The U.S. Nuclear Regulatory Commission Approach to Modeling and Simulation of Advanced Non-LWRs;

Preparing for the Next Nuclear Renaissance

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ABSTRACT

Nuclear reactor thermal-hydraulics has always been a challenging technical area as the industry has developed both conventional and passively cooled light-water reactors and improved fuel designs. The nuclear industry has continuously improved and evolved in order to improve safety and economics. However, in spite of improvements in performance there remains the need for enhanced safety and to adapt to ever changing market conditions.

The U.S. nuclear industry is responding by proposing a wide variety of advanced non-light water reactors; gas-cooled, liquid metal cooled, molten salt cooled, and what may be termed "micro" reactors cooled using heat pipes. Both fast and thermal spectrum reactor designs are being developed, with fuels ranging from TRISO and metallic to liquid fuel salts. While the thermal-hydraulics of the fluids proposed for these new designs is relatively well understood, modeling and simulation of these new designs and their behavior during hypothetical accident scenarios represents a challenge to both design and licensing.

U.S. Nuclear Regulatory Commission (NRC) has developed a vision and strategy to assure that the NRC is ready to review potential applications for non-light water reactor technologies effectively and efficiently. An important part of that strategy is development of the capability to perform confirmatory analysis of these designs, and to ensure that analysis capability supports an accelerated review schedule. This paper describes the modeling and simulation approach the NRC is developing for non-LWRs. The codes and rationale for their use in confirmatory analysis is discussed, as well as some of the phenomena and processes that must be incorporated into the analysis.

KEYWORDS: Thermal-hydraulics, nuclear safety, licensing, design basis event analysis

1. Introduction

The nuclear industry is at a crossroads. Over the past several years some plants have pre-maturely shutdown and entered into decommissioning. Other plants are clearly stressed economically and are considering pre-mature closure. The market forces are harsh. Yet, there are compelling reasons for the nuclear industry to continue to provide an important contribution to the energy sector and possibly even expand. Global efforts to address climate change and to meet the energy needs of developing nations are likely to need new sources of clean, carbon-free energy. Currently, nuclear plant provide roughly 20% of the electrical power capacity in the U.S., and represent approximately 60% of carbon-free production. If world-wide demand for increased carbon-free electrical production continues nuclear must certainly play a role.

In the U.S. there has been considerable interest in advanced non-light water reactors as a means to provide carbon-free energy, and for their potential to replace existing, yet aging, conventional nuclear units. A significant number of conventional plants are expected to retire over the two decades. Replacement of this capacity is necessary, and offers an opportunity to new nuclear capacity.

Numerous designers are responding to this opportunity. Advanced non-LWR designs utilizing a variety of coolants; gas, liquid metal, molten salt, are under development. Preliminary estimates are that these new designs can improve safety, improve economics, and provide capabilities not possible with conventional designs.

The U.S. NRC is responding to this changing environment by updating its licensing process and by preparing for its review of non-LWRs. The NRC has developed a vision and strategy to assure that the NRC is ready to review potential applications for non-light water reactor (non-LWR) technologies effectively and efficiently [1]. The strategy consists of six strategic areas:

(1) staff development and knowledge management
(2) analytical tools
(3) regulatory framework
(4) consensus codes and standards
(5) resolution of policy issues
(6) communications

The purpose of this paper is to discuss and focus on strategic area #2; analytical tools. As pointed out in the vison and strategy document, "The staff must have adequate computer models and other analytical resources to conduct its review of non-LWR designs in an independent manner. The emphasis in the staff's approach is to leverage, to the maximum extent practical, collaboration and cooperation with the domestic and international community interested in non-LWRs with the goal of establishing a set of tools and data that are commonly understood and accepted." While not a specific requirement of a review for Design Certification, the staff frequently relies on its analytical tools to perform independent confirmatory studies of a new or modified design. However, this is often a critical step in a successful review and approval as the confirmatory findings inform the staff on safety margin and adequacy of an applicant's submittal. The use of codes for confirmatory analyses represents one of most frequent uses of the NRC's codes. The role of confirmatory analysis has been documented in the Office of New Reactors' (NRO) 'Confirmatory Analysis Job Aid" [2]. Confirmatory analyses are used by staff to obtain insights on the results of a licensee's or applicant's analyses and provide additional confidence in the staff findings. Confirmatory analyses are a useful and recommended tool when:

- Novel design features are involved and sufficient historical regulatory basis associated with NRC review and approval of such design features does not exist;
- The licensee or applicant deviates from an acceptable method (i.e., proposes an alternative method) cited in NRC guidance and the licensee's or applicant's design bases documents, and justification provided within the application, raises fundamental safety concerns;
- The staff determines it is necessary to confirm the licensee's or applicant's prediction of responses to postulated accidents for a structure, system and/or component; and
- The staff determines it is necessary to confirm the licensee's or applicant's conformance to NRC guidance and compliance with NRC regulations.

The following sections discuss the analytical tools, the basis for selection, and development approach. Because confirmatory analysis is integral to the licensing process, the NRC's approach to preparing for the review of non-LWR designs is also discussed.

2. Licensing Preparedness

The NRC's review and licensing processes are intended to be flexible and allow interactions related to a wide variation in design development and deployment strategies. However, the vast majority of experience with regards to licensing has been for light water reactors. The NRC recognized this potential difficulty and regulatory uncertainty, and engaged with stakeholders on two projects relevant to modeling and simulation of non-LWRs; the Licensing Modernization Project (LMP), and the Advanced Reactor Design Criteria (ARDC).

The Licensing Modernization Project is being led by Southern Company, coordinated by the Nuclear Energy Agency (NEI), cost-shared by the Department of Energy (DOE) with active participation by the NRC. The objective of the LMP is to develop technology-inclusive, risk-informed, and performance based regulatory guidance for licensing non-LWRs. A draft of the LMP was released [3] providing guidance on several issues important to regulatory requirements. Of significance to modeling and simulation is the selection of events. The LMP describes a systematic process for identifying and categorizing event sequences as anticipated operational occurrences (AOOs), design basis events (DBEs), or beyond-design-basis events (BDBEs). The primary determinate for categorizing events is the estimated frequency of the event sequence, rather than a deterministic characterization of events. Probabilistic Risk Assessment (PRA) models are expected to be developed and refined as the design process progresses and the licensing basis documents are developed. For traditional modeling and simulation analysis, this tends to "blur" the boundary between AOO, DBE, and BDBE and may require an overlap in analysis capabilities. For the present discussion, DBE analysis assumes that at the end on an event the core remains intact while in BDBEs there has been significant core disruption and release of fission products.

The second project of significance to modeling and simulation is the development of the Advanced Reactor Design Criteria (ARDC). In 2013, the DOE and the NRC established a joint initiative to address a key portion of the licensing framework essential to advanced reactor technologies. The initiative addresses the "General Design Criteria for Nuclear Power Plants," Appendix A to 10 Code of Federal Regulations (CFR) 50, which were developed primarily for LWRs, by adapting them to the needs of advanced reactor design and licensing. These General Design Criteria (GDC) are the basis for review of a design. The NRC staff evaluates the design based on these criteria, and seeks to ensure that the application clearly shows how these criteria are satisfied. In 2018 the NRC issued its ARDC [4], which addressed the design criteria for two specific non-LWR design concepts: sodium cooled fast reactors (SFRs), and modular high temperature gas-cooled reactors (MHTGRs).

The representative LWR regulatory areas in the ARDC where applicants and the NRC depend on modeling and simulation tools, computer codes, and computational analysis are shown in Figure 1. The codes shown here are those traditionally used in LWR analysis. Accordingly, analyses are performed to demonstrate compliance with the regulatory requirements, and modeling and simulation tools used in NRC confirmatory analyses can be effective in helping the staff to evaluate the importance of various phenomena and overall safety margins.

There are however, more designs under consideration than sodium cooled fast reactors, and modular high temperature gas-cooled reactors. Currently, there are organizations designing many other types of reactors including molten salt cooled reactors, and what might be termed “micro” reactors cooled by heat pipes. In the next section, the variety of design types are characterized as this is defines analysis requirements for an NRC tool set.

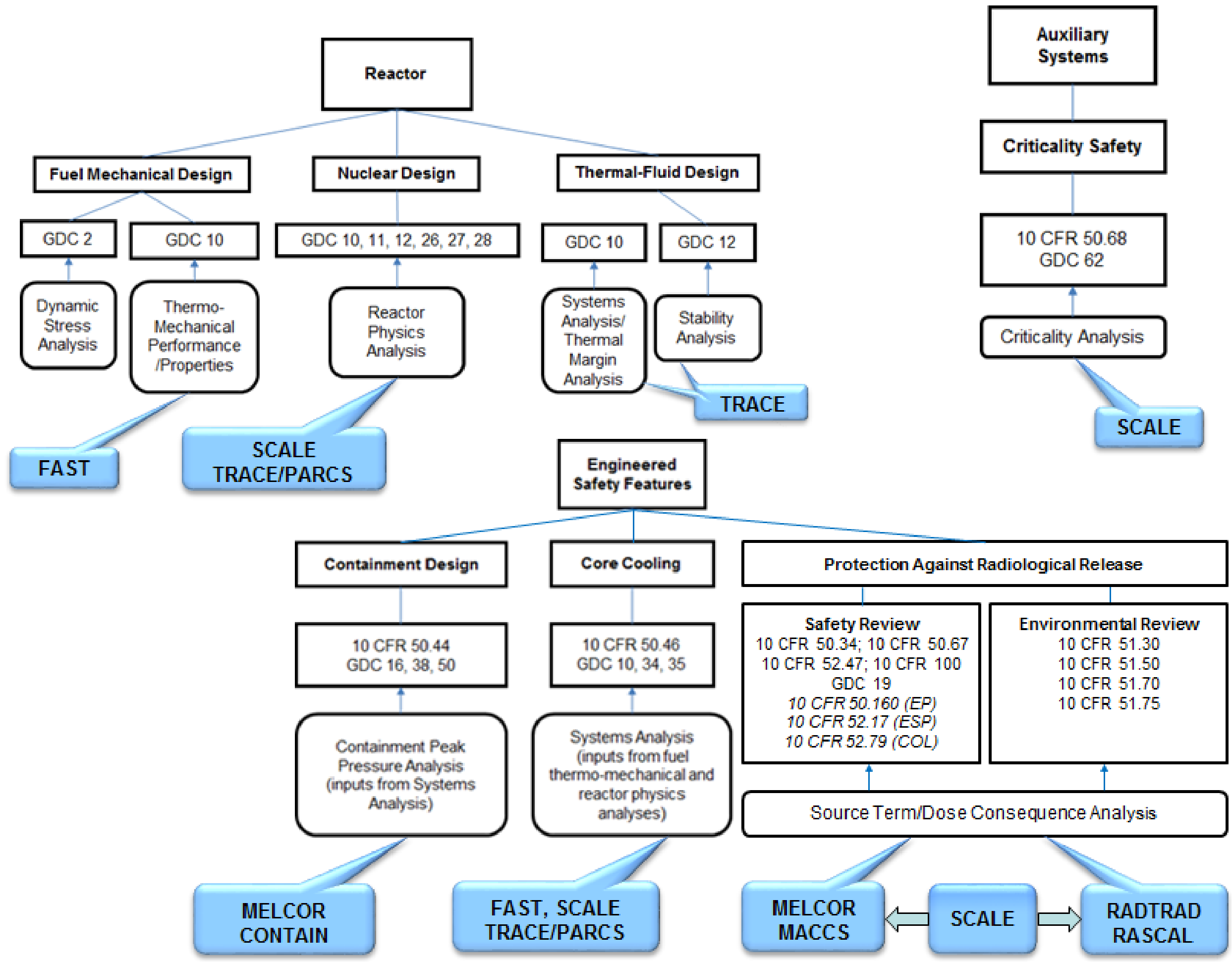

Figure 1. Confirmatory Analysis and General Design Criteria for LWRs.

3. Classification of Design Types

There are numerous designs under development, and each is unique in the specifics. However, to approach code applicability, these designs have been characterized into specific "bins." The designs currently under consideration can be characterized into ten generic design types. Table 1 lists these designs, and the expected fuel type for each design. There are numerous organizations that have expressed an interest in obtaining a license from the NRC, but to avoid proprietary information none are named here. These designs can be further categorized into four general design type: gas-cooled reactors, liquid-metal reactors, heat-pipe cooled "micro" reactors, and molten salt reactors.

Table 1. Characterization of non-LWR Designs

Plant Type Number	Description	Fuel
1	High Temperature Gas Cooled (HTGR); prismatic core, thermal spectrum	TRISO (rods or plates)
2	Pebble Bed Modular Reactor (PBMR); pebble bed core, thermal spectrum	TRISO (pebbles)
3	Gas Cooled Fast Reactor (GCFR); prismatic core, fast spectrum	SIC clad UC (plates)
4	Sodium Cooled Fast Reactor (SFR); sodium cooled, fast spectrum	Metallic (U-10Zr)
5	Lead Cooled Modular Reactor (LMR); lead (or lead-bismuth) cooled, fast spectrum	Nitride
6	Heat Pipe Cooled Reactor (HPR); heat pipe cooled "micro" reactor, fast spectrum	Metallic (U-10Zr)
7	Molten Salt Cooled Reactor (MSR); prismatic core, thermal spectrum	TRISO (plates)
8	Molten Salt Cooled Pebble Bed Reactor (MSPR); pebble bed core, thermal spectrum	TRISO (pebbles)
9	Molten Fluoride Salt Reactor (MFSR); fluoride fuel salt, thermal spectrum	Fuel salt (liquid)
10	Molten Chloride Salt Reactor (MCSR); chloride fuel salt, fast spectrum	Fuel salt (liquid)

Each of these design types involves physical phenomena that are unique to its geometry, coolant, fuel, and spectrum. Modeling and simulation of these for design basis events requires an Evaluation Model (EM) that is developed and validated for the phenomena expected in a particular transient. The NRC follows the process described in Regulatory Guide 1.203 [5] to develop its own codes, as well as to provide guidance to applicants and reviewers. A flow diagram for the process is shown in Figure 2.

The EM development process is relatively straight-forward however, in the case of non-LWRs there are two complications that both applicants and the NRC will need to address. One of the first steps is identification of the event scenarios that the EM will be applicable to. First, the scenario should be defined at the initiation of code development. Central to code development is a clear understanding of the physical phenomena that must be simulated by the codes within the EM. A Phenomena Identification and Ranking Table (PIRT) is therefore important step at the initiation of an EM effort. For LWRs the accident scenarios are well-known and defined as "Chapter 15" and "Chapter 19" events for DBE and BDBE respectively. However, for non-LWRs following the Licensing Modernization Project approach, PRA insights are necessary and expected.

Event scenarios for non-LWRs will differ from traditional "Chapter 15" and "Chapter 19" events due to their unique characteristics and the expected large safety margins. The new designs are likely to be able to withstand multiple failures and/or require fewer safety significant components (SSCs) in order to mitigate an accident. The analysis capability being defined in this paper is intended to simulate events in non-LWRs up to those conditions that lead to core disruption and release of fission products. While a scenario involving multiple failures is traditionally considered a beyond design basis event, for non-LWRs they may be acceptable as part of the design basis and will be analyzed by codes proposed in this paper.

The event scenarios that the codes for confirmatory analysis will be used for will include, but not be limited to, are the following hypothetical events in gas-cooled, liquid metal cooled, molten salt, and "micro" reactors:

Gas-Cooled Reactors

- pressurized loss of force cooling (P-LOFC) accident
- de-pressurized loss of force cooling (D-LOFC) accident
- reactivity-induced transients, including ATWS events

(Events that involve air-ingress and significant oxidation of the graphite, water-ingress, transport and release of "graphite dust" will be simulated with a traditional severe accident code.)

Liquid Metal Reactors

- loss of coolant with and without scram
- loss of forced flow
- unprotected loss of flow
- unprotected loss of heat sink
- reactivity-induced transients, including ATWS events

Molten Salt Reactors

- loss of forced flow
- unprotected loss of flow
- inadvertent reactivity insertion transients, including ATWS events
- loss of coolant
- over-cooling events (leading to partial solidification)
- station blackout
- loss of heat sink

Heat Pipe Cooled "Micro" Reactors

- loss of heat sink
- inadvertent reactivity insertion transients, including ATWS events
- localized heat pipe failure
- cascading loss of heat pipes
- seismic event (causing reactivity increase)
- events related to coupling the reactor to the power conversion unit
- monolith temperature and stress under normal operating conditions
- monolith temperature and stress under postulated accident conditions.

(The "monolith" is the structural component that supports the fuel and control rods.)

In order to identify codes for development and validation, the Phenomena Identification and Ranking Table (PIRT) represents a critical step in the process. As described in Regulatory Guide 1.203 [5] the

PIRT should be used to determine the requirements for physical model development, scalability, validation, and sensitivities studies. Ultimately, the PIRT is used to guide any uncertainty analysis or in the assessment of overall EM adequacy. The PIRT is not an end in itself; rather it is a tool to provide guidance for the subsequent steps. Because of this central role in code development, PIRTs applicable to non-LWRs are discussed in the next section.

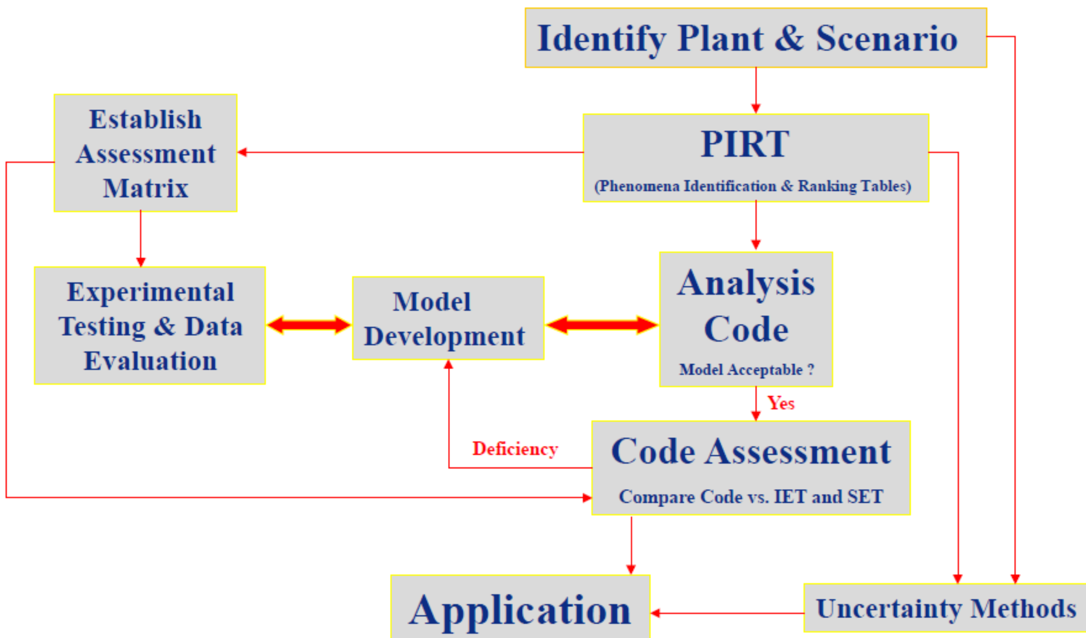

Figure 2. NRC Code Development Process.

4. Non-LWR Dominant Physical Phenomena

Table 1 provides a categorization of non-LWR designs as a means of identifying the Evaluation Models that will need to be developed for confirmatory analysis. While ten design types are listed, it is convenient for the purposes of discussing important phenomena to consider four overall design types; gas-cooled, liquid metal cooled, molten salt, and "micro" reactors cooled by heat pipes. Phenomena important to each design have been identified for each of these design type, and in some cases applicable PIRTs are available. This section discusses the references used by the NRC to help establish code requirements for non-LWRs.

Phenomena for gas-cooled reactors, both prismatic and pebble bed, were the subject of a comprehensive effort related to the Next Generation Nuclear Plant (NGNP). Expert panels considered five interrelated subject areas including thermal-fluids, neutronics and accident analysis, high temperature materials, nuclear grade graphite, process heat and cogeneration, and fission product transport [6]. With regards to design basis event analysis, phenomena and processes of major concern due to their importance in analysis and relatively low knowledge level included:

- core coolant bypass flows,
- power/flux profiles,
- outlet plenum flows,
- reactivity-temperature feedback coefficients,
- emissivity for the vessel and reactor cavity cooling system,
- reactor vessel cavity air circulation and heat transfer, and
- convection/radiation heating of upper vessel.

A separate PIRT was also developed for TRISO fuel [7], which discussed mechanisms and phenomena for fission product release and the evaluation of fuel performance. This report is relevant to design basis event analysis due to its identification of fission product release mechanisms and thus helps establish figures-of-merit that segregate design basis and beyond design basis analysis.

For sodium fast reactors, there has been significant efforts made to understand the phenomena and processes of importance to modeling and simulation of various accident scenarios. While not specifically a PIRT for design basis events, a recent study into technology "gaps" was completed by a panel of experts [8]. The objective of the study was to evaluate the status of knowledge for accident analysis of sodium fast reactors (SFRs), and the panel considered a broad range of phenomena expected to be important in licensing SFR designs. Because of a fairly large quantity of experimental information, the panel was able to conclude that the knowledge level for SFRs was good and that there are no major technology gaps in preparing a safety case for an advanced SFR, so long as one stays with known technology. Potential gaps were acknowledged for advanced simulation of coupled neutronic/fluid flow dynamics, supercritical CO2 power conversion, and high minor-actinide content fuel.

The study identified several phenomena important to DBE analysis of SFRs, including:

- single phase transient sodium flow
- thermal inertia
- pump-coast down
- sodium stratification
- transition to natural convection core cooling core flow
- decay heat generation
- reactivity due to mechanical changes in core structure
- reactivity feedback at high power

The NRC sponsored an additional review that considered sodium fast reactors, in addition to lead- and lead-bismuth cooled systems [9]. The findings in [8] were confirmed, and largely extended to other liquid metal designs, and developed a comprehensive list of phenomena and processes that must be modeled in analysis of a liquid metal reactor.

While important phenomena and "gaps" in the technologies are well established for gas-cooled and liquid metal reactors, molten salt reactors represent a new and unique challenge. PIRTs applicable to these designs have however recently been developed. For a molten coolant salt design, a PIRT was developed considering two events; station blackout and a simultaneous withdrawal of control rods [10]. An accompanying study [11] considered phenomena involved in the molten salt reactor design and the need for a tight coupling between some phenomena. The two studies [10, 11] discussed physical phenomena and "gaps" in knowledge for the particular design which utilized a relatively well known fluoride salt and TRISO fuel in the form of plates.   Phenomena of importance which were recognized as having "gaps" in the knowledge base included:

- thermophysical properties of coolant salt (conductivity and viscosity)
- wall friction in the core
- core flow asymmetry
- upper and lower plenum mixing
- safety system component performance
- chimney natural circulation and performance

These studies were considered by the NRC, which then sponsored an expert panel to consider molten fuel salt reactors. This panel investigated and developed PIRTs for both fluoride (thermal spectrum) and chloride (fast spectrum) fuel salt reactors [12-14].   This study that in addition to the phenomena in a coolant salt, there are two main challenges are introduced with liquid fuel: delayed neutron precursor motion, and strong coupling to salt composition. Hence, it is necessary to take into account the movement of delayed neutron precursors into and out of the core, and the transit times of the fuel, fission products, and transmutation and chemistry products through the primary system.   For a fuel salt reactor design the phenomena of importance included:

- delayed neutron precursor motion
- salt chemical composition
- neutron absorption in fuel salt
- physical properties
- convective heat transfer
- primary system flow resistances
- structural material performance (swelling and expansion)
- tritium production and transport (fluoride salts)

A major recommendation of this study [12] is that modeling and simulation of MSRs will require development of computational tool(s) capable of tracking chemical inventories of constituents throughout the primary loop of the reactor facility. A comprehensive evaluation of fuel salt composition will likely require the modeling of salt chemistry, thermodynamics, mass transport, and addition and removal of chemical species.

Finally, for heat pipe cooled "micro" reactors, four individual PIRTs have been developed for a special purpose 5 MWt design [15]. The PIRTs are for reactor accident and normal operations, heat pipes, (3) materials, and power conversion.   The PIRTs expressed concerns with:

- monolith thermal stress
- single heat pipe failure
- machining and inspection of the monolith
- heat pipe performance
- reactivity and core criticality

These PIRTs will likely need to be reconsidered as each of these reactor designs mature and new experimental information is made available.

References 6-15 provide a preliminary basis for modeling and simulation requirements for a non-LWR code suite. These PIRTs will likely need to be reconsidered as each of these reactor designs mature and new experimental information is made available.

This section has listed a set of phenomena and “gaps” that are likely to require efforts to resolve in the development of a confirmatory analysis capability. A comprehensive review of these entire PIRTs are beyond the scope of this paper, yet there are some findings that are of particular importance with respect to code requirements and future develop for non-LWRs. Phenomena that are significant and “new” and with increased importance for non-LWRs relative to conventional LWRs include but are not limited to:

- Thermal stratification and thermal striping
- Thermo-mechanical expansion and effect on reactivity
- Large neutron mean-free path length in fast reactors
- Transport of neutron pre-cursors (in fuel salt MSRs)
- Solidification and plate-out (MSRs)

Codes that simulate non-LWRs need to account for these phenomena as well as many others that are important to design basis events. Development and assessment of these tools will certainly be necessary, but the basic code requirements defined in these PIRTs should be satisfied with the code suite the NRC is developing for confirmatory analysis.

5. DBE Code Selection Criteria

Several other criteria factor into the selection and development of NRC confirmatory codes for non-LWR analysis. First and foremost, the codes must either possess the capability for simulation of non-LWR physical phenomena or be amenable to development and enable the necessary models and features as identified in the available PIRTs. Additional criteria involve the “multi-physics” nature of non-LWRs, computational requirements, regulatory independence, and the impacts on staff.

One of the factors identified as necessary for advanced non-LWRs [11] is that it will be necessary to have a tight coupling between several analysis codes because of the feedback between physical phenomena. Fast reactors for example, require a tight coupling between neutronics, thermal-hydraulics and structural changes to account for rapid changes in power due to reactivity feedback that can occur with changes in temperature. Thermo-mechanical expansion of the core is an important phenomenon that must be accounted for because of its impact on neutron leakage. In some designs this leakage can be the largest source of negative reactivity as a core heats up.

In general, for design basis type events three types of codes are necessary; fuel performance, neutronics, and system thermal-hydraulics. Peripheral codes for cross-section determination and computational fluid dynamics may be needed as part of what might constitute an Evaluation Model for analysis of some designs. Therefore, because of the need to provide feedback between the phenomena, a computing environment in which codes can be efficiently coupled together is needed. Coupling codes can also reduce development costs by making use of analysis capabilities that are within the set of coupled codes, but not specifically within a given code. The "down-side" of coupling is that time-step and control can be limited by the slowest or least computationally efficient code in the set. Thus, in the evaluation of the codes an examination of computational efficiency is necessary.

The expected review schedule is highly uncertain. Several potential applicants have indicated that they intend to make submittals for NRC review as early as (late) 2019 and some have suggested commercial operation before 2025. In order to be ready for confirmatory analysis with an aggressive licensing schedule, it is important that the codes selected for design basis event analysis be essentially complete so that code validation can begin as early as 2019. Especially for these expected early applicants, the codes need to be "out of the box" ready.

Computation resources are an important consideration, as well. The NRC (and possibly some applicants) currently has limited access to high-end computational platforms consisting of thousands of CPUs. If new codes, or nature of non-LWR analysis requires exceptionally high geometric or temporal resolution then it may be necessary to improve the capability of existing platforms or gain access to high end resources. To avoid the complication of requiring high performance computing systems, there is a clear advantage if a code can execute quickly on the desktop systems or if the computational models can be simplified so that they can execute efficiently on relatively small systems.

Training and expertise is a concern, as new codes represent a challenge to the efficient and effective use of staff. Non-LWR technologies are diverse, and are not as well known to NRC staff as are those for conventional light-water reactors. Thus, there is a learning curve" with each technology as well as another "learning curve" for each code that the staff is not familiar with.

A final consideration is that of regulatory independence. Confirmatory calculations made by the staff are intended to enable the staff to perform its review. By independently modeling and simulating an applicant's design, the staff gains the expertise and knowledge to fully understand the design and when necessary request additional information from the applicant to justify the safety case. Comparing NRC confirmatory calculations to those of an applicant often show considerably different behavior, and allow the staff to question the reason for those differences and if they are safety significant. For non-LWRs, there will be a shortage of experimental information relative to LWRs. A concern, if both the NRC and applicants use the same code(s) is that both code(s) might be "tuned" to the same set of information. It may be difficult to claim independence in such a case.

6. The Comprehensive Reactor Analysis Bundle

The proposed code suite for non-LWR confirmatory analysis makes use of existing NRC codes, and integrates them with several codes developed through the DOE NEAMS program. The codes in some cases have multiple applications. That is, the intent is that they be used for more than one design type. There is some redundancy built into the proposed structure to allow for options in order to meet unanticipated pre-applicant issues.

Figure 3 presents a schematic showing the full suite of non-LWR codes, known as the Comprehensive Reactor Analysis Bundle (CRAB). In Figure 3 the NRC developed codes are shown in gold, while those produced by the DOE are shown in light blue. For each reactor design type, only a subset would be active as part of a given Evaluation Model.

Codes that are expected to play a role are the NRC developed or sponsored codes TRACE [16], FAST [17], PARCS [18] and its associated code for cross sections SCALE [19]. Codes developed by DOE such as MOOSE [20], BISON [21], PRONGHORN [22], SAM [23], and MAMMOTH [24] are expected to be utilized extensively. The expectation is that CFD analysis be a necessary component of analysis and would be done using a commercially available code such as FLEUNT [25] or possibly the DOE code Nek5000 [26]. SERPENT [27] is a reactor physics code developed at the VTT Technical Research Centre of Finland, capable of calculating cross-sections and performing detailed Monte Carlo simulations.

Three systems level thermal-fluids codes are used: TRACE, PRONGHORN, and SAM. The TRAC and RELAP Advanced Computational Engine (TRACE) is a systems thermal-hydraulic code developed by the NRC and is well-validated for transient two-phase flow. TRACE has the capability to simulate fluid flow and heat exchange in a thermal system and has the ability to model a variety of working fluids including helium, sodium and some molten salts. TRACE will most likely be used for simulation of secondary and safety systems where water is the working fluid. PRONGHORN is a MOOSE-based multi-physics reactor analysis application developed at Idaho National Laboratory to model gas-cooled reactors. The System Analysis Module (SAM) is a MOOSE-based system analysis tool being developed at Argonne National Laboratory for advanced non-LWR safety analysis. It can provide fast-running, whole-plant transient analyses capability for sodium fast reactors and other liquid metal design. Both PRONGHORN and SAM can also simulate some molten salt coolants.

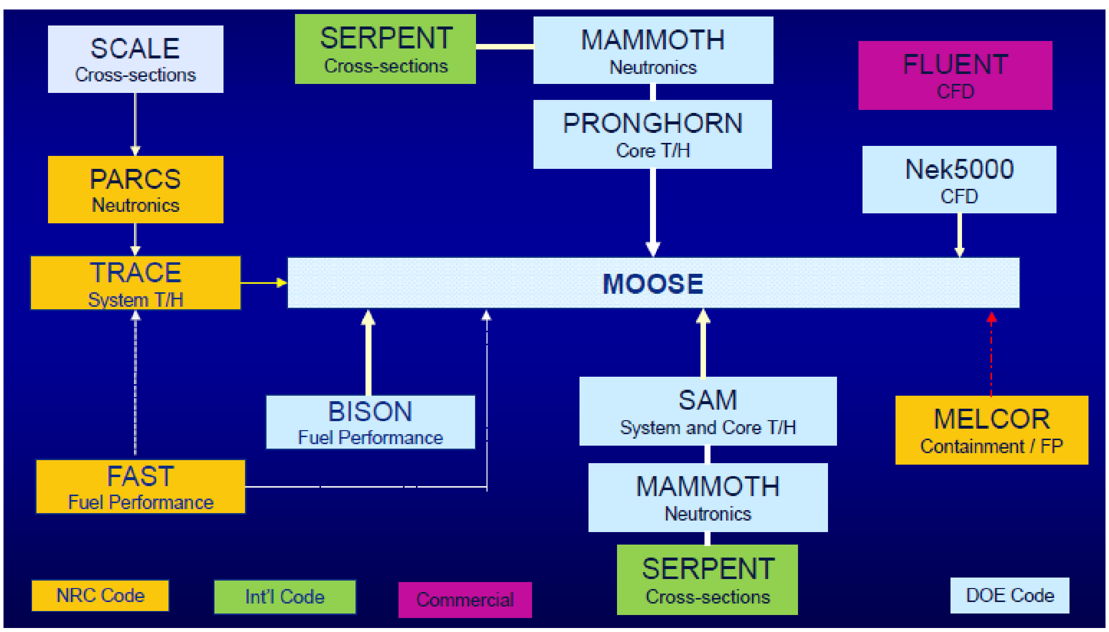

Figure 3. The Comprehensive Reactor Analysis Bundle (CRAB) for Analysis of Design Basis Events in non-LWRs.

Two fuel performance codes are integrated into CRAB: BISON and FAST. While both codes offer considerable capability, BISON will likely be the tool of choice for TRISO fuel while FAST is used for conventional oxide fuel. For metallic fuels, either or both may be applied depending on the success of validation efforts and the need to understand uncertainties in simulation results.

Reactor kinetics will be covered by MAMMOTH and SERPENT, or by PARCS and SCALE. The NRC's traditional approach using PARCS and SCALE will be used for designs where diffusion theory is sufficiently accurate, however for most non-LWR designs MAMMOTH and SERPENT will be applied when transport theory is needed to obtain accurate power distributions.

Part of the rationale for this suite of codes involves the 'multi-physics" nature of non-LWR analysis which requires that the codes operate in a coupled computational environment. Coupling allows for a rapid feedback if one physical package affects another. The need for a tight coupling between codes is probably most apparent for molten fuel salt reactors where the reactor kinetics depends on the local temperature for reactivity feedback and the flow field for tracking delayed neutrons. Other reactor designs (gas-cooled and liquid-metal) may not require as tight a temporal coupling however because of the much longer mean free neutron path-length a tight coupling exists in the core-wide power and temperature distributions.

As alluded to previously, thermal-mechanical effects can be important in some non-LWR simulations. If the core expands, changes in the core geometry will affect neutron leakage and the reactivity. The expansion however depends on the temperature distribution in the core and support structures. In some designs, in particular the "micro" reactors cooled by heat pipes, the support structure temperature depends on thermal conduction from the fuel element.

Both code coupling and evaluation of thermal-mechanical evaluation can be accomplished using MOOSE (Multiphysics Object Oriented Simulation Environment). MOOSE is a finite element based computational platform that enables a user to model and simulate a wide range of problems involving heat transfer and solid mechanics. The finite element approach allows for an unstructured mesh so that any geometric form can be modeled. This formulation is significant in that it allows MOOSE to model geometries than cannot be accurately modeled with conventional NRC codes. Some of the non-LWR applications where MOOSE is necessary include the annular fuel in some heat-pipe cooled designs and the structures supporting the fuel elements and heat pipes. Supporting structures in liquid-metal fast reactor designs can also be modeled with MOOSE to provide the radial expansion in the affected fuel assemblies.

MOOSE further has been used as a platform to couple other analysis codes. The NRC's TRACE code has been coupled into the MOOSE framework and verification is continuing. This now allows the use of DOE codes to simulate the non-LWR primary system while TRACE is used for secondary and tertiary systems as well as Reactor Cavity Cooling Systems, some of which involve a Rankine cycle and two-phase water.

7. Severe Accident and Consequence Analysis

The main objective of this paper is to outline the NRC's approach to event scenarios that may be part of the design basis for a non-LWR. These events are those which lead to little damage to the core and fission products are retained in the fuel and core. The NRC is also proposing code development efforts to extend NRC's traditional modeling and simulation capabilities for severe accident progression, source term, and consequence analysis for non-LWR technologies [28]. For non-LWR confirmatory analysis the NRC intends to use MELCOR [29] for accident progression and source term analysis, MACCS [30] for consequence analysis, and SCALE [19] for radionuclide inventories.

MELCOR is the state-of-the-art computer code developed by Sandia National Laboratories for NRC to perform nuclear reactor severe accident progression and source term analyses. MELCOR is a flexible, integrated computer code designed to characterize and track the evolution of severe accidents, and the transport of associated radionuclides within a confinement such as a containment or building. MACCS (MELCOR Accident Consequence Code System) code suite is the NRC's computer code system for probabilistic consequence analysis. MACCS models atmospheric releases of radioactive materials into the environment and the subsequent consequences of such releases. MACCS is the only tool for probabilistic modeling of all the technical elements of the Level 3 PRA Standard including radionuclide release, atmospheric transport and dispersion, meteorology, protective actions and site data, dosimetry, health effects, economic factors, and uncertainty.

The results of MELCOR source term analysis and MACCS insights would in turn be used by radiation protection and health physics codes RASCAL [31, 32] and RADTRAD [33, 34].

8. Summary and Conclusions

The NRC is rapidly expanding its capabilities to understand and perform confirmatory analyses for advanced non-LWRs. Since development of the 2016 "Vision and Strategy" document, the NRC is addressing needs in the licensing process, applicable (industrial) codes and standards, and modeling and simulation capabilities. The strategy for analysis codes is one that will use both NRC conventional codes along with several of the advanced tools being developed by DOE. This approach is expected to provide an analysis scheme that is technically superior to what could be achieved with the NRC codes alone, and helps ensure readiness to support a design review. In Figure 1, the application of conventional NRC codes to General Design Criteria was shown. Figure 4 shows this with the expectation of confirmatory analysis for non-LWRs. With development of the codes, validation, and eventual plant model generation, these codes are expected to play an important role in the NRC review process.

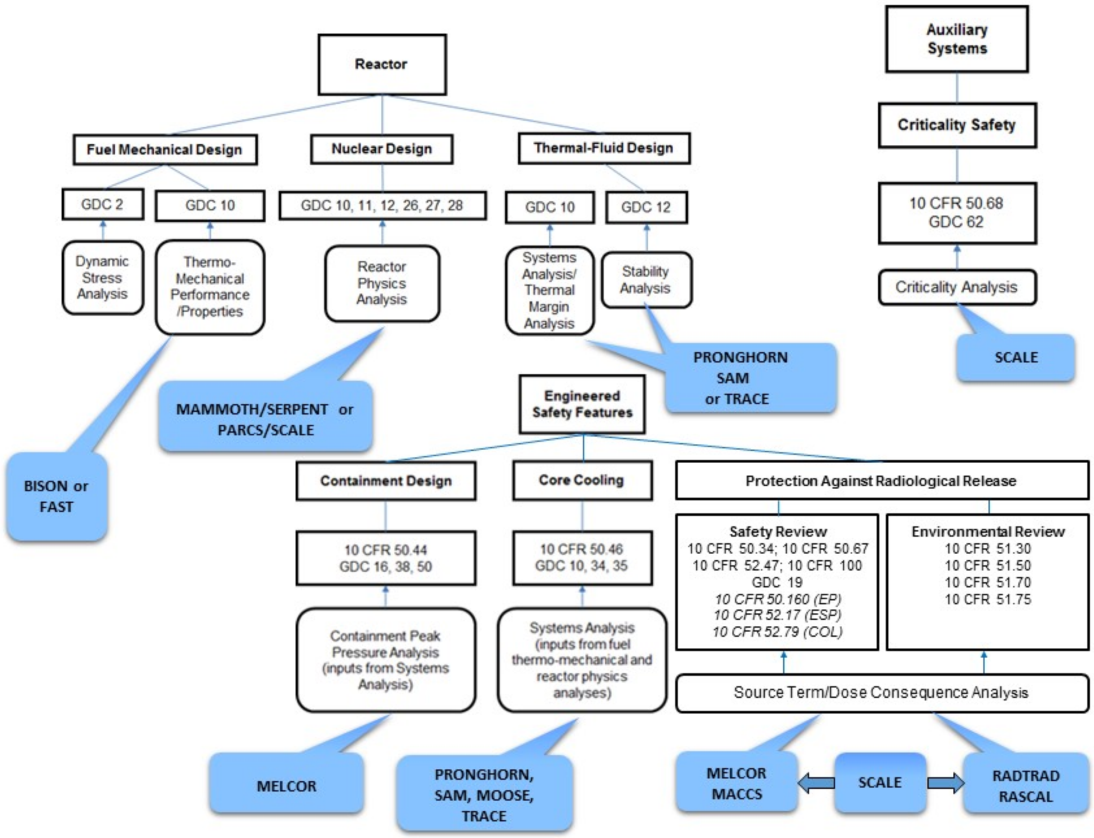

Figure 4. Confirmatory Analysis and General Design Criteria for Advanced non-LWRs.

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