Task 4.c: Analysis Capability Development – Neutronics

Neutronics calculations are an integral part of the design-basis accident analyses required by Chapter 15 of NUREG-0800, because they provide decay heat rates, core power, and reactivity values used by thermal hydraulic and fuel performance codes. Neutronics analysis is also required for the quantification of nuclide inventory for severe-accident/consequence analyses required by Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100 and for evaluations supporting 10 CFR 50.68, "Criticality Accident Requirements," 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," and 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." Neutronics analysis is also performed to support decisions for spent fuel pool (SFP) loading, for confirming fluence calculations necessary to quantify vessel embrittlement, shielding analyses to support as low as reasonably achievable (ALARA) objectives, calculations of SFP decay heat rates and dose for human reliability analyses, core power/reactivity for transient calculations, assembly decay heat rates for cask loading, etc.

The Nuclear Regulatory Commission's (NRC's) main neutronics codes are SCALE which provides a computational capability to evaluate nuclear systems, and GenPMAXS/PARCS to evaluate time dependent core performance. SCALE is used by the NRC staff to support licensing reviews by performing criticality safety evaluations of enrichment and fuel fabrication facilities, developing lattice physics parameters for reactor operations, performing safety evaluations for transport and storage, as well as use in spent fuel pools, severe accidents and input into probabilistic risk assessment (PRA). PARCS is used as a core simulator that supports thermal hydraulic reviews under design basis scenarios. GenPMAXS reads the lattice physics parameters from SCALE (nodal averaged cross sections and kinetic parameters) and converts the nuclear data into the format form that is required by PARCS.

Scoping Study – A review of the current NRC neutronics packages, SCALE and GenPMAXS/PARCS, will be necessary to understand the modifications, if any, that will be required to characterize accident tolerant fuel (ATF) for the whole fuel cycle. These codes need to be reviewed against the unique features of the ATF fuel designs (e.g., coated-Zirconium, doped-UO_2, FeCrAl, U_3Si_2, SiC), and for enrichments of higher than 5 weight percent U-235 (up to 20 weight percent). The SCALE code suite covers the following functional areas: nuclear data and methods, Monte-Carlo methods, isotope decay, depletion and activation methods, reactor physics methods, and sensitivity and uncertainty methods. The GenPMAXS/PARCS code covers reactor operations and transient performance.

Consideration of the needs for each of these functional areas will be included in the scoping study. The scoping study will necessarily involve a review of the fuel cycle and the associated impact of ATF's designs.

Code Architecture Updates – The required infrastructure development activities for SCALE and GenPMAXS/PARCS involve any code modifications and/or enhancements identified in the scoping study for implementation.

Expected development activities include:

- Coupling of SCALE and PARCS with Fuel Analysis under Steady-state and Transients (FAST) in order to receive detailed fuel data (such as temperatures and geometry) while providing intra-pin radial power profiles and axial rod power profile to FAST (to eliminate the need for FAST to develop new correlations)
- SCALE geometry enhancements to support non-cylindrical fuel for both 3D Monte Carlo analysis and 2D lattice physics calculations
- SCALE input and modeling investigations and enhancements to model coated cladding, where coat thickness is the less than 100 microns
- Updates to the energy group structures (both the fine multi-group structure that is used in SCALE and the collapsed broad group structure that is used by PARCS for core calculations), updates to the nuclear data library, nuclear methods development (at the lattice and nodal level) to enable pin power reconstruction for non-cylindrical fuel, and input interfaces

Model Development – The required development activities for SCALE and GenPMAX/PARCS involve any code modifications and/or enhancements identified in the scoping study for implementation. Expected development activities include:

- Evaluation of the depletion and activation effects for ATF fuel and cladding compositions
- Evaluation of the impact of thermal hydraulic and fuel performance effects that may be more or less important for ATF such as thermal expansion, heat capacity, thermal conductivity, swelling and gap closure, etc.
- Evaluation of a different cross section parameterization methodology and the development of different fuel temperature averaging techniques in order to better characterize Doppler feedback for the different carbide, ceramic, and metallic fuels under consideration
- Evaluation of the methodology for thermal-hydraulic calculations with steady-state PARCS calculations (PARCS includes PATHS, a simplified drift flux formulation, for thermal-hydraulic feedback). This will include the accommodation for more extreme axial discontinuities and heterogeneities, the evaluation of the coupling of the fuel and wall temperature to the determination of bulk fluid temperature, the evaluation of the applicability of the current constitutive relationships (void fraction quality models, subcooled quality, wall friction factor, and two-phase flow friction factors)
- Sensitivity and uncertainty assessment, including assessment of modelling techniques, for ATF candidates
- Evaluation of tritium release through advanced cladding materials such as SiC cladding

(Note: parts of these activities can be started before the receipt of necessary data)

Code Assessment and Validation – Code verification and validation are important elements of the SCALE and GenPMAXS/PARCS software quality assurance (SQA) program. All new updates, modification, enhancements, etc., must be assessed against test data. These would include test data from post irradiation examinations (e.g., destructive assay of fuel and/or clad to validate depletion), but would be finalized by the gap analysis.

Within the code assessment and validation task, testing of the combined code sequence for a particular application will also be required. Within reactor operations space, the following computer codes will have to be tested together, SCALE-GenPMAXS/PARCS-TRACE for example.

The milestones are listed in the table below with their related trigger or needed input, lead time, and schedule driver.

“Evolutionary” ATF concepts (e.g., coated cladding, doped pellet, FeCrAl).

Activity	Data Needs and Inputs	Duration	Needed By
Scoping Study	Short duration task		
Code Architecture Updates	Completion of previous milestone	1 year	-
Model Development	Data as required from scoping study	1 year	-
Code Assessment and Validation	Data as required from scoping study	1 year	Safety Analysis Submittal
Lead Time for Neutronics Analysis Capability for Evolutionary ATF	1-2 years		

“Revolutionary” ATF concepts (e.g., SiC, U_3Si_2, metallic pellets, or solid rods).

Activity	**Data Needs and Inputs**	**Duration**	**Needed By**
Scoping Study	Short duration task		
Code Architecture Updates	Completion of previous milestone	2 years[2]	-
Model Development	Data as required from scoping study	2 years[2]	-
Code Assessment and Validation	Data as required from scoping study	2 years[2]	Safety Analysis Submittal
Lead Time for Neutronics Analysis Capability for Revolutionary ATF	2-3 years		

[2] Tasks can be worked on in parallel

Technical Lead: RES/DSA/FSCB