U.S. NRC Research Activities on Severe Accident Progression and Source Term Analysis

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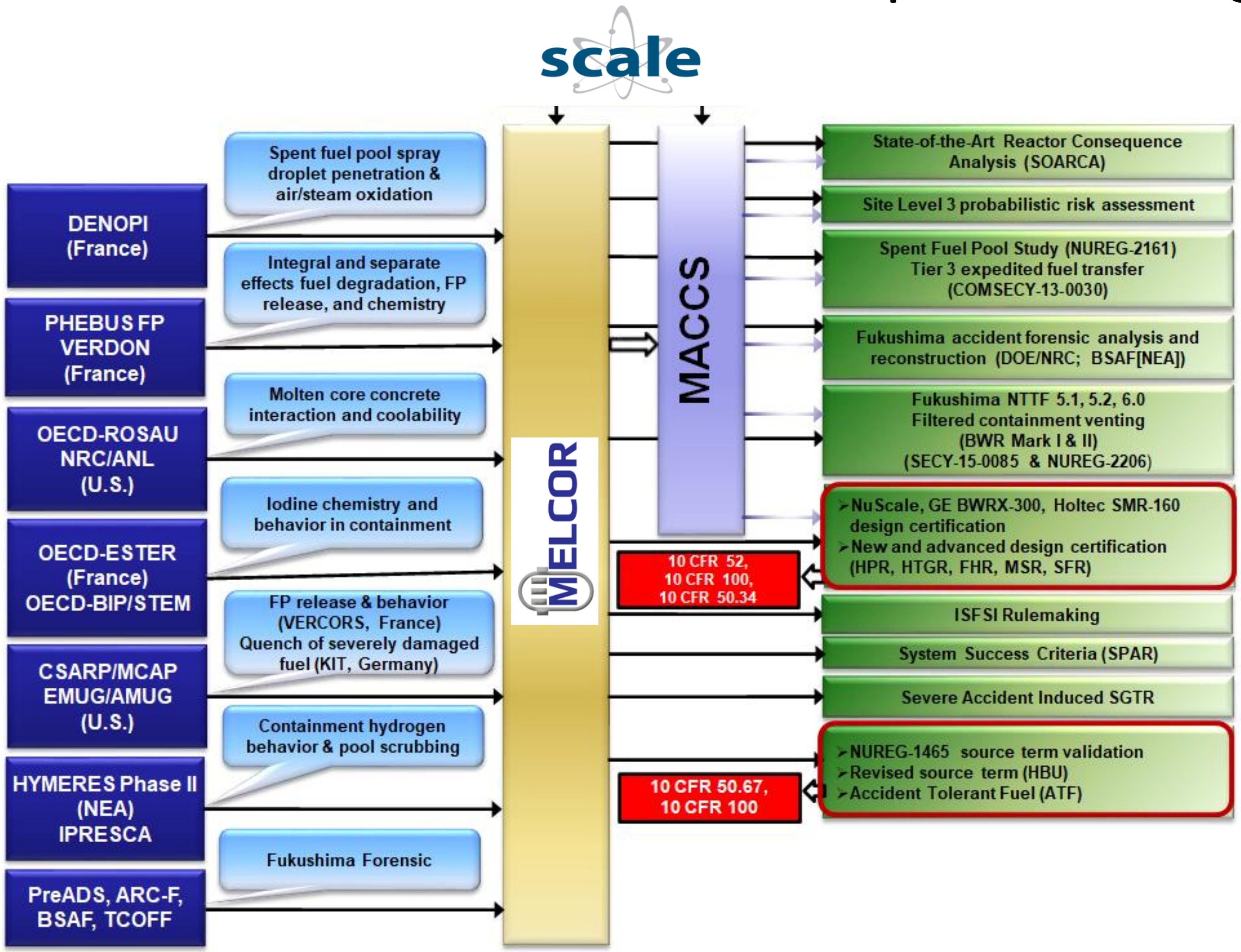
Office of Nuclear Regulatory Research

European MELCOR User Group Meeting
Brno, Czech Republic
April 7-11, 2025

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Severe Accident Code Development & Regulatory Applications

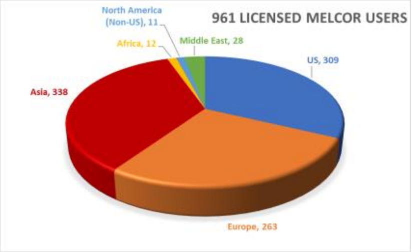


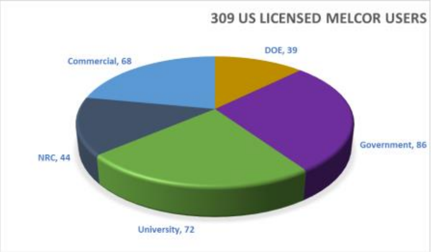
What Is It?
MELCOR is an engineering-level code that simulates the response of the reactor core, primary coolant system, containment, and surrounding buildings to a severe accident.

Who Uses It?
MELCOR is used by domestic universities and national laboratories, and international organizations in around 30 countries. It is distributed as part of NRC's Cooperative Severe Accident Research Program (CSARP).

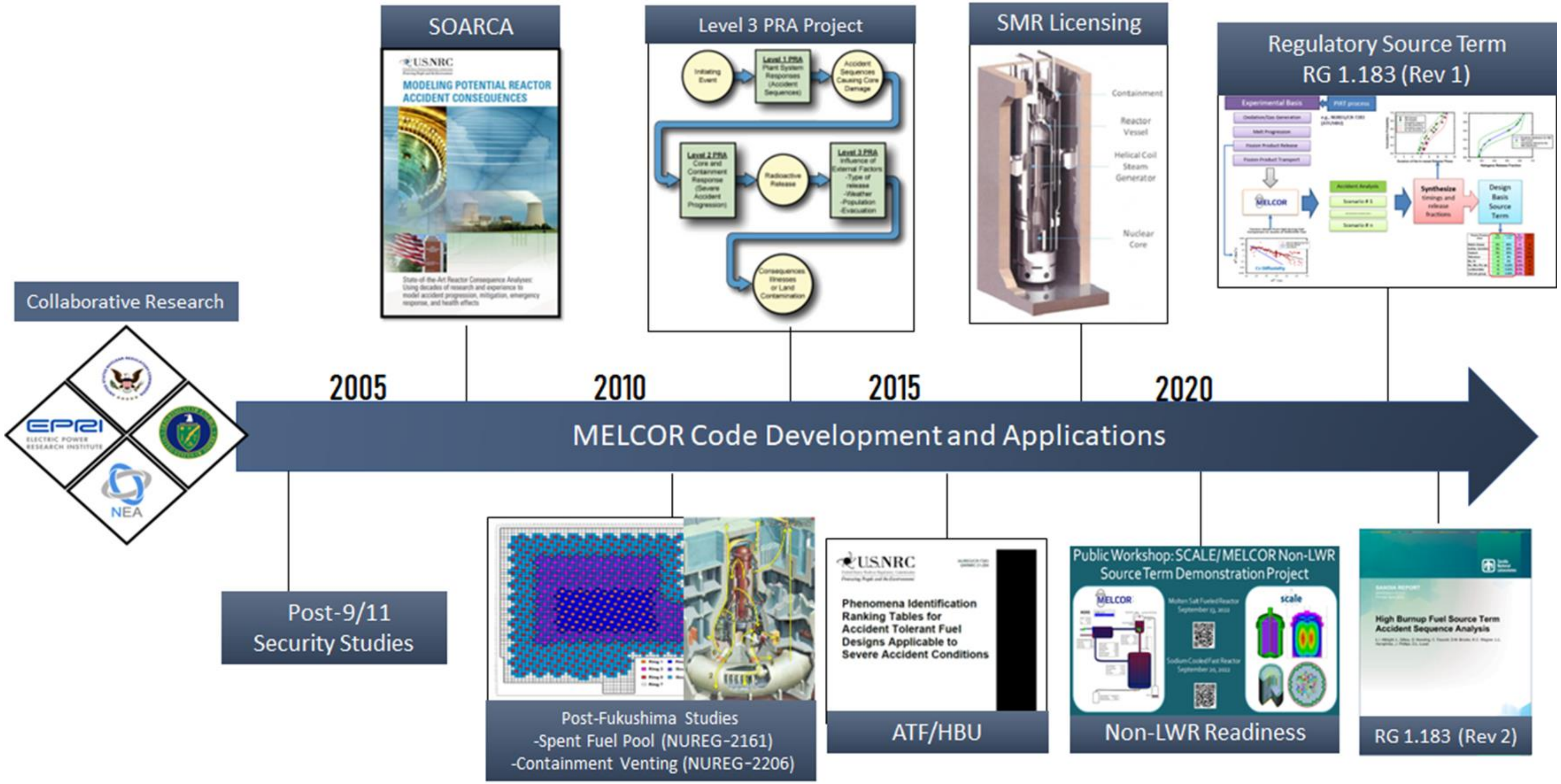
How Is It Used?
MELCOR is used to support severe accident and source term activities at NRC, including the development of regulatory source terms for LWRs, analysis of success criteria for probabilistic risk assessment models, site risk studies, and forensic analysis of the Fukushima accident.

How Has It Been Assessed?
MELCOR has been validated against numerous international standard problems, benchmarks, separate effects (e.g., VERCORS) and integral experiments (e.g., Phebus FPT), and reactor accidents (e.g., TMI-2, Fukushima).





Severe Accident Code Development & Regulatory Applications



NRC International Severe Accident Experiments

- **1988-2010** — IRSN: Phébus-Fission Products Program
- **2005-2010** — IRSN: Phébus-International Source Term Program
- **2006-2019** — NEA: Behavior of Iodine Project (BIP)
- **2005-2016** — IRSN: Experimental Program for Iodine Chemistry Under Radiation (EPICUR)
- **2011-2019** — NEA: Source Term Evaluation and Mitigation (STEM) Project
- **2013–2018** — NEA: Benchmark Study of the Accident at Fukushima (BSAF) Project
- **2017–2021** — NEA: Preparatory Study on Analysis of Fuel Debris (PreADES) Project
- **2019–2021** — NEA: Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima (ARC-F)
- **2019–2023** — EU: Management and Uncertainties of Severe Accidents (MUSA)
- **2020–2024** — NEA: Experiments on Source Term for delayed Releases (ESTER); Reduction of Severe Accident Uncertainties (ROSAU)
- **2022-2024** — NEA: Thermodynamic Characterization Of Fuel debris and Fission (TCOFF-2)
- **2023–2026** — NEA: Fukushima Accident Information Collection & Evaluation (FACE)
- **2024–2028** — NEW — NEA: POol during Loss of Cooling Accident (POLCA)
- **2024–2028** — NEW — NEA: Fission Products Behavior under Severe Accident with New Fuel Concepts (FORESEEN)
- **2024–2029** — NEW — NEA: COrium Properties for reactor Simulation and uncertainties (COPS)

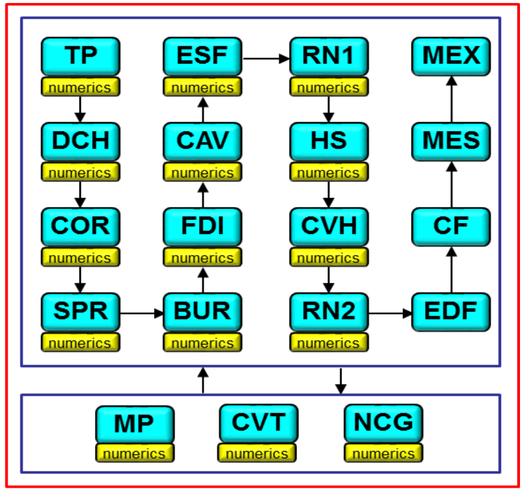
SASPAM
SEAKNOT

MELCOR User Groups & Technical Meetings

Cooperative Severe Accident Research Program (CSARP) – June/U.S.A
MELCOR Code Assessment Program (MCAP) – June/U.S.A
European MELCOR User Group (EMUG) Meeting – Spring/Europe
Asian MELCOR User Group (AMUG) Meeting – Fall/Asia

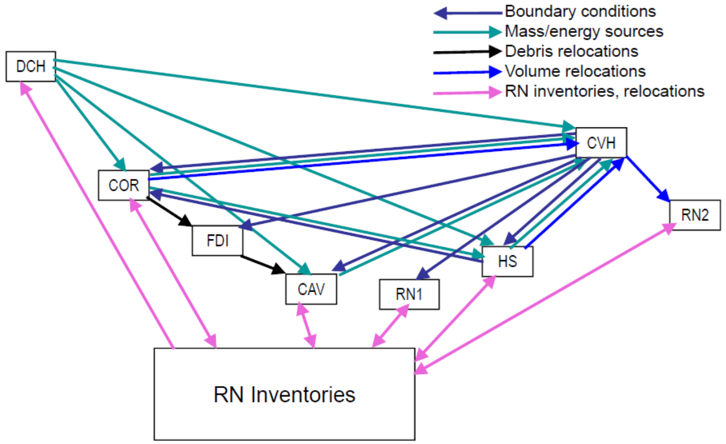
MELCOR Modernization

- Generalized numerical solution engine
- Hydrodynamics
- In-vessel damage progression
- Ex-vessel damage progression
- Fission product release and transport

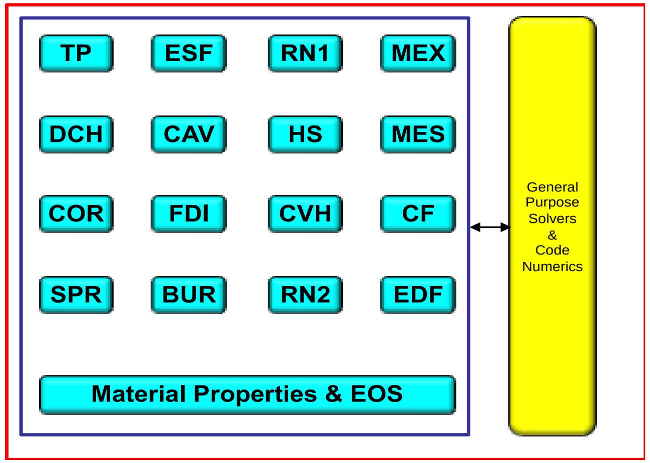


TP = Transfer Process
DCH = Decay Heat
COR = Core
SPR = Containment Spray
BUR = Gas Combustion
FDI = Fuel Dispersal Interaction
CAV = Cavity (MCCI)
ESF = Engineered Safety Features
MP = Material Properties

RN = Radionuclide
HS = Heat Structure
CVH = CV Hydrodynamics
EDF = External Data File
CF = Control Function
MES = Special Messages
MEX = Executive
CVT = CV Thermodynamics
NCG = Non Condensible Gas

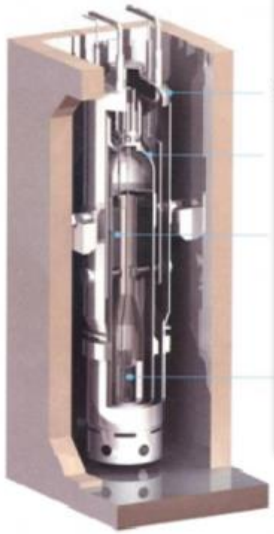


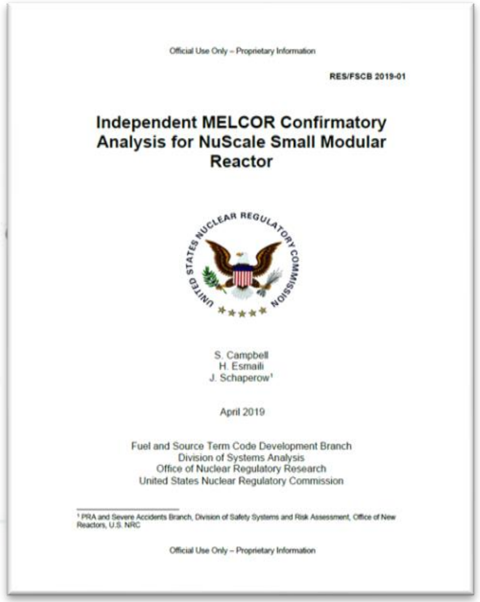
Separate **Physics** & **Numerics**



SMR Licensing Support

NuScale





Official Use Only – Proprietary Information

RES/FSCB 2019-01

Independent MELCOR Confirmatory Analysis for NuScale Small Modular Reactor

S. Campbell
H. Esmaili
J. Schaperow[1]

April 2019

Fuel and Source Term Code Development Branch
Division of Systems Analysis
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

[1] PRA and Severe Accidents Branch, Division of Safety Systems and Risk Assessment, Office of New Reactors, U.S. NRC

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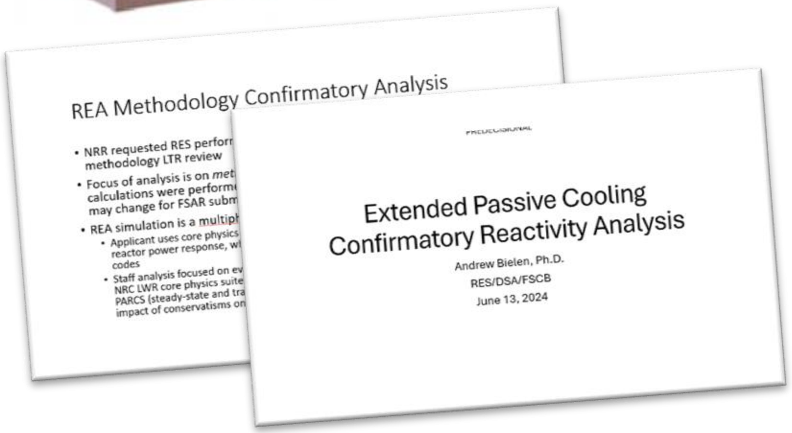
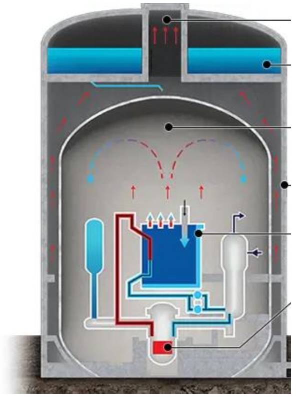
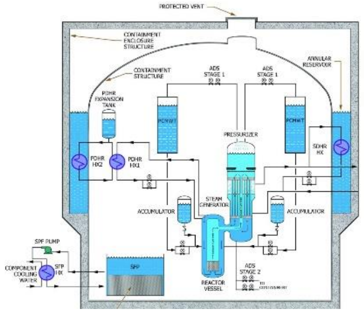


PARCS MELCOR scale

- Neutronics (Chapter 4, 15)
- Containment Peak Pressure (Chapter 6)
- Severe Accident (Chapter 19)



Westinghouse AP300



Holtec SMR-160

RES/FSCB 2019-01

Independent Confirmatory Analysis for BWRX300

S. Campbell
H. Esmaili
J. Schaperow[1]

April 2019

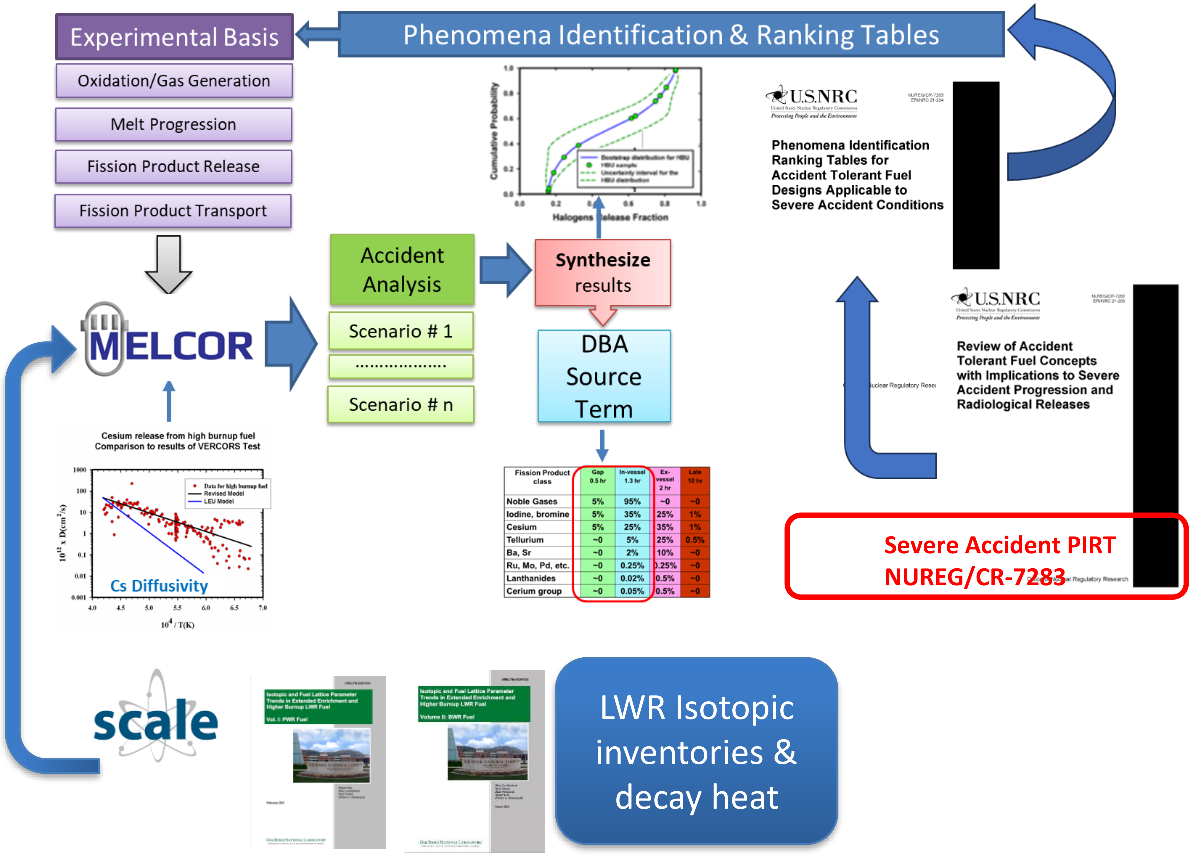
Fuel and Source Term Code Development Branch
Division of Systems Analysis
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

ACKNOWLEDGEMENTS

[1] PRA and Severe Accidents Branch, Division of Safety Systems and Risk Assessment, Office of New

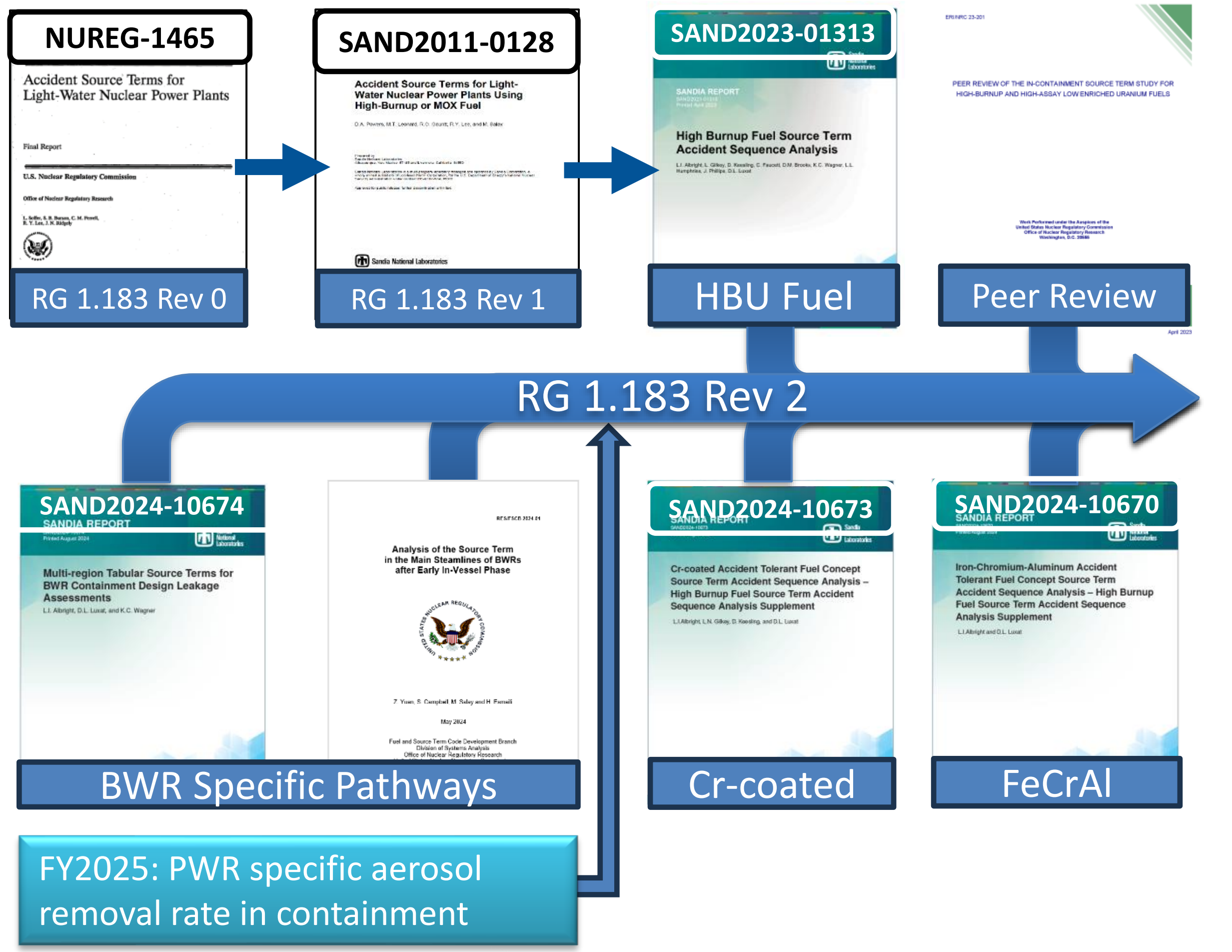
Regulatory Source Term for Advanced Fuels

Process of Developing Regulatory Source Term



Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0

Evolution of Technical Basis for Regulatory Guide (RG) 1.183



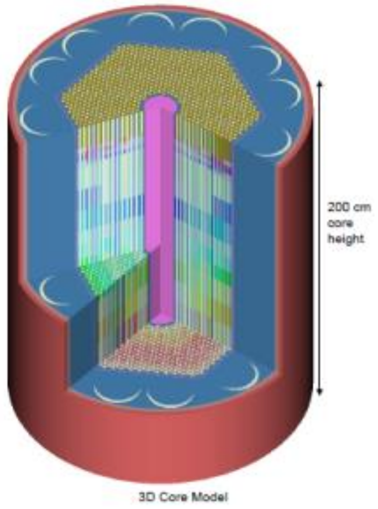
Pre-Application Activities for Advanced Reactors

High Temperature Gas Reactors	Light Water Reactors	Molten Salt Reactors / Molten Chloride Fast Reactors	Sodium Cooled Fast Reactors	Other Designs/ Not Specified
Energy Northwest	Deep Fission	Abilene Christian University	* ARC Clean Technology	Aalo Atomics
General Atomics	GE-Hitachi BWRX-300	Kairos Power, LLC	Oklo Inc.	Duke Energy - Belews Creek, NC
General Atomics Electromagnetic Systems	* Last Energy	Natura Resources	TerraPower & GE - Hitachi Natrium	Japan Atomic Energy Agency
* Radiant Industries, INC.	SMR, LLC (Holtec)	TerraPower, LLC		Texas A&M University - RELLIS Campus
* Terra Innovatum (SOLO)	TVA - Clinch River Nuclear Site	Terrestrial Energy USA, INC.		* Westinghouse eVinci
* University of Illinois at Urbana-Champaign - NANO Nuclear Energy Inc.	Westinghouse AP300			
X-Energy, LLC (XE-100)				
X-Energy, LLC (XENITH)				

** Indicates Microreactor Design*

Severe Accident Workshops

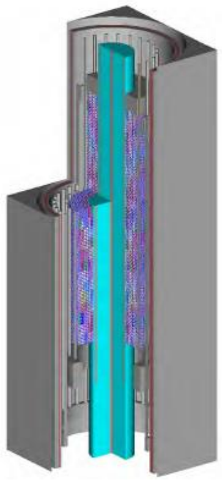
June 29, 2021



Heat Pipe Reactor
INL Design A

- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with metallic U fuel (19.75 wt.% U-235)
- Reactivity controlled via control drums

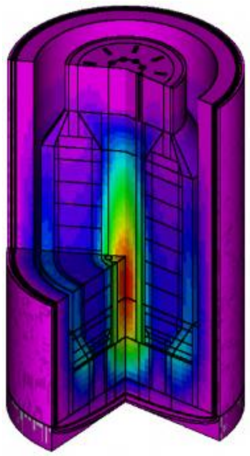
July 20, 2021



High-Temp. Gas Cooled Reactor
PBMR-400

- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTIHM)

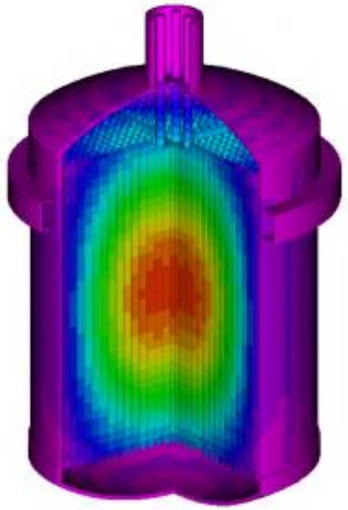
September 14, 2021



Molten Salt-Cooled Reactor
UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- FLiBe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

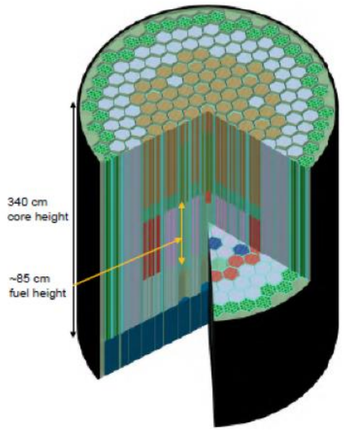
September 13, 2022



Molten Salt-Fueled Reactor
MSRE

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with liquid dissolved fuel in molten salt (34.5 wt. % U-235)

September 20, 2022

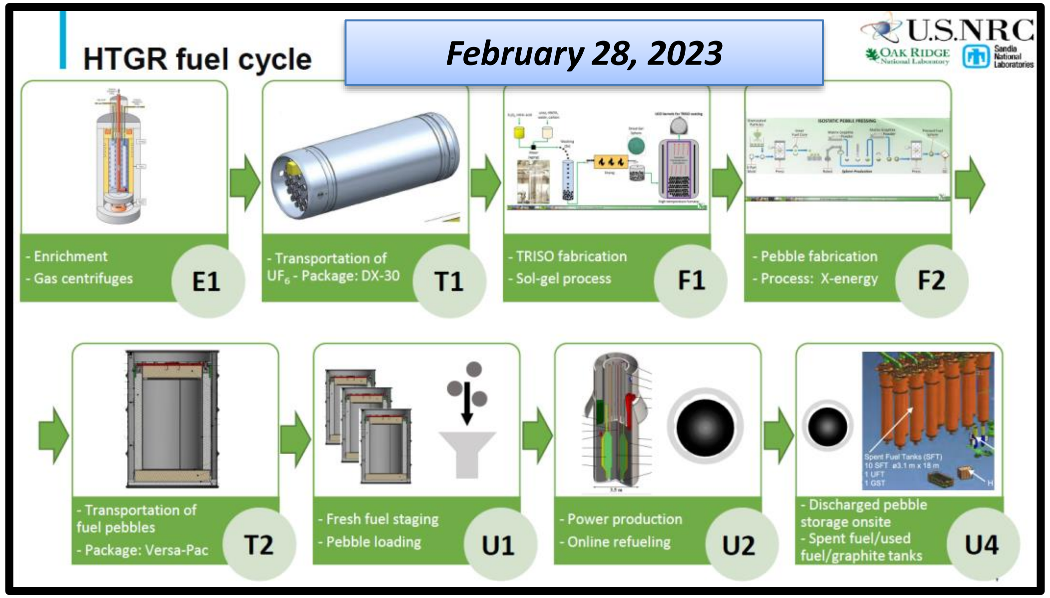


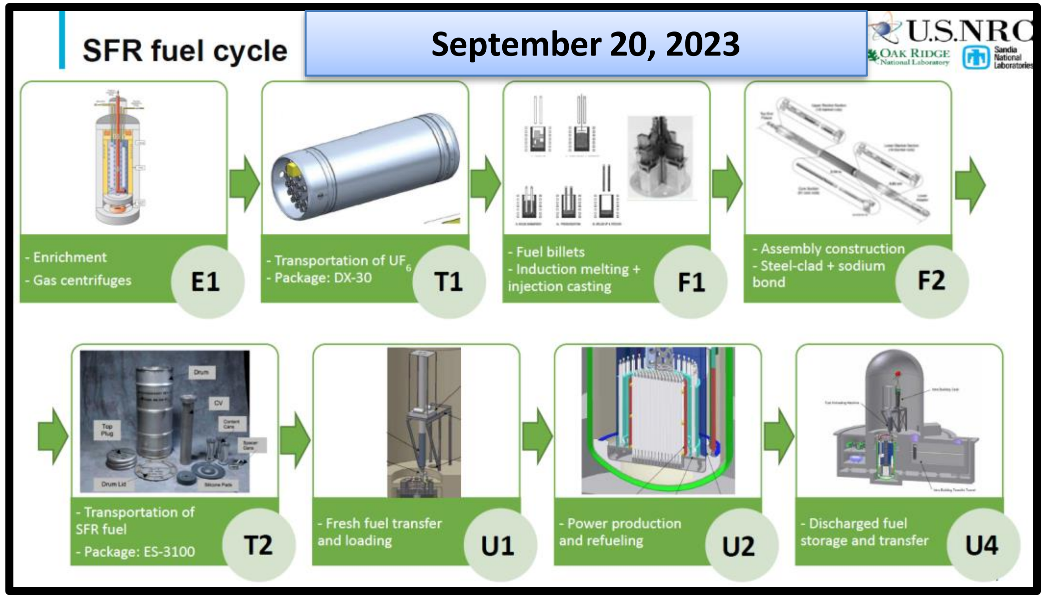
Sodium-Cooled Fast Reactor
ABTR

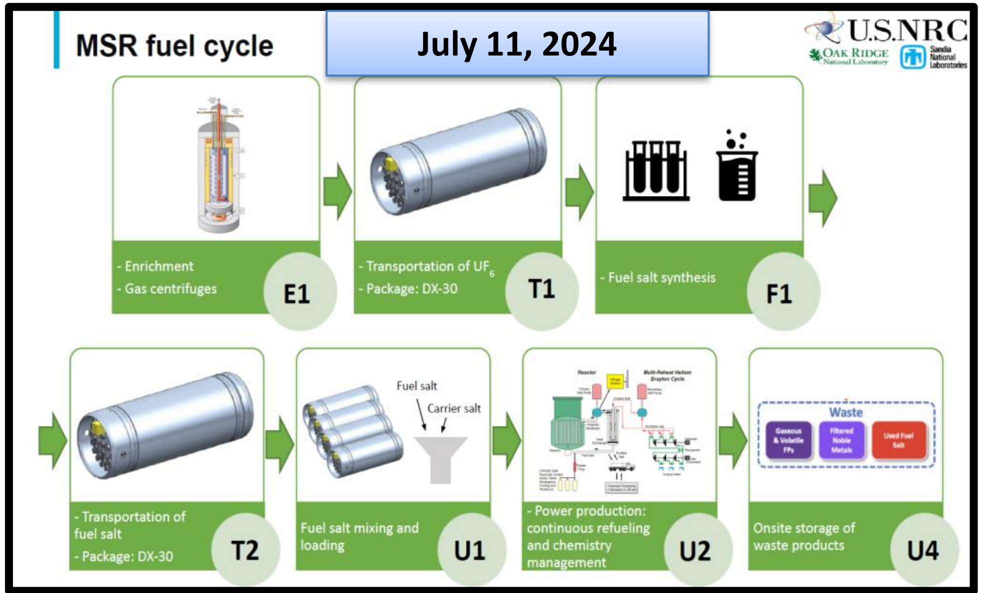
- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel
- Liquid sodium coolant

Public workshop videos, slides, reports at the NRC advanced reactor source term webpage

Fuel Cycle Workshops

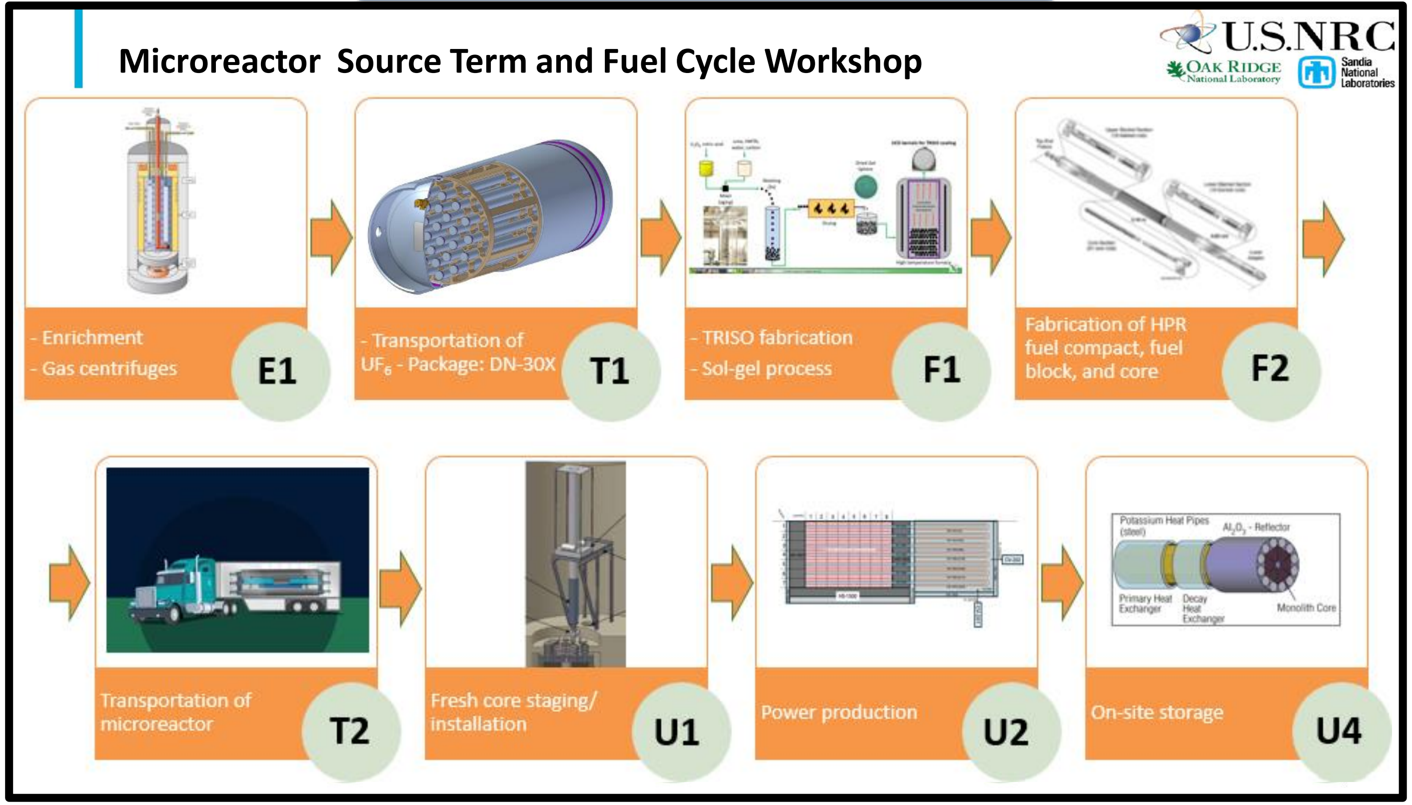


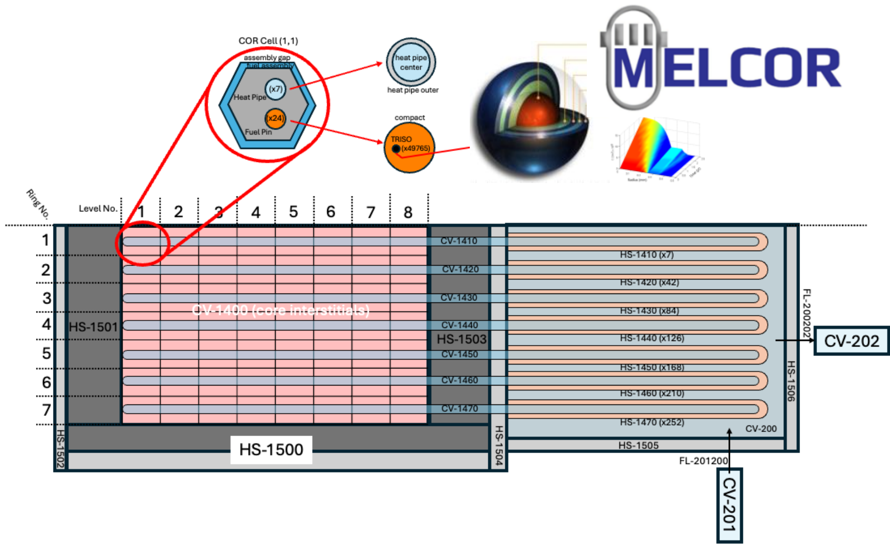


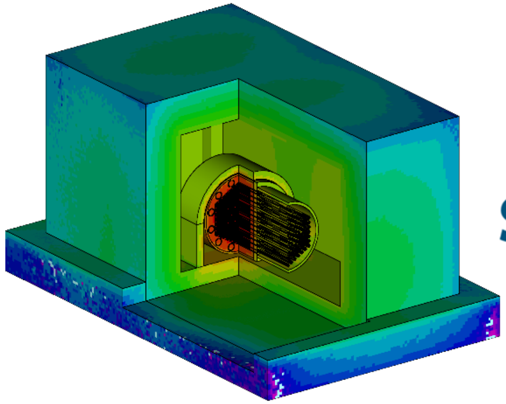


Recent Microreactor Workshop

March 26, 2025



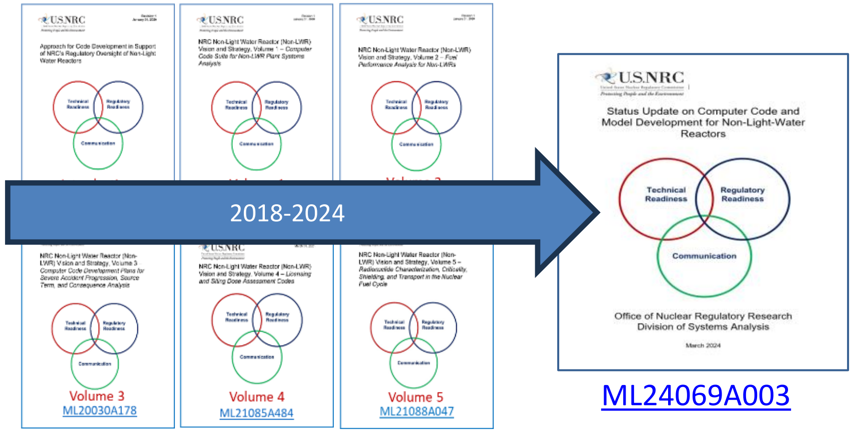




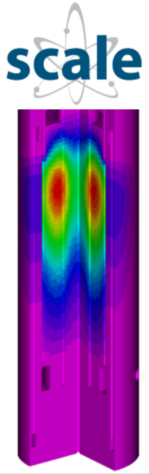
scale

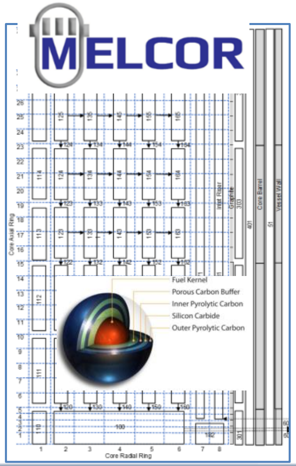
Non-LWR Model
Application to Licensing Reviews

Developed Efficient Strategy and Plan

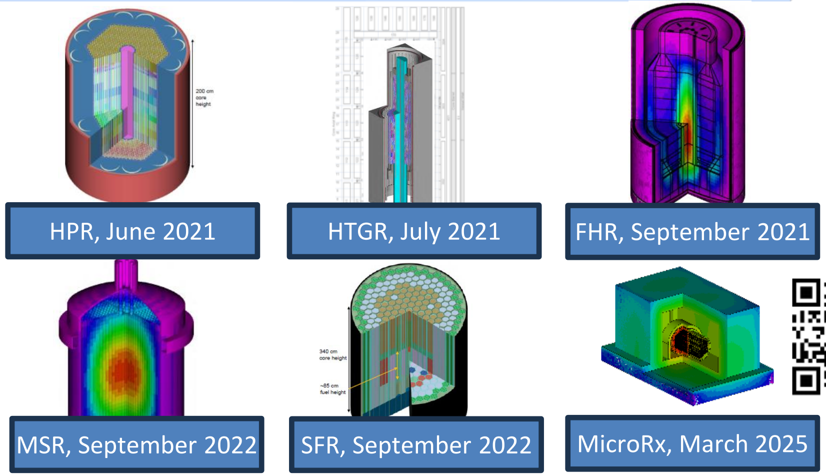


Develop Reference Plant Models

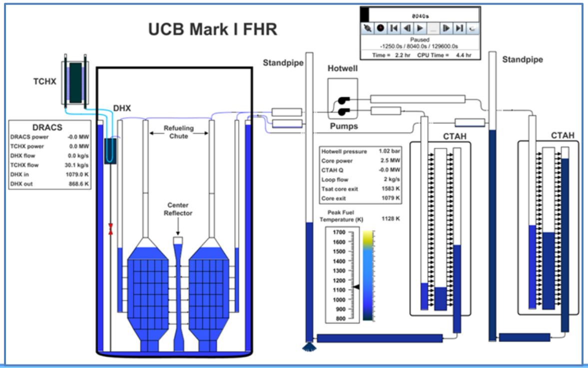




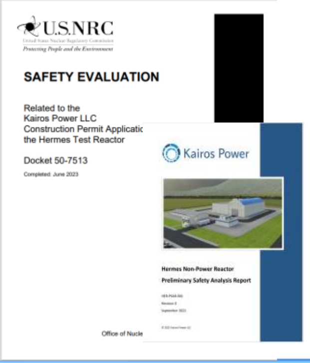
Expertise & Show Readiness

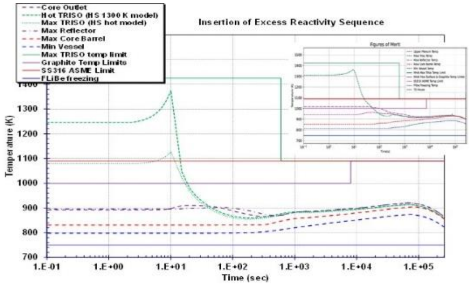


Modify Reference Plant Model

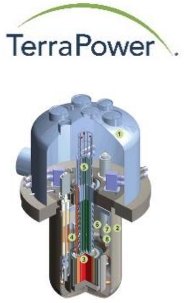


Support Hermes CP Review





Current & Future Support



ACRS April 2024: "NRC is in a good position to support technical reviews of advanced reactor design applications anticipated in the near future"

2025 CSARP/MCAP meetings

- June 2-6, 2025
- Bethesda Maryland, USA

2025 MELCOR Workshop

- June 9-13, 2025
- Bethesda Maryland, USA