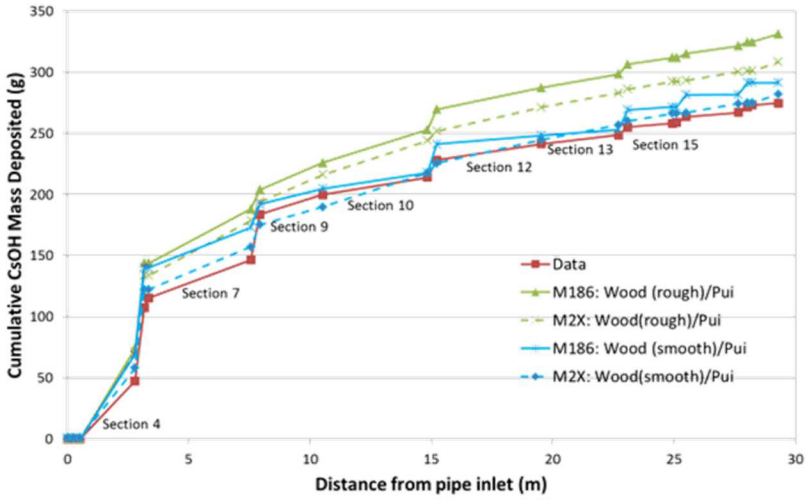
Radionuclide Package

- Tracks the release and transport of
  - Fission product vapors & aerosols
  - Non-radioactive masses such as water, concrete, etc.
  - Traces hosted by other materials
    - Negligible volume and heat capacity
- Aerosol physics
  - MAEROS
- Agglomeration of aerosols
  - Several mechanisms cause collisions and sticking to produce larger particles
    - Brownian diffusion
    - Differential gravitational settling
    - Turbulent agglomerating by shear and inertial forces
- Hygroscopic effects
- Condensation & evaporation
  - TRAP-MELT
- Deposition on surfaces
  - Modeled as always sticking to surfaces contacted
  - Several mechanisms drive aerosols to surfaces
    - Gravity
    - Brownian diffusion
    - Thermophoresis
    - Diffusiophoresis
    - Turbulent deposition
- Pool Scrubbing
  - SPARC
- Validation
  - ABCOVE, ACE, AHMED, DEMONA, LACE, LOFT, PHEBUS, POSEIDON, STORM, …

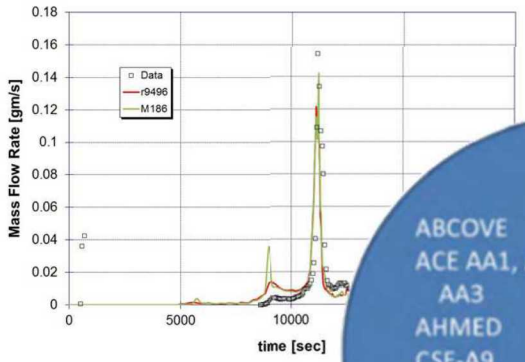
AHMED – Hygroscopic Effects



LA3 – Turbulent Deposition



MELCOR Code Validation Tests



ABCOVE
ACE AA1, AA2, AA3
AHMED
CSE-A9
DEMONA
FALCON 1&2
LACE LA1 & LA3
LACE-LA4
Marviken ATT-4
Poseidon
RTF ISP-41
STORM
VANAM-M3
VERCORS
VI (ORNL)

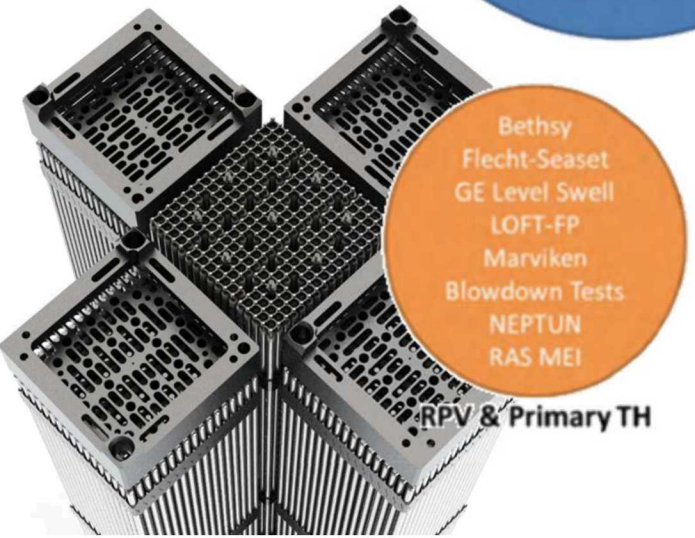
RN Physics / Transport

CORA-13
DF-4
FPT1 & FPT3
LHF/OLHF
LOFT-FP2
MP1 & MP2 (SNL)
PBF-SFD
Quench 11
VERCORS
VI (ORNL)

Core Heatup & Degradation

Integral / Accidents

FPT1 & FPT3
TMI-2
Fukushima



Bethsy
Flecht-Seaset
GE Level Swell
LOFT-FP
Marviken
Blowdown Tests
NEPTUN
RAS MEI

RPV & Primary TH

VERCORS
VI (ORNL))

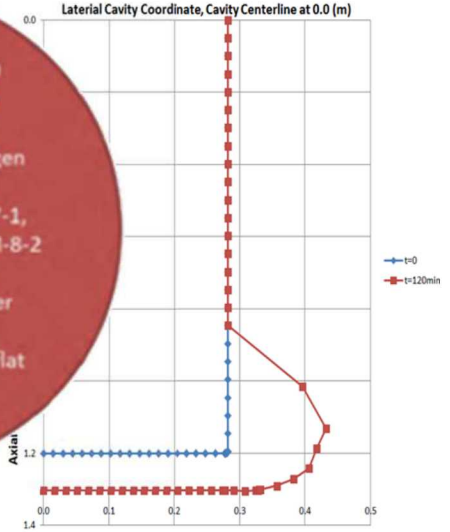
RN Release

IET-DCH
OECD-MCCI
SURC

Ex-Vessel Debris

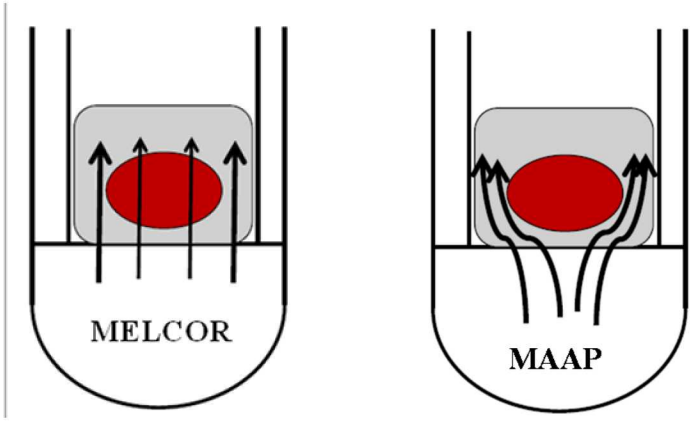
CSE-A9
CSTF Ice Condenser Test
CVTR
DEHBI
GE Mark II Suppression Pool
HDR E-11
HDR V44
IET 1 thogh IET7 and IET 9
JAERI Spray Tests
NST Hydrogen Burn
NUPEC M-7-1, M-8-1, M-8-2
PNL Ice Condenser Tests
Wisconsin flat plate

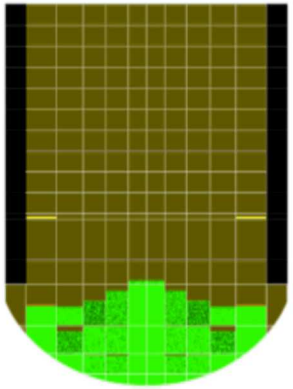
Containment

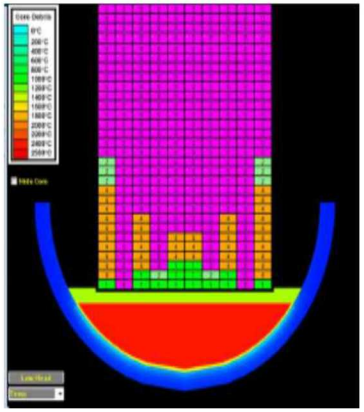


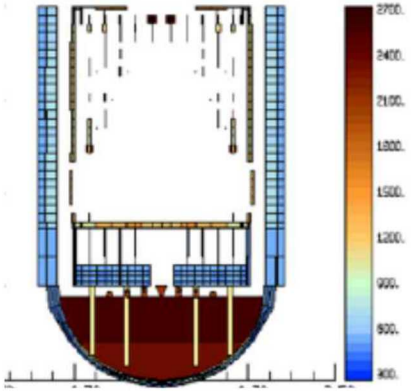
Core Degradation Modeling – Cross walk comparisons

- Differences in assumption of permeability of debris crust
  - MELCOR – flow blockage model with permeable crust
  - MAAP –impermeable crust
- Fuel rod collapse modeling
  - MELCOR/MAAP – fuel rod collapse model
  - ASTEC – Rods melt to form magma but no collapse





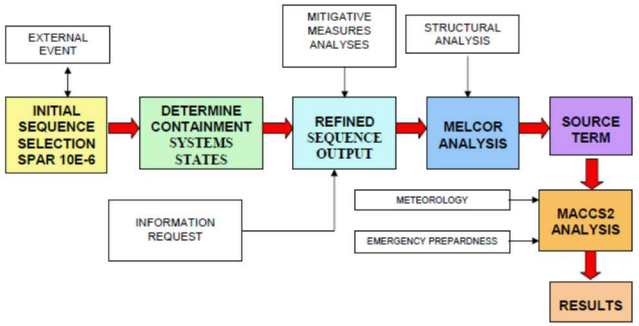




MELCOR Applications

- Forensic analysis of accidents – Fukushima, TMI
- State-of-the-art Reactor Consequence Analysis- SOARCA
- License Amendments
- Risk informed regulation
- Design Certification
- Preliminary Analysis of new designs
- Support of International Regulatory Bodies
- Non-reactor applications
  - Leak Path Factor Analysis
  - DOE Safety Software "Toolbox" code

SOARCA PROCESS



MOX and HBU Release Modeling

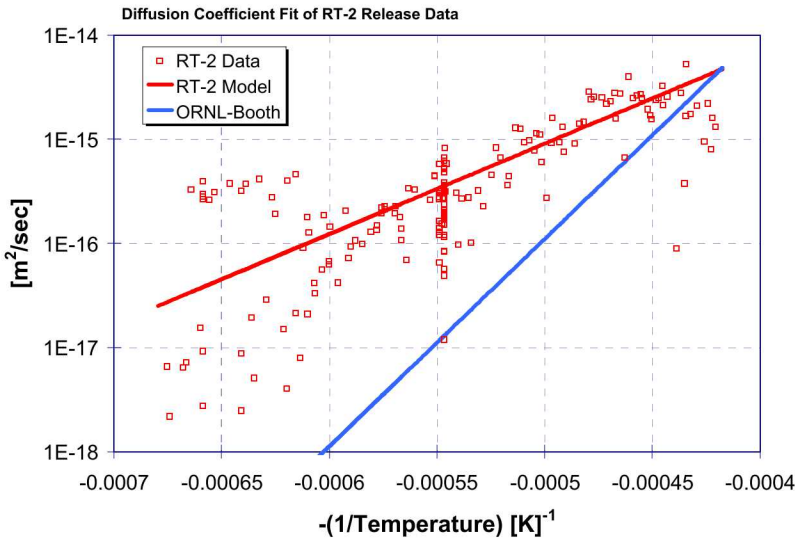
$$F(t) = 6\sqrt{\frac{Dt}{\pi a^2}} - 3\frac{Dt}{a^2} \quad for \quad \frac{Dt}{a^2} \le 0.1547$$

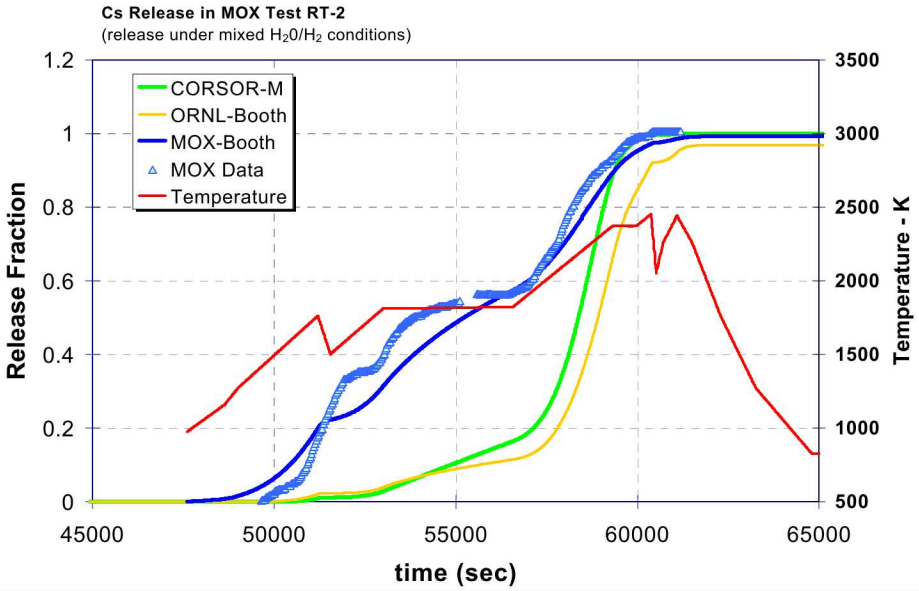
or,

$$F(t) = 1 - \frac{6}{\pi^2}\exp\left(\frac{-\pi^2 Dt}{a^2}\right) \quad for \quad \frac{Dt}{a^2} > 0.1547$$

fractional release vs D(T) and time

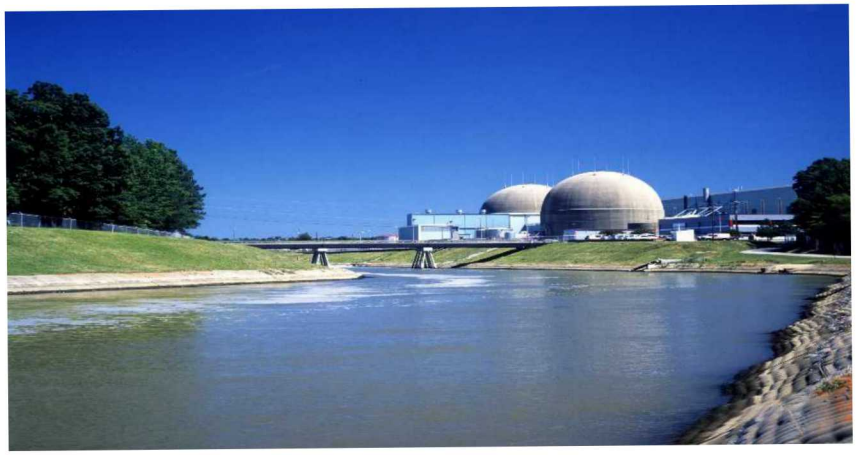
- French VERCORS data used to determine D(T) for MOX and HBU
- MOX and HBU show higher release rate from fuel
- Re-fitted D(T) captures release well





SOARCA Study

Peach Bottom Atomic Power Station

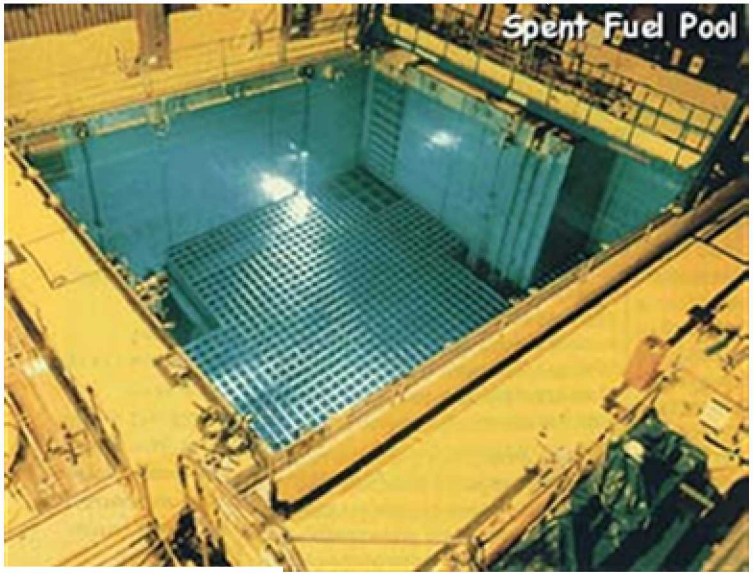


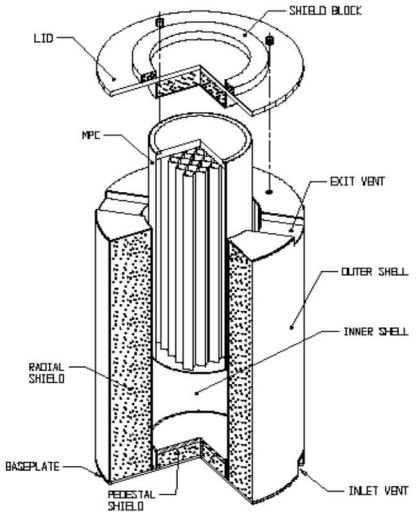
Surry Power Station

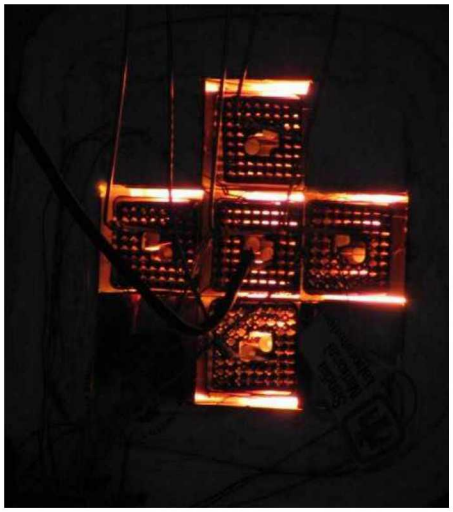
Sequoyah Nuclear Generating Station

Spent Fuel

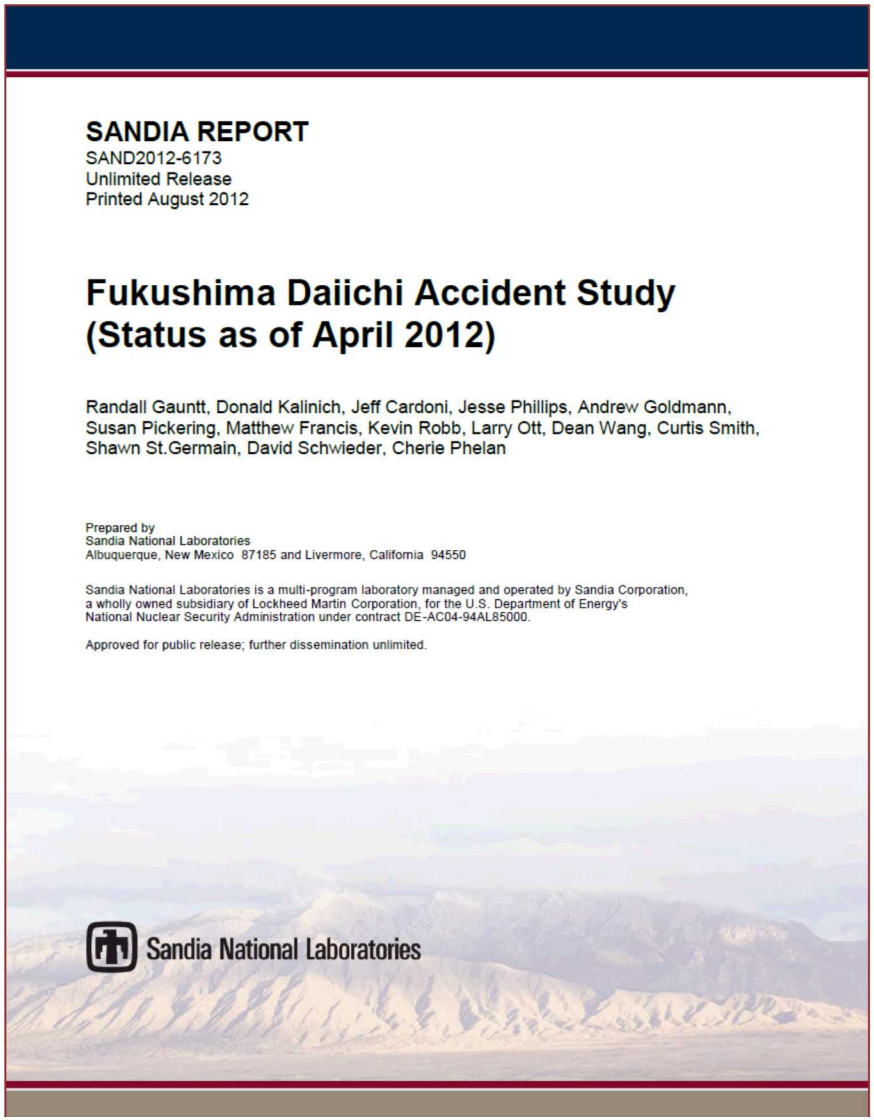
- Spent fuel pool risk studies
  - Improvements to phenomenological models for air oxidation and thermal radiation have been made
  - Improved modeling in MELCOR for BWR geometry
  - PWR analyses next
- Multi-unit accidents (large area destruction)
  - Reactors (multi unit)
  - Spent fuel pools
  - Complex source terms for MELCOR and MACCS
- Dry Storage
  - Dry casks analyses with MELCOR and MACCS







Fukushima Accident Reconstruction Study



SANDIA REPORT
SAND2012-6173
Unlimited Release
Printed August 2012

Fukushima Daiichi Accident Study (Status as of April 2012)

Randall Gauntt, Donald Kalinich, Jeff Cardoni, Jesse Phillips, Andrew Goldmann, Susan Pickering, Matthew Francis, Kevin Robb, Larry Ott, Dean Wang, Curtis Smith, Shawn St.Germain, David Schwieder, Cherie Phelan

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185 and Livermore, California 94550

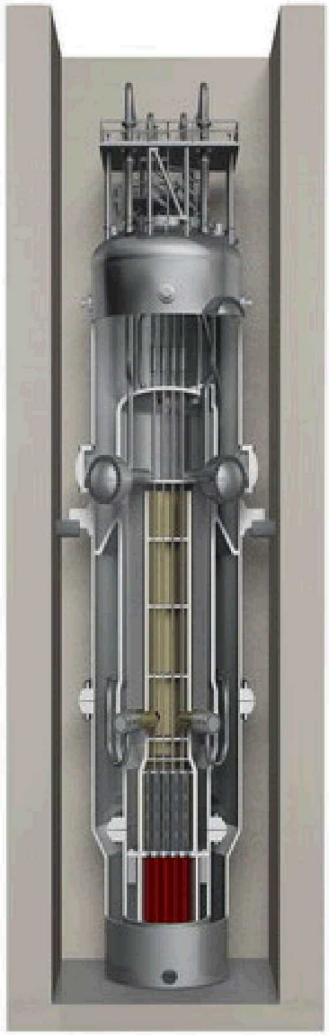
Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Sandia National Laboratories

- DOE commissioned an accident reconstruction study
- MELCOR used in a multi-lab effort led by Sandia
- Fukushima accident sequences proceeded quite similar to SOARCA Peach Bottom LTSBO and STSBO cases
- Studies continue with OECD BSAF project

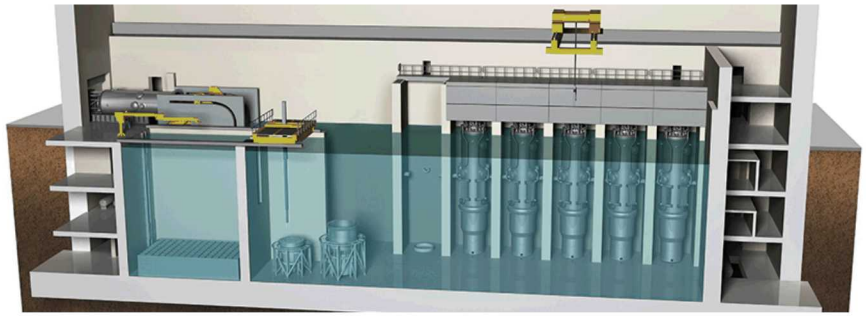
Advanced Reactors NuScale SMR (Reactor)

- New MELCOR Models
  - Multiple vessels (reactor pressure vessel and containment vessel)
  - Added new shroud model for heat transfer from reflectors
  - Multi-HS radiation enclosure model
  - Aerosol re-suspension model
  - Zukauskas heat transfer coefficient (external cross-flow across a tube bundle)
  - Simplistic bubble swell model
  - Multiple fuel rod types in a COR cell

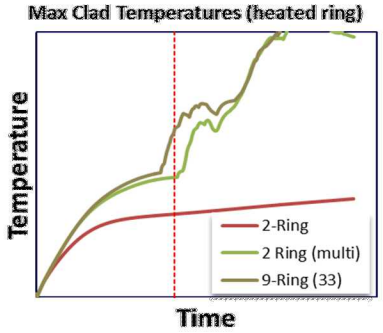


Advanced Reactors NuScale SMR (Spent Fuel Pool)

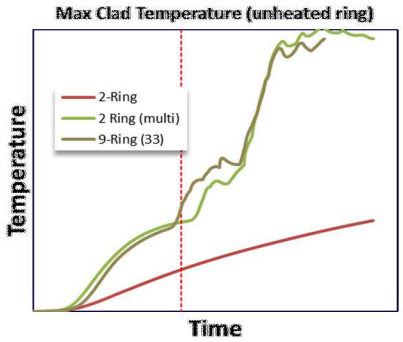
- New Models
  - Multi-rod model to allow adequate simulation of radiation between rods
  - New air oxidation model (Paul Scherrer Institute)
  - MELCOR modeling of multiple connected spent fuel pools
  - Release and tracking of release for regions of different burnup
- Validation
  - Validation against Sandia Spent Fuel Pool experiments

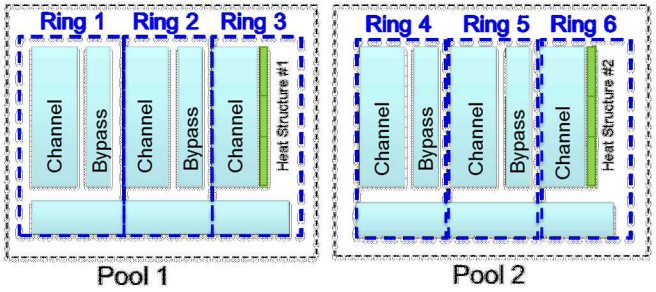


Hot Assembly



Cold Assembly



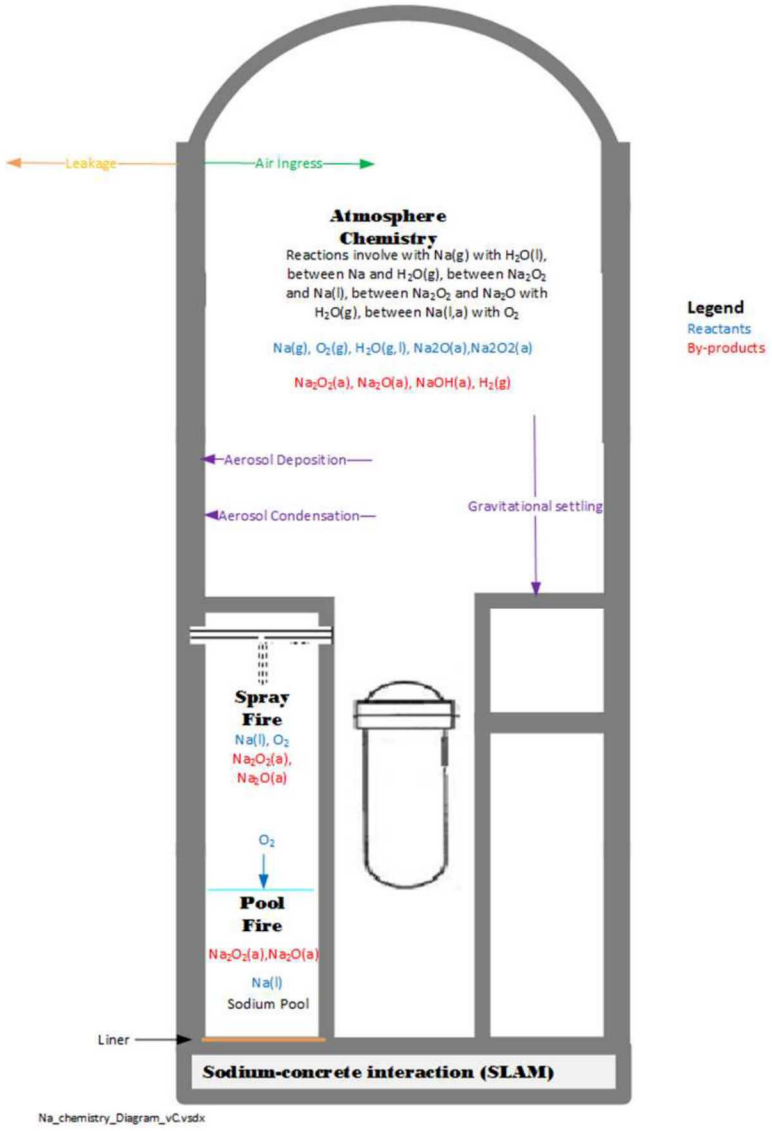


Advanced Reactors Sodium Fast Reactors (Containment Analyses)

Sandia National Laboratories

New MELCOR Models

- Sodium Working Fluid
- Atmosphere Chemistry Models
- Spray Fire Models
- Pool Fire Models
- Sodium-concrete Interaction Models



HTGR Capability

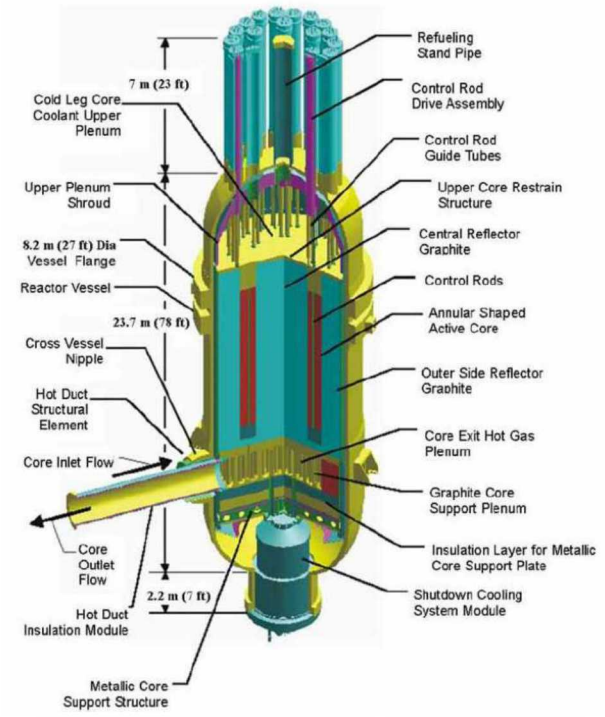
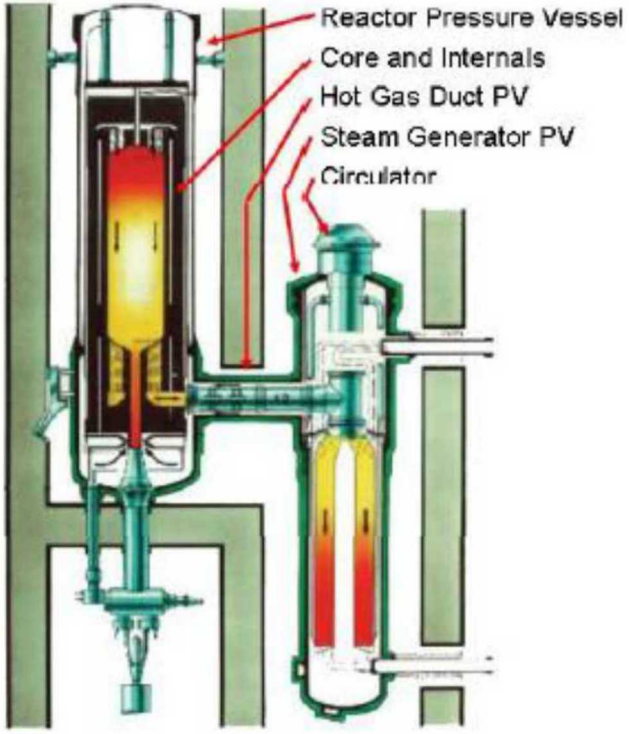


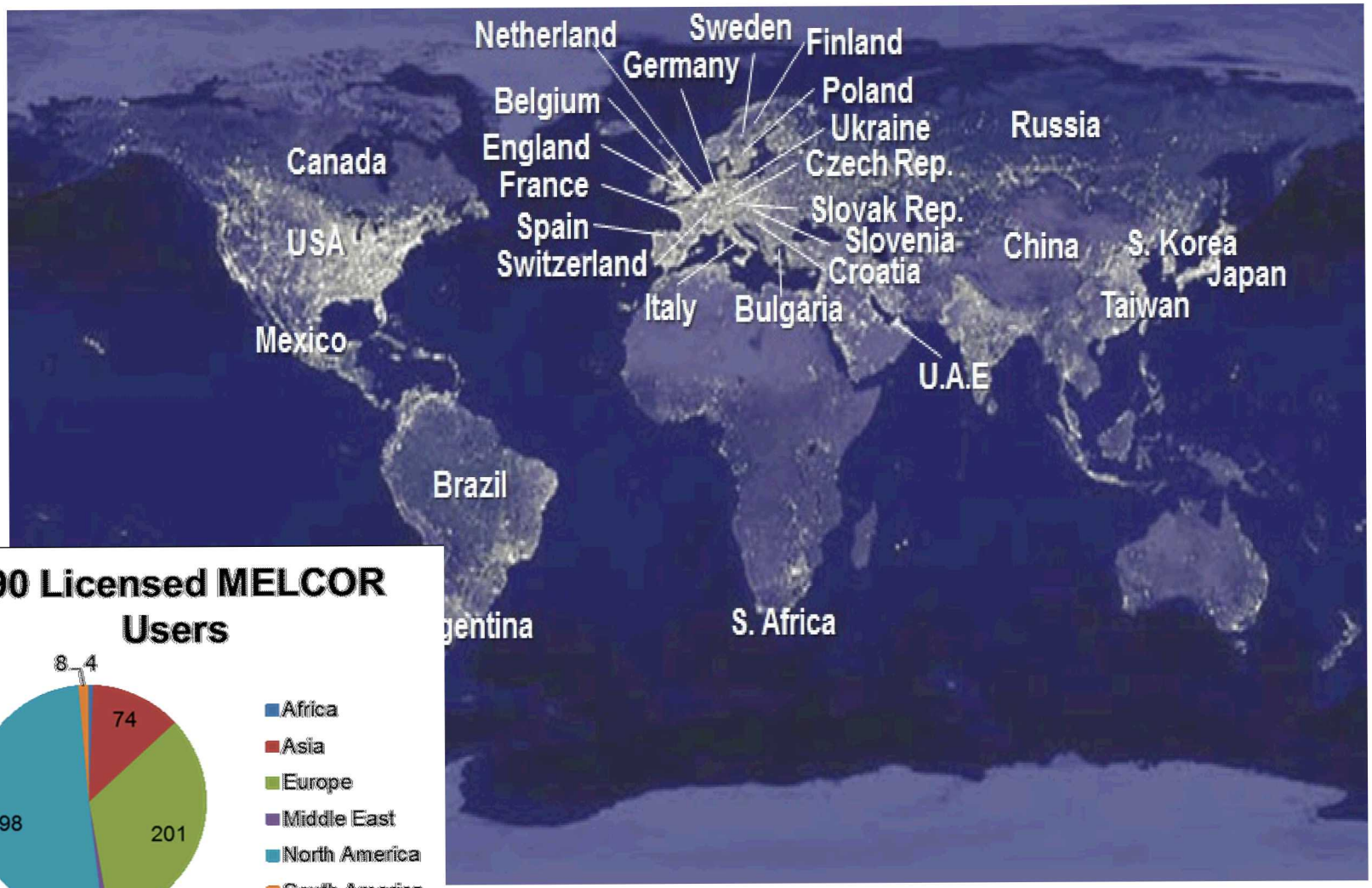
Figure 1: VHTR Nuclear Heat Source based on prior U.S. DOE Prismatic Modular HTGRs



- Pebble bed
- Prismatic
- Graphite oxidation
- Triso fuel failure model

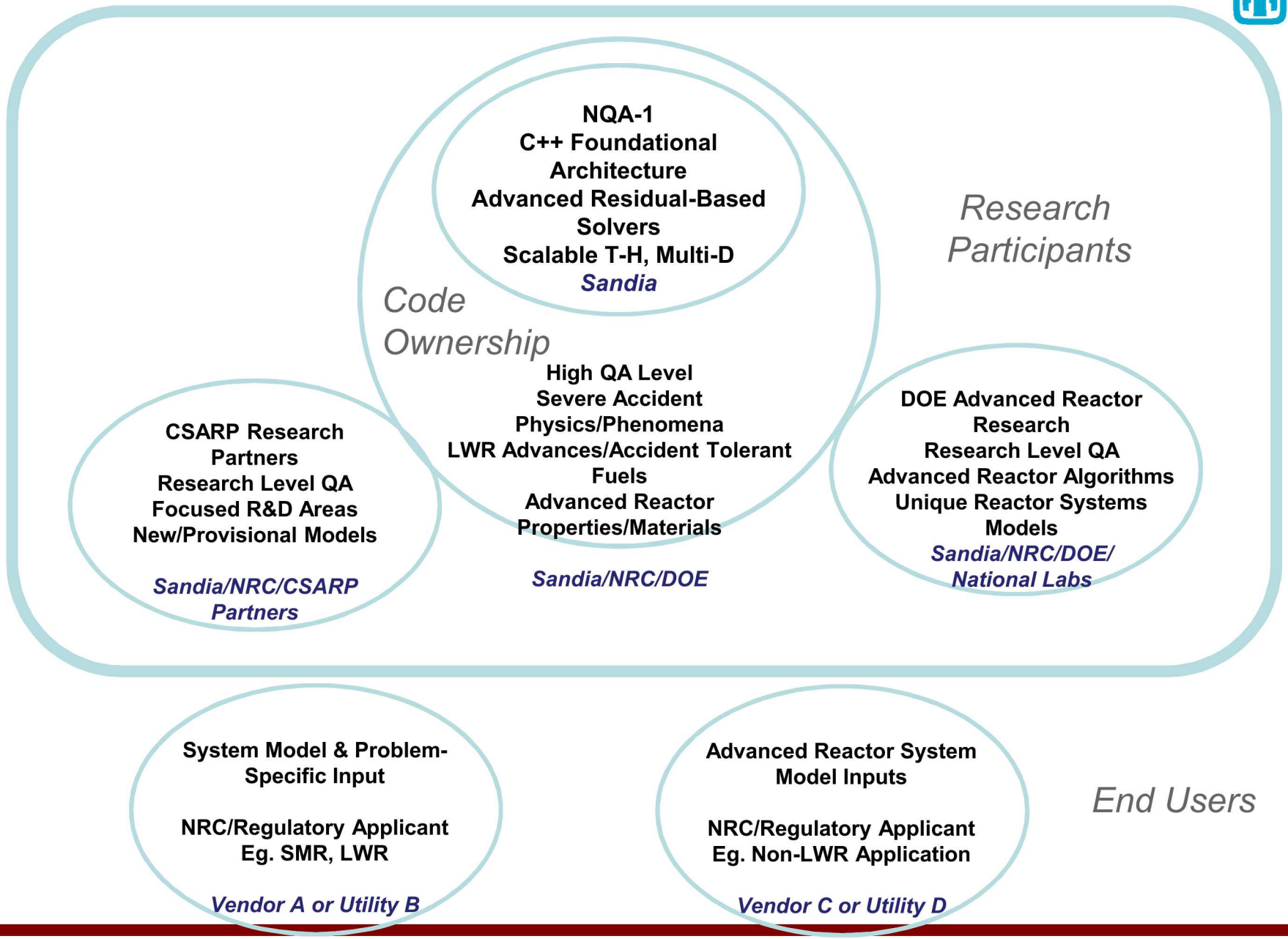
International Use of MELCOR

Sandia National Laboratories



MELCOR 3 Layered Software Design

Sandia National Laboratories



Sandia National Laboratories

Advanced Software Design Encapsuled for Ease of Maintenance and Development Separation of Physics and Solvers

Modern Software "Orthogonal" Design Separates Functions

Legacy Software Hopelessly Intertwined Physics and solvers

MEMORY Hardware optz.

CORE

THERMO

RESIDUALS FOR MASS, MOM, ENERGY

SOLVERS

T-H

SOLVERS N-K, etc.

PHYSICS 1

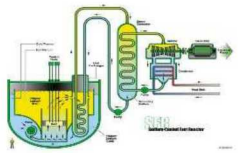
PHYSICS 2

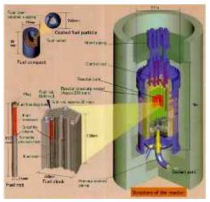
WORKING FLUIDS EOS, HX, friction

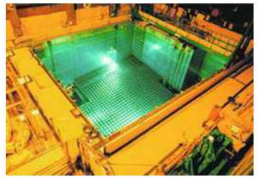
SPECIAL MODELS Pumps, turbines, acc.

Accident Tolerant And Advanced Fuels

Advanced Reactors & Spent Fuel Pools

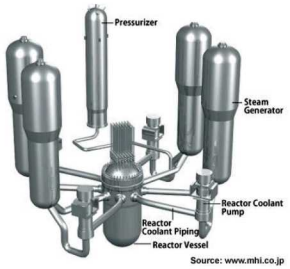




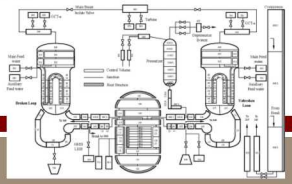


MELCOR 3 Leverages Advances in Software Design, Numerical Solvers, Evolving Hardware With Lower Development Costs & Expanded Capabilities

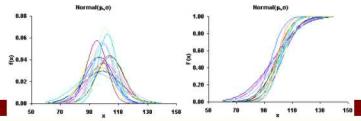
Scalable Thermal-Hydraulics Mass, Momentum, Energy 3-Eq to 9-Eq



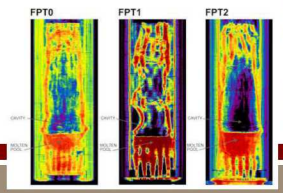
Scalable Dimensionality 0-D to 3-D



User Defined Accuracy and Uncertainty Quantification



Extensive Existing Phenomenological Validation Basis



Sandia National Laboratories

end