RIC 2024 Hybrid

U.S. Nuclear Regulatory Commission
36th Annual Regulatory Information Conference

ADAPTING TO A CHANGING LANDSCAPE

MARCH 12-14, 2024

Bethesda North Marriott Hotel
and Conference Center
Rockville, Maryland

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Application of Computational Tools for Advanced Nuclear Technologies

Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

FAST, SCALE, and MELCOR have been used to support NRC research, licensing, and oversight activities for more than four decades. The NRC continues to update and improve our computer codes and analysis methodologies due to the recent interest in advanced nuclear technologies such as accident tolerant fuel (ATF) small modular reactors (SMRs) and advanced, non-light water reactors (non-LWRs). This exhibit describes new features and capabilities that have been added to FAST, SCALE, and MELCOR to accommodate these new technologies. Examples of how these updated computational tools are applied for these advanced nuclear technologies are also provided. These include SCALE/MELCOR modeling of the Hermes nonpower test reactor for the recently approved construction permit application, development of regulatory source term for HALEU/HBU/ATF fuels, and the NRC's non-LWR demonstration projects.

*This digital exhibit does not necessarily represent the views of the NRC.

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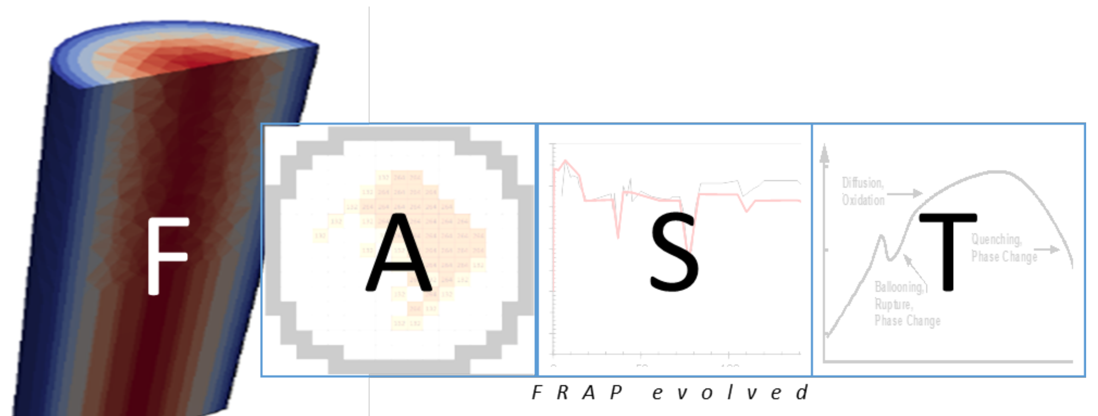
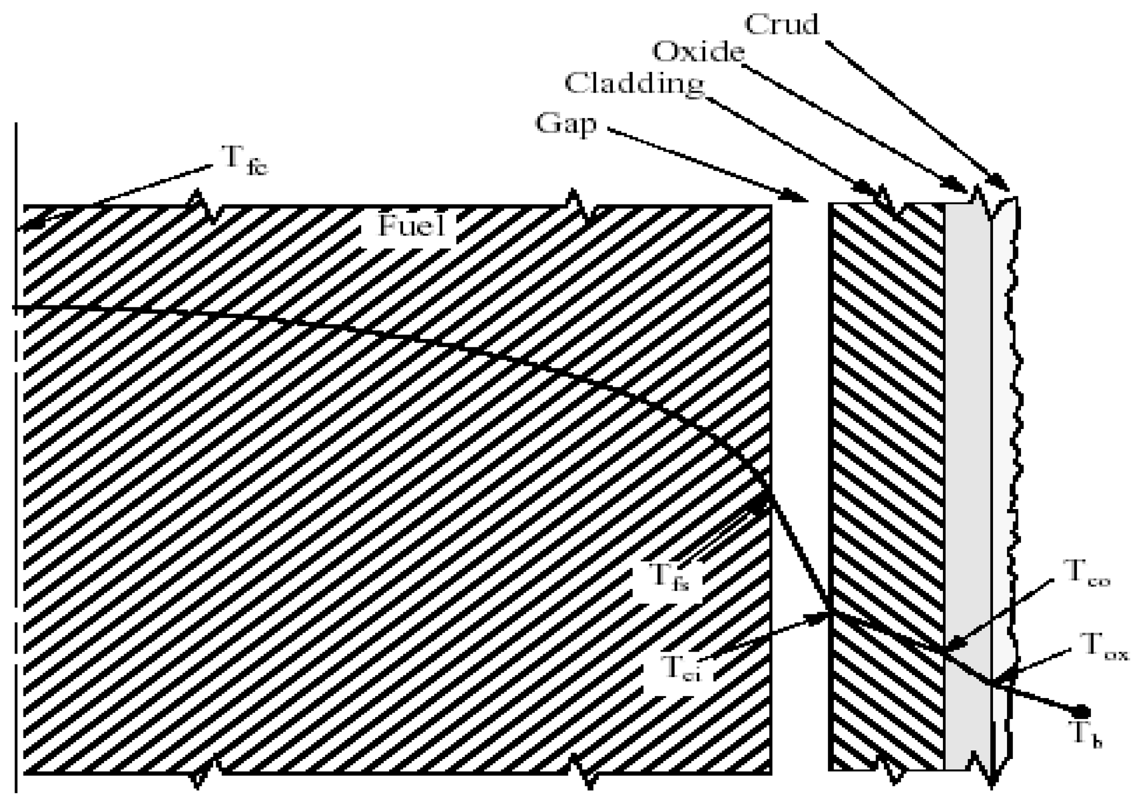
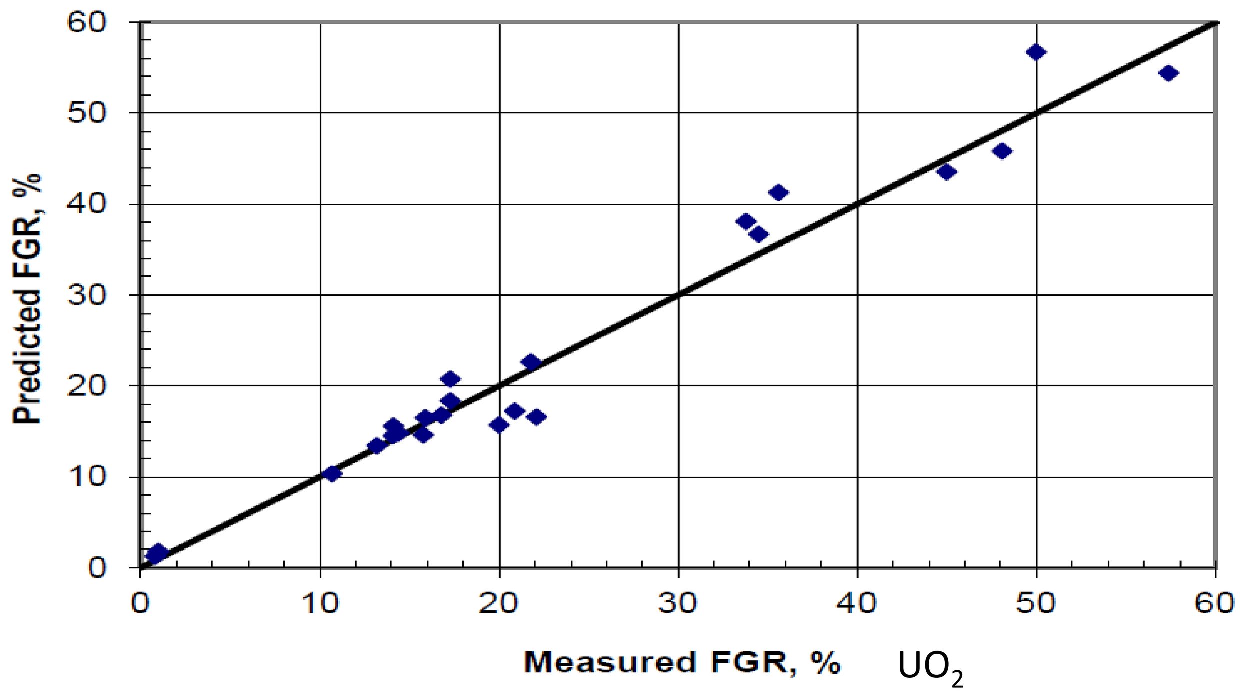


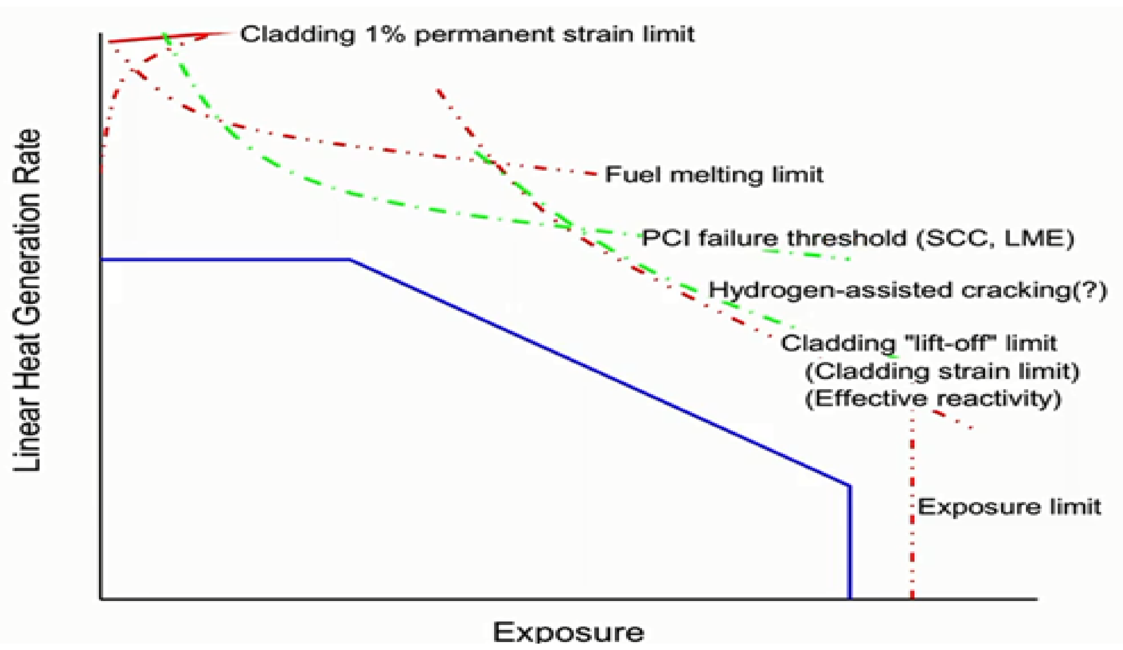
FAST Code Development and Applications

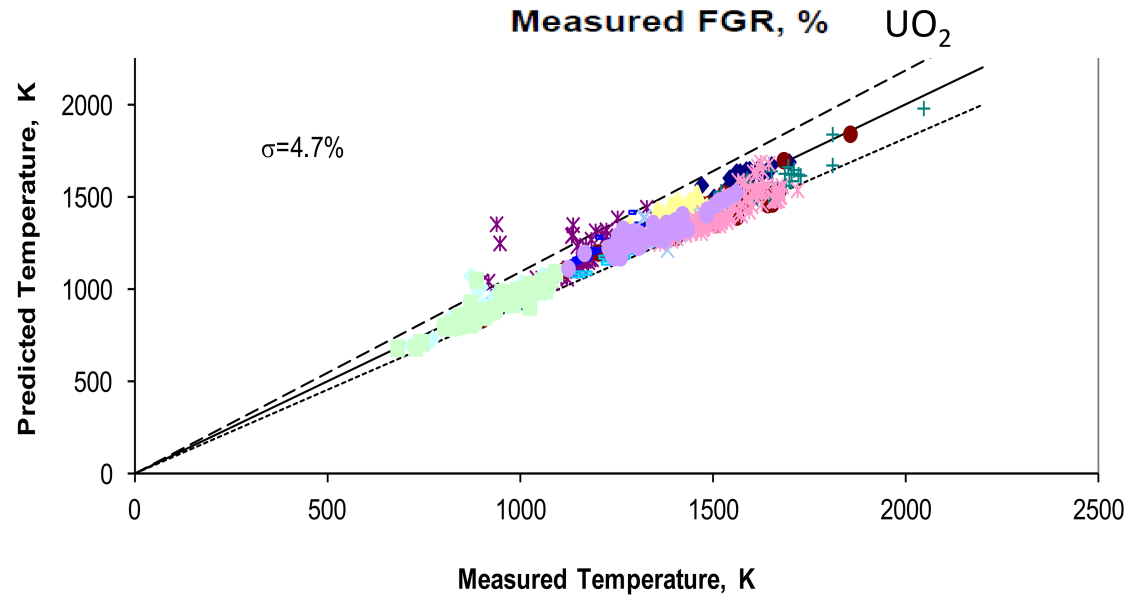
What Is It?
FAST (Fuel Analysis under Steady-State & Transients) calculates the thermal-mechanical response of nuclear fuel under steady-state and accident conditions.

How Is It Used?
FAST is used to support licensing reviews by assessing specified acceptable fuel design limits, evaluating vendor fuel codes and methods, and providing initial conditions for design-basis accident analysis. It is also used to perform spent fuel analyses.



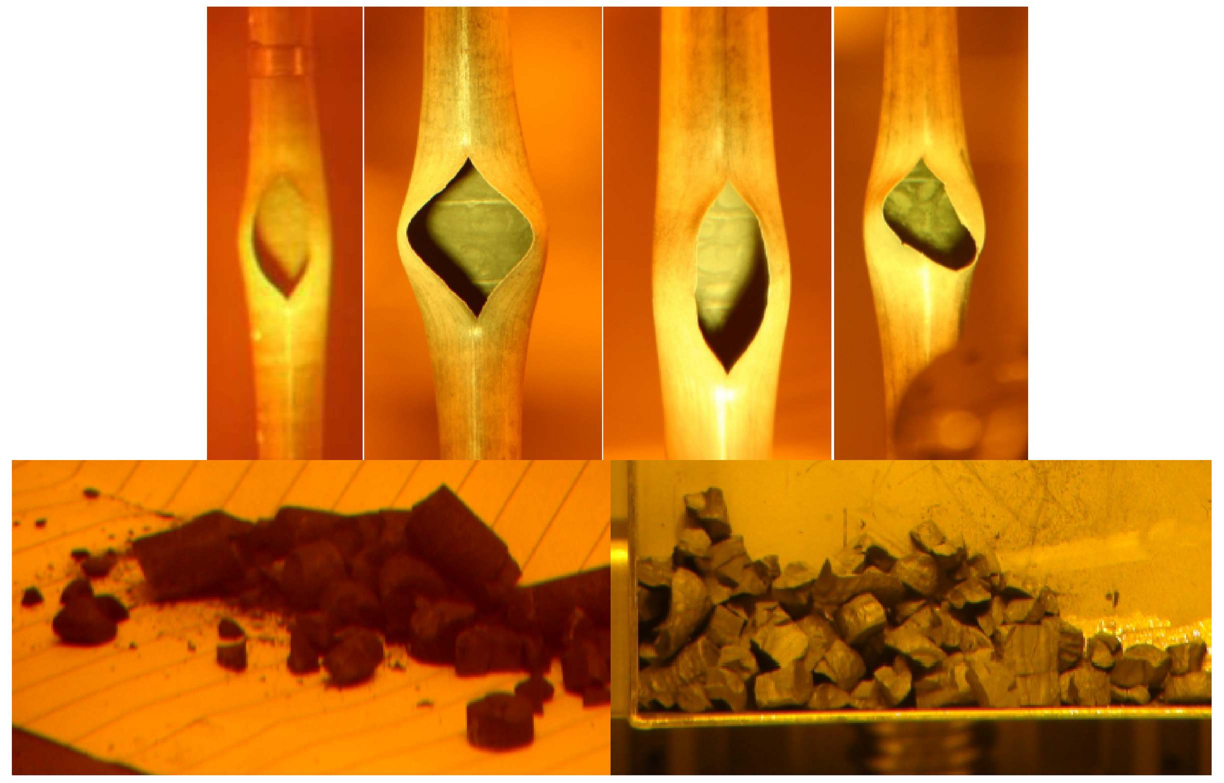






Halden Reactor Project	Studsvik Cladding Integrity Project (Phase I - V)	Second Framework for Irradiation Experiments (FIDES-II)	Cabri International Project	OECD/NEA's QUENCH-ATF Program	Lead Test Assemblies & Lead Test Rods Programs	DOE's Advanced Gas Reactor Program (AGR)	DOE's Sibling Rod Program
1980s-Present	**2004-Present**	**2021-2024**	**2000-Present**	**2021-Present**	**Ongoing**	**2002-Present**	**2016-Present**

Who Uses It?
FAST is used by more than 75 domestic and international organizations, including other regulatory bodies, technical scientific organizations, and utilities, for safety and core reload applications.



How Has It Been Assessed?
FAST is built on more than 30 years of assessment stemming from the FRAPCON/FRAPTRAN codes, as well as experience with fuel vendor codes and data. It offers more than 200 assessment cases that cover the UO_2/zirconium fuel system, and new cases added for metallic fuels.

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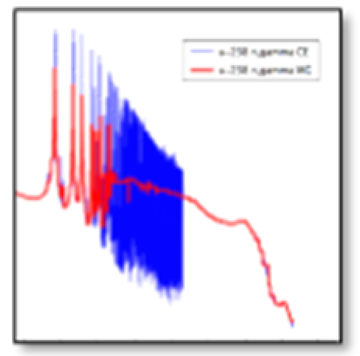
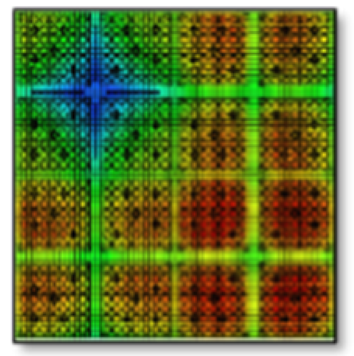
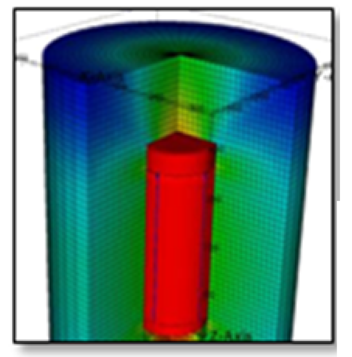
SCALE Code Development and Applications

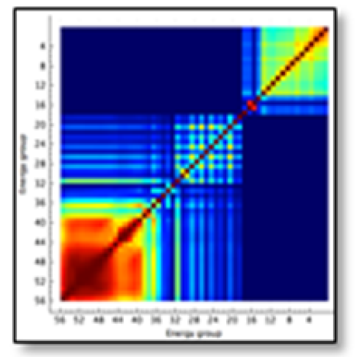
What Is It?
The SCALE code system is a modeling and simulation suite for nuclear safety analysis and design. It is a modernized code with a long history of application in the regulatory process.

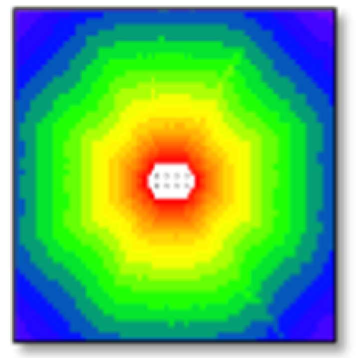

Nuclear data

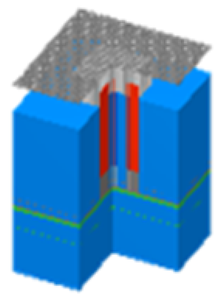

Reactor physics

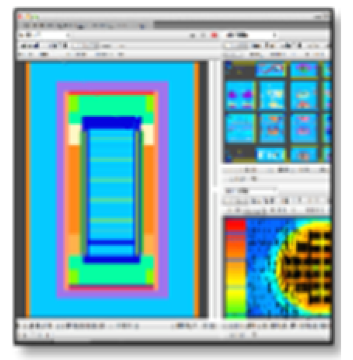

Criticality safety


Radiation shielding

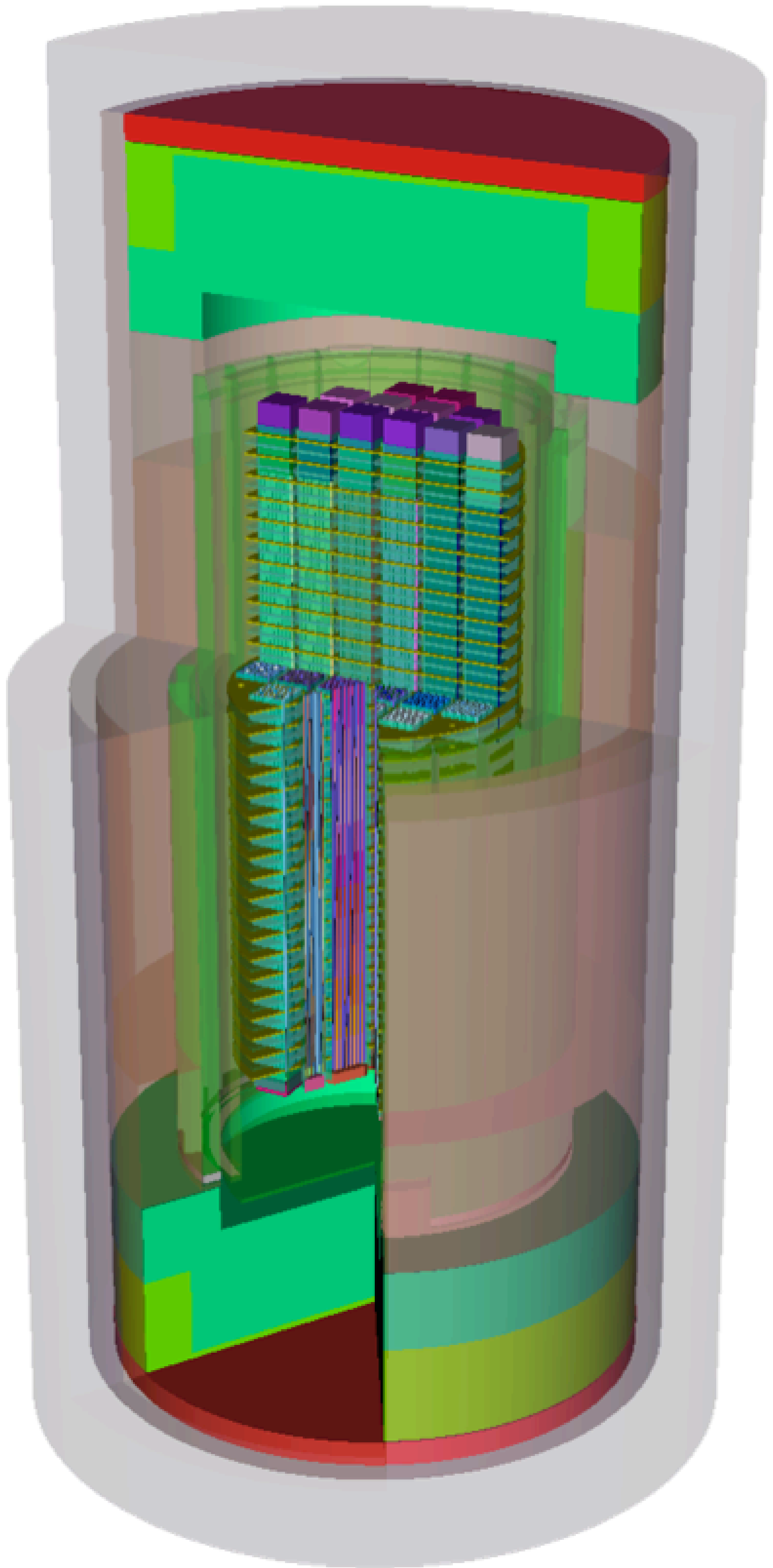

Sensitivity/ uncertainty

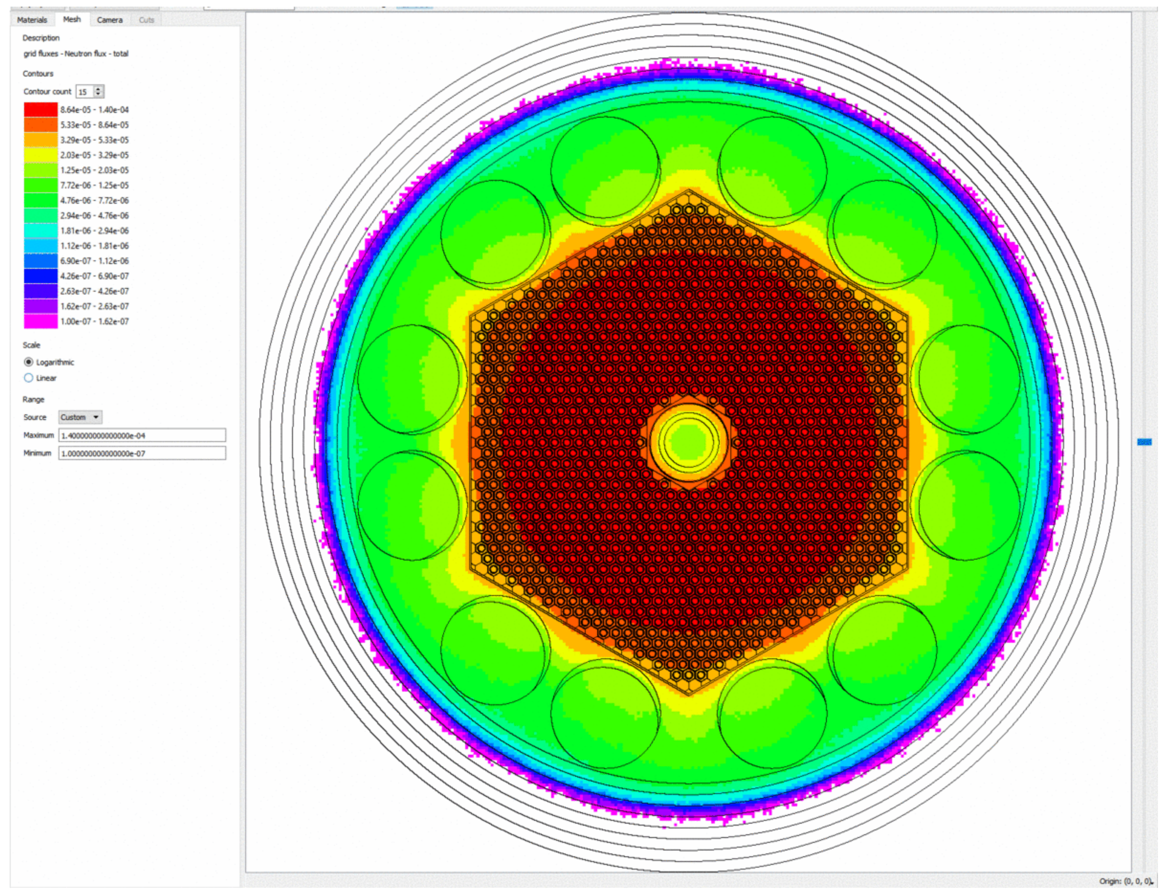

Hybrid methods

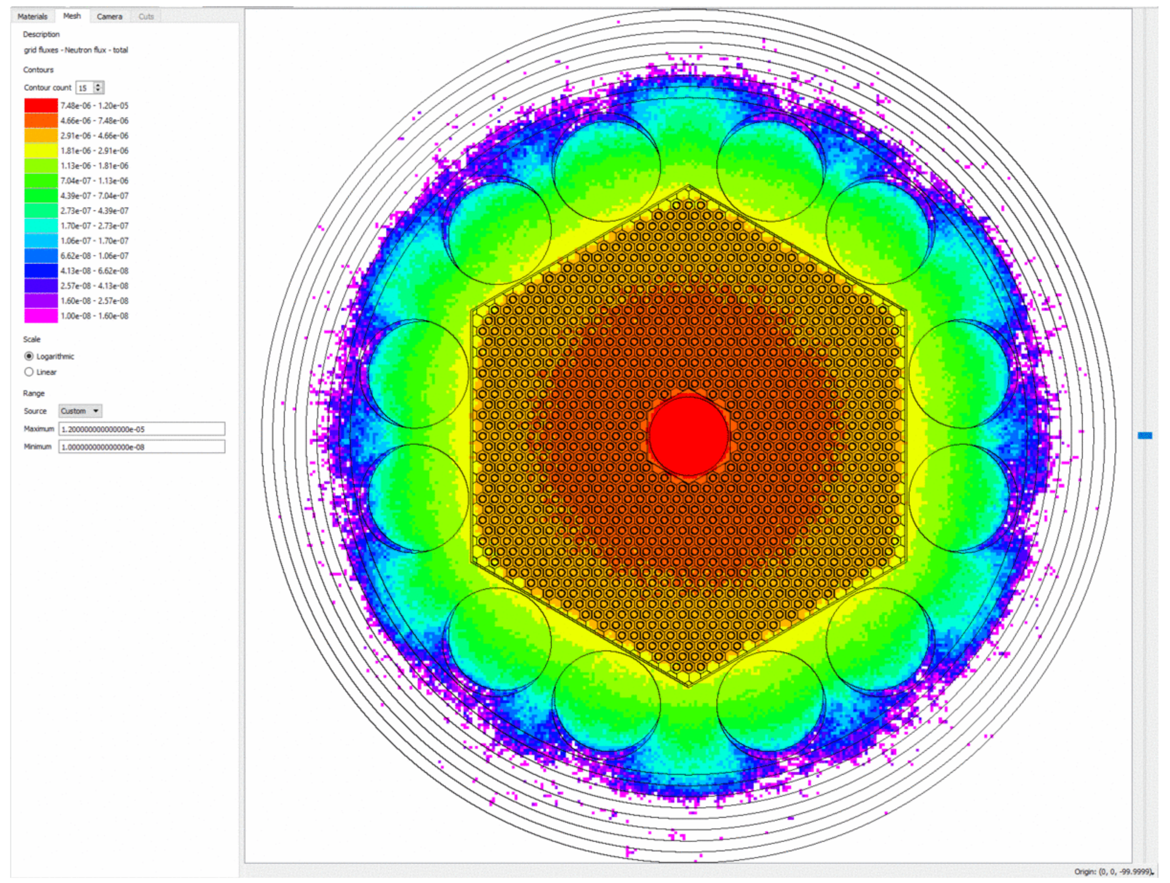

Verification/ validation


User interfaces

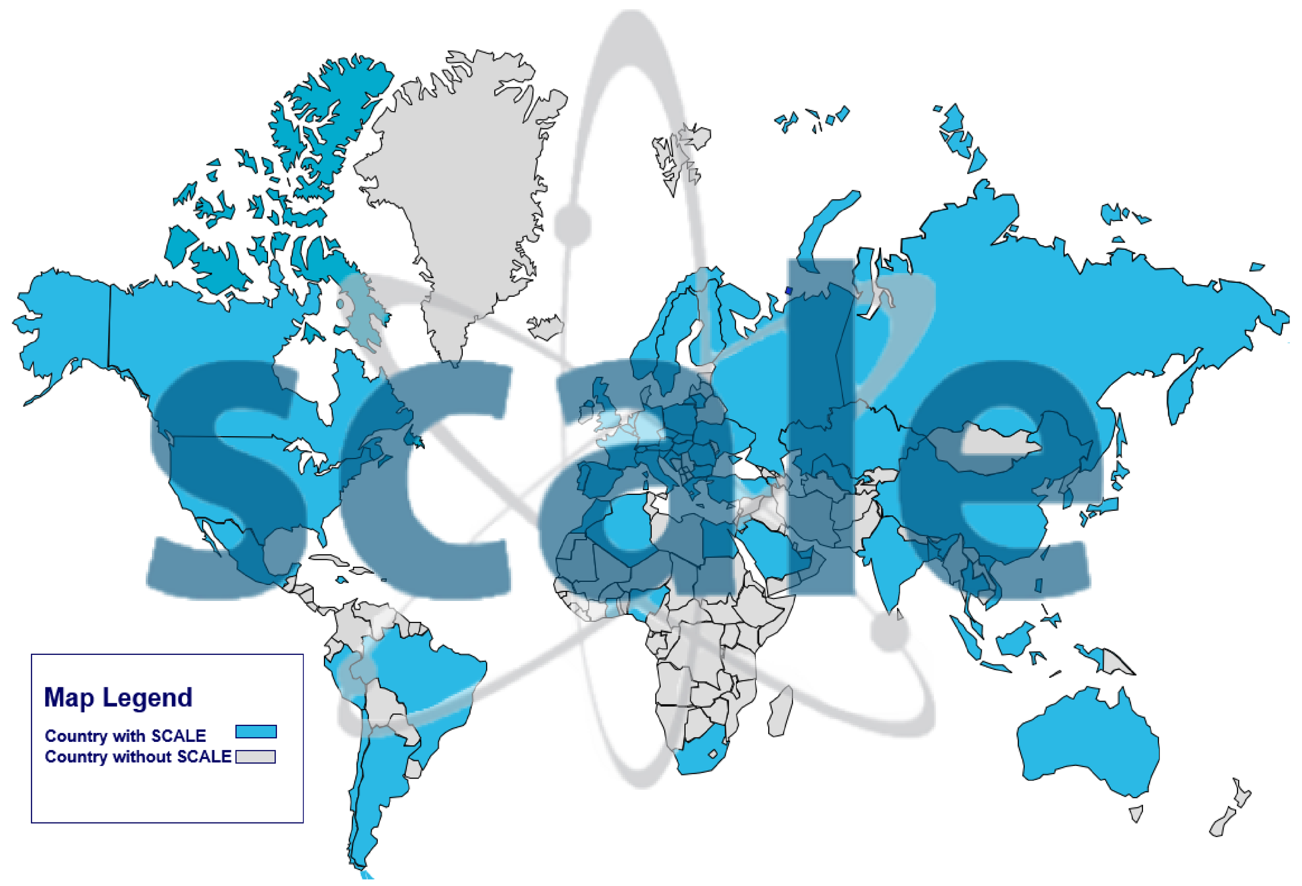
How Is It Used?
SCALE is used to support licensing activities (e.g., analysis of spent fuel pool criticality, generating nuclear physics and decay heat parameters for design-basis accident analysis, and review of consolidated interim storage facilities, burnup credit).







Who Uses It?
SCALE is used by the NRC and in 61 countries (about 11,000 users and 33 regulatory bodies).



How Has It Been Assessed?
SCALE has been validated against numerous critical experiments that cover a range of fuel and moderator materials and geometries, and against measured PWR and BWR spent fuel isotopic composition and decay heat measurements.

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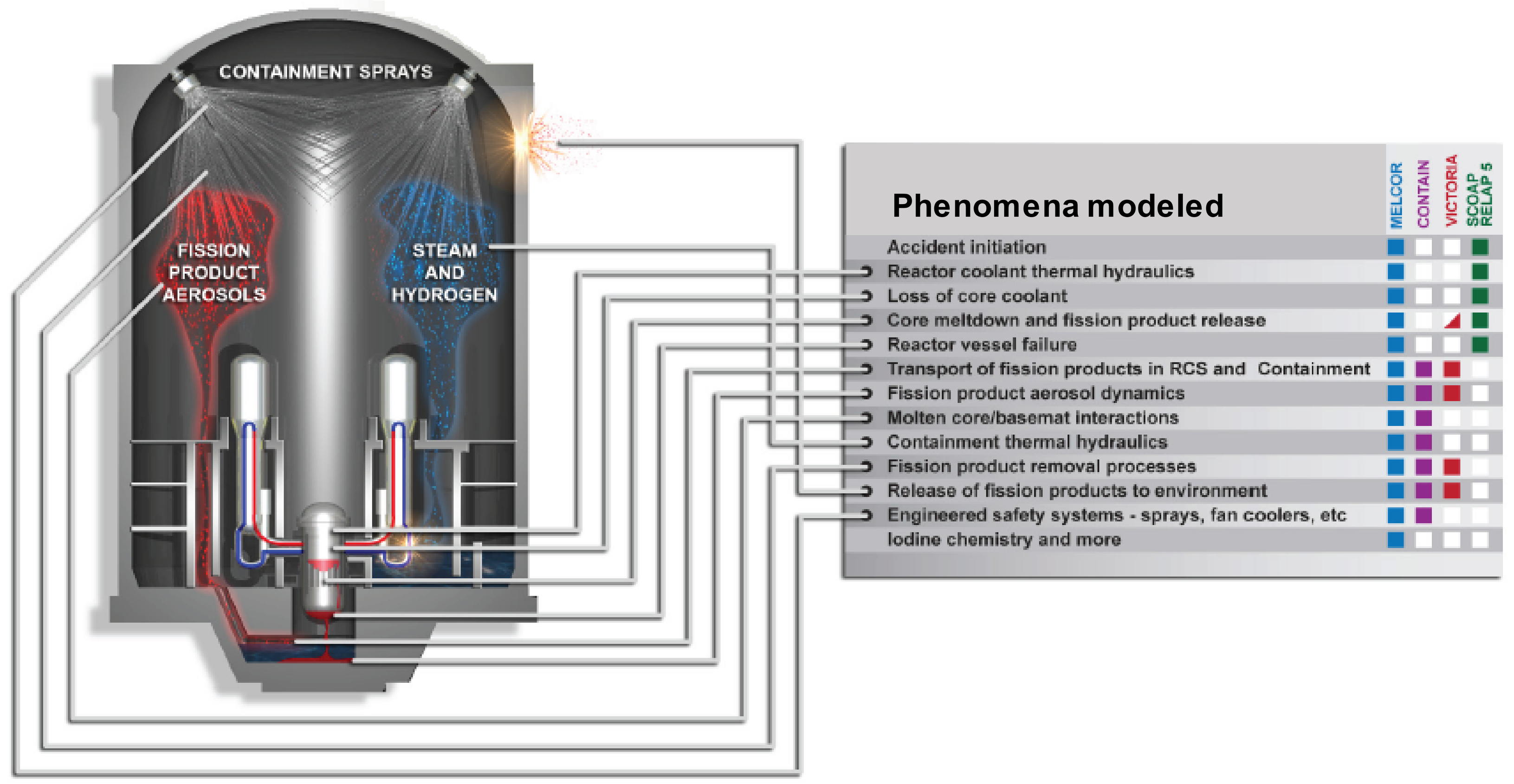
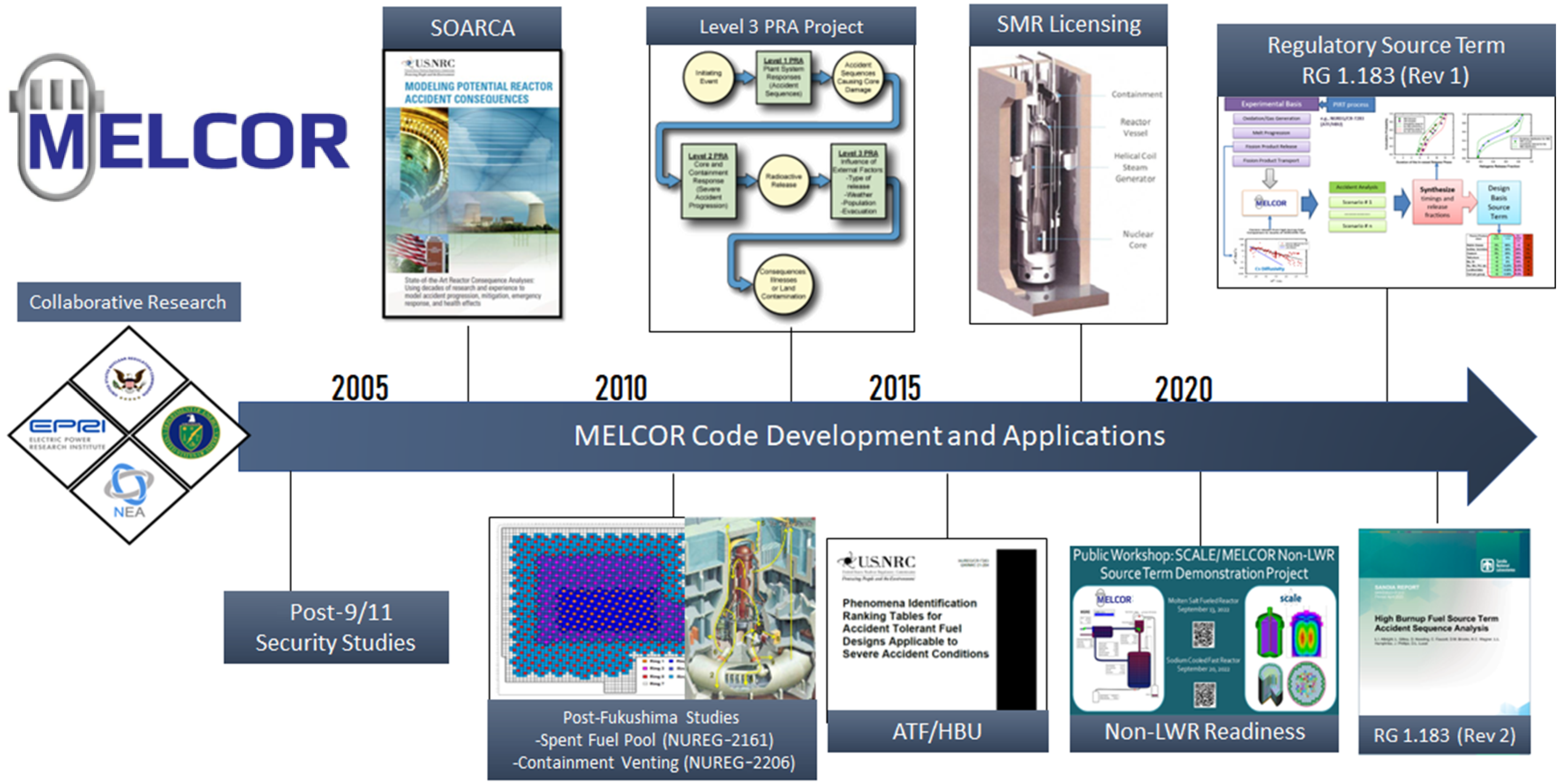
MELCOR Code Development and Applications

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.



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



Phébus-Fission Products & Source Term Program	Behavior of Iodine Project (BIP)	Experimental Program for Iodine Chemistry Under Radiation (EPICUR)	Source Term Evaluation and Mitigation (STEM) Project	Benchmark Study of the Accident at Fukushima (BSAF) Project	Management and Uncertainties of Severe Accidents (MUSA)	Experiments on Source Term for delayed Releases (ESTER) Reduction of Severe Accident Uncertainties (ROSAU)	Thermodynamic Characterization Of Fuel debris and Fission (TCOFF-2)	Fukushima Accident Information Collection & Evaluation (FACE)
1988-2010	**2006-2019**	**2005-2016**	**2011-2019**	**2013-2018**	**2019-2023**	**2020-2024**	**2022-2024**	**2023-2026**

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

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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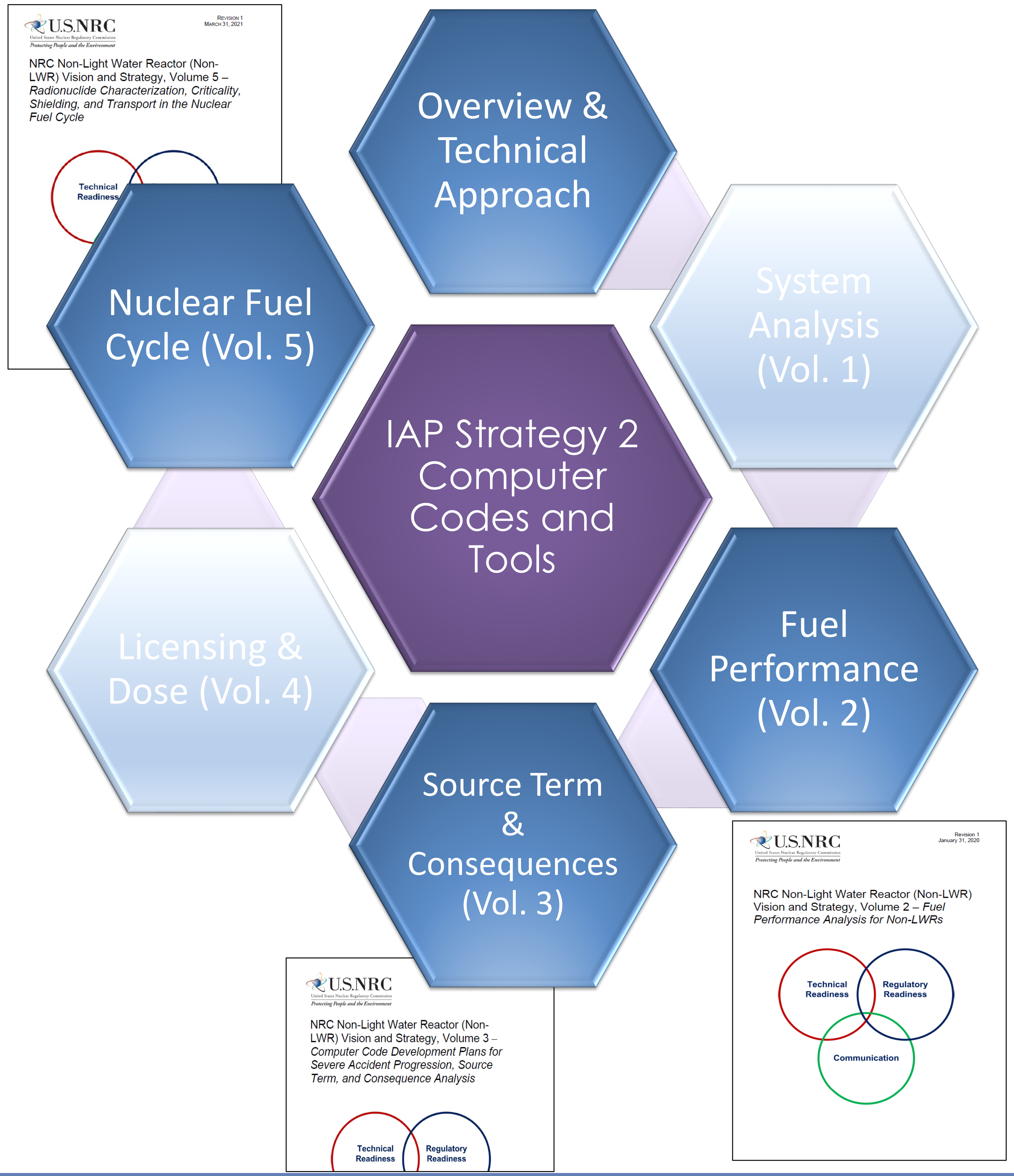
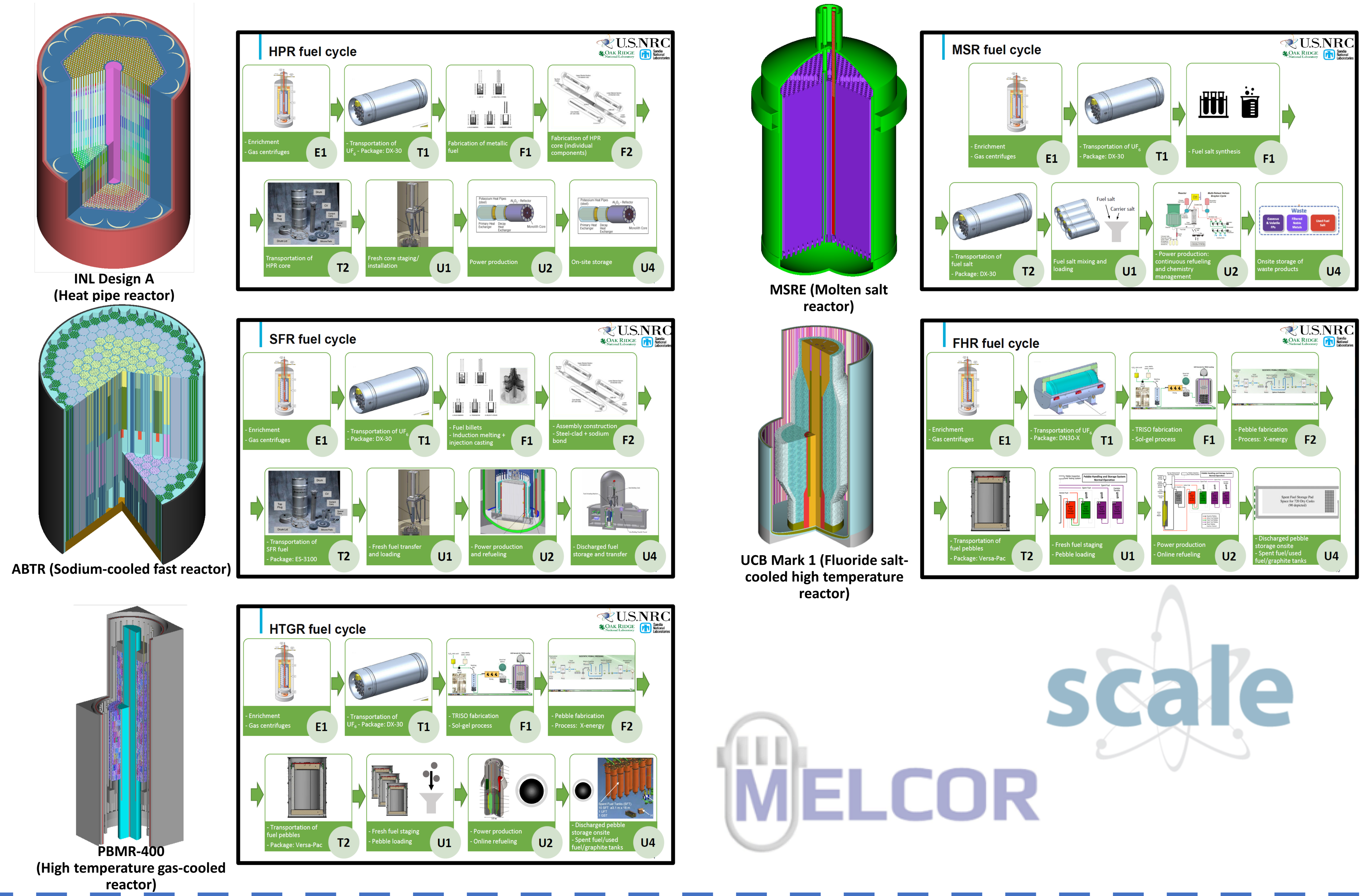
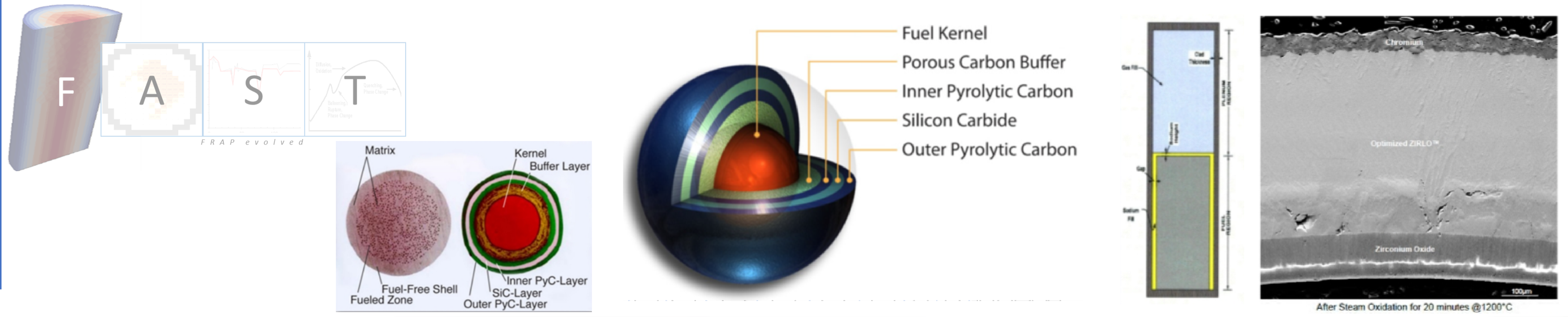
Non-LWR Demonstration Projects & Licensing

Planning

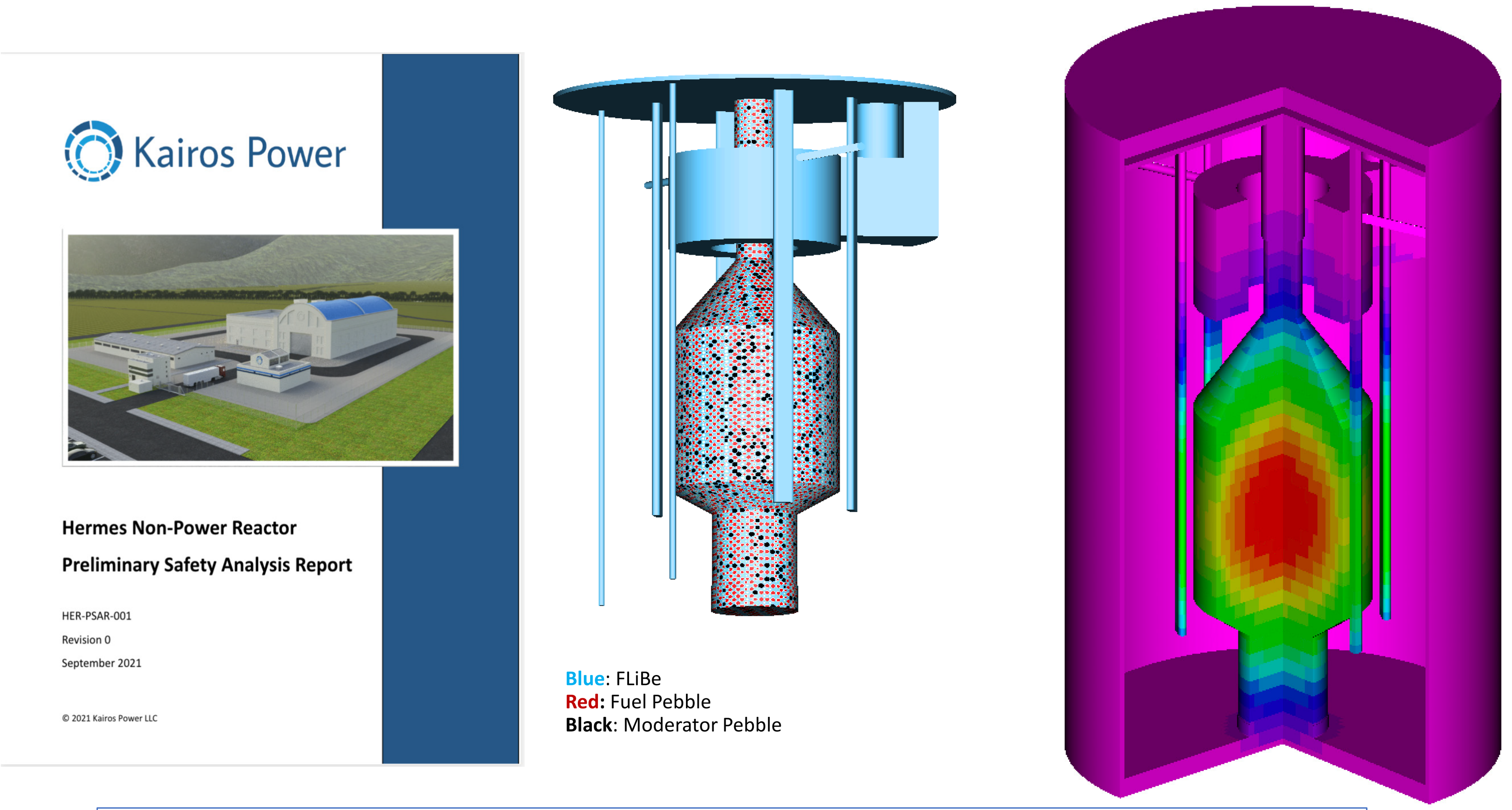


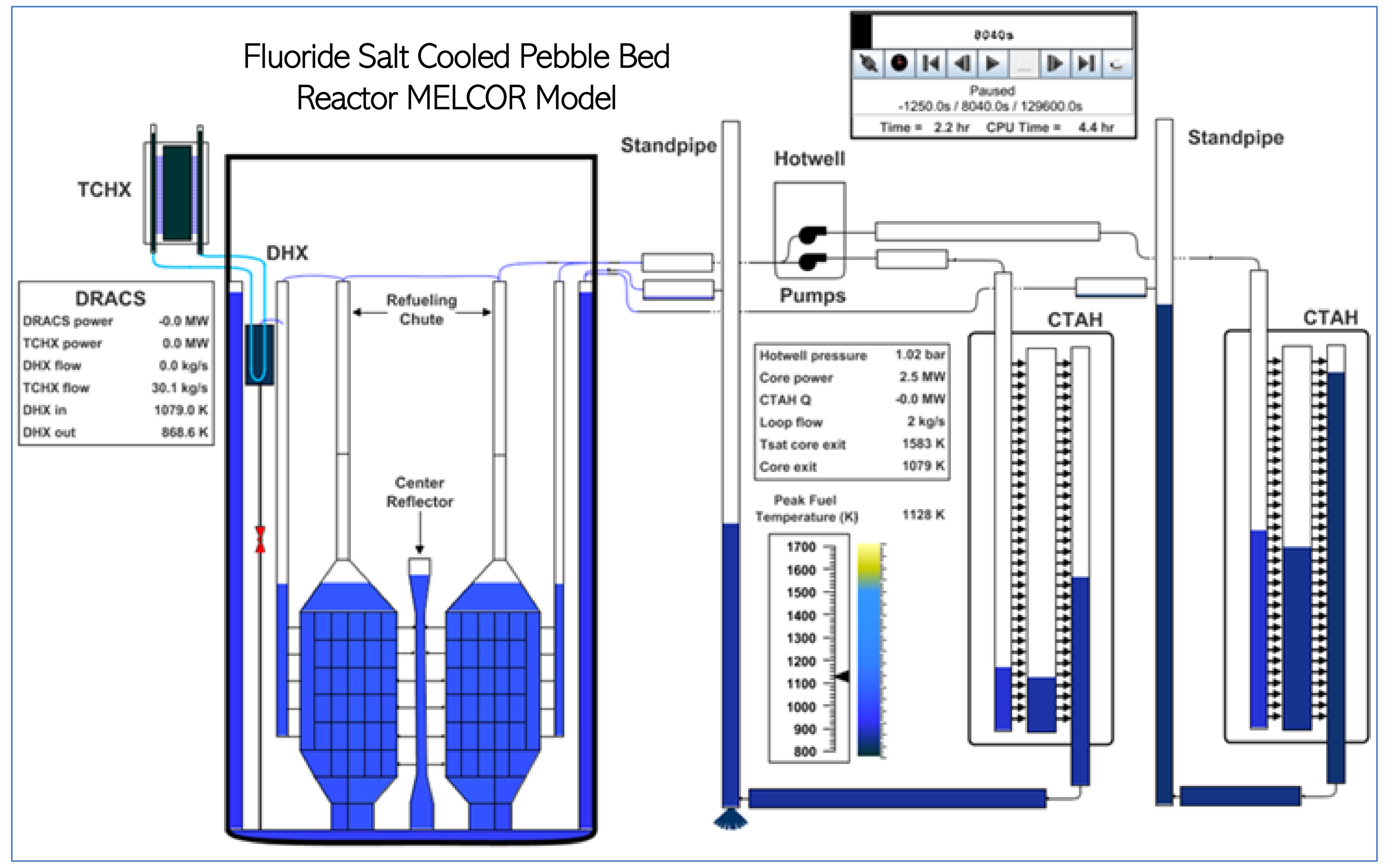
Implementation





Applications





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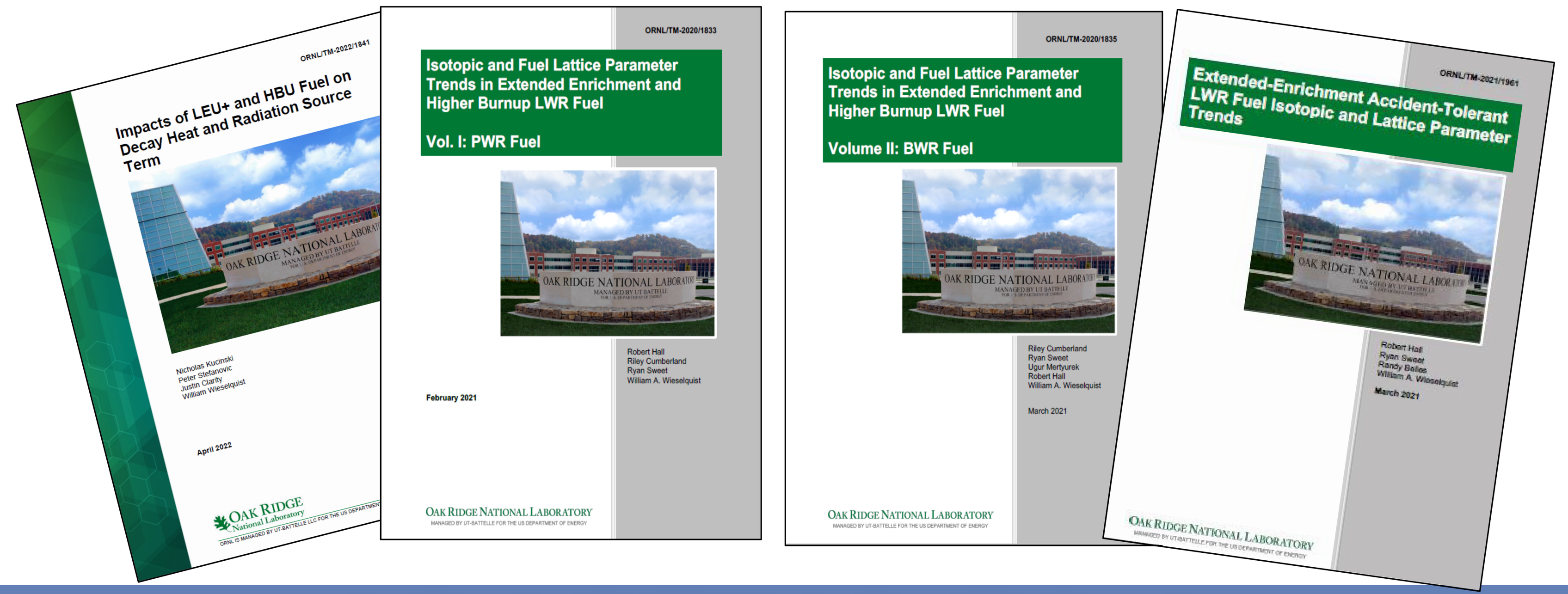
Regulatory Applications of Tools for ATF/HBU/EE

Neutronics Fuel Cycle Research

Front-End (Criticality Safety)

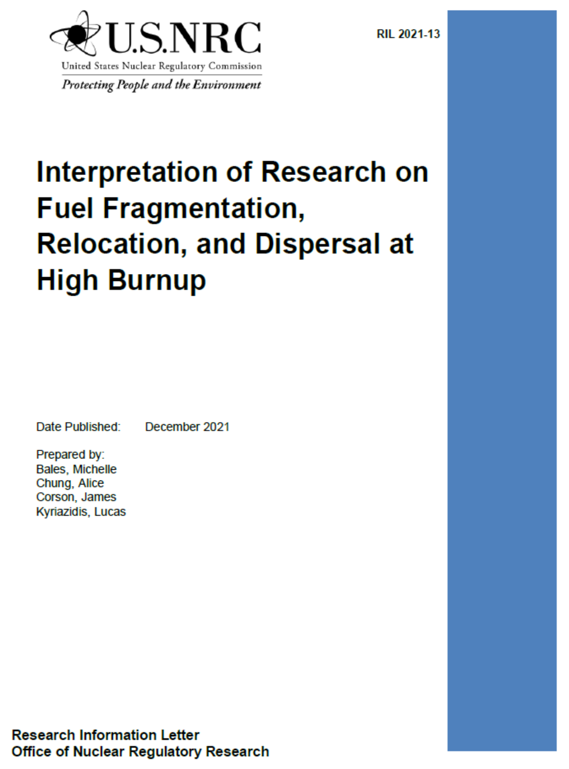
In-Reactor & Power Production

Shielding, Dose, and Decay Heat

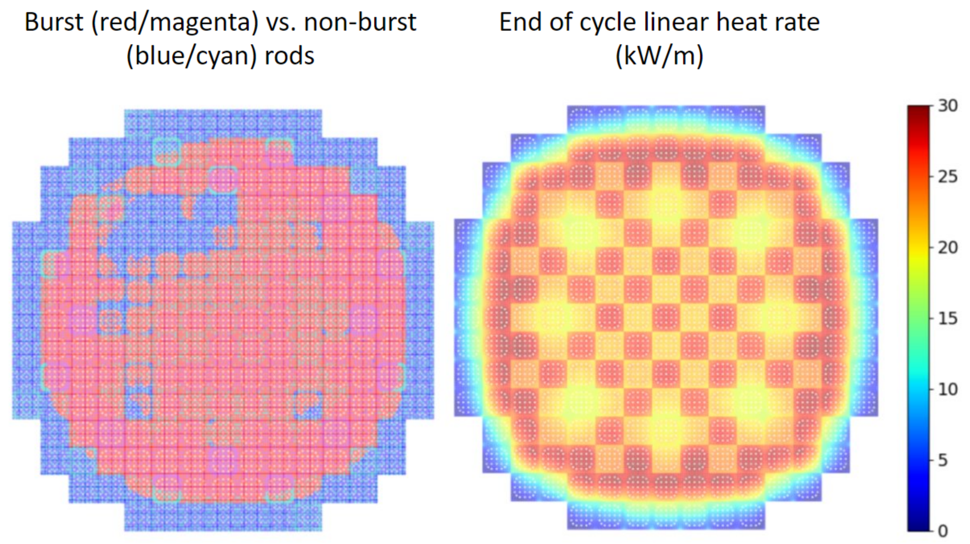


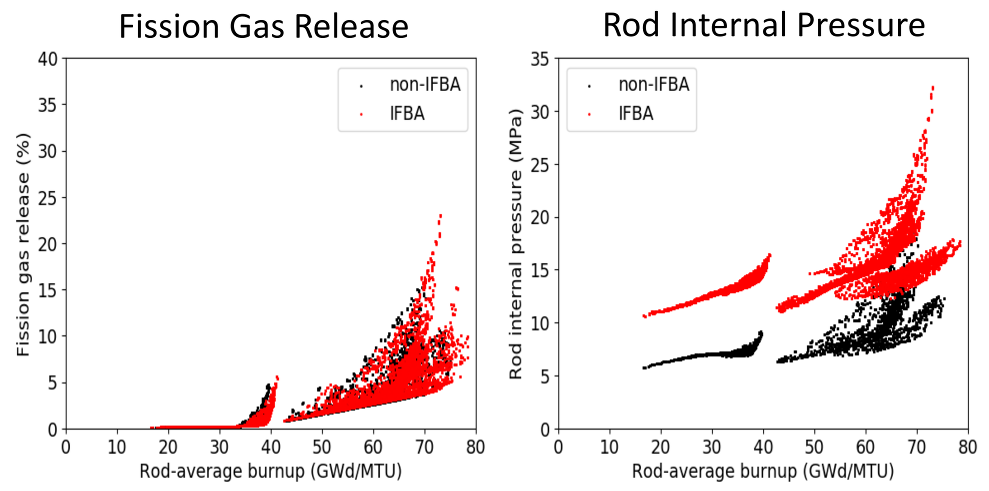
Fuel Fragmentation, Relocation, and Dispersal (FFRD)

Developing FFRD Models Based on Experimental Programs



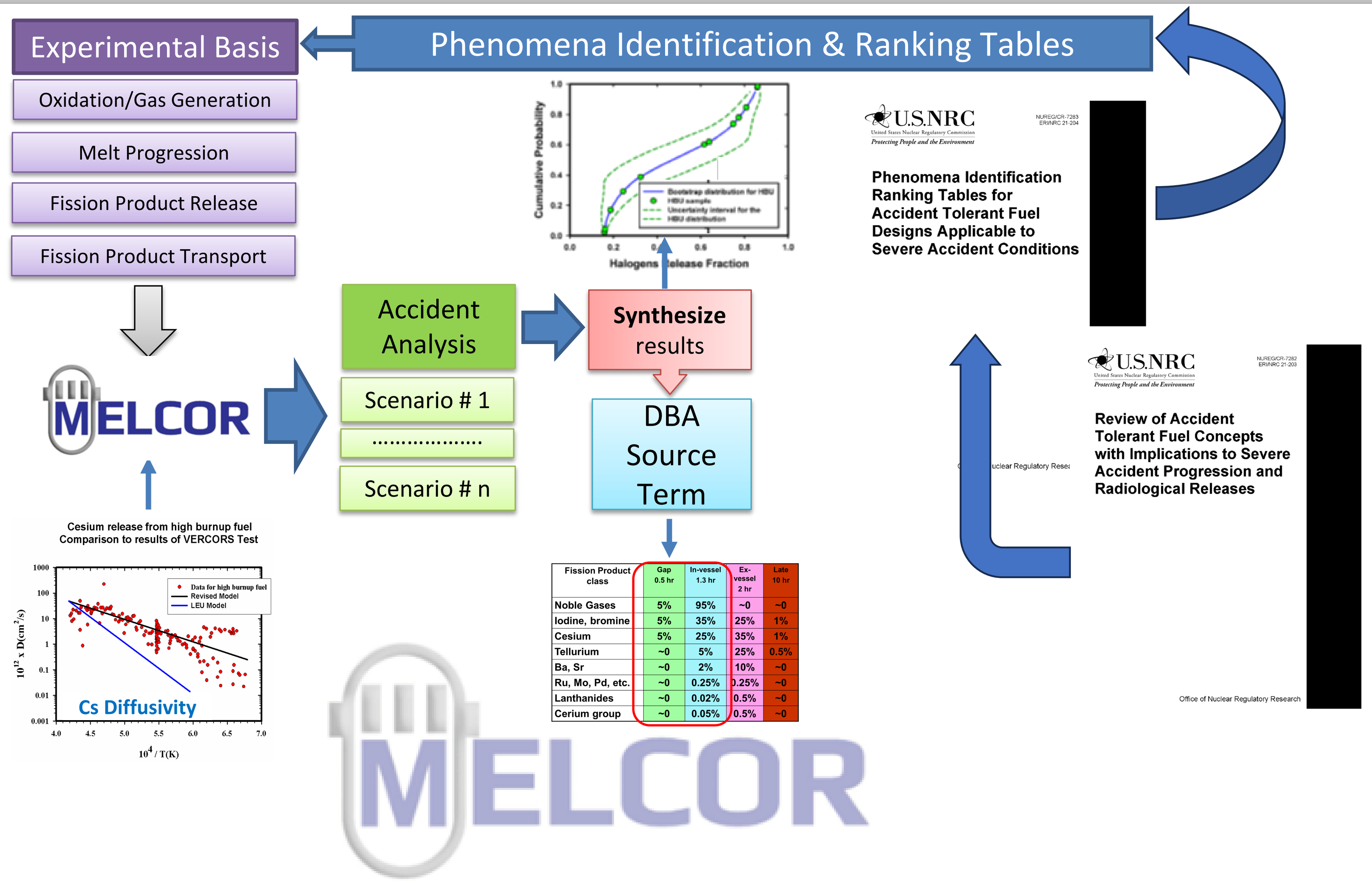
Application of FFRD Insights to Full Plant Simulations Using NRC Tools



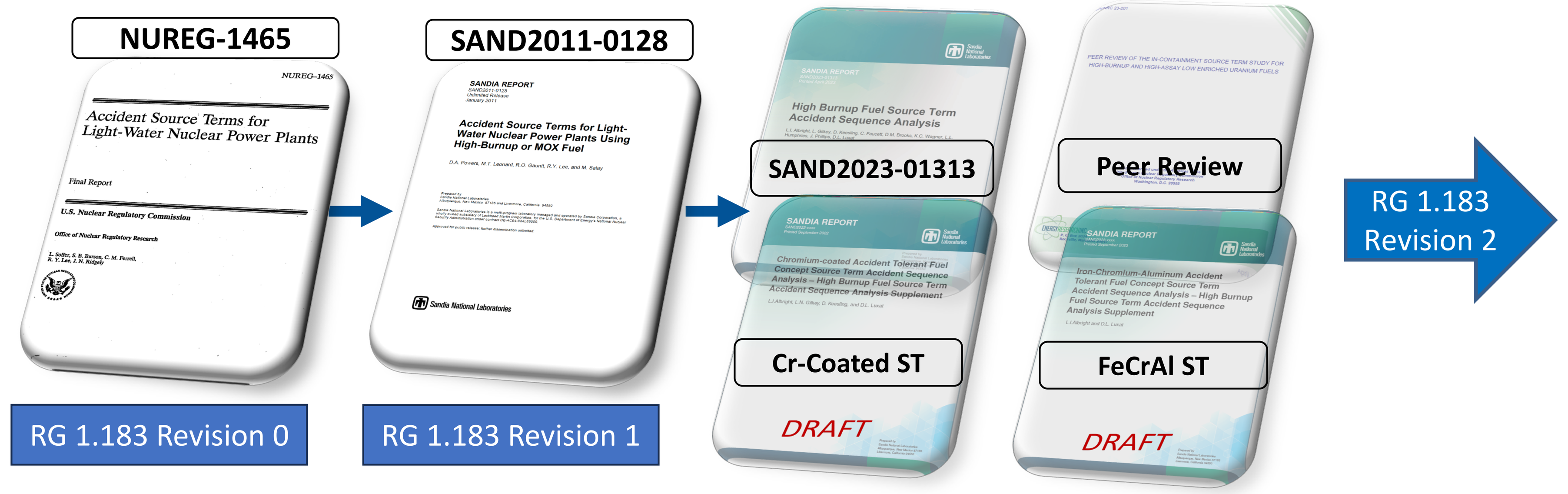


Regulatory Source Term

Process of Developing Regulatory Source Term



Evolution of Technical Basis for Regulatory Guide (RG) 1.183



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Additional Information

For questions or comments about material in this presentation, please contact Lucas Kyriazidis & Dr. Shawn Campbell in the NRC Office of Nuclear Regulatory Research, at Lucas.Kyriazidis@nrc.gov & Shawn.Campbell@nrc.gov.

For code documentation, a selected list of publications, and contact information, please visit the following websites:

FAST

scale

MELCOR

Non-LWR Demonstration Project