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

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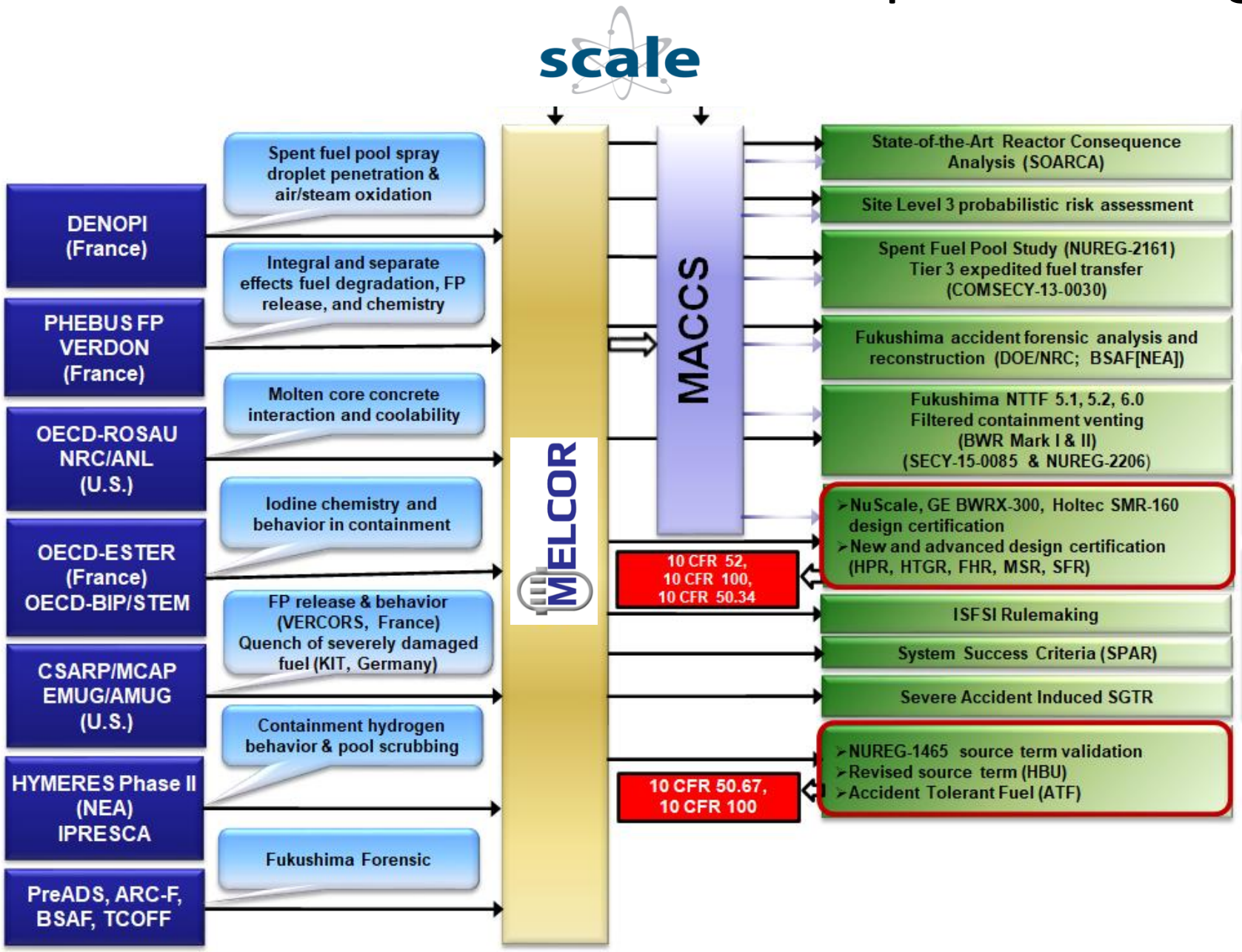
Office of Nuclear Regulatory Research

European MELCOR User Group Meeting
Rome, Italy
April 15-18, 2024

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

Severe Accident Code Development & Regulatory Applications

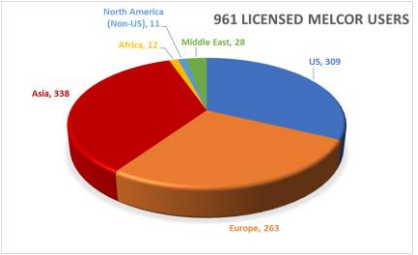


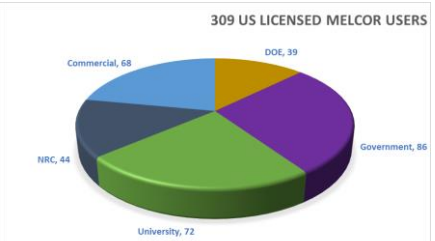
What Is It?
MELCOR is an engineering-level code that simulates the response of the reactor core, primary coolant system, containment, and surrounding buildings to a severe accident.

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).





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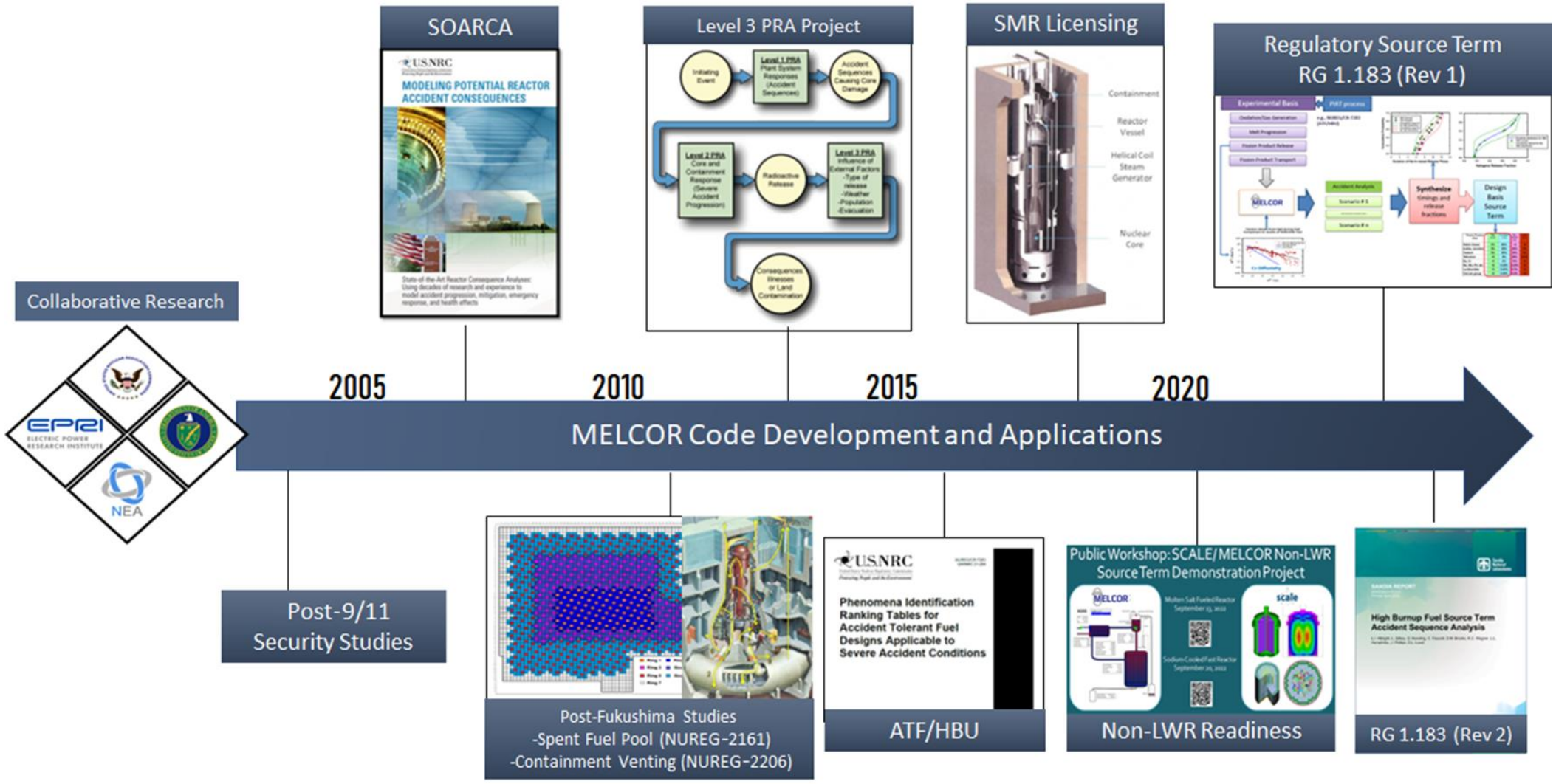
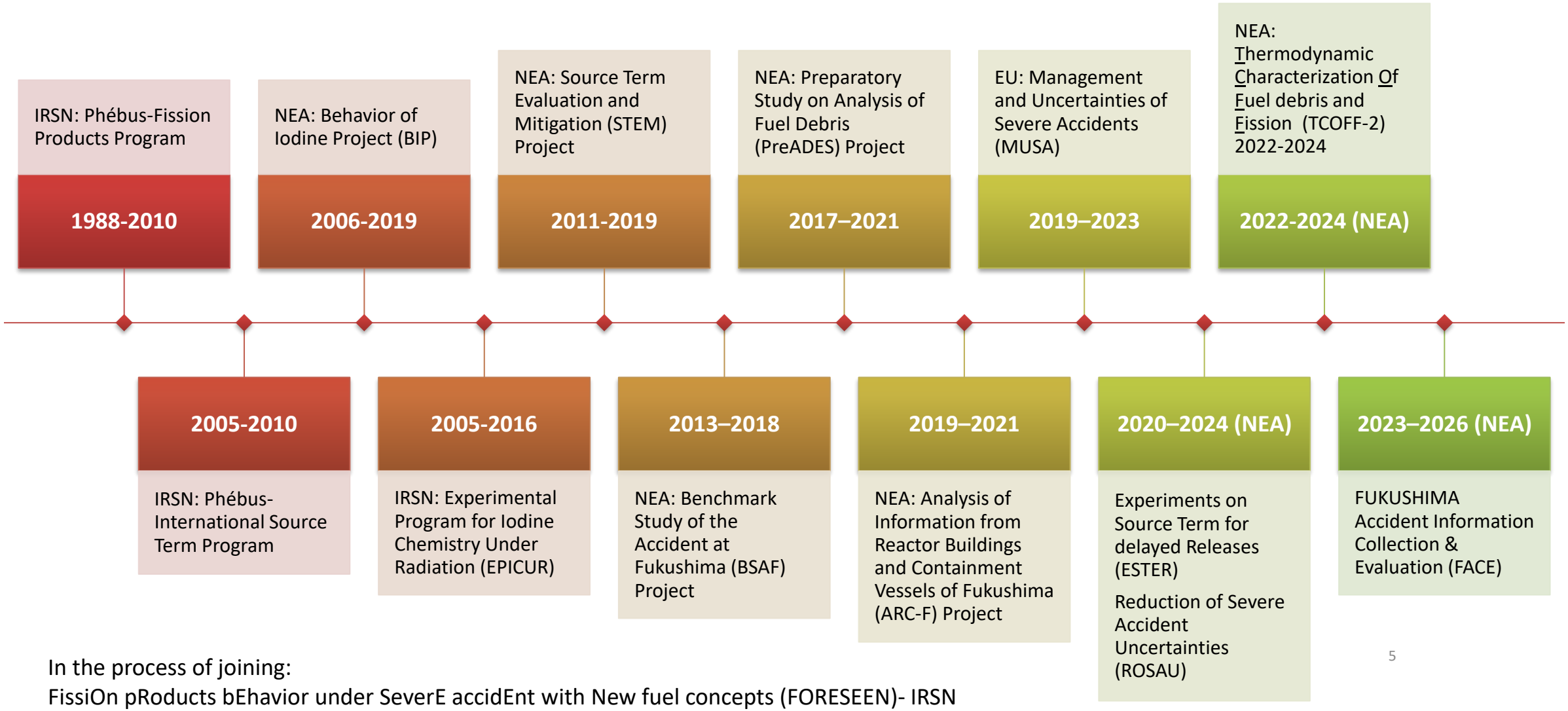


MELCOR User Groups & Technical Meetings

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

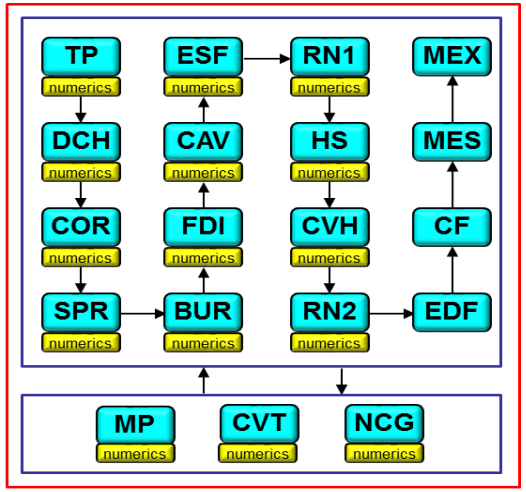
International Severe Accident Projects



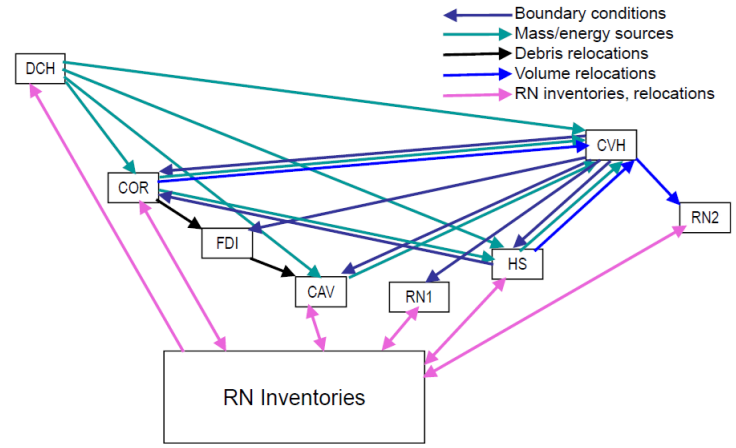
In the process of joining:
FissiOn pRoducts bEhavior under SeverE accidEnt with New fuel concepts (FORESEEN)- IRSN
POol during Loss of Cooling Accident (POLCA) - IRSN
COrium Properties for reactor Simulation and uncertainties (COPS) - CEA

MELCOR Modernization (2024)

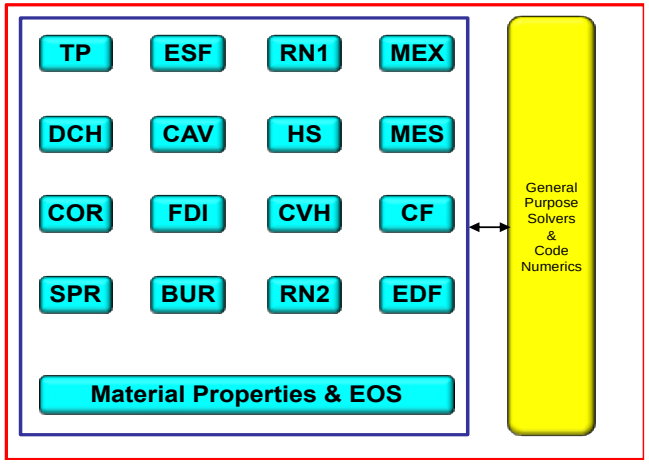
- Generalized numerical solution engine
- Hydrodynamics
- In-vessel damage progression
- Ex-vessel damage progression
- Fission product release and transport



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**



Source term for HBU/ATF

Experimental Basis ← PIRT

Oxidation/Gas Generation

Melt Progression

Fission Product Release

Fission Product Transport

MELCOR

Accident Analysis

Scenario # 1

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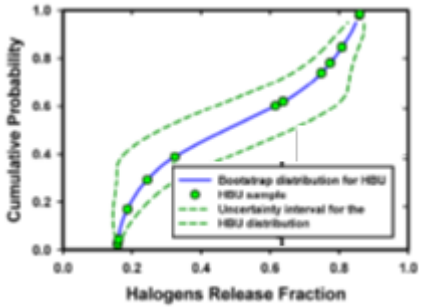
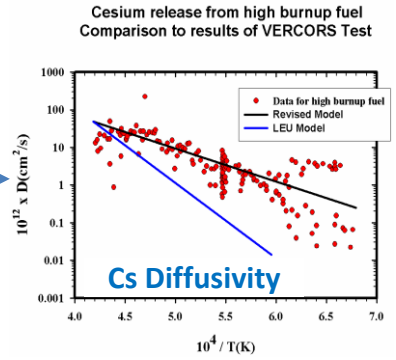
Scenario # n

Synthesize timings & RF

DBA Source Term

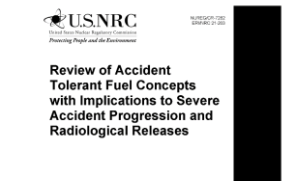




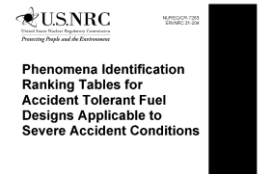
Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0

NUREG-1465 (RG 1.183)

Panel of international severe accident experts Phenomena Identification and Ranking Tables (PIRT) that addressed significant phenomenological issues to improve MELCOR

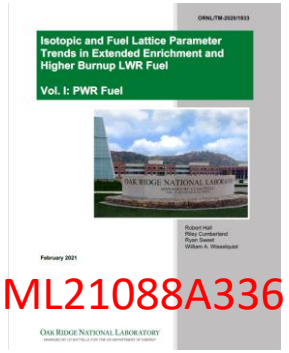


NUREG/CR-7282



NUREG/CR-7283

HBU/extended enrichment – SAND2023-01313



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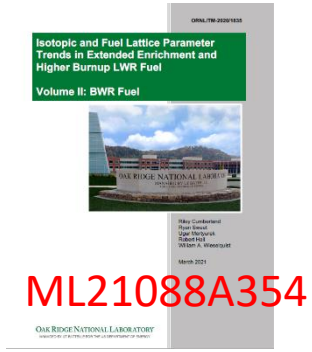

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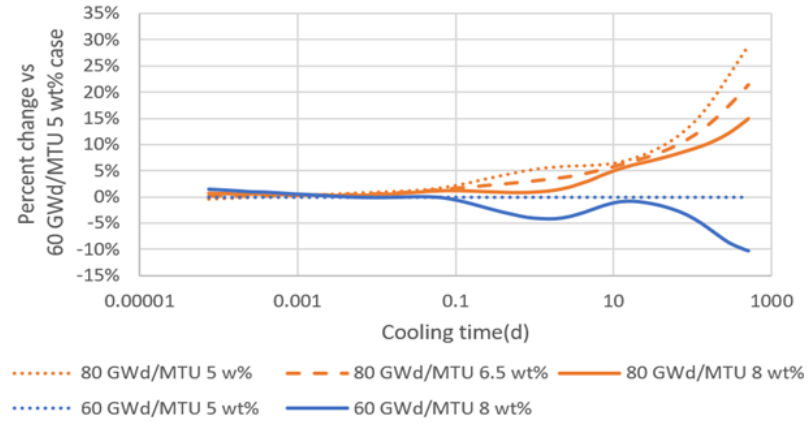


Advanced Nuclear Technology Research

Five Major Types of Non-LWRs Analyzed for Source Term:

2021

- Heat Pipe Reactor – INL Design A
- High-Temperature Gas-Cooled Pebble-bed Reactor – PBMR-400
- Molten-salt-cooled Pebble-bed Reactor – UCB Mark 1

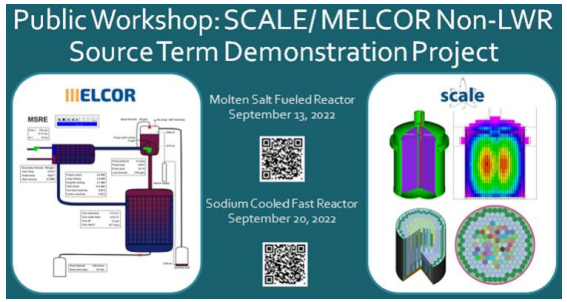
2022

- Molten-salt-fueled Reactor – MSRE
- Sodium-cooled Fast Reactor – ABTR

Also Evaluating the Non-LWR Fuel Cycle

2023

- High-Temperature Gas-Cooled Pebble-bed Reactor – PBMR-400
- Sodium-cooled Fast Reactor – ABTR



SCALE/MELCOR non-LWR source term demonstration project	

• Heat-pipe reactor workshop on June 29, 2021
  • Slides
  • Video Recording EXIT
• SCALE report
• MELCOR report
	June 29, 2021

• High-temperature gas-cooled reactor workshop on July 20, 2021
  • Slides
  • Video Recording EXIT
• SCALE report
• MELCOR report
	July 20, 2021

• Fluoride-salt-cooled high-temperature reactor workshop on September 14, 2021
  • Slides
  • Video Recording EXIT
• SCALE report
• MELCOR report
	September 14, 2021

• Molten-salt-fueled reactor workshop on September 13, 2022
  • Slides
  • Video Recording EXIT
• SCALE report
	September 13, 2022

• Sodium-cooled fast reactor workshop on September 20, 2022
  • Slides
  • Video Recording EXIT
	September 20, 2022

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

SCALE input models available here.

MELCOR input models available upon request.

2024 CSARP/MCAP meetings

- June 3-7, 2024
- Bethesda Maryland, USA
- Registration:
  - CSARP/MCAP Registration

MELCOR