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ONGOING SEVERE ACCIDENT ANALYSIS ACTIVITIES AT SANDIA NATIONAL LABORATORIES

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ABSTRACT

Since the Three Mile Island Accident Unit 2 Accident, Sandia National Laboratories (SNL) have been a world leader in the phenomenological understanding and modeling of severe nuclear accidents. The majority of these efforts have been at the request of the Unites States Nuclear Regulatory Commission (NRC), including a wide array of experimental and modeling activities.

Currently, SNL develops the state-of-the-art severe accident analysis software MELCOR, the MELCOR Accident Consequence Code System (MACCS), the Radiological Assessment System for Consequence Analysis (RASCAL) and the Response Technical Tools (RTT) program for the NRC. These analytic codes have undergone significant revision and have had major releases following the severe accidents at Fukushima Daiichi to ensure all appropriate phenomena are captured.

The SNL supported Severe Accident software is continuously improved and benchmarked through involvement in a number of collaborative international programs and domestically funded research. A recent OECD/NEA effort designed to glean information from entries into the Fukushima Daiichi containments and conduct supplemental experiments have provided valuable insights. Of particular note is the now completed BSAF project, which provided the first opportunity for benchmarking MELCOR against a three-week long accident transient. Additional activities include the TCOFF, PreADES, ROSAU and ARC-F projects. Ongoing U.S. sponsored activities include SOARCA modeling activities of Peach Bottom, Sequoyah and Surry with MELCOR 2.2.

A robust SNL experimental program to characterize the behavior of spent nuclear fuel. After significant experiments to support spent fuel pool characterization, new experiments are being performed to better understand the behavior of dry cask storage options.

Lastly, both globally in within the U.S., a renewed regulatory and governmental interest in advanced non-light water reactors requires the expansion of software to capture non-LWR specific phenomena. SNL actively engaged in developing MELCOR to capture key phenomena necessary to accurately model non-LWR severe accidents taking into account lessons learned over the past four decades of severe accident research at SNL.

KEYWORDS
Severe Accident, MELCOR, Sandia National Laboratories

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1. INTRODUCTION AND SCOPE

This paper provides brief descriptions and introductions to four separate, ongoing SNL initiatives. The first is the continued development of MELCOR 2.2 for advanced reactors and accident tolerant fuel. The second is the development for the Response Technical Tools software for nuclear power plant accident monitoring for the NRC. The third topic addressed is SNL involvement in international programs at the OECD/NEA that focus on informing decommissioning at Fukushima Daiichi. Information from decommissioning will also provide key validation points for MELCOR and other severe accident analysis programs. The fourth area covered is experimental work on dry cask storage, particularly thermal-hydraulic and stress corrosion cracking.

2. MELCOR 2.2: ADVANCED REACTORS AND ACCIDENT TOLERANT FUEL

Since its inception, the modular framework of MELCOR has made it capable to model non-LWR problems without drastic alteration to the fundamental solution models. MELCOR has been used in radionuclide leak path evaluations, spent fuel evaluations and fusion reactor modeling. Over the past several years there has been increased interest in the United States for both advanced reactors and Accident Tolerant Fuel (ATF) has led to the implementation of models to support the necessary calculations required for the licensing of such technical advancements. [1]

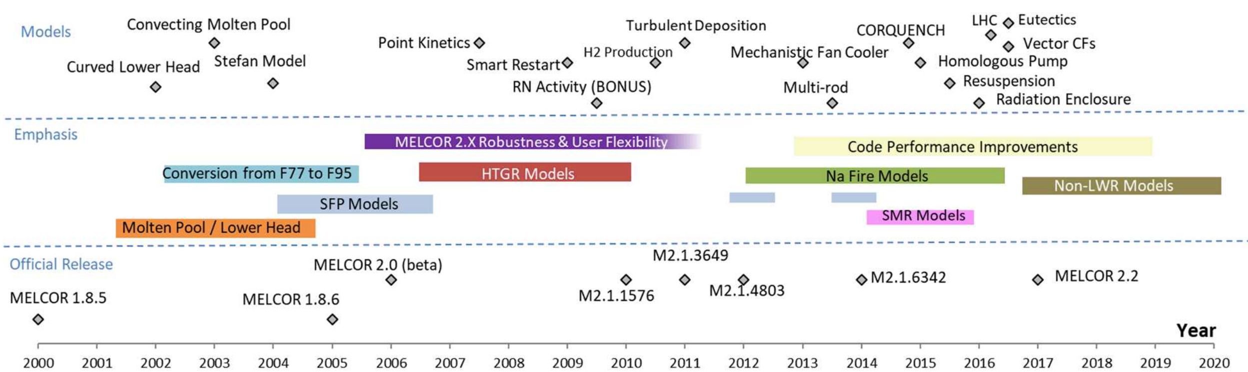


Figure 1. MELCOR Code Development Timeline

Advanced reactor models are currently being implemented to support the calculations required for molten salt reactors (MSRs), fluoride-salt-cooled high-temperature reactors (FHRs), high-temperature gas reactors (HTGR), sodium fast reactors (SFRs), small modular reactors (SMRs) and micro nuclear reactors. This work has been for both the NRC and DOE to support the design and licensing efforts of multiple domestic companies. Similarly, industry and DOE interest in ATF has led to the scoping and implementation of relevant ATF models within MECLOR. [1]

For HTGR, several additional models and physics were required to be added to MELCOR, including new components, point kinetics neutronics and TRISO fuel failure models. Relevant additions to the code include: [1]

- Insertion of helium gas properties
- Development of an accelerated steady-state initialization
- Two-sided reflector (RF) component
- Modified Fuel components (PMR/PBR)
- Point kinetics neutronics model for use in transients
- TRISO fuel failure and subsequent fission product diffusion, transport, and release
- Implementation of a cavity cooling system (ultimate heat rejection)
- Expanded oxidation models to include graphite oxidation (in O_2, H_2O)

- Graphite dust generation, transport (deposition and resuspension)

With these additions, MELCOR is able to model key accident scenarios including loss of coolant flow at high pressure accident (PLOFC), depressurized loss of coolant/flow accident (DLOFC), air ingress and subsequent graphite oxidation, anticipated transient without SCRAM (ATWS) and other reactivity insertions. [1]

To support the development and deployment of sodium fast reactor technologies sodium properties have been added to MELCOR, including the equation of state and thermos-mechanical properties. The implementation of these properties has also allowed for the implementation of other sodium-related components including sodium heat pipes. Significant work has also been performed to incorporate the capabilities of CONTAIN-LMR into MELCOR, providing a starting point for source term assessments. Models include: [1] [2]

- Sodium pool fire model
- Sodium spray fire model
- Atmospheric chemistry model
- Sodium-concrete interaction

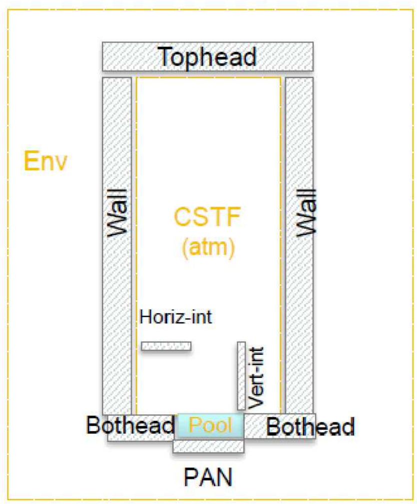


Figure 2. MELCOR representation of ALCOVE AB1 test for pool fire model

The modeling of salt-based reactors presents a new and unique challenge to MELCOR. For FHR designs significant amounts of the core models for HTGRs can be applied; however, for fuel-in-coolant MSRs new radionuclide transport and chemistry models are being. To date, the properties for LiF-BeF2 have been added to the source code including the equation of state and thermal-mechanical properties. This serves as a starting point for code validation and the implementation of other fluids and necessary models. [1]

May different ATF technologies have been explored and advocated by vendors, DOE and utilities. The most promising of these technologies are currently being scoped for implementation into the MELCOR framework. Additionally, previous work on SiC for fusion applications is being used as a starting point for this technology. [1]

3. INVOLVEMENT IN INTERNATIONAL PROGRAMS AND ACTIVITIES

In an effort to improve the capabilities of the MELCOR software code and enhance the accident source term program area, researchers at researchers at SNL, at the request of the NRC, are participating in multiple Organisation for Economic Collaboration and Development Nuclear Energy Agency

(OECD/NEA) activities in which severe accident. Programs in which SNL has recently participated include:

- BSAF: Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant [3] [4]
- TCOFF: Thermodynamic Characterisation of Fuel Debris and Fission Products Based on Scenario Analysis of Severe Accident Progression at Fukushima-Daiichi Nuclear Power Station [5]
- PreADES: Preparatory Study on Analysis of Fuel Debris [6]
- ROSAU: A Proposal for Experiments and Analysis to Reduce Severe Accident [7]
- ARC-F: Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima Daiichi Nuclear Power Station [8]

This paper provides a brief description of the SNL contribution to the BSAF, TCOFF and PreADES projects.

3.1. Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF Project)

The NRC and SNL, sponsored by the NRC, has participated in the BSAF project since it was originated in 2012. The first phase of this analysis was completed in 2015, and is published in “Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF Project) Phase I Summary Report.” This presented analyses focused on the reactor and containment for the first six days of the accident. [3] However, it became clear that to fully capture the atmospheric dispersion it was necessary to include the reactor building in the model and expand the calculation duration from six day to three weeks. These extended simulations constituted institutions contributions to BSAF Phase II, which is currently concluding. [3] [4] As a result of involvement in this activity, MELCOR code performance and runtime were enhanced, allowing three-week long core-damage simulations to be run in a few days – a significant improvement over previous capabilities. MELCOR was the first software that was able to perform such simulations. [3] [4] One such MELCOR calculation can be seen in Figure 3 for Fukushima-Daiichi Unit 2, showing significant agreement between MELCOR predictions and plant data.

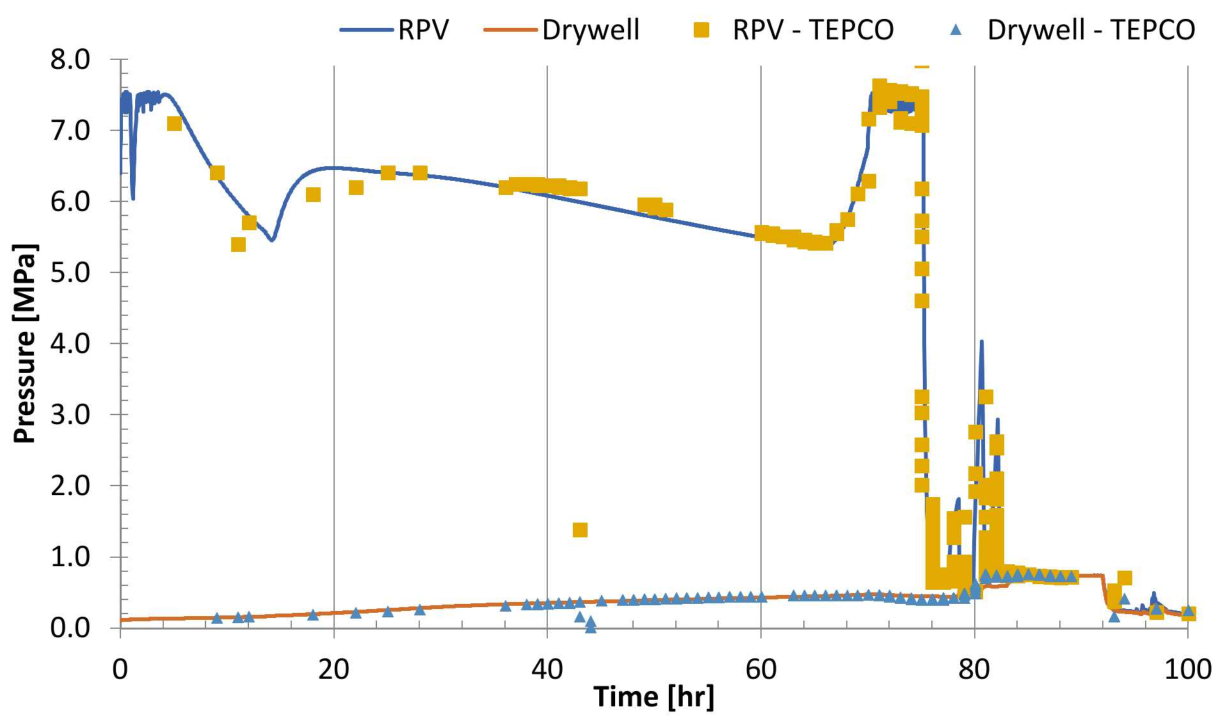


Figure 3. Fukushima-Daiichi Unit 2 BSAF Phase II calculation, comparing MELCOR results to TEPCO plant data.

Benefits to the MELCOR program include the improvement of software based on the insights gained from participating in the OECD/NEA BSAF Phase I and Phase II Projects. These highlighted several key areas where MELCOR could be (and subsequently was) enhanced by improving model robustness and implementing new dedicated models to better capture phenomenological behavior and key boundary conditions that drive source term. Specific examples include:

- Model corrections and numerical improvements to the MELCOR quench model were developed and implemented and have significantly improved the robustness of the code for reflood conditions.
- A temporal relaxation model was introduced within the code. Many physical processes in MELCOR are modeled by correlation based relationships developed from steady-state experiments. This has made it significantly easier to perform forensic analysis of core oxidation and relocation behavior for our analysis and improved code robustness.
- MELCOR crust formation and molten pool/crust formation modeling with a focus on steam permeability to severely damaged core regions and its effects on hydrogen generation, sensible heat gain and convective heat loss from such degraded regions.
- MCCI and corium spreading models which impact both the rate of both fission product release and combustible gas generations. [3]

3.2. TCOFF

The TCOFF project provides a vehicle for international nuclear severe accident and safety experts to interact with the scientific thermodynamic community. In particular, the knowledge base for the thermodynamic databases used to model key processes such as fuel relocation and core degradation are being expanded through a combination of simulation and modeling with thermodynamic tools. This work is being done in collaboration with severe accident experts analyzing analogous problems with higher-level, less-intensive system level codes including MELCOR and ASTEC. [4]

Thermodynamic evaluations are being performed to better characterize the following aspects of severe accident progression by analyzing them as quasi-equilibrium systems, which thermodynamic codes are designed to solve:

- fuel melting,
- molten core relocation,
- fission product behavior,
- fuel debris chemical and phase composition, and
- thermodynamic evaluation to determine the materials that potentially exist. [4]

Severe accident experts are, within this project, advising the thermodynamic experts on the most important binary and ternary systems to examine and potential materials present during the core degradation, thus expanding the overall thermodynamic database. Of particular interest is expanding the understanding of the U-Zr-O system under severe accident conditions. Within MELCOR, this ternary system is modeled as a pseudo-binary system (UO_2/ZrO_2) or with a single eutectic-informed melting temperature. Thermodynamic experts can, within the framework of TCOFF, perform thermodynamic calculations to evaluate and verify the current eutectic and pseudo-eutectic currently implemented within MELCOR. Work will also support future development by the NRC and SNL more accurate and comprehensive U-Zr-O eutectic models, should such models be proved necessary. [4]

3.3. PreADES

The PreADES project is focused on providing information to decommissioning authorities, including TEPCO and NDF, about potential end-states of the Fukushima-Daiichi damaged reactors. Experts are characterizing the locations of degraded materials, providing information on the fraction of fuel debris that remain in-vessel, relocated ex-vessel and spread to outer areas of the containment. By knowing the elemental composition of degraded materials within the containment, the decommissioning authorities can better prepare techniques for the removal of fuel removal. [5]

Such information would allow for the construction of a “map” of the containment of the reactor buildings, containments and reactor pressure vessels of the degraded reactors. This would include necessary information such as material hardness, density and neutronic characteristics. [5] Severe accident code analysts have been asked to provide input based on the most up-to-date simulation they have performed of the three reactors, preferably by applying an uncertainty analysis methodology. This would allow the full spectrum of possible end state to be captured for further characterization. [5]

As their contribution to this project, SNL is performing a miniature uncertainty analysis to identify representative MELCOR realizations that can be used as a starting point for thermodynamic calculations. One such calculation is show in Figure 4, showing the total amount of particulate debris within the lower plenum of 400 individual Fukushima Daiichi Unit 1 calculations, varying core degradation parameters.

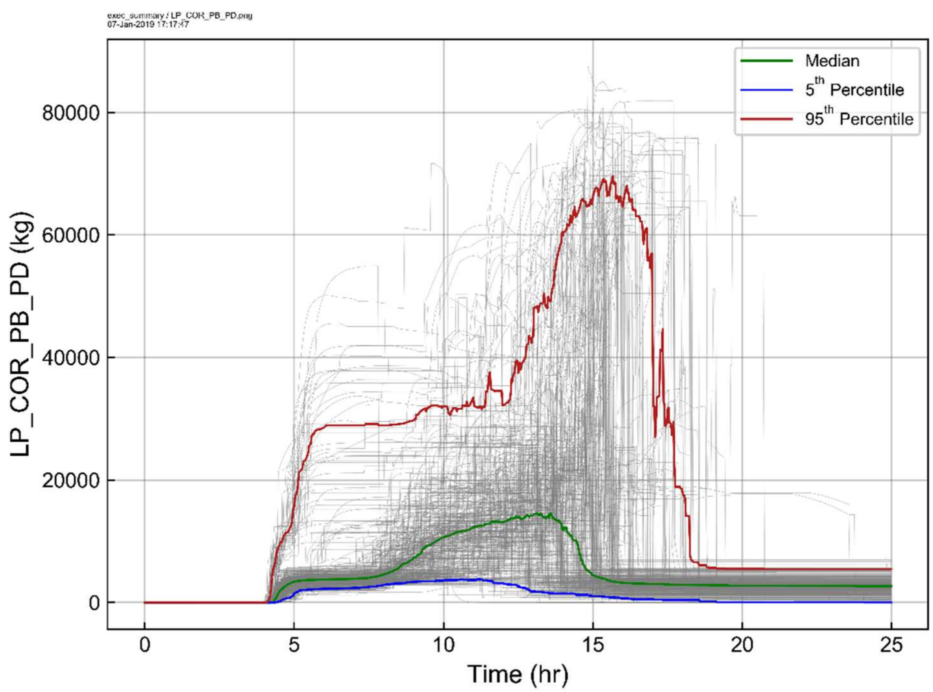


Figure 4. Particulate debris mass within the lower plenum for 400 separate Fukushima-Daiichi Unit 1 calculations, varying COR package material relocation parameters

Participation within this program provide SNL and the NRC access to the most up-to-date information from the entries to the containments of the tree damaged reactors. This provides key information for informing future MELCOR model development in the areas of:

- Core degradation,
- Pressure vessel upper internals modeling,

- Lower head modeling,
- Structures below the lower head,
- MCCI,
- Debris spreading upon relocating to the containment,
- Fission product release and distribution, and
- Relocation behavior of core internals. [5]

4. DEVELOPMENT OF THE RESPONSE TECHNICAL TOOLS SOFTWARE

In order to support the nuclear power plant accident response mandates of the (NRC), the Response Technical Manual (RTM) [8] was transformed into an automated and easy-to-use software program, named Response Technical Tools (RTT). [9] The RTM is a publicly available document, which provides a prediction as to the status of a nuclear reactor during a severe accident. The RTM is based on previous experiments and experience with severe accident phenomena. [8] The RTT is based on section A of the RTM, which is devoted to an assessment of core damage during a severe accident. It is also informed by state-of-the-art analyses and software programs, such as MELCOR and MAAP. RTT evaluations can be used to track and predict, at a very coarse level, the progression of a severe accident in nuclear power plant. The goal of this code is to automate the simple, yet very time consuming, calculations that are necessary to evaluate the status of a reactor during accident. This allows an analyst to spend more time making an evaluation as to the status of the plant and not performing simple calculations. The assessment is broken into five separate steps:

- Assess Critical Safety Systems
- Core Uncovery Determination
- Estimate Timing of Core Damage
- Assess Core Damage State
- Deflagration Assessment [9]

Upon an initial assessment, continued monitoring can be performed in the program until the end of the accident. The initial assessment screen can be seen in Figure 5.

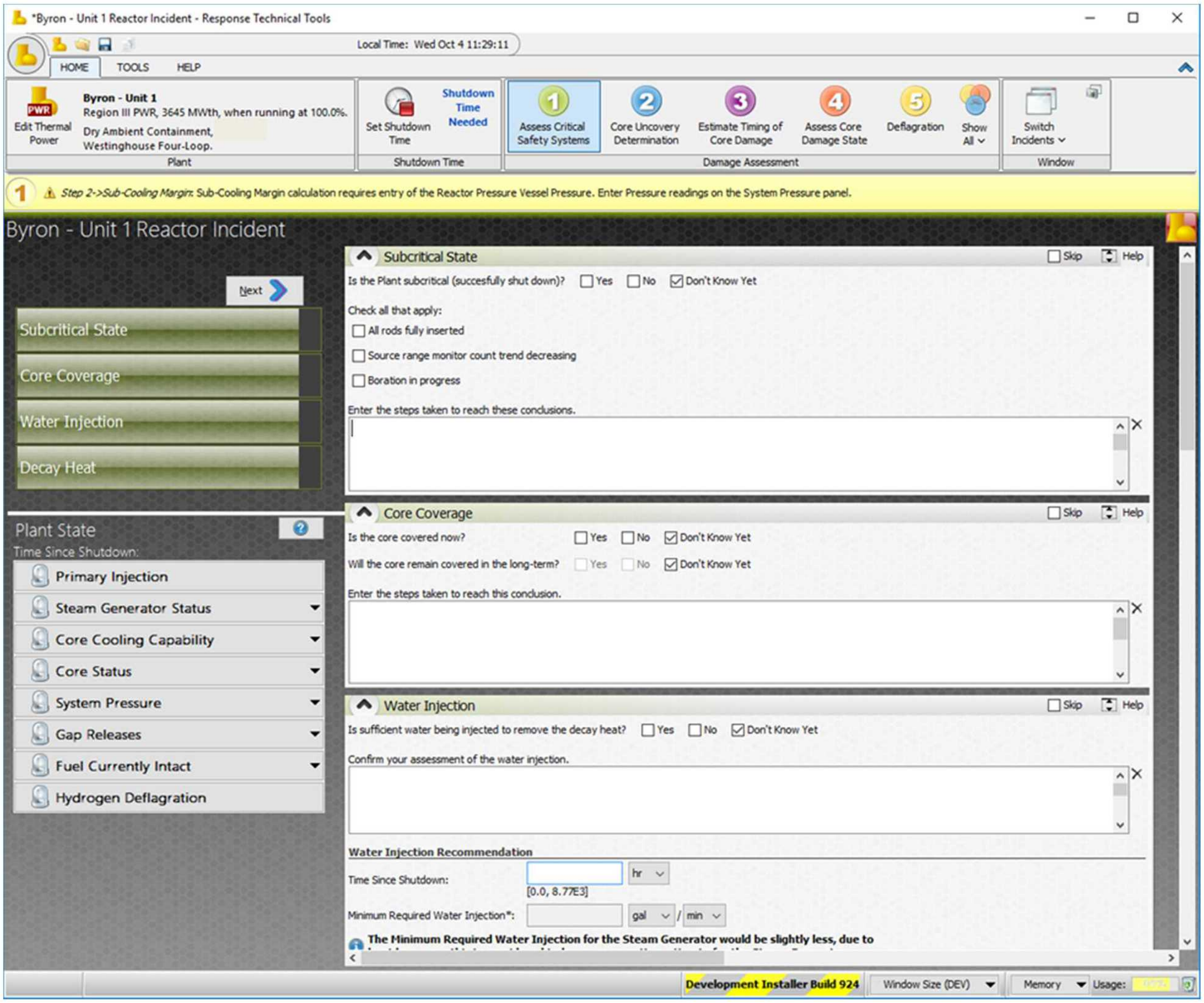


Figure 5. Step One Elements and Plant State.

All calculations performed within the RTT code are based off those detailed in the RTM. Among the important plant status calculations, are the following:

- Project the status of core and extent of core damage (See Figure 6.)
- Predict time of core dryout
- Possibility of a H2 explosion
- Determine whether the core can be cooled [9]

The inputs for all these calculations would be provided to the code user by the operator of a reactor undergoing a severe accident.

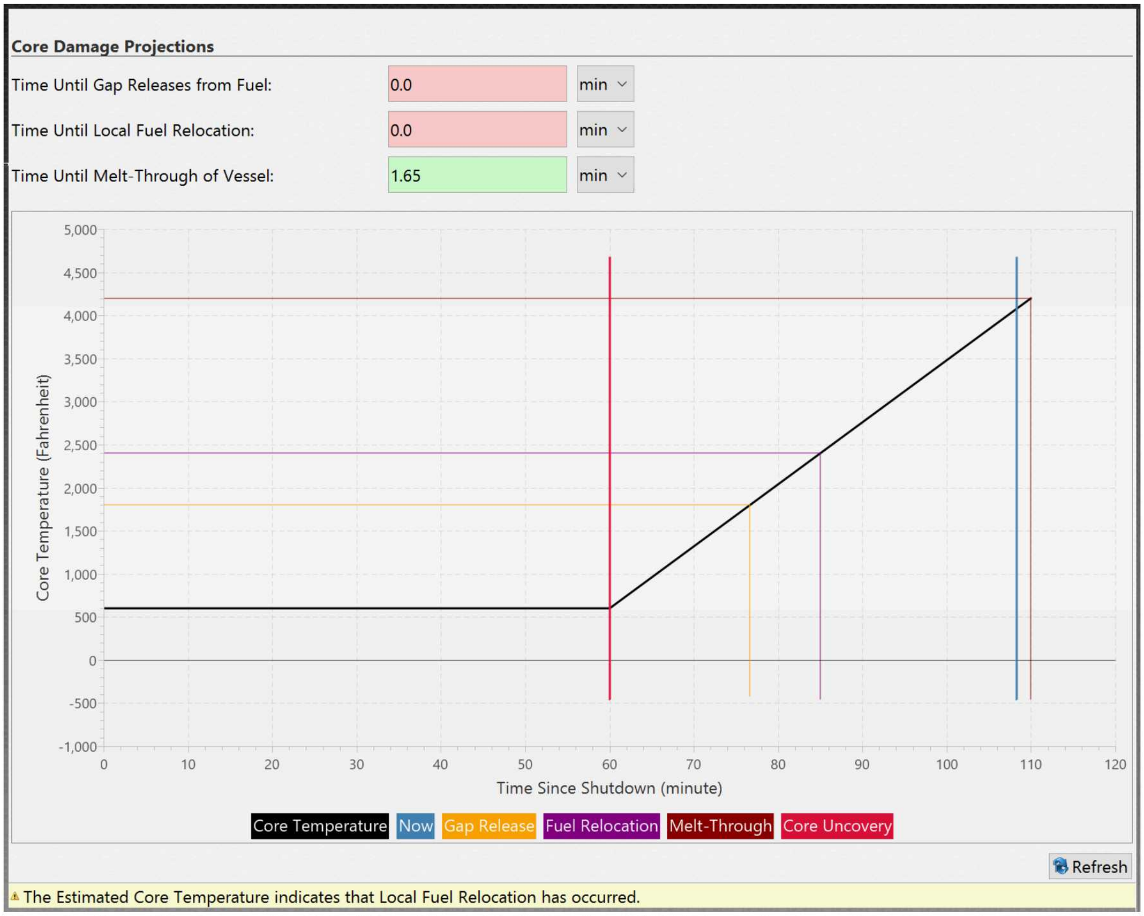


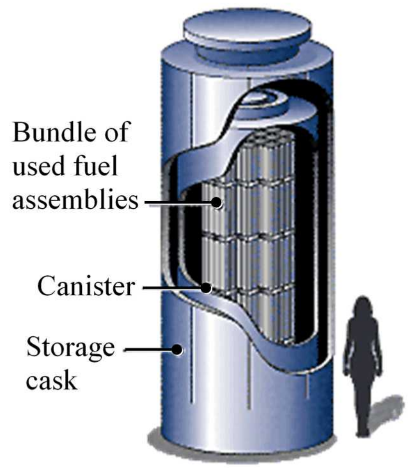
Figure 6. Core Damage Projections Provided by the RTT, based upon the time of core uncovery.

As RTT is undergoing further development, the current primary effort is to update old methods to incorporate modern research and add new predictive features. The latest released revision of RTT is version 1.3, released on 01/30/2017. [9]

The RTT allows a user to track the progress of an accident in a nuclear power plant from the point of initiation through the point of containment breach and release to the environment. It is to be used by reactor safety experts in order to provide an accurate and relevant assessment of the status of a nuclear power plant a consequence analysis team. The consequence analysis team, using a tool such as the Radiological Assessment System for Consequence AnaLysis (RASCAL). [9]

5. SPENT FUEL ACTIVITIES

One of the largest experimental programs at SNL in the past decade has been the characterization of spent nuclear fuel (SNF) under accident conditions. SNF is initially stored in pools of water for cooling where the water also provides radiation shielding. As these pools get closer to capacity, dry storage systems are becoming the primary alternative for interim storage. After sufficient cooling in pools, SNF is loaded into a canister and placed inside a storage cask, where the canister is welded shut. Then the dry storage cask system is decontaminated and dried, and the system is ultimately sent to a storage location. Recently a large campaign to characterize the behavior of SNF in spent fuel pools under various loss of coolant accident (LOCA) conditions was performed, characterizing oxidative behavior up to and including cladding ignition. [10] Now the focus of SNL research has shifted to better characterizing dry cask storage systems (DCSS). [11] [12]



Source: https://www.nrc.gov/waste/spent-fuel-storage/diagram-typical-dry-cask-system.html

Figure 7. Typical vertical dry storage cask.

5.1. Thermal-Hydraulic Characterization

Recently the thermal-hydraulic response of commercial DCSS has been evaluated numerically and then validated against experiments performed at SNL for casks stored both above and below ground. Carefully measured data sets generated from testing of full sized casks or smaller cask analogs are widely recognized as vital for validating these models. Recent advances in dry storage cask designs have significantly increased the maximum thermal load allowed in a cask in part by increasing the efficiency of internal conduction pathways and by increasing the internal convection through greater canister helium pressure. These same canistered cask systems rely on ventilation between the canister and the overpack to convect heat away from the canister to the environment for both aboveground and belowground configurations. While several testing programs have been previously conducted, these earlier validation attempts did not capture the effects of elevated helium pressures or accurately portray the external convection of above and below ground canistered dry cask systems. [11]

To do this, a test apparatus simulating a modern dry cask was constructed and operated to produce first-of-a-kind, high-fidelity transient and steady-state thermal-hydraulic data sets suitable for CFD model validation. An existing electrically heated but otherwise prototypic BWR Incoloy-clad test assembly was deployed inside of a representative storage basket and cylindrical pressure vessel that represented the canister. Simulated decay power was scaled to mimic the desired range of prototypic dimensionless groups. Test configurations for both vertical aboveground and belowground storage cask systems were tested. A wind machine was used to test the effect of wind speed on the peak cladding temperature and induced air mass flow rate in the belowground configuration. Power levels between 0.5 and 5.0 kW at pressures between 0.3 kPa and 800 kPa for differing wind providing a validation matrix of over 40 unique data sets. Fourteen data sets for the aboveground configuration were recorded for powers and internal pressures ranging from 0.5 to 5.0 kW and 0.3 to 800 kPa absolute, respectively. Similarly, fourteen data sets were logged for the belowground configuration starting at ambient conditions and concluding with thermal-hydraulic steady state. Over thirteen tests were conducted using a custom-built wind machine. This addition to the dry cask experimental database signifies a substantial addition of first-of-a-kind, high-fidelity transient and steady-state thermal-hydraulic data sets suitable for CFD model validation. [11]

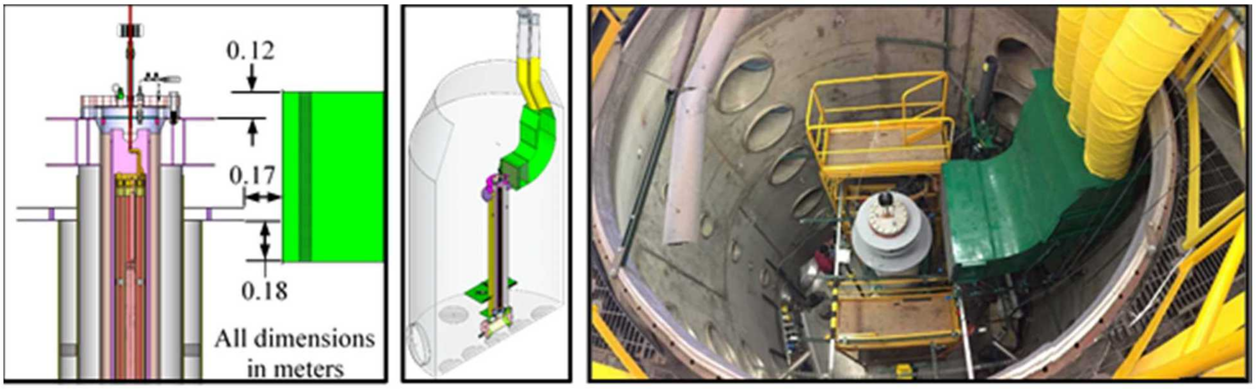


Figure 8. Layout of the cask simulator and wind machine for cross-wind testing.

5.2. Particulate Retention and Subsequent Stress Corrosion Cracking

A second study in the DCSS campaign at SNL has focused on developing methods for quantifying potential releases through a stress corrosion crack (SCC) that could form casks. Typically, the dry storage casks are made of stainless steel, with the open volume between the canister and the cask allowing passive ventilation from outside air, which can impart dust that collects on the surfaces of the canister. As the SNF cools, salts contained in the dust may deliquesce to form concentrated brines, which may contain corrosive species such as chlorides. These species are capable of causing localized corrosion, called pitting. With sufficient stresses, these pits can evolve into stress corrosion cracks, which could penetrate through the canister wall and allow communication from the interior of the canister to the external environment [13].

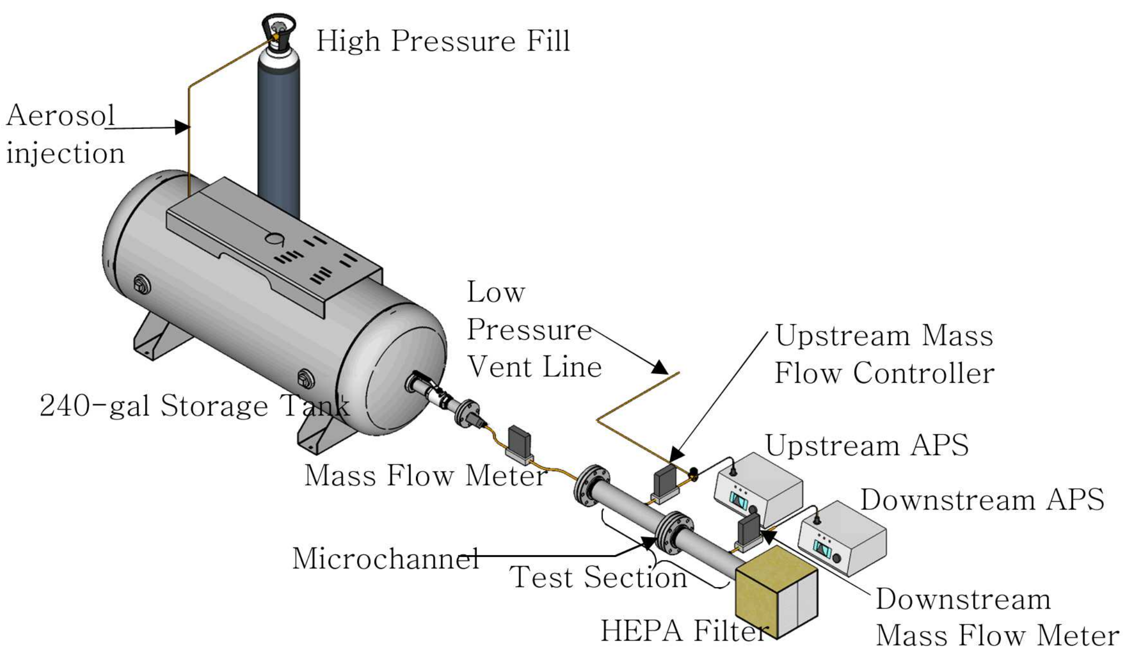


Figure 9. General layout of the experimental apparatus.

The flow rates and aerosol retention of an engineered slot with characteristic dimensions similar to those in SCCs were explored. Pressure differentials covering the upper limit of commercially available dry cask storage systems were studied (up to 700 kPa). These data demonstrated a new capability to characterize flow and aerosol transport through stress corrosion cracks under well-controlled boundary conditions. [2] Improvements to the test apparatus and more prototypic SCC are being explored for future testing.

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Dry cask work on stress corrosion cracking and aerosol retentions was conducted under the Department of Energy Spent Fuel and Waste Science and Technology campaign.

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