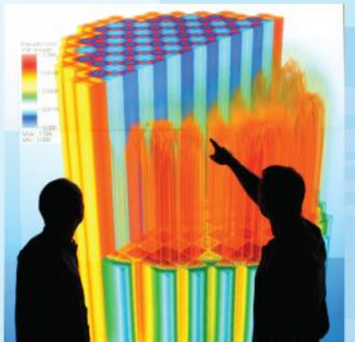
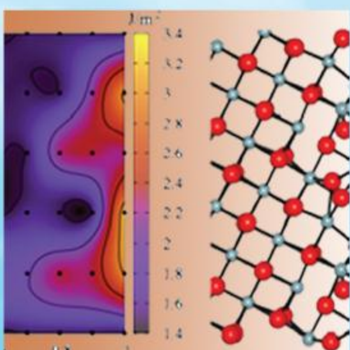
CASL-U-2011-0236-002

Power uprates and plant life extension

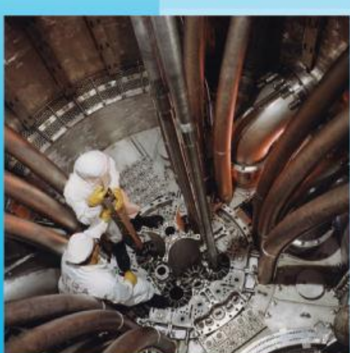


Engineering design and analysis

Science-enabling high performance computing



Fundamental science



Plant operational data

CASL
Consortium for Advanced Simulation of LWRs

IC Workflow Project: Final Report

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REVISION LOG

Revision	Date	Affected Pages	Revision Description
0	12/31/2011	All	Original Draft Report
1	1/25/2012	All	Minor Revision
2	3/30/2012	All	Include data from Phase 2

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CONTENTS

ACRONYMS

AMA	CASL Advanced Modeling Applications Focus Area
AO	Axial Offset
AOA	Axial Offset Anomaly
AREVA	AREVA NP, Inc.
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CASL	Consortium for Advanced Simulation of Light Water Reactors
CFD	Computational Fluid Dynamics
CILC	CRUD Induced Local Corrosion
CIPS	Crud Induced Power Shift
CPR	Critical Power Ratio (BWRs)
CRUD	Corrosion Related Unidentified Deposits (also, Chalk River ….)
DNB	Departure from Nucleate Boiling
Dominion	Dominion Virginia Power
Duke	Duke Energy Corp.
EOC	End of Cycle
EPRI	Electric Power Research Institute
FAD	Fuel Assembly Distortion
FSI	Fluid-Structure Interaction
GNF	Global Nuclear Fuels
GSE	GSE Systems, Inc.
GUI	Graphical User Interface
IC	CASL Industry Council
LWR	Light Water Reactor
M&S	Modeling and Simulation
NPP	Nuclear Power Plant
ORNL	Oak Ridge National Laboratory
PWR	Pressurized Water Reactor
R&D	Research and Development
RCS	Reactor Coolant System
Rolls-Royce	Rolls-Royce Group plc
SMR	Small Modular Reactor
TBD	To Be Determined
T/H	Thermal-Hydraulics
TVA	Tennessee Valley Authority
UQ	Uncertainty Quantification
UT	Ultrasonic Fuel Cleaning
VERA	Virtual Environment for Reactor Applications
VRI	CASL Virtual Reactor Integration Focus Area
Westinghouse	Westinghouse Electric Company

INTRODUCTION

The Workflow Project is a joint research task between the CASL Advanced Modeling Applications (AMA) and the Virtual Reactor Integration (VRI) focus areas, and the Industry Council. The purpose of the project is to solicit the expertise of the IC member organizations to inform CASL staff and provide feedback for the development of CASL products. The goal is for CASL staff to understand existing workflows at end-user organizations in order to achieve compatibility of CASL products with current needs and future plans. CASL is committed to developing and delivering advanced capabilities and solutions that will allow the nuclear power industry to solve complex problems and improve the operating performance and efficiency of light water reactors. Through the R&D being performed by CASL, advanced computational methods and software is being developed and embodied in the Virtual Environment for Reactor Applications (VERA). The information collected through this project will provide additional considerations that must be included in the development of VERA to ensure that it meets the needs of the nuclear industry. As will be discussed below, key additional areas that must be considered is the usability of the system in an engineering environment including the user input, quality assurance, output of results, computational runtime, and integration into existing engineering analysis processes. This document contains a summary of information gathered through a series of interviews and meetings with IC organizations.

In addition to developing information on workflow to support VERA development, this project also provides an opportunity to collect input on the definition of a “pilot projects” that can be early demonstrations of CASL-developed capabilities to problems of interest to industry.

OBJECTIVES

CASL has brought together experts in many research and development areas for development of its M&S capabilities. The AMA staff provides a critical link to the nuclear industry and provides a broad, but not comprehensive, level of knowledge of industry practices and needs. The VRI staff provides state-of-the-art software capabilities to deliver advanced interfaces, data structures, and software integration for an end user with whom they are not necessarily familiar. Therefore, the staff desires a much better understanding of the various ways the product might be used, the types of results that will be most useful, the types of resources that will be available, and the types of user interfaces that will be the most effective for industry engineers and analysts.

Therefore, there are three high level objectives for the Workflow Project, outlined below.

1. **Obtain input from IC members** on the potential uses of VERA and potential methods of incorporating it into industrial analyses and processes for a variety of analysis types.

2. **Highlight important activities, capabilities, and insights** that IC members feel are vital to ensuring CASL will successfully provide a tangible benefit to the nuclear power industry.

3. **Provide detailed workflow of important activities** to AMA & VRI for education and development purposes.

Is should be noted that the content of this report is geared towards PWR analysis methodology and CASL’s challenge areas of CIPS, CILC, and GTRF. While the physics involved in these areas exists in any LWR, the applicability to BWRs is limited. As the CASL project progresses to more areas outside of PWRs, further insight will be sought from BWR vendors and utilities, as well as operators of other potential reactor types.

PARTICIPANTS

In order to meet the Workflow Project objectives, many members of the IC volunteered to participate in the information exchange with CASL. The project has been divided into two phases, with Phase 1 mostly consisting of U.S. LWR utilities and fuel vendors. These interviews focused on fuel design, core reload, and performance issue analyses. This was done first specifically to focus on the CASL challenge problems and early pilot applications of VERA releases. Phase 2 discussions addressed gaps in the workflow project results from the perspective of other IC members, sometimes with a different focus than Phase 1 participants. The primary project participants, by phase, are provided in Table 1 below.

Table 1 Project Participants

<table>
<tr><th></th><th>IC Participants (IC Representative)</th><th>CASL Participants</th></tr>
<tr><td>Phase 1</td><td>Duke Energy (Scott Thomas)
Westinghouse (Sumit Ray)
Dominion (John Harrell)
AREVA (Chris Lewis)
GNF (Russell Stachowski)</td><td rowspan="2">John Gaertner, EPRI, IC Chairman
Andrew Godfrey, ORNL, AMA
Scott Palmtag, Core Physics, Inc., AMA/VRI
Jess Gehin, ORNL, AMA Lead</td></tr>
<tr><td>Phase 2</td><td>TVA (Dan Stout, Rose Montgomery)
GSE (Zen Wang, Steven Freel)
Rolls-Royce (Alan Copestake)</td></tr>
</table>

Each Phase 1 meeting occurred at the IC member organization (here forth called the “member”) locations and was attended by numerous persons in each organization, including such experts in core reload design, neutronics, thermal-hydraulics, fuel mechanical performance, new fuel design, safety analysis, and licensing. Each meeting was approximately three hours in length. The Phase 1 meetings occurred over a one month period from mid-October to mid-November, 2011. Questionnaires and notes were collected and used to develop an overall general summary of the information obtained in these meetings without specific reference to any individual or organization.

The Phase 2 meetings were performed during February 2012 via teleconference or email exchange. A summary of the findings contained in this report was presented to the IC at the meeting on March 7th, 2012, when all IC members were given an opportunity to obtain the report and provide additional feedback. Therefore, this final report contains all the data obtained from each phase of the project and following the initial presentation of the topic to all IC members.

QUESTIONAIRE

Each Phase 1 meeting was approached as an informal interview with the IC member. The format of the meeting was to allow the IC member organization to provide an overview of their processes for performing an analysis of their choosing. A list of questions was provided and used as a discussion guide, but strict adherence was not enforced. Rather, an open dialogue and information exchange was encouraged, and this typically addressed the needed information in addition to details that were not in the questionnaire. The questions used as reference are provided below.

1. Describe the workflow for the current analysis.
2. Is the analysis primarily safety related? Operation? Business/Financial?
3. Does the analysis include iterations, sensitivity studies, determination of uncertainties?
4. What types of QA or validation is required?
5. What computer codes are used? On what platform(s)?
6. What computer resources are available and/or planned?
7. Who/what interfaces with the codes? How are the codes coupled and/or data transferred?
8. What type of output or visualization is required? Desired?
9. How modular is the processes? What are the needs
10. How much “wall time” (total time duration) is typically allotted to perform the analysis?
11. How much staff time is needed? How much computer time is needed?
12. Is there a time bottleneck in the current process?
13. What issues do you have with the current analysis tools and workflows?
14. How could the workflow for this analysis be improved with higher fidelity methods? Tighter couplings?
15. What additional capabilities or features would be most useful?
16. What are the training requirements for obtaining qualification to perform the analysis?

The Phase 2 meetings did not attempt to adhere to the questionnaire but rather addressed comments and gaps in the original Phase 1 report.

CODES AND METHODS

The M&S tools utilized for analyses are typically general purpose, engineering-grade tools employing tried-and-true single physics methodologies. They are typically single processor applications developed in FORTRAN or C languages. Input and output are predominantly in a simple ASCII format. These methods and tools have been developed, validated, and in some cases licensed for application independently from other codes (i.e. with little coupling). Over time, each code has evolved independently, dependant on resource availability and technical needs, resulting in a somewhat difficult environment for implementing enhancements and physics couplings. Furthermore, tools that are qualified and licensed for safety-related applications that undergo little, if any, changes would require significant effort to relicense. There are a few exceptions to this description, such as the commercial CFD codes, which have only been used in non-licensed, very specific applications.

There are many obvious advantages to the nature of the current tools. Foremost, they are very fast and require little computer resources by modern day computing standards. They are developed in languages with solid standards that remain unchanged and/or retain backwards compatibility over time and across platforms. Because they are typically general purpose, they can be employed in many different analyses in a modular fashion by merely implementing utility codes to interpret the output of one code for the input of the next. In this way, one single code can be applied for many different types of analyses. Finally, the workflows using these codes can be fairly easily automated to reduce human error using scripting languages. In some instances, the automation can create the input for the first code, execute it, post-process the output for input into the second code, and so on. And though these codes are fundamentally not written for parallel computing, the analysis environment typically involves dozens to thousands of independent executions that can be done on parallel processors, effectively accomplishing the same wall time goal. These codes have been developed for use in engineering analyses such that they are easily used and require minimal setup time, have robust solution algorithms, and provide results in formats that are easily interpreted. These methods and codes have been used to achieve an excellent record of plant operation.

The drawbacks to the current M&S tools employed by the industry are also fairly obvious, as these are part of the drivers of the CASL program. The codes typically perform lower-order physics due to being designed for much less computer resources than are available today. These methods are sometimes decades old and often require substantial approximations that must be addressed by uncertainties/safety factors and biases. Furthermore, the value of reduction of these uncertainties has typically remained undetermined even though higher-fidelity solutions are possible. The coupling between codes, in terms of communication and iterative execution, is very limited, and information and margin may be lost at each step in the process because of simplifying assumptions used for information exchange. In terms of exploration, the nature of the programming and output format limit the ability to augment the program with detailed graphical results or interactive execution. The codes are designed and qualified to predict coarse reactor quantities for nominal conditions, but lack the fidelity or flexibility to simulate off-nominal, complex, or small scale local phenomena. Finally, the application methodology of these codes is to unrealistically propagate error allowances/safety factors from one discipline to the next, rather than a more realistic, integrated best estimate approach.

Core Simulator

Perhaps the most heavily relied upon software product is the reactor core simulator (not to be confused with a real-time plant simulator used for training operators). This is a general purpose engineering tool for best-estimate predictions of reactivity, power distribution, fluence, instrument response, etc., reflecting the basic physical aspects of the nuclear reactor core. For most analyses, this tool either directly provides the quantities needed for the analysis, or it provides the problem boundary conditions. Due to fuel management strategies used in nuclear power plants, detailed analyses typically require the simulation of 3-4 entire fuel cycles prior to performing the target calculations. This is needed to obtain a good estimate of quantities such as fuel exposure (isotopic), cladding stress, accumulated fluence, etc. The core simulator determines and permits archiving and restart of the operation history of the reactor core.

Industry core simulators employ nodal neutronics methods, which have been in use for decades and have proven to be very accurate and reliable for coarse quantities during steady-state operation. The neutronics method is performed in two steps. First, two-dimensional infinite lattice physics calculations are performed in many neutron energy groups for each fuel type in each unique fuel assembly for average and perturbed reactor conditions. The results of these calculations are parameterized into tables of few-group (usually two or three neutron energies) homogenized macroscopic neutron cross sections and stored in a library file. This library is typically built in advance of performing any analyses such that this step need only be performed once. The second piece of the neutronics is the three-dimensional core calculation based on the parameterized cross sections applied in a few-group nodal diffusion-based calculation of the neutron flux. This is typically performed using homogenized six inch nodes, and rod-level power and exposure is reconstructed from the infinite lattice-based cross sections.

Because the majority of the nuclear cross section treatments and isotopic depletion calculations are performed in the lattice physics codes, and because the full core portion of the calculation is a coarse mesh homogenized solution, these methods can execute a single full core statepoint in a few seconds (2-10) on a single processor. A complete parameterization of the lattice cross sections may take 10 minutes, but this is done only once per fuel type, and prior to performing any engineering analyses. The two main disadvantages of this methodology is that local quantities are not directly computed in the full-core calculation (2D pin power reconstruction is used) and the spectral differences which can accumulate over time between the 3D core analysis and the pre-generated cross sections. For many situations, these issues result in relatively small errors, however they cannot be used to provide more detailed information (such as sub-pin power distributions) and accommodate detailed spectral changes within and between fuel assemblies.

The core simulator software also provides thermal-hydraulic (T/H) feedback at the nodal (or sub-assembly) level. This is required to accurately calculate the neutron flux and power distribution for the neutronics as the reactor operates and depletes over time. Though this can be a low-order solution to the T/H conditions (particularly for PWRs), it is completely coupled to the neutronics solution for accurate feedback and an iterative converged solution.

Other very important features of the core simulator are its concise, geometry-based input and output. This allows the user to easily perform a variety of analysis types and executions, such as critical power search, control rod movements, power maneuvers, accident analyses, etc. The application range of this tool is nearly endless, yet the accuracy at the ends of this range must be clearly understood, and the validation basis of the tool is firmly established prior to its implementation.

The core simulator tools are typically validated versus a variety of industry standard problems, such as critical experiments, code comparisons, and actual reactor operating data. They are safety-related and licensed by the regulator for a particular application. During this process, the uncertainty in reactivity and power distribution are calculated for thousands of data points, and this uncertainty is applied for subsequent analyses. There is no built-in uncertainty quantification or propagation.

Figure 1 contains a depiction of the core simulation components and how it is integrated into the CIPS analysis workflow. The core simulator is shown on the left as the tool which generated the assembly powers for the analysis.

Sub-Channel Thermal-Hydraulics

Typically separate from the core simulator is the detailed thermal-hydraulics simulation tool. This is typically used for analysis of limiting fluid conditions, usually at a rod or sub-assembly level, for prediction of DNB or CPR (dryout). Setup of these models can be cumbersome and time consuming, especially for large geometries such as full reactor cores. For this reason, this process is typically automated such that data can be moved from the core simulator output to the sub-channel code quickly and error free. Like the core simulator, this code is well developed, validated, and documented, and executes on a single processor very quickly (minutes for a full core coarse mesh solution with ~100 axial levels). Also, this tool can be setup and applied to a variety of analyses and with varying levels of fidelity, depending on the needs of the analysis.

A few industry organizations have been coupling coarse-mesh sub-channel T/H with core neutronics for decades. This provides important benefits such as sub-cooled boiling, two-phase flow, and cross flow (between sub-assemblies). However, given the current nodal neutronics methods, it is somewhat inconsistent to further increase the level of sub-channel fidelity without a corresponding increase in the neutronics (i.e. the sub-channel code cannot be fully coupled to the neutronics due to the two-dimensional parameterization of the cross sections). In addition, especially for PWRs, most in the industry do not perceive this tight coupling as required or particularly useful for current analyses, though they recognize the benefits (LOCA, for instance) if there were a sufficient technical or economical driver to implement. In general, for steady-state PWRs, a manual one-way coupling is sufficient and results in only small inconsistencies between core and sub-channel T/H results. This may not be the case for BWRs or SMRs.

CRUD and Chemistry

Sub-channel fluid conditions are provided to the CRUD/Chemistry code along with primary system sources, chemistry information, CRUD release factors, etc. Industry PWR organizations use the standard EPRI BOA methodology for risk reduction of CIPS and CILC. BOA calculates the CRUD thickness and boron mass deposited on each fuel assembly throughout the cycle. BOA is a modern code and provides multiple interfaces (ASCII and GUI). It is both a code and analysis methodology, and the GUI assists the user in proper application of the tool.

BOA calculates CRUD deposition over an entire fuel cycle (for 15-20 timesteps) in about 15 minutes. The methodology does not directly predict CIPS or CILC, but determines a relative level of risk to these phenomena as compared to a previous reference or benchmark cycle. It is used specifically for calculation of CRUD deposition, but is also being used for other analyses of primary system chemistry changes.

The EPRI BOA methodology is well established and been in practice for nearly a decade for many organizations. BOA has been validated against hundreds of fuel cycles. Therefore, the code and the analysis methodology rely heavily on industry experience and recommendations.

Unlike the previous codes, which are basically single physics methodologies, BOA couples the technical areas of thermal-hydraulics, materials, and chemistry. This means that fewer users can become experts in setting up the code and evaluating the results, because it requires a more broad level of understanding and more training and experience than a single physics methodology.

Some members do not regularly perform analyses of CRUD deposition and are not particularly concerned with this type of tool because CRUD does not limit core reload design or operation. Also, some believe the BOA methodology requires too much ‘tuning’, resulting in inaccuracy and excessive conservatism. Utilities that have not experienced CIPS or CILC are less likely to be concerned with this capability.

CFD Thermal-Hydraulics

CFD analyses are not prominent in the nuclear industry, due to limited computer resources and a financially limited economic environment. The most prevalent use of CFD is by the fuel vendors. New fuel designs are evaluated for T/H performance with CFD, such as GTRF evaluations, fine mesh CRUD deposition calculations with BOA, and to analyze flow mixing beneath the reactor core. Commercial CFD codes are significantly parallel and require substantial memory for performing problem meshing. They are substantially more evolved than other nuclear related M&S, and can have very good user interfaces and visualization capabilities. CFD analysis of large problems can also be prohibitively slow for schedule-driven production work like core reload analysis. Fortunately, most analyses do not require this level of fidelity and CFD is generally only applied when absolutely necessary.

Nuclear industry use of CFD in-core is basically limited to sub-assembly analyses. This process may take over a month for problem setup and solution on several hundred computer cores, and requires a significant amount of user training and expertise. CFD codes are not typically used for licensing grade calculations, but for understanding special phenomena not predicted by lower order methods. Some in the industry have used CFD to model fluid mixing beneath the reactor core. It is not used for two-phase flow or large full core problems. Note that CFD is utilized in many instances for analysis elsewhere in the in the nuclear plant system.

In general, it is financially prohibitive to purchase, setup, and execute CFD models, and most members cannot make a compelling business case to support the personnel/expertise, training, and computer resources required. In general, fuel vendors have more incentive to use these tools than utilities for a limited number or applications.

Others

Other minor codes, utilities, and post-processors are used during most analyses. Examples are tabulation of cross sections, fuel management optimization, output data format conversion to another codes input format, automation of execution sequences, output data assimilation and summary, automation of input creation for parametric studies, etc. This are typically home grown utilities that are developed as needed to quickly and easily provide connectivity of the methods codes in the workflow.

System codes such as RELAP, RETRAN, or TRACG are not considered in this report. Also, fuel mechanical codes were not discussed, which are required to provide fuel temperatures for the core simulator.

Table 2 Approximate Runtimes of Current Codes (CIPS)

Codes	Number of Cases	Single Case CPU-sec	Full Cycle CPU-min
2D Lattice Code	5000 – 10000 per cycle	1 – 5	~ 100
3D Core Simulator	~ 20 per cycle; 1000s per reload	10 – 30	5 - 10
Sub-Channel T/H	4000 per cycle; 80000 per reload	?	10 - 20
CRUD	20 per cycle; 400 per reload;	< 60	~ 15
CFD T/H	?	?	?

SAMPLE WORKFLOWS

Each IC participant provided a unique perspective on engineering analysis and workflow. It is clear that the workflows at different organizations for similar problems are not the same. Organizations have different resources, different responsibilities, and different customers. They experience different types of problems at their plants and interact with the regulator differently. Perhaps most importantly, organizations are managed in different business environments where often decisions are heavily based on financial considerations. These drive organizations to develop their own independent capabilities and processes, and to meet their own needs as inexpensively as possible.

The most obvious commonality between organizations is the need and desire to have general purpose capabilities. These tools are adapted and combined for specific analyses, and can continue to adapt when new issues or problems arise. Almost all analyses start with full core reactor simulation, or it is desired to do some type of multi-scale evaluation beginning with a lower order full core screening evaluation. Ultimately, the full core analysis must be performed to understand the range of boundary conditions and nuclear and T/H states encountered by different regions in the reactor core.

Other generic characteristics of industry workflow are:

- Typically ASCII interfaces to engineering-grade codes in Unix environment
    - Few GUIs, except for BOA, CFD, etc
- Data is manually transferred via ASCII or binary files
- Minimal coupling or communication between codes
- Homegrown post-processors often combine/reformat data for transfer to other codes

- Minimal code parallelization is used. Most clusters are limited in size, and must be shared amongst the entire organization.
- Minimal output visualization is used (other than CFD, which has limited applications). General purpose visualization tools are sometimes homegrown to analyze 3D datasets and 1D/2D line graphs. Commercial codes such as Tecplot and Matlab are sometimes employed.
- Analyses may include 100s or 1000s of full core simulations
- Analysis problem space is often limited due to limited CPUs, time, or personnel
- Time bottleneck is often verification and documentation, not actual computations
- Uncertainty quantification is not performed in most analyses (but is generally performed in a one-time uncertainty analysis as needed to support margin and safety assessments). Each M&S application requires the determination of unique error allowance/safety factors.
- All methods used have high validation pedigree

In all locations, there is a clear desire to have more resources, more personnel, and more analytic (higher fidelity) capabilities. There are always unanswered questions, yet seldom is the business case developed that justifies answering them. Important issues arise unexpectedly, and must be solved quickly. There is minimal time for research, complicated model development, or long runtimes. There is a prevalent opinion that, in most cases, more capability will only result in more work - not necessarily faster work, less work, or better answers.

Though there is no single specific workflow for all locations, this section provides details for some sample analyses that were discussed during the course of the IC meetings.

General Reactor Core Simulation

Simulation of the entire reactor core is a dominant part of most analyses performed by core designers and reactor engineers. Even the analyses that are performed on a smaller scale (such as at the level of a single assembly) require some assumptions or iteration to determine if the smaller scale problem is indicative of the full core limiting location. A general purpose tool also provides the user with an ability to explore and solve unanticipated challenges. Some sample analyses that rely on the core simulator are:

- Core loading / shuffle pattern optimization
- Minimizing impact of CRUD depositions (CIPS / CILC) through risk assessment
- Evaluation of FAD / Channel Bow Risk
- Determination of cycle energy requirements (set fresh fuel enrichment and fuel batch size)
- Prediction of maximum fuel exposure, corrosion, and rod internal pressure
- Control rod maneuvering and insertion limits (PCI)
- Physics parameter calculation (temperature coefficient, rod worths)
- Ensuring shutdown margin (ability to force the reactor subcritical)
- Calculation of reactor trip setpoints / operating limits
- Core follow / exposure accounting
- Verification of as-loaded core through startup testing
- Reactor surveillance (criticality and power distribution monitoring)
- Calculation of control rod / incore instrument depletion or fluence
- Prediction of plant response to power/rod maneuvers
- Core response to plant accident
- Confirmation of tech. spec. limits for reload cores

- Calculation of input for plant computer and real-time operations simulators
- Investigation of anomalous plant behavior
- Investigation of local power and T/H conditions after fuel failure
- Initialization of plant conditions prior to transient analysis
- Calculation of the neutron/gamma source term for component lifetime and shielding analyses
- Advanced fuel design

The following steps roughly describe the core simulation workflow for basic depletion calculations. This is the process of setting up and modeling the operation of the reactor, in order to establish the fuel power history, isotopic concentrations, and save a full core restart file such that future analysis can be performed (for instance, the next cycle depletion). This is somewhat depicted on the left side of Figure 1.

1. Fuel geometry specifications and typical reactor conditions are obtained.
2. ASCII inputs are built for each unique 2D fuel lattice and burnable poison combination in an assembly. This could be about 10-15 lattices per cycle (this step may be automated).
3. The lattice physics code is executed for each lattice given the ASCII input. The code is serial but each lattice can be run in parallel if enough processors are available. Approximately 5000-10000 statepoints are calculated in a cumulative CPU time of about a few hours (potentially 10-20 minute wall time).
4. The output files of the lattice code are collected, and a post-processor utility code may be needed to tabulate all of the calculated cross section data for each fuel type and store in a library file. (These files could be read directly by the core simulator). This code generally has a small ASCII input and executes rapidly.
5. The ASCII input for the nodal core simulator is prepared by the user based on plant operating characteristics, actual plant data, and core loading pattern. The cross section file from step 4 is provided and each assembly and lattice type in the core is identified and assigned a cross section identifier.
6. For BOC, the core simulator is executed to perform the fuel shuffling from the previous cycle (incorporating fuel exposure and other information from previous cycles) and calculate the zero power critical boron concentration. Other startup test values could also be calculated, such as rod worth or temperature coefficients. This job may take 10-30 seconds.
7. For cycle depletion, the user provides (in the same or second input), the core average power, flow, inlet temperature, and control rod pattern verses time throughout the cycle.
8. The nodal simulator is executed once given the entire cycle information. The code internally calculates the coolant density distribution simultaneously with the flux neutron distribution for each statepoint, calculates the exposure increments, and continues until it reaches EOC. This may take 5-10 minutes to run the entire cycle. The code writes a restart file after the last case (or as needed).
9. The ASCII output can be checked for peak pin powers, boron concentrations, convergence errors, etc. Core results can be post-processed for a variety of subsequent analyses. Values are compared to core design objectives and operating limits.
10. The next cycle input can be prepared as was done in step 5. Using the EOC restart files, steps 5-9 can be repeated for each subsequent fuel cycle. Steps 1-4 are performed each time a new or unique fuel type is implemented.
11. For official calculations, all of the inputs and jobs are documented, independently verified, and archived. This step takes the longest of the entire process.

In general, the cross sections are prepared in advance for each fuel type. The complete cycle models can be run in the 5-10 minutes time frame on a single core. For cycle N accuracy, cycles N-3, N-2, and N-1 should be modeled with consistent methods prior to running cycle N. A single statepoint can execute in 10-30 seconds. During core reload design, hundreds or thousands of full core calculations may be executed while optimizing fuel location, enrichment, and burnable poison loading. When setting operating limits, 18,000 cases may be run in a period of approximately 10 CPU-hours.

CIPS

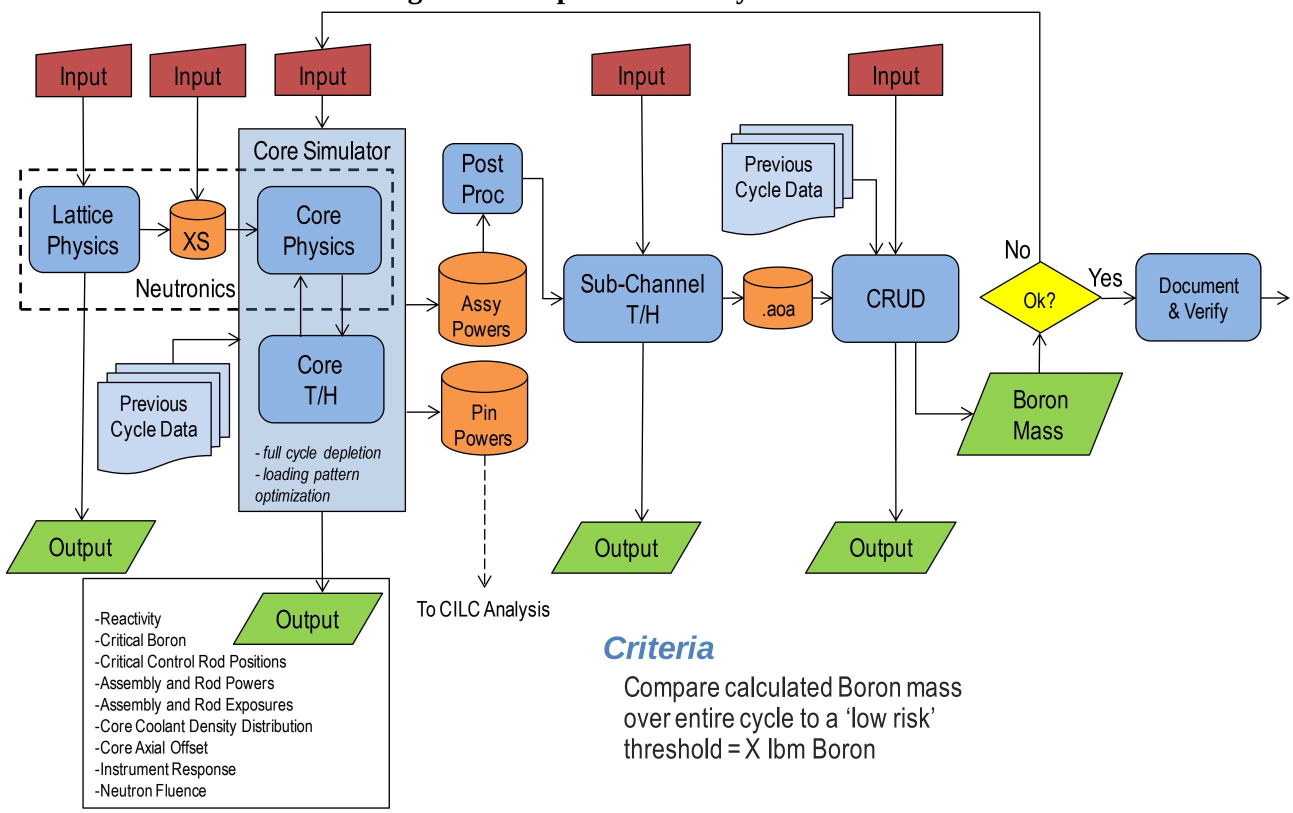
Analysis of the risk of CIPS is an important part of some of the members' core reload design processes. This analysis is a relatively small part of the approximately 1-2 month engineering/design process to develop a core reload loading pattern, which is only one of 30-40 different analyses or tasks that are performed over a 12-18 month period to design, analyze, and install a reload core into an operating NPP. The members interviewed in this project apply the recommended BOA methodology developed by EPRI.

Preliminary scoping studies for the next fuel cycle of a given unit are begun approximately 18 months prior to scheduled startup. The core designer evaluates potentially dozens of loading patterns and determines the merits of each based on many rules and limits for the design. Each candidate pattern is depleted nominally to the planned cycle length prior to performing the CIPS analysis. Depending on the performance of the design relative to boron deposition and fuel cycle economics, the fine tuning of the design may become a manually iterative process between the core designer and CIPS analyst until a final pattern is selected.

Application of the BOA methodology requires several executions of the problem workflow in order to perform a single cycle analysis. First, the methodology must be applied to a previous cycle known to have CIPS, or alternately one that is felt is a limiting reference case (such that the reference may be to a cycle that is known not to have CIPS). This establishes the CRUD release factor(s) that force the code to predict the recommended threshold of boron mass. Prior to analyzing the design cycle, Cycle N, the currently operating cycle, analysis of Cycle N-1 must also be completed. This is another important step for adjusting the CRUD source term and for developing an accurate distribution of CRUD on the reinsert fuel for Cycle N. However, most of the adjustments come in the designed cycle N.

Figure 1 provides a graphical depiction of the CIPS workflow for one pass through the analysis. General descriptions of each step in the workflow are provided below the figure. Focus is given to interactions with codes, decision points, data transfer, etc. This discussion is limited in detail in order to be clear and concise.

Figure 1 Sample CIPS Analysis Workflow



1. The reactor core neutronics simulations are supported by macroscopic cross section libraries that are generated prior to performing this analysis. This data is provided by a 2D lattice physics code, as described in the previous discussion on the core simulation workflow.
2. The core reload design scoping process considers many candidate loading patterns and evaluates each of them over the entire cycle length against dozens of pre-established rules or criteria. One of these criteria is the risk of developing CIPS, as determined by the predicted mass of boron deposited on the fuel rods.
3. The workflow for general reactor core simulation is performed as described in the previous section. Hundreds of quarter or full core cases are evaluated for each candidate pattern to optimize the pin peaking, fuel costs, safety margin, and operational risks such as CIPS. For each pattern chosen for CIPS analysis (approximately 10% of the preliminary designs), a full cycle depletion is performed (15-20 cases) and the 3D assembly power distribution is output to a file. A cycle depletion takes approximately 5-10 minutes on a single processor.
4. The output core simulator power distributions are post-processed by a utility code to generate the ASCII input to coarse mesh sub-channel T/H analysis. This is considered coarse here because it is performed on a sub-assembly level rather than with pin-by-pin detail.
5. An automation code creates the burnup-dependent sub-channel inputs from a manually created base input deck. All the decks are ASCII. The base deck may take several days to create.
6. The sub-channel code calculates the detailed T/H conditions and sub-cooled boiling for each channel at each statepoint based on the provided axial power distributions. This requires input of loss coefficients, best estimate flows, and temperatures. There is no feedback from the sub-channel code back to the neutronics. These jobs take 10-20 minutes to complete on a single processor. This piece may be performed by an expert core designer, or may be handed off to the T/H or BOA expert.
7. The sub-channel T/H code produces a file formatted for BOA input (part of the EPRI BOA methodology) called a .aoa file.

8. BOA takes sub-channel fluid conditions, system sources, assembly CRUD histories from previous cycles, boron concentration, lithium, etc., and can calculate the CRUD thickness and boron deposition for each statepoint throughout the cycle. Input is based on EPRI recommended methods.
9. BOA output is used to compare to risk threshold(s):
    a. Maximum CRUD thickness or change of thickness (CILC screening)
    b. Core-wide boron loading of XX lbm
    c. Assembly boron loading XX lbm per assembly (not required, suggested by EPRI)
    d. Much more results are available from BOA, such as nickel mass, etc.
10. BOA evaluation is performed for each depletion step, typical 15-20 per cycle. This requires approximately 15 minutes per cycle on a single processor.
11. CIPS analysis is performed for each viable loading pattern, perhaps 15-20 analyses per reload design, plus a final "official" analysis after pattern selection. Total is approximately 400 BOA statepoints per reload.

Methodology Discussion

The current BOA methodology provides a quantitative prediction of CIPS risk based on the system level CRUD inventory, mass evaporation, and boron deposition. It simultaneously combines heat transfer, chemistry, and corrosion effects, and provides a mechanism to account for CRUD sources carried over from previous cycles. The radial mesh is only 2x2 per assembly, but some members believe that an increase in fidelity (i.e. pin-by-pin) will not provide any significant improvement in accuracy. PWR rod powers are typically very uniform in an assembly, and boron must be deposited on a large population of fuel rods in order to produce a core global axial shift in power. Therefore, it is felt that for CIPS, a sub-assembly representation of the axial boron deposition is sufficient.

The BOA methodology requires analysis of a plant specific reference or benchmark cycle in order to define the risk threshold for subsequent cycles. The CRUD release factor, CREL, is adjusted so that the reference cycle predicts the correct magnitude of CIPS. For NPPs which have not experienced CIPS, the factor is tuned to the recommended boron mass threshold (which can be changed based on the level of risk that one is willing to accept for a particular core design). This can be an overly conservative penalty. In these cases, the most aggressive core design that has not had CIPS sets the value of CREL to maintain the same threshold. In this way, the members conservatively assume that the reference cycle was close to CIPS occurring, and the risk of CIPS occurring in the design cycle is less than the reference cycle. Essentially this means that ***the methodology is not predicting CIPS but is ensuring that the design cycle is within the operating experience of the reactor (in terms of boron deposition).***

In the latest version of BOA, the nickel/iron ratio can also be adjusted as well as CREL. The procedure for setting up the benchmark cycle and determining the tuning factors is provided in the EPRI methodology. These are then carried forward to evaluate the risk for future cycles. Some members suggested that the largest uncertainty the CIPS calculation is with the CREL parameters. Changes in the primary system can invalidate the previous reference cycles from which the CREL value is determined. For instance, changes in the pH program, RCS piping surface area, and the addition of zinc injection will alter the CRUD source terms for the reactor. Therefore, it is important that the design cycle be similar in terms of CRUD sources and release rates to the benchmark cycle. In addition, some plants undergo CRUD bursts at EOC or use ultrasonic cleaning approaches to remove CRUD. The amount of CRUD removed during these processes is uncertain, but must be included in the analysis.

The calculated boron deposition over the fuel cycle is directly fed back to the core design process. Assembly level results are also available from BOA such that the core designer can perform small design changes to address the increased local risk. This could be accomplished by increased burnable absorber or assembly cross-core shuffles. In addition to predicting the boron loading, it is critical to provide the core designer relative local data to inform small adjustments in the loading pattern.

One member has found that the current BOA methodology is accurate in predicting total core boron deposition, but differences have been noted when compared to the measured CRUD distribution on the fuel. For instance, BOA might predict more CRUD on feed assemblies, yet more activity is measured from reinsert fuel from the previous cycle.

The current methodology performs well for avoiding CIPS. However, the members might be willing to be more aggressive and design/operate with some localized boron deposition (aka mild CIPS) if the analysis methodology was shown to be accurate and that the phenomenon was limited to only a few high powered locations, not core-wide. This may require a best-estimate approach to the simulation with accurate feedback (boron deposition in neutronics with depletion possibly).

The current BOA analysis is largely performed for scoping analyses and is not safety related. Only the final pattern is analyzed with independent verification and documentation. The CIPS analysis is performed mainly for operational and financial concerns. CIPS is limiting for only some of the member plants. Some members employ ultrasonic fuel cleaning and do very little CIPS analysis, but this issue could be more important for future plant changes in operation (such as power uprates).

The BOA methodology has been heavily validated by others in the industry (ERPI, etc) against 30-40 cores with CIPS, or possibly hundreds of cycles. The members follow the EPRI Fuel Reliability Guidelines and put their limits and criteria into engineering procedures. Use of CASL tools in this process would require some reevaluation and procedure modifications, as well as extensive validation.

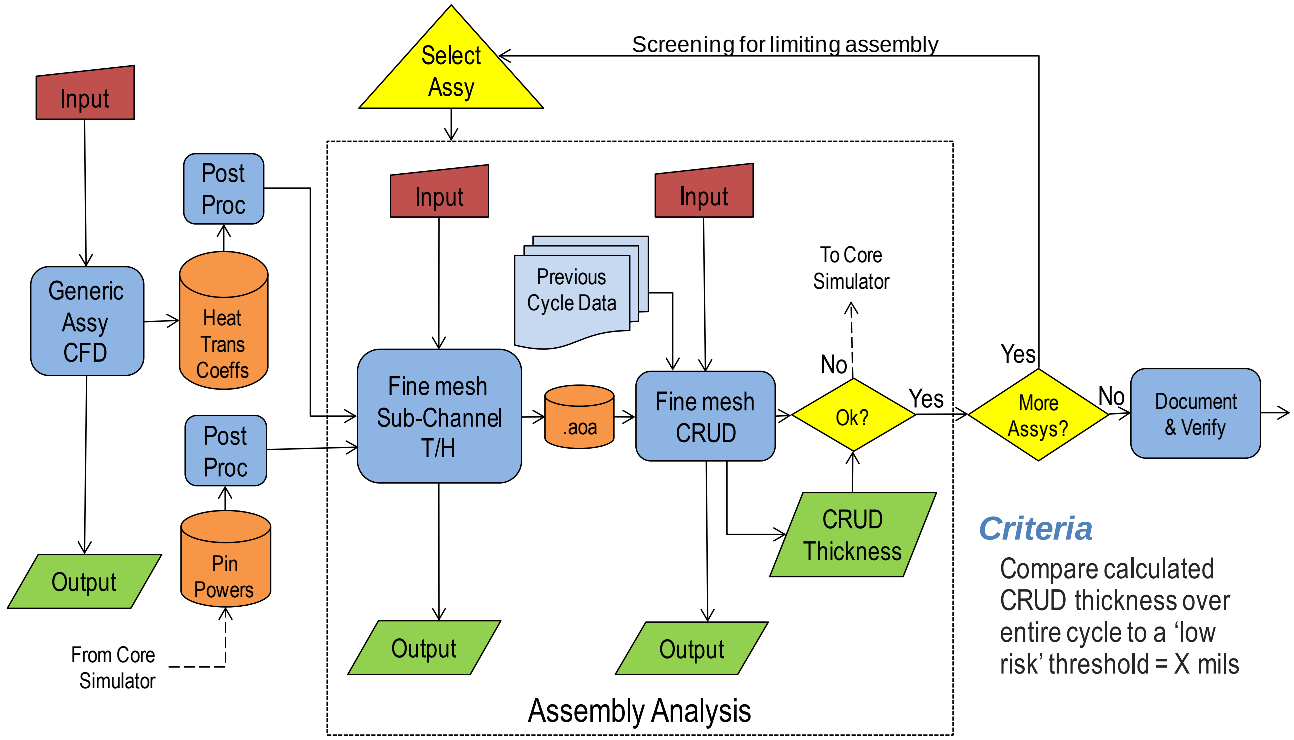
CILC

CILC analysis follows the CIPS analysis at only a few member organizations. The BOA methodology used for CIPS is continued for detailed CILC analysis, but CILC requires a much higher level of fidelity for fuel rod power and local T/H conditions. Only approximately five instances of CILC failures have occurred in the industry, so many members do not consider this as a high risk when performing their core reload designs. In general, other core design practices, peaking limits, and the coarser CIPS analysis prevent the need for detailed calculation of CILC risk. However, for the members that do perform the CILC check, it is a very resource consuming process with the current tools.

A CILC failure can occur on a single rod depending on the amount of CRUD deposited, so this check is a rod-by-rod process and thus requires more automation, longer runtimes, and larger input files. Using the same BOA methodology, the analysis adds pin-by-pin powers from the core simulator, azimuthally dependent heat transfer coefficients for each rod from a CFD calculation, and rod-by-rod T/H and boron deposition using so called fine mesh sub-channel and CRUD models.

Figure 2 provides a graphical depiction of the CILC workflow following the coarse mesh CIPS analysis. General descriptions of each step in the workflow are provided below the figure. Focus is given to interactions with codes, decision points, data transfer, etc.

Figure 2 Sample CILC Analysis Workflow (Continued from CIPS)



1. The coarse mesh BOA results from the CIPS analysis in the previous section are used to screen the candidate loading pattern for CILC risk.
2. A CFD model of the fuel assembly type is used to determine the variation of heat transfer coefficients around each fuel rod. This process can add an extra month or more to the core design process and requires much more computer resources. Commercially available CFD codes are used for the analysis.
3. A set of assemblies suspected to be limiting are selected for fine mesh CILC analysis. Each assembly has to be analyzed separately to determine the most limiting location for the core. The fine mesh criterion is different from the coarse mesh screening criteria.
4. A fine mesh sub-channel T/H model is built using the fine mesh pin powers from the core simulator (calculated during CIPS analysis) and using the distribution of heat transfer coefficients calculated by the CFD model. The pin powers calculated by the neutronics code may not include any azimuthal variation, so there is a small inconsistency between the fidelity of the methods.
5. The BOA model uses the sub-channel results and input .aoa files, which can contain both coarse mesh data (like CIPS) and the local fine mesh data for the selected assembly. A post-processing code creates the combination .aoa files for BOA.
6. BOA is executed for the fine mesh assembly problem and the CRUD thickness is compared to the limit. Then the process continues for the next assembly.
7. The process can continue for hours or days for just one case. The entire analysis can take several months, which can be too late to go back and redesign the core loading pattern.
8. Documentation of this analysis is also very time consuming.

CILC analysis requires high fidelity multi-physics coupling at the sub-rod level over long time scales. The codes and methods needed require significant computer resources and engineering expertise. Many members do not perform the analysis at all. Considering only a few failure events have occurred, it's difficult to justify the development efforts required to truly solve this problem from a first-principles approach. However, for additional power uprates, CILC may be more of a concern and the additional analysis may be necessary.

Performing the CILC analysis and evaluating the BOA results requires considerable expertise and requires qualitative engineering judgment. It requires multi-discipline interaction and coordination between experts on different teams. Not many engineers or core designers can be well trained in all these areas.

The CIPS and CILC analyses rely heavily on collected industry experience and recommendations. ***The EPRI BOA methodology is not just a software product***. The members will be reluctant to replace BOA unless the current EPRI guidelines are adapted to the new product.

GTRF

The analysis of GTRF risk is typically performed by fuel vendors and not utilities. The workflow is complicated and is a combination of CFD analysis and physical testing of spacer grid designs. These tests are performed as best as possible to cover the range of application conditions, but are limited and cost-prohibitive, and are not performed on irradiated fuel. CFD analysis is used to augment the testing process and to help narrow down what designs are actually tested. As best as possible, the tests are performed to simulate the largest range possible of potential operating conditions, but in reality there are an infinite number of unique conditions and flow fields in an operating reactor core and it is not possible to test all scenarios.

Micro scale tests are performed very infrequently, typically after failure conditions are already known. The current analysis techniques can be inconsistent at different scales, and it is difficult to predict on a system scale level where the micro scale failures will occur. Unfortunately, full core CFD analyses with commercial codes have not been practical. As new reactors are brought online, the uncertainty in the flow fields and conditions that lead to GTRF will increase, as many of the current criteria are based on operating experience with the existing fleet.

It is critical to understand the flow fields and the loading conditions around each spacer grid in the reactor. Due to core power distributions, cross flow, baffle jetting, inlet flow mixing, fuel assembly distortion, etc, the number of perturbations is significant. There is currently no good way to screen a full core over many cycles to determine what local conditions will produce a failure. Therefore, current analyses only have a micro-level perspective of the problem, and make gross assumptions about the core conditions that can produce it.

The current GTRF analysis is primarily concerned with determining the small scale flow field with CFD (turbulent excitation) and material components (Fluid-Structure Interaction (FSI) effects). The flow fields change in the fuel over time (power distribution, fuel shuffling, core location), and several different wear behaviors are possible, the prediction of which is highly empirical. The problem requires the understanding of mechanics, flow, materials, and vibration, all on different scales and over an extended period of time. Ideally, full core simulation over multiple cycles to calculate the fluid flow and neutron fluence would provide the ability to screen each core location for GTRF risk at a macro scale, honing in on the limiting locations for micro scale analysis.

A better understanding of failure mechanisms and core conditions is needed to be able to perform better tests and reduce the uncertainty in grid performance under operating conditions. This could also reduce the number of tests needed.

Data retention and quality control is very important to the workflow. Human performance measures drive the process to automate steps and remove the human interface when possible. Generation of large datasets on remote computers creates I/O constraints. The engineer needs to be able to analyze the problem results, often graphically, make decisions, and then resubmit jobs, and finally store the data and input, without being placed in a time bottleneck waiting for data to transfer across the internet.

The training and qualifications to perform this analysis are extensive, and only a few people have the required multi-discipline expertise and experience.

ANALYSIS TYPES AND FINANCIAL IMPACTS

The CIPS, CILC, and GTRF analyses support core and fuel design and avoidance of operational concerns. They are not safety related, but do often require verification and documentation of the behavior of the final design.

The general core simulation capability feeds the vast majority of members' calculations to some degree, and therefore can be used in safety related, operational, or financial calculations. Safety related applications tend to be highly scrutinized in terms of verification, validation, and independent quality assurance. CFD analysis is used sparingly because of the large expense in engineering time to setup the calculations and long calculation times required for the analysis, and the substantial personnel and computer resources required. In addition, some participants do not trust CFD enough, due to lack of sufficient validation, to provide benchmarks for verification of lower-order methods.

In general, all participants agreed that safety analyses to support changes in plant operation and limits are most important for the industry, but also these are the least likely to be influenced by CASL due to the licensing requirements that go along with that application. Therefore, CASL should focus on operational issues that do not require licensed codes or methods. Some members feel that, after several decades of analyses, there is no more margin to be found in this area.

The financial impact of CIPS in terms of the fuel costs to 'detune' optimized loading patterns is estimated to be $400k to $1M per cycle to some members. Others feel there is no impact. The design and analysis of a nuclear reactor faces multiple constraints, and operators' priorities can change quickly when a new constraint appears. Note that the fuel cost is relatively small compared to the revenue generated by a NPP at full power.

No appreciable cost was attributed to CILC or GTRF risk mitigation. Certainly occurrences of these phenomena could be very costly to the member organization, but it doesn't appear to be a significant financial impact to avoid to the risk. Core designers follow rules to limit GTRF risk on the core periphery which may result in a less efficient loading pattern than desired. Grid testing can be expensive.

Engineering time is relatively inexpensive compared to revenue. The extra analysis required is not a significant financial factor.

Fuel customers are willing to pay more for fuel that is more likely to avoid CIPS/CILC/GTRF. The primary concern is not necessarily the fuel failure but avoiding loss of electrical generation.

The major economic drivers are related to plant performance and efficiency, not necessarily fuel issues. Perhaps the greatest financial value is provided by the general purpose tools and capabilities, being that some of the biggest risks are problems that have yet to occur – things that aren't expected. There is value in versatility.

Improved analysis and training is hard to justify financially to utilities. Important drivers in these organizations are safety, electrical generation (production), and regulation. It is sometimes very difficult to directly tie M&S improvements to these three areas.

SENSITIVITIES & UNCERTAINTIES

Uncertainty Quantification (UQ) is not embedded in any of the workflows discussed. Sensitivities and uncertainties are typically calculated at the outset of establishing the analysis methodology, and these results are applied as conservatisms to the limiting parameters, such as effects of manufacturing tolerances, etc. There is no dynamic calculation of uncertainty in any of the processes. However, it should be considered that there are limits on the accuracy that can be gained by increasing the precision of methods, and extrinsic factors may result in an exceedingly complicated uncertainty quantification that may not always be warranted.

For CIPS and CILC, the sources of the system CRUD inventory are largely unknown and critical to accurate simulation. More than 25-30 lbs of CRUD material may exist in the primary system and its distribution is not very well known. There is also a large uncertainty in general corrosion of the RCS over a large period of time, such as 15 years for steam generator tubing. Corrosion rates and release factors are empirical and currently are used as 'tuning' parameters in the analysis.

Carryover CRUD from previous cycles may be removed through ultrasonic cleaning (UT) and CRUD bursts. Approximately 50% total efficiency has been assumed, but this value could have a large variation. This is a very important input to this analysis, as the lack of fuel cleaning alone could result in CIPS at some member NPPs. It is sometimes approximated based on visual inspections, measurements of localized CRUD scrapes, or CRUD mass in the cleaning system. This value needs to be better characterized, as its importance could dominate any enhancement with high-fidelity CRUD deposition models, or higher fidelity neutronics or T/H.

The heat transfer partitioning on surfaces with CRUD is not well understood. Some of the heat flux generated by the fuel is transferred via mass evaporation, while some via convection. This distribution is uncertain, as well as its dependence on CRUD thickness, CRUD morphology, fuel surface heat flux, and local chemistry.

Limited availability of test data other than actual plant experience. This is needed to benchmark predictive models. Better tests can result in lower uncertainties.

For GTRF, the core conditions and flow fields local to the failure are mostly unknown. Also, the failure mechanisms are highly empirical based on the limited test data that is available. The flow fields may be sensitive to geometry changes from FAD. Need to quantify the sensitivity of cross-challenge problem effects.

Geometry effects of channel and assembly bow are not well quantified. This is partly due to the limitations in nodal methods, and could be more easily quantified with higher fidelity methods.

Additional significant uncertainties identified include:

- Limited availability and large uncertainty in plant data (makes it difficult to drive sophisticated models).

- The aggregate uncertainties in reactor analysis are dominated by the reactor system, not the core or vessel. This is demonstrated with the new LOCA best estimate methodology.

- For uncoupled analyses, the source of uncertainty is dominated by the boundary conditions (how well does the input conditions represent the actual variability under actual operating conditions?).

- The uncertainty in local pin power distributions at off nominal conditions due to the use of nodal neutronics methods needs to be evaluated.

- The rollup of uncertainties may lead to unrealistically large results. Over-estimation of actual aggregate uncertainties is very problematic.

- The uncertainty in nuclear cross sections should be considered when developing and validating higher order transport and Monte Carlo methods for full core calculations

- Higher fidelity does not necessarily equal less uncertainty, or more accuracy. Don't always need a scalpel to make important improvements in aggregate analyses.

RESOURCES

Most member organizations are running small Unix-based clusters for their analyses. Utilities tend to have smaller systems of < 100 CPU clusters of ~12-16 CPUs per machine with minimal parallelization. Fuel vendors, who tend to have a higher volume of analysis, have somewhat larger systems and more parallel computers with higher memory, such a 500-600 CPUs and single machines up to 256 cores. One member uses a Cray HPC platform with up to 400 cores available for parallel computing. Another member has a 512 core Linux HPC platform, and is planning a 2-3 times expansion.

BOA is executed on the PC, and data is transferred from compute clusters via direct drive mapping (such as Samba or NFS).

All members expressed interest in purchasing larger computers for higher fidelity analysis, if a clear economic benefit was demonstrated. However, it is unlikely industry organizations, especially utilities, will purchase supercomputers, even if the tool is very valuable. This may be somewhat do to the structure of business organizations, the separation of the analysis teams from the operations teams, and the difficulty of relating analysis improvements to safety or electrical generation.

CIPS, CILC, and GTRF are each highly specialized cross-discipline analyses that require significant training and experience. There are typically only a few subject matter experts for these analyses, who need to support a much larger number of engineers performing core reload design and other analyses for potentially many NPPs.

Significant time and resources are spent on verification and documentation of analyses, sometimes more than actually performing the calculations.

The current business environment is very lean and there are limited personnel and computers available for implementing new codes and methodologies without first making a solid business case. A clear financial incentive must be demonstrated in order for the organization to invest in these resources. One success path would be to allow the industry organization access to the larger computers at ORNL such that the value of the CASL tools can be clearly demonstrated and prototype analyses can be used to drive financial decisions.

INTERACTIONS

Most of the members' workflows involve simple human interaction with engineering-style analysis tools, usually with command line interfaces or scripting files (except for CFD and BOA). Inputs and outputs are predominantly ASCII, and data is manually transferred from code to code via ASCII or binary files which are produced or post-processed from other codes. There is little coupling involved (no feedback), and for many cases it is not apparent that improved coupling, T/H to neutronics for instance, will result in much improvement or financial savings. This process can be automated for repetitive analyses that do not require human intervention.

BOA has a useful GUI for input. The GUI not only provides an easy interface, but also functions as a guide for applying the correct solution methodology and following recommended guidelines for the problem. BOA also provides EXCEL plots as output. There is no other graphical output, and this is not viewed as important to the analysis.

The commercial CFD codes used have output visualization capability which is important to understanding and digesting the large amounts of data produced by the high fidelity simulation. This often results in highlighting a phenomenon the engineer may not have considered.

Commercial plant training simulators provide advanced 3D video game graphics and animations for visualizing and understand plant responses. These are not just “movies”, but high-end multi-dimensional representations or the results being produced by the physics codes. This makes it easier and faster for the user to observe, evaluate, and adjust for the next scenario.

Code interactivity, in terms of detailed graphical interaction or interactive execution, can be a useful tool for researchers, students, and for exploratory and communicative purposes. However, interactivity itself is also viewed as a potential source of human performance errors and needs to be balanced against the strict regime of safety related activities.

Reload core design involves a repetitive, iterative approach to loading pattern optimization. As patterns are found that meet all the core design criteria, the assembly powers are often passed to the CIPS/CILC subject matter expert, who proceeds to setup and execute 15-20 statepoints with the sub-channel T/H code and BOA. For minor limit violations, the designer may need to perform a few cross-core shuffles or increase the burnable poison loading in a few fresh locations. For neutronics, this evolution can be accomplished in a minute or so, and this type of task will be performed hundreds or thousands of times during a core design. The BOA analysis can subsequently follow this minor neutronics change in about 15 minutes. The core designer can use the assembly level output provided by BOA (boron mass) to target the limiting locations in the design. This process will iterate until the core designer has satisfied all the design criteria and shown low CIPS risk. For a typical core design, BOA will be run 15-20 times (each run will have 15-20 depletion statepoints), and these do not include efforts to establish the cycle N-1 BOA model from the previous cycles. Therefore, for scoping analyses, the runtime required for this analysis is very small. Very fast runtimes are required for production activities.

OTHER TECHNICAL INTERESTS AND COMMENTS

This section contains some comments unrelated to specific workflows that were discussed by the IC participants:

- Concerned about wear resulting from fuel assembly interactions with the core baffle
- Limited concern about PCI for PWRs. PCI risk is typically related to control rod movements being performed at operating conditions, which is a characteristic of s fraction of PWRs (B&W plants), BWRs, SMRs, etc.
- DNB can be limiting for radiological analyses (dose). This affects power uprates.
- Reactor coolant system concentrations of boron and lithium can be limiting
- Some members felt that the some analyses would not benefit from higher fidelity neutronics. The limiting assumptions and uncertainties are driven by measurement data (or lack thereof), lack of validation bases, or input from the NPP that is unknown or widely uncertain. There is greater uncertainty in materials, chemistry, and system models than in the neutronics results.
- Interested in the full core CFD model for a variety of one-time R&D or confirmatory studies
- Interested in higher fidelity models in general for validation of current licensed methods and for analysis of current and future problems which have yet to be explained, such as:
    - Hot leg streaming - One hot leg could cause trip due to being 18-20 degrees off from the others
    - Lower plenum anomalies
    - Inlet coolant flow redistribution during accident conditions
    - Boron precipitation during LOCA
    - Radial power distribution anomalies – needs independent verification of methods and requires a 'jump in' to cycle N-2, without going back to cycle 1
    - Future anomalies will require general purpose tool, not challenge problem specific
- Need to move beyond nodal neutronics methods – need new generation of toolsets
- V&V of CASL methods is extremely important. The CASL CIPS tool needs to be validated to the 30-40 CIPS cycles like what has been done with BOA.
- Some members are not currently too interested in CILC (basically avoided by using a conservative approach to CIPS and by UT cleaning)
- Prefer an integrated system view rather than high-fidelity core models.
- Source range detector sensitivity – predict out-of-vessel behavior with subcritical multiplication
- Seismic analyses – are input and techniques appropriate?
- Interaction with NRC – not pushing current plants as much but are tougher with new plants. NRC wants more details and more justification.
- Interested in confirmatory analyses for Critical Heat Flux testing. Need a reliable high-fidelity prediction to avoid increased testing.
- Interesting in data sharing - publishing of UQ results, even if performed with proprietary methods.
- Interested in better logic and controls models for a better prediction of plant real system response.
- More interested in a general purpose toolkit than specific challenge problem solutions. Allows exploration of new issues and a new level of learning that has not been possible before.
- Interested in the details and functionality of the coupling environment and methodology, even for existing M&S.

  Interested in moving towards best-estimate plus uncertainty methodologies of coupled disciplines, rather than using a deterministic worst-case approach where each discipline generates its own set of safety factors, resulting in an undue pessimism being brought to design calculations.

POTENTIAL PILOT PROJECT CONCEPTS

The following projects were mentioned as potential sources of an early success for CASL, or a potential project for collaboration with the IC member. Each member expressed a willingness to cooperate with CASL in some form in the near term.

- GSI-191 - Debris accumulation on the PWR sump screen
- CIPS analysis for cycle with moderate to severe CIPS
    - Demonstrate less tuning required
    - Demonstrate core design savings with new methods
- GTRF
    - Identify high risk core locations based on core wide flow fields calculated by CFD
    - Learn about unknown local variations in flow for cores with known GTRF risk.
- Other applications of full core CFD model
    - Hot leg streaming
    - Lower plenum anomalies
    - Inlet coolant flow redistribution
- Calculate baffle flows and temperatures
- Post-LOCA cladding integrity
    - Applicable to PWRs and BWRs
    - Multi-physics, multi-time scale
    - Model fission gas release, hydrogen pickup
- Application of coupling environment/framework for outside M&S or training simulators

SUMMARY OF ADDITIONAL MEMBER INSIGHTS

The following items are a summary of additional insights provided by the IC member organizations. There are obviously a wide variety of opinions of CASL's objectives and of what will and will not be useful. Either way, it is important for CASL to be aware these insights and understand the member's point of view.

Versatililty

- A general purpose toolkit is most important. A capability to investigate current and future problems with a new approach. Provide the advantage to learn and explore the physics of the reactor in new ways.
- Interested in general purpose core simulation capability for V&V of current methods and obtaining a better understanding of current and future anomalies that are plant-specific. Interested in higher fidelity and better coupling but not necessarily for production work.
- Additional capabilities (other than analysis) may be training and/or engineering assisted design.
- In the nuclear industry, issues and challenges tend to be undervalued until a particular organization is affected directly. This results in a certain uncertainty about what problems are important and which ones will be important in the future, and likewise an uncertainty about what M&S tools are needed for current and future analysis.

CIPS

- The current EPRI BOA methodology does not predict or prevent CIPS, but provides a gross qualitative way to ensure that the future design is within the CIPS operating experience of the reactor, based on a previous reference cycle
- There may be little value in higher fidelity neutronics or thermal hydraulics in CIPS analysis. The current approach is to avoid the risk and that has been successful for many years. Should the utility desire to design cores with localized boron deposition, or mild CIPS, the higher fidelity, coupled feedback, and extensive validation would be required. However, the uncertainty in the CRUD sources will likely prevent any improved best-estimate (or first principle) approaches.

CILC

- CILC has low occurrence in the industry (possibly as low as 5 instances), and requires a significant amount of resources to predict (currently requires CFD and possibly fine mesh neutronics, T/H, and chemistry/corrosion). It is not obvious that this is worth a significant amount of CASL's focus, other than as a demonstration of capability. CASL should weigh both the probability and the consequence of this phenomenon and consideration of occurrences for future plant uprates.
- CILC is not limiting (or even interesting) to some members (utilities). It is not a significant financial impact to avoid.
- The CIPS/CILC analyses rely heavily on collected industry experience, which provided the basis for the EPRI methodology and recommendations. In this manner, BOA is not just a software product, but a complete methodology with 30-40 cycles of validation. CASL will need to address this scope in its final product.

GTRF

- Seemingly endless problem – infinite number of core conditions (flows, vibrations, etc) and must account for wear over 1-3 cycles in different core locations under different flow fields.
- Knowing the steady state flow field at every location in the reactor is 90% of the problem. Need to screen the full core problem for limiting wear location and then analyze that location with higher fidelity.
- Current reactor vessel conditions are basically known, but next generation plants could offer surprises. Knowing the detailed local flow conditions in the core is key to avoiding GTRF.

- Real challenge is model validation and demonstration of benchmarks.
- For baffle jetting, need to benchmark VERA against a CE or B&W reactor with a history of GTRF on the core periphery.

BWRs

- Should not embed PWR assumptions in the methods or interfaces (such as structural, chemistry, etc.)
- Certain elements are more connected between PWRs and BWRs. This should be identified early.
- Interested in when and how an advanced two-phase flow sub-channel analysis methodology will be developed for BWR fuel bundles.
- Interested in early application of 3D transport coupled to two phase T/H.

Expertise

- CIPS, CILC, and GTRF are unique multi-discipline engineering problems, requiring knowledge in reactor physics, thermal-hydraulics, materials, and chemistry. Few people have this expertise, and it requires a long time to develop. Currently, organizations tend to be divided into separate functional areas, but as the analysis of reactors becomes more coupled, the software tools need to assist the engineers to solve these multi-physics problems.

Productivity

- Solve the science – make new revelations. Obtaining a better understanding is more valuable than providing a production tool. If the CASL tools demonstrate more positive margin, the member will want to understand the difference and then develop a process of accounting for the effect in the existing methodology.
- Production tools are required to be very fast. Up to 400 BOA cases may be executed during the preliminary core design process which might include thousands of full core neutronics cases, each running in less than a minute on a single computer core. It is not clear how the CASL tools could be beneficial to this process.
- Coupled sub-channel T/H is fine for many applications and is much quicker than CFD.

Licensing Bases

- Tools to solve real problems must maintain licensibility.
- Not much margin to be gained. Most feel that the plants have been taking to their limits over the last 30 years by sharpening pencils and brute force.
- Tools don't need to be licensable, but need to be accepted as accurate benchmark.

Challenge Problems

- VERA is needed for the problems that haven't occurred yet. A tool with general application with demonstrated performance in specific technical areas (such as challenge problems) is important to future applications.
- FAD affects the fluid flow field and possibly the pin power distributions. These may lead to different results for CIPS, CILC, and GTRF. Has CASL considered challenge problem interactions?
- Today's M&S tools often only provide a relative gauge of risk to a particular problem, and often fail to identify the actual margin to its occurrence. Changes are often characterized as small and insignificant, but without knowing the actual margin, the aggregate changes over time may result in unexpectedly encountering the problem or exceeding a limit.

Resources

- The economics of nuclear power is dominated by the electrical generation of the reactor. The costs associated with fuel inefficiency, margin gain, dealing with operational issues, and maintaining high capacity factors are very small compared to the value of keeping the plant online and at full power. The value added by CASL is not the value of the uprate, per se, but merely the incremental cost of dealing with CIPS when uprated. While the operational challenge problems are important, efforts focused on improving the economics of nuclear energy should be focused on capacity factors, uprates, new reactors, and other methods of increasing the net output of the nation's fleet of NPPs.

Interests

- Interested in moving beyond nodal methods for neutronics – need a new generation of tools. Quantification of the limitations of current nodal methods and training simulators is important.
- An integrated system view is more important than high-fidelity models. The uncertainty in models for plant response to accident scenarios is much larger than the uncertainty in pin power distribution, for instance. This is not really CASL's focus.
- Would like VERA to be validated with a dynamic application range and accepted for use by NRC. Can the code consistently perform with an understanding of the application range and uncertainty quantification?

Interfaces

- The BOA version 3 GUI is useful because it has the EPRI BOA methodology embedded. It is not just a software product.
- It is critical for high fidelity analyses to present the results and data to the analyst in a way which supports accurate decision making. Visualization can be important in aiding in this data digestion. Graphical software can expose a phenomenon that the engineer would otherwise not have recognized.
- Potential end users may already have invested significantly in the development of models and input for current codes. It would be valuable to be able to use the existing models directly or provide a conversion capability to limit the amount of resources wasted on developing models repetitively.
- Visualization is important for complex phenomenon. If you can see it, you can understand it more easily.
- In a safety related environment, the independent verification of GUI-based input is problematic. There must be an audit trail of input, output, and information generation and transfer.

Philosophy

- To do something important, CASL needs to provide practical solutions to current industry problems. Solutions are not found in simulation alone, but through three areas:
    1. Modeling and Simulation – CASL focus
    2. Analysis – Applying the M&S to real problems
    3. Problem evaluation – Understanding and implementing methodological and licensing bases changes based on new analyses
- A higher fidelity, coupled tool does not by itself provide a problem solution, nor necessarily result in lower uncertainties.
- Use should NOT be limited by proprietary data. Nothing in the interface should be proprietary. If the code has IP hooks, some members will NOT use it.
- Ideally, VERA could use fast adequate models, and then use of the VUQ tools could tell the user where the model or method needs refinement.
- Need comparable and consistent solutions on different geometric scales (pin-grid-assembly-core) and different time scales (milliseconds-minutes-years).

- 'New' is bad. Many power plant organizations and other areas directly performing safety related tasks are adverse to change.
- To develop a new way of observing and learning from nuclear reactor simulation, the workflow of going from what is known to what needs to be known is quite different and should supersede the workflows of current analyses and processes.

CONCLUSIONS

The Industry Council Workflow Project has been very successful in meeting its core objectives established at its inception. IC members willingly volunteered to participate and were gracious hosts for CASL staff during the Phase 1 meetings. They provided a great amount of detailed information about the current analyses performed in their organizations. They also provided professional opinions, both support and criticisms, for the CASL goals, highlighting what capabilities they felt are important to achieving a successful product for the nuclear industry. All of this information has been captured in this report, with an effort to collect and organize similar ideas. This document will be a resource for CASL staff to learn about industry practices and to evaluate its development goals and project plans against the opinions of those it aims to support.