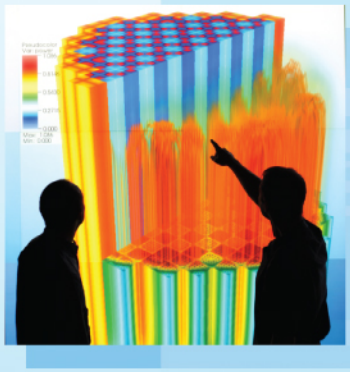
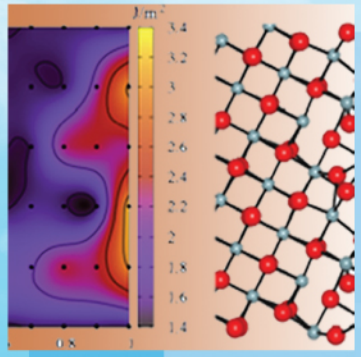
CASL-U-2014-0185-000

Power uprates
and plant life extension



Engineering design
and analysis

Science-enabling
high performance
computing



Fundamental science

Plant operational data

Consortium for Advanced Simulation of LWRs

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VERA-CS Validation Plan

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U.S. DEPARTMENT OF ENERGY | Nuclear Energy

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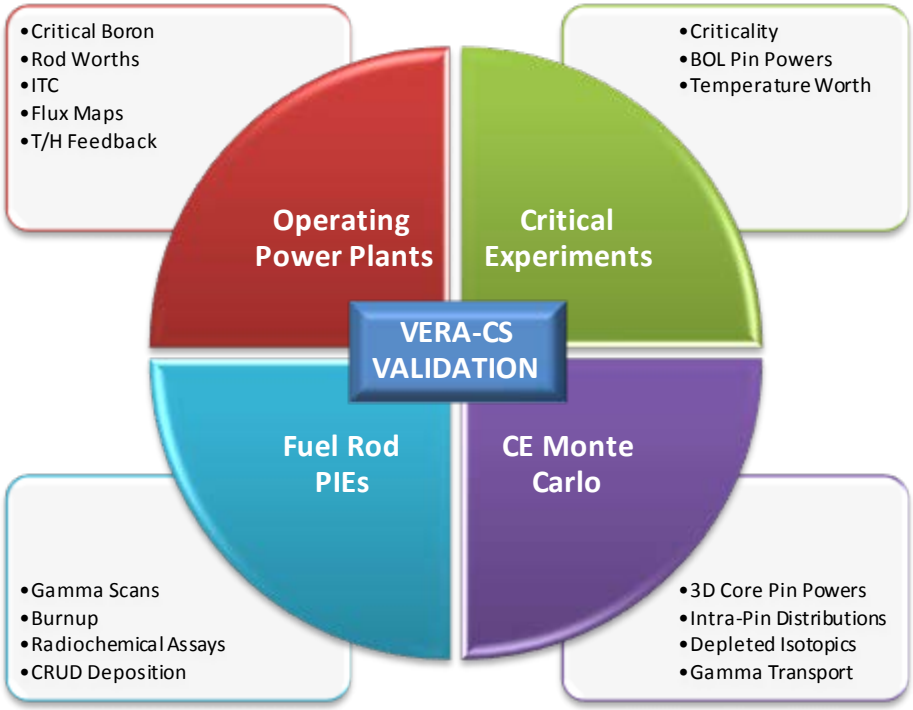
EXECUTIVE SUMMARY

The VERA core simulator, VERA-CS, provides the capability for steady-state reactor core simulation for operating nuclear power plant (PWR) conditions across multiple fuel cycles. This capability parallels that of industry core simulators which are typically licensed by the NRC for core design, core surveillance/monitoring activities, and safety-related calculations that support accident analyses. The licensing basis for these codes for these applications is a rigorous process of validating the software against measured data and quantifying the resulting biases and uncertainties. This document provides an initial plan for performing similar activities with VERA-CS, in a manner consistent with benchmarking activities performed in the nuclear industry.

Compared to industry methods, VERA-CS employs high-fidelity physics approaches and direct feedback couplings to obtain a more accurate and consistent solution for pin-wise quantities in a reactor core (as opposed to nodal method results based on pin power and exposure reconstruction). However, the measured data for validation of core simulators are limited in spatial fidelity and application space, resulting in the need for a somewhat combinatorial approach to overall validation.

The goal of this validation plan is to establish confidence in VERA-CS for simulating the standard evolutions of normal power plant operations, and to demonstrate this confidence to CASL stakeholders, including end users and project sponsors. It includes four validation components, shown, shown below:



Successful completion of a broad range of these components will demonstrate that CASL has provided a reliable and accurate core simulation capability that can be used by the nuclear power industry for reactor core analysis. The accuracy of the pin-wise quantities and isotopics will be inferred from a combination of the above comparisons. These results will also form the initial foundation for an end user's own licensing basis calculations, should the need arise.

It is not the intent of this plan to perform all of the needed benchmarking and uncertainty calculations to support the licensing of VERA-CS by CASL for any particular application.

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ACRONYMS

2D	Two-Dimensional
3D	Three-Dimensional
AIC	Ag-In-Cd control rod material
ANL	Argonne National Laboratory
APSRs	Axial Power Shaping Rods
ARO	All-Rods-Out
ATM	Approved Testing Material
BAF	Bottom of Active Fuel
BEAVRS	Benchmark for Evaluation and Validation of Reactor Simulations
BEP	Bottom End Plug
B&W	Babcock & Wilcox
BOL	Beginning-of-Life (fresh fuel)
BOC	Beginning-of-Cycle
BWR	Boiling Water Reactor
CIPS	Crud Induced Power Shift
CASL	Consortium for Advanced Simulation of Light Water Reactors
CE	Continuous Energy or Combustion Engineering
CEA	Commissariat à l'Energie Atomique or Control Element Assembly
CET	Core Exit Thermocouple
CNS1	Catawba Nuclear Station Unit 1
CZP	Cold Zero Power
DBW	Differential Boron (Reactivity) Worth
DRWM	Dynamic Rod Worth Measurement
DOE	U.S. Department of Energy
EFPD	Effective Full Power Days
EOC	End-of-Cycle
FSI	Flux Suppression Insert
GE-VNC	General Electric - Vallecitos
GWd/MT	Gigawatt-Day per Metric Ton (of initial heavy metal loading)
HEDL	Hanford Engineering and Development Laboratory
HFP	Hot-Full-Power
HZP	Hot-Zero-Power
ICSBEP	International Criticality Safety Benchmark Evaluation Project
IFBA	Integral Fuel Burnable Absorber
INL	Idaho National Laboratory
IRPHEP	International Reactor Physics Experiment Evaluation Project
ITC	Isothermal Temperature Coefficient (of Reactivity)
JAERI	Japan Atomic Energy Research Institute
JSI	Jožef Stefan Institute
KAPL	Knolls Atomic Power Laboratory
LANL	Los Alamos National Laboratory
LEU	Low Enriched Uranium (implies UO_2)
LTA	Lead Test Assembly
LWR	Light Water Reactor
M&S	Methods and Software

MCC	Materials Characterization Center at PNL
MIT	Massachusetts Institute of Technology
MNS1	McGuire Nuclear Station
MOX	Mixed Oxide Fuel
NEK	Nuklearna Elektrarna Krško
NNSA	National Nuclear Security Administration
NRC	U.S. Nuclear Regulatory Commission
OD	Outer Diameter
OECD/NEA	Organization for Economic Co-operation and Development Nuclear Energy Agent
OLCF	Oak Ridge Leadership Computing Facility
ONS	Oconee Nuclear Station
ORNL	Oak Ridge National Laboratory
PCM	Percent mille (1/1000th of a percent)
PIE	Post Irradiation Exam
PHI	Physics Integration Focus Area
PLSA	Part-Length Shielded Assembly
PNL	Pacific Northwest Laboratory
PPM	Parts per million (usually for soluble boron)
PWR	Pressurized Water Reactor
RCA	Radiochemical Assay
RCF	Reactor Critical Facility (at RPI)
RCCA	Rod Cluster Control Assembly
RPI	Rensselaer Polytechnic Institute
RPV	Reactor Pressure Vessel
SNU	Seoul National University
SPERT	Special Power Excursion Reaction Test
SPND	Self-Powered Neutron Detector
T/H	Thermal-Hydraulics
TMI-1	Three Mile Island Nuclear Station Unit 1
TPBAR	Tritium Producing Burnable Absorber Rod
TVA	Tennessee Valley Authority
VERA	Virtual Environment for Reactor Applications
VERA-CS	VERA (Reactor) Core Simulator
WABA	Wet Annular Burnable Absorber
WBN1	Watts Bar Nuclear Plant Unit 1
ZPPT	Zero Power Physics Tests

1. PURPOSE

The Consortium for Advanced Simulation of Light Water Reactors (CASL) is developing a collection of methods and software (M&S) tools known as VERA, the Virtual Environment for Reactor Applications. The core simulator component of VERA, referred to as VERA-CS, provides the capability for pseudo-steady-state simulation of Pressurized Water Reactor (PWR) cores for operating nuclear power plant conditions across multiple fuel cycles, including plant startup testing, full power operations, power maneuvering or load follow, and finally core reload and fuel discharge. This capability parallels that of industry core simulators, which are typically licensed by the U.S. Nuclear Regulatory Commission (NRC) for core design, core surveillance/monitoring activities, and safety-related calculations that support accident analyses. The licensing basis for these codes for this application is a rigorous process of validating the software against measured data and quantifying the resulting biases and uncertainties. This document provides an initial plan for performing similar activities with VERA-CS, in a manner consistent with, but not fully inclusive of, benchmarking activities performed in the nuclear industry.

The purpose of this validation plan is to establish confidence in VERA-CS for simulating the standard evolutions of normal power plant operations, and to demonstrate this confidence to CASL stakeholders, including end users and project sponsors. It includes four validation components, shown in Figure 1-1, and described in the subsequent text:

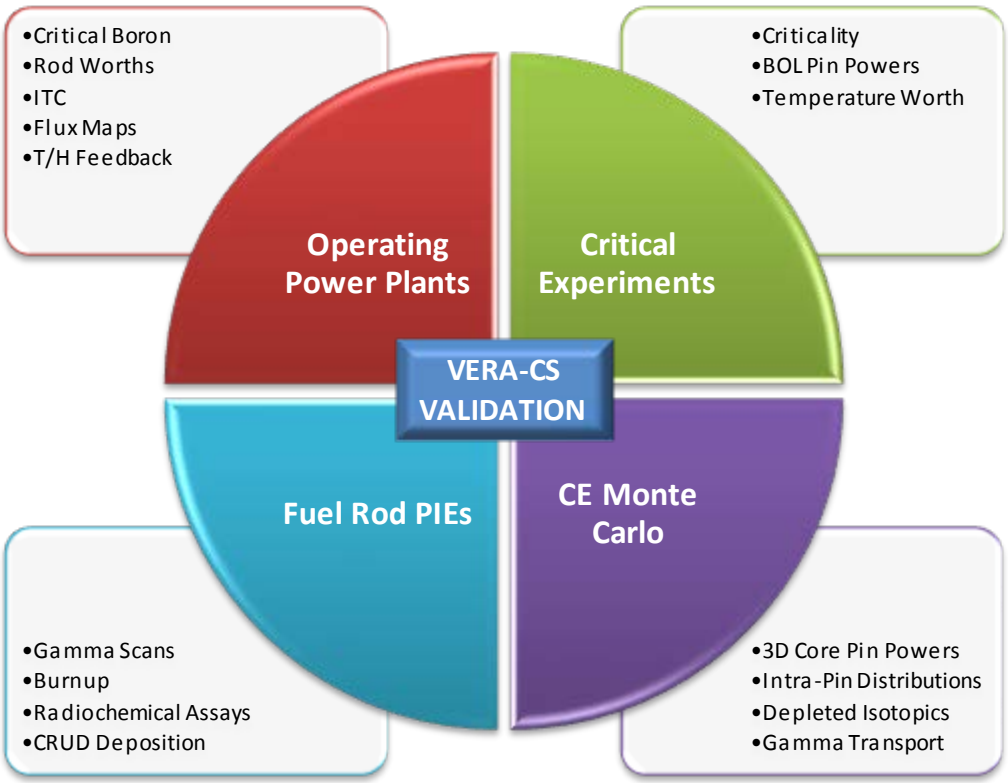


Figure 1-1 Four Components of VERA-CS Validation

VERA-CS results will be compared to the following sources:

1) Measured data from operating nuclear power plants (Section 3). This includes critical soluble boron concentrations, beginning-of-cycle (BOC) physics parameters such as control rod worths and temperature coefficients, and measured fission rate responses from in-core instrumentation.
2) Measured data from experiments with small critical nuclear reactors (Section 4). This includes critical conditions, fuel rod fission rate distributions, control rod or burnable poison worths, and isothermal temperature coefficients.
3) Measured isotopics in fuel after being irradiated in a nuclear power plant (Section 5). This includes gamma scans of ^{137}Cs activity, burnup based on ^{148}Nd concentrations, and full radiochemical assays (RCA) of the major actinides and fission products.
4) Calculated quantities on fine scales from continuous energy (CE) Monte Carlo methods (Section 6). This includes 3D core pin-by-pin fission rates at operating conditions, intra-pin distributions of fission and capture rates, reactivity and pin power distributions of depleted fuel, and support for other capabilities such as gamma transport and thick radial core support structure effects, for which there is currently no known measurements to benchmark against.

The validation will be performed using the integrated core simulator product whenever possible, including the VERA common input and output files and T/H couplings when needed. While this may be impractical for some activities (i.e. a non-standard experiment geometry), the validation should encompass the entire product as much as possible. Validation of single physics such as thermal-hydraulics and fuel mechanics are not included in this validation plan (as is typical for the validation and licensing of core simulators). The “neutronics-only” validation cases (i.e. critical experiments) included are those which can be simulated with the VERA-CS input, considered to be reactors without power and not needing feedback from the other physics. Validation of the thermal-hydraulics and fuel mechanics outside of reactor conditions will be performed elsewhere.

Successful completion of a broad range of each of these components will demonstrate that CASL has provided a reliable and accurate core simulation capability that can be used by anyone in the nuclear power industry for reactor core analysis. In particular the typical quantities produced by core simulators, reactivity, power distribution, and physics parameters will be substantially qualified. These results will also form the initial basis for an end user’s own licensing basis calculations, should the need arise.

It is not the intent of this plan to perform all of the needed benchmarking and uncertainty calculations to support the licensing of VERA- CS by CASL for any particular application.

2. VALIDATION MATRIX

The VERA-CS validation assessment matrix compares the required capabilities, features, and application range of VERA-CS to the proposed benchmarking activities. Its purpose is to ensure that enough activities are performed to provide confidence that VERA-CS is capable of accurately and reliably performing the functions of a steady-state core simulator, and additionally that its advanced features are also reliable. The capabilities desired for coverage are listed on the left, and the validation activities, described in detail in Sections 3 through 0, are shown across the top. Coverage is indicated by an 'X' in the corresponding row and column positions.

Ideally, all capabilities should be covered by at least one activity, to some degree. Note that some features are confirmed indirectly, such as fuel temperature or coolant density feedback, because measurements may not exist. However, in these cases, successful prediction of the fission rate distribution indirectly validates the predicted T/H conditions of the problems, to the degree obtainable from that activity. More detailed validation of the individual physics capabilities (neutronics, thermal-hydraulics, fuel mechanics, etc.) will be performed elsewhere.

Due to budget and time constraints, it is not intended that ALL activities listed in the matrix will be performed, but only a minimum of those which are required to provide confidence to the CASL stakeholders that VERA-CS is reliable and accurate for these types of analyses. The optional activities (mostly critical experiments) are shaded. In general, the priorities for the activities for each component are decreasing from left to right, meaning the cases on the left side of each section be performed first.

The VERA-CS validation assessment matrix is provided on the following page in Figure 2-1.

Capabilities	Operating Power Plants								Critical Experiments																Post-Irradiation Exams					CE Monte Carlo			
	Watts Bar	BEAVRS	Catawba	McGuire	Westinghouse 3-Loop	Krsko	B&W-Type	CE-Type	B&W	Helstrand	KRITZ	DIMPLE	VENUS	IPEN/MB-01	RPI	SPERT III	Strawbridge & Barry	Saxton	CREOLE	EPICURE	CAMELEON	CROCUS	JAERI TCA	ICSBEP	Catawba MOX LTAs	Three Mile Island	MALIBU	Robinson	Calvert Cliffs	Pin-by-Pin Fission Rates	Intra-Pin Distributions	Depleted Isotopics	Misc. Applications
PWR Types																																	
Westinghouse 4-Loop	X	X	X	X																					X					X			
Westinghouse 3-Loop					X								X															X					
Westinghouse 2-Loop						X																								X			
Babcock & Wilcox (B&W)							X																			X							
Combustion Engineering (CE)								X							X														X				
Fuel Assembly Types																																	
17x17	X	X	X	X	X				X										X						X					X	X	X	
16x16						X																								X			
16x16 CE								X	X						X																		
15x15					X																						X	X					
15x15 B&W							X		X																	X							
14x14 CE								X							X														X				
Mixed Oxide Fuel (MOX)			X								X		X					X	X	X					X		X						
Burnable Poison Types																																	
Pyrex	X	X	X	X	X	X			X											X										X			
IFBA	X		X	X	X	X																			X					X			
WABA	X		X	X	X																				X					X			
Solid B4C-AL2O3							X	X																	X	X			X				
Gadolinia							X		X				X							X	X		X			X	X						
Erbia								X							X																		
Control Rod Types																																	
AIC	X	X	X	X	X	X	X		X					X				X		X	X					X				X			
B4C	X		X					X	X					X						X	X	X											
Hybrid	X		X																						X					X			
Gray							X														X					X				X			
Hafnium					X																												
Incore Instrument Types																																	
Moveable	X	X	X	X	X	X																			X					X			
Fixed	X						X	X	X																	X							
Radial Reflector Types																																	
Thin Baffle	X	X	X	X	X	X	X	X				X	X												X	X				X			
Thick (Heavy) Shroud																																	X
Physics Components																																	
Neutron Transport	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Gamma Transport																																	X
Coolant Density Feedback	X	X	X	X	X	X	X	X			X								X				X		X	X	X	X	X	X			
Fuel Temperature Feedback	X	X	X	X	X	X	X	X			X								X				X		X	X	X	X	X	X	X		
Isotopic Depletion	X	X	X	X	X	X	X	X																	X	X	X	X	X		X	X	
Xenon Concentration	X	X	X	X	X	X	X	X																								X	
Shutdown Decay	X	X	X	X	X	X	X	X																	X	X	X	X	X				
Physics Results																																	
Reactivity	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X					X	X	X	
Assembly Power Distribution	X	X	X	X	X	X	X	X																	X					X			
Pin Power Distribution									X		X	X	X	X	X			X	X	X	X									X	X	X	
Intra-Pin Power Distribution																															X		
Pin Burnup Distribution																									X	X	X	X	X		X		
Intra-Pin Burnup Distribution																													X		X		
Incore Instrumentation Response	X	X	X	X	X	X	X	X	X				X						X						X	X							
Excore Instrumentation Response			X	X								X	X																				X
Control Rod Worth	X	X	X	X	X	X	X	X	X									X		X	X	X			X					X			
Temperature Coefficient	X	X	X	X	X	X	X	X			X			X					X				X		X								

Validation Activities

Figure 2-1 VERA-CS Validation Assessment Matrix

3. OPERATING POWER PLANTS

Measurement data from operating nuclear power plants provides the best and broadest range of core simulator validation data. The CASL consortium is working with several stakeholders who own and/or operate PWR power plants and may be willing to collaborate for validation of VERA-CS. These may include:

- CASL partners: Tennessee Valley Authority (TVA)
- CASL Industry Council members: AREVA, Dominion, Duke Energy, and Exelon
- Partner (ORNL, Westinghouse) customers or collaborators
- Publicly available specifications

The following sections describe the measurements that are typically available for code comparison for each fuel cycle of an operating PWR. In most cases, existing power plants have been operating for 10-30 fuel cycles depending on the age of the plant. For the older plants, early cycle data may not be available.

Critical Soluble Boron Concentration

PWR's control excess reactivity with boric acid, H_3BO_3, dissolved in the primary coolant. The measured boron concentration (along with the regulating bank position, inlet temperature, and power level) provides a reliable metric for validation of calculated reactivity. The boron concentration is typically measured very often, perhaps daily or more frequently for power maneuvers, and during specific startup tests. An example of a typical boron letdown curve for a fuel cycle is provided in Figure 3-1 below. Because the change in boron can be slow (due to the large volume of the primary system), instantaneous reactivity changes are compensated by small movements in the regulating control bank.

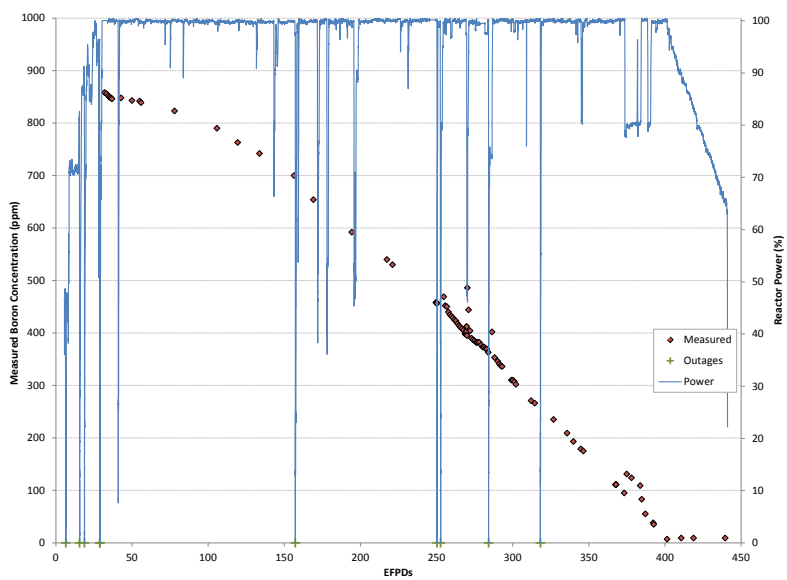


Figure 3-1 Example Critical Soluble Boron Letdown over a Fuel Cycle [3]

A complication to the use of measured boron for reactivity is the depletion of ^{10}B in the coolant, which occurs through thermal neutron capture. Letdown/makeup to the system is relative small, resulting in a continuous decrease in the concentration of ^{10}B atoms relative to ^{11}B over time. If the boron is not refreshed relatively frequently (during outages and large power maneuvers), this effect can become significant on the boron comparisons, since the titration does not discriminate on the

particular isotopes. In order to compensate for this effect, infrequent mass spectrometry is performed on system samples to determine the isotopic content of the coolant. This may occur only monthly, or quarterly, and likely during shutdowns for criticality testing. These ^{10}B measurements can be used to adjust the measured values to ensure consistent comparisons with methods codes (or to adjust the code calculation itself). In general, some compensation for ^{10}B depletion will effectively reduce the measurement uncertainty and improve reactivity comparisons to the measurements.

Analysis of critical boron concentration comparisons over the entire fuel cycle indirectly provides confidence in the isotopic depletion models. A similar analysis performed for a large power maneuver or power coastdown can help validate the coolant and fuel temperature feedback and short term fission product concentrations, and can demonstrate if power-dependent biases exists. Though these are indirect confirmations, there are no other measured data to support validation of these models directly.

3D Power Distribution

PWR's utilize an in-core detector system to measure the 3D core-wide power distribution based on normalized instrument responses. The measurements are derived from electrical signals produced typically by fission chambers that are either moveable or fixed. The moveable system provides a very detailed axial power distribution but only at planned intervals (usually about one month between "flux maps"). The fixed systems are limited to only a small number of axial positions (such as five or seven) but the measurements are taken continuously with time. In either case, only about 1/3rd of the fuel assemblies in the core are instrumented, with the detectors being located in the centermost tube of the assembly. Examples of the core locations of in-core instruments and the processed instrument responses from a flux map are shown in Figure 3-2 and Figure 3-3, respectively.

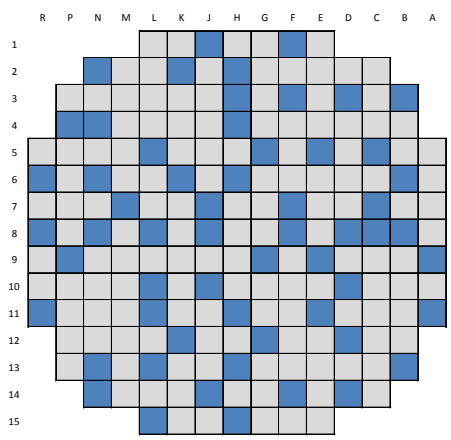


Figure 3-2 Example Instrumented Core Locations [3]

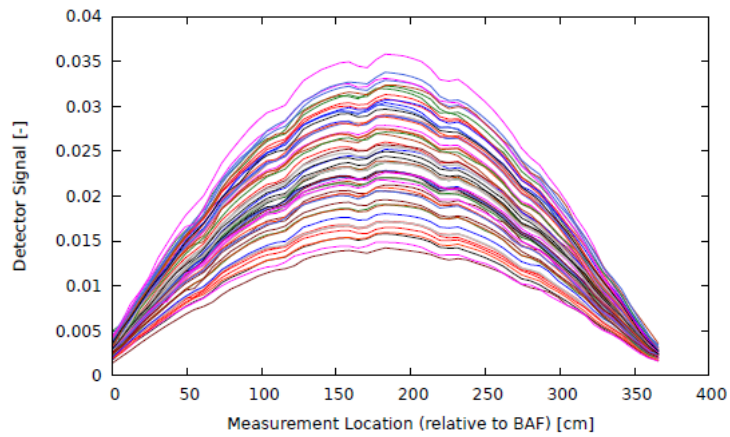


Figure 3-3 Example Moveable In-Core Instrument Signals [7]

For core surveillance activities, the measured fuel assembly powers are inferred from the measured instrument responses and pre-calculated signal-to-power ratios. However this step can be avoided because VERA-CS will directly calculate the normalized instrument response, permitting a direct comparison to the normalized signals. Furthermore, the movable system may produce hundreds of axial points per location, providing an ideal detailed axial comparison to VERA-CS that includes fuel boundaries, poison tips, and spacer grids.

Measured flux maps are taken at regular intervals throughout the cycle to confirm the reactor core is behaving consistently as predicted, and to confirm that the core power distribution agreement is within the uncertainties applied in the safety analyses. Maps are also taken at the beginning-of-cycle (BOC) during power escalation testing to ensure the reactor core is loaded properly and all the poison and control elements are properly positioned. Comparisons to these distributions provide indirect validation of the integrated models in the core simulator.

BOC Startup Physics Tests

Startup physics testing, also known as zero power physics testing (ZPPT), is performed at the beginning of each PWR fuel cycle after the reactor core is refueled. The primary goals of this testing are:

a. To confirm the reactor core is loaded as designed (i.e. detect fuel miss-loadings)
b. To ensure the reactor core meets technical specification limits for temperature reactivity coefficients
c. To validate the reactivity parameters and uncertainties assumed in safety and accident analyses performed for the fuel cycle (particularly the reactivity worths)

The measured quantities resulting from these BOC hot-zero-power (HZP) (isothermal, no xenon, peak samarium, fresh boric acid) tests are:

- Criticality based on regulating control bank position and soluble boron concentration
- Control bank reactivity worths
- Differential soluble boron worth (DBW)
- Isothermal Temperature Coefficient (ITC)

Though these are coarse, core average quantities that do not require high-fidelity modeling and simulation to predict, the startup physics test results are some of the most important validation sources for core simulators. Successful comparison to this data indirectly confirms the localized fuel depletion on the reinsert fuel assemblies, the accuracy of highly rodded conditions, and temperature dependence of the cross section data. Plants using Dynamic Rod Worth Measurement (DRWM) can provide ex-core detector responses for the rod worth tests that could also be used for validation.

BOC Power Escalation Testing

At the beginning of each PWR fuel cycle, a slow and controlled power escalation is performed, stopping at various points (i.e. low, intermediate, full power levels) to perform power distribution tests. Critical boron concentrations (with fresh ^{10}B abundance) are also measured during the power ascension, such that reactivity comparisons can be made given a detailed enough power history. Successful prediction of these parameters verses time and power level indirectly confirms the thermal-hydraulic and fuel temperature models in VERA-CS, as well as time-dependent calculation of the core average xenon concentration. The comparison to the power distributions at each power level also validates the calculated xenon distribution verses time. Finally, the measured radial power distributions also help to validate the shutdown decay of the reinsert fuel assemblies in the new cycle, especially any that were discharged from previous cycles (very long decay times).

Mid-Cycle Estimated Critical Conditions

In addition to BOC criticality, data is also collected each time the reactor is returned to critical following a reactor shutdown. The number of these tests depends on the operation of the cycle, but can provide cycle burnup-dependent criticality measurements, such as the critical boron and control bank positions. Like the hot-full-power (HFP) boron measurements, these HZP criticals can confirm that the accuracy of the core reactivity calculated by VERA-CS is not changing with cycle exposure.

Power Maneuvering

Occasionally operating plants perform power maneuvers to support maintenance activities, perform equipment testing, load following, or to continue operating with some equipment out of service. Though these maneuvers are seldom, they provide valuable data on plant reactivity and power distribution (for fixed in-core detector systems) for a variety of reactor power levels and control bank positions. To accurately predict the critical boron concentrations as a function of time, the neutronic feedback from thermal-hydraulics and fuel temperatures must the accurate as a function of power level, and the time-dependent xenon concentration and distribution must be also be calculated correctly. In addition, at the lowest power levels, the reactor will typically utilize deeper control rods so that errors in control rod worth will be evident. Without power maneuvers, startup power escalations, and coastdown data, all of the validation data will typically be only at 0% and 100% full power with no data in between.

Additional measurements may be available on a plant specific basis. Secondary quantities for comparison may include quadrant power tilts, core/assembly exit coolant temperatures, HFP temperature coefficients, etc.

The measured power plant data is essentially global core quantities and measured signals from a limited number of moveable or fixed in-core fission chambers. These provide the validation basis for VERA-CS over coarse spatial and time scales, but provide very little in terms of validation of small scale quantities such as pin-wise distributions of power, burnup, temperature, etc. Thus, this plant data alone is not enough for a thorough validation and needs to be supplemented with the other three components of this plan, which follow in the subsequent sections.

3.1 Watts Bar Nuclear Plant

The Watts Bar Nuclear Plant in Spring City, TN is owned and operated by the Tennessee Valley Authority (TVA), a CASL core partner. Watts Bar was selected as CASL's "Physical Reactor" for initial benchmarking activities. Unit 1 was the last commercial nuclear unit to come online in the 20th century, and Unit 2 will be the first to come online in the 21st century. Groundbreaking on the Watts Bar site occurred in 1972 [1]. Because Unit 1 has been operating for decades, and Unit 2 is under construction, they are treated separately in this document.

Figure 3-4 Watts Bar Nuclear Plant [1]

3.1.1 Unit 1

Watts Bar Nuclear Unit 1 (WBN1) is a traditional Westinghouse 4-loop PWR with an ice condenser containment design, one of the most common reactor designs in the U.S. today. It is currently licensed to 3459 MW_{th} power [2]. It began commercial operation in May of 1996 [1], and is currently operating Cycle 13.

WBN1 has 193 fuel assemblies of the 17x17 type, has used Pyrex, IFBA, and WABA burnable poisons, and has 57 AIC/B_4C hybrid rod cluster control assemblies (RCCAs). It has a moveable in-core detector system for power distribution measurement. WBN1 is also the only commercial reactor in the U.S. to contain Tritium Producing Burnable Absorber Rods (TPBARs) for the DOE/NNSA's tritium program [3].

The Unit 1 (and 2) reactor core layout and fuel assembly designs are shown in Figure 3-5 and Figure 3-6 [4].

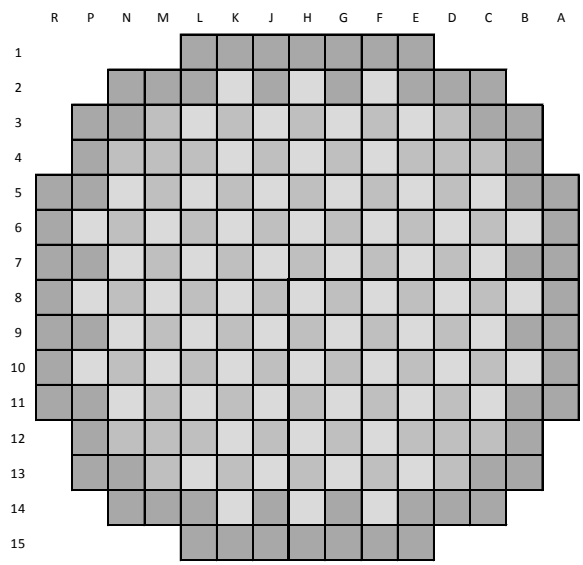


Figure 3-5 Westinghouse 4-Loop Core [4]

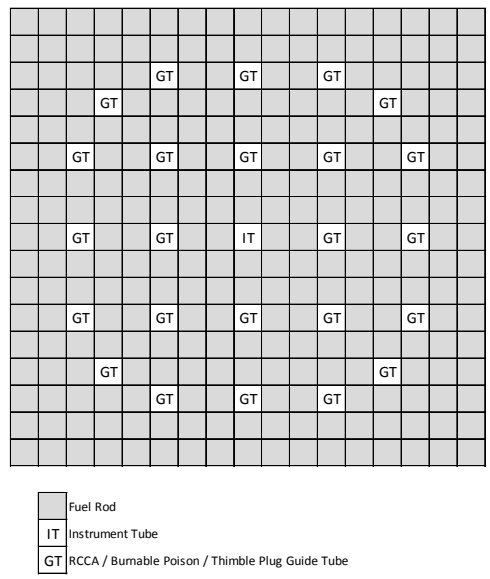


Figure 3-6 17x17 Fuel Assembly [4]

Because TVA is a CASL core partner, all the detailed data needed to perform a thorough validation of VERA-CS should be available. However, the detailed 600 axial level flux measurements are not available and some of the data from the early cycles is unavailable (such as startup history or ^{10}B isotopics) [3].

3.1.2 Unit 2

Watts Bar Nuclear Unit 2 (WBN2) is currently under construction and scheduled to be completed in December 2015. The initial startup of Unit 2 will provide detailed data for VERA-CS validation – no other plant has started up with a clean core in almost 20 years, so instrument and test data availability and quality is expected to be very high.

The reactor design and fuel assembly type for Unit 2 are identical to Unit 1 [4]. One significant difference between the two units is WBN2 will utilize fixed in-core instrumentation with Vanadium self-powered detectors. Each instrument location will have five detectors and one Core Exit Thermocouple (CET) [5, 6]. Also, TPBARs are not currently planned for WBN2.

3.2 BEAVRS

The Benchmark for Evaluation And Validation of Reactor Simulations (BEAVRS) is a publicly available reactor specification provided by the Massachusetts Institute of Technology (MIT) Computational Reactor Physics Group [7]. It contains two cycles of detailed geometry and measurements from an unnamed utility's PWR. The BEAVRS reactor is a traditional Westinghouse 4-loop PWR very similar to WBN1, with core and fuel layouts as shown in Figure 3-5 and Figure 3-6. The three region core loading and fuel enrichments are also similar to WBN1. The control rod type is AIC and only Pyrex is used as a discrete burnable absorber, in different lattice patterns than WBN1.

The measured data provided for BEAVRS includes Cycles 1 and 2 ZPPT results, power escalation and HFP measured flux maps, and HFP critical boron concentration measurements for both cycles. The power history for each cycle is provided, but the regulating bank history is not. The flux map data provided is the processed 61 level data.

Because BEAVRS is a public release from an unnamed utility, its data is limited and support is not readily available for problems or questions. Also, it is unlikely to be continued to any more cycles, which limits the long term value that could be gained (as opposed to benchmarking against a plant that is still operating, in cooperation with an end user). Nevertheless, this benchmark is becoming an industry standard for validation of advanced codes [8, 9, 10, 11] and VERA-CS should be able to easily perform this analysis.

3.3 Catawba Nuclear Station

Catawba Nuclear Station Units 1 and 2 (CNS1 and CNS2) are also traditional Westinghouse 4-loop PWRs with ice condenser containments [2, 12]. These reactors are operated by Duke Energy 18 miles from Charlotte in York, SC. The units are nearly identical, beginning operation in 1985 and 1986, respectively. Each unit is currently licensed to operate at 3411 MW_{th}. Current operating fuel cycles are Cycle 22 for Unit 1 and Cycle 20 for Unit 2.

Figure 3-7 Catawba Nuclear Station [12]

The initial reactor and fuel designs are nearly identical between WBN1, BEAVRS, and CNS1 and CNS2. They are sister plants of the same vintage, though WBN1 is about a decade newer. There are 193 fuel assemblies, 53 hybrid AIC/B_4C control rod clusters, and 58 movable in-core detector locations [13]. However, CNS1 is notable in that it operated two recent cycles with four mixed oxide (MOX) lead test assemblies (LTAs) derived from weapons grade plutonium. It was also one of the earliest PWRs to exhibit signs of CIPS (Cycle 8). CNS has utilized both Westinghouse and

AREVA 17x17 fuel assemblies, with burnable poisons including Pyrex, IFBA/WABA, and boron carbide/alumina matrix, Al_2O_3-B_4C [14].

Duke Energy is a member of the CASL Industry Council. Duke also operates seven nuclear plants, and historically has performed nearly all the design and licensing work for Catawba, McGuire, and Oconee, including M&S validation activities [13]. The VERA-CS validation will benefit significantly from the expertise and data available from Duke Energy, and the inclusion of Catawba with both CIPS and MOX fuel is a natural choice for benchmarking with VERA-CS. More discussion of the MOX LTA program is included in Section 5.1.

3.4 McGuire Nuclear Station

McGuire Nuclear Station Units 1 and 2 (MNS1 and MNS2) are also traditional Westinghouse 4-loop PWRs with ice condenser containments owned and operated by Duke Energy [2]. The McGuire units are nearly identical to their sister reactors at Catawba. They are located 17 miles from Charlotte in Huntersville, NC. The units are nearly identical, beginning operation in 1981 and 1984, respectively [15]. Each unit is currently licensed to operate at 3411 MW_{th}. Current operating fuel cycles are Cycle 24 for Unit 1 and Cycle 23 for Unit 2. One difference between McGuire and Catawba is that MNS1 uses AIC-only control rods, not hybrids [16].

Figure 3-8 McGuire Nuclear Station [15]

For validation purposes, the reactors at McGuire can be considered to be very similar to those at Catawba. Validation with McGuire data may be optional, but the data would likely be available along with the data from Catawba through collaboration and data exchange with Duke Energy.

3.5 Westinghouse 3-Loop-Type Nuclear Power Plant

A Westinghouse 3-loop PWR has not yet been selected for validation. Examples of such a plant are H.B. Robinson and Shearon Harris (Duke Energy), North Anna and Surry (Dominion), Beaver Valley, Farley, Turkey Point, and V.C. Summer [2].

For instance, North Anna has a thermal power rating of 2940 MW_{th}. The two units began commercial operation in 1978 and 1980, respectively. The reactor core contains 157 17x17 fuel assemblies (Figure 3-10)[17], 48 AIC control rod clusters, with 50 movable in-core detector locations. The 17x17 fuel assemblies are essentially the same as those used in the 4-loop plants (Figure 3-6). For burnable poisons, IFBA is currently used, but discrete solid B_4C-$A1_2O_3$ rods have been used previous cycles, as well as Pyrex [18, 19].

Figure 3-9 North Anna Power Station [17]

Surry is also a Westinghouse 3-loop design, but it is licensed to 2587 MW_{th} [2] and uses 15x15 fuel (Figure 3-11). It has 48 full-length AIC control rod assemblies, and initially had 5 part-length control rod assemblies (which have since been removed). Flux Suppression Inserts (FSI) containing hafnium have also been used in peripheral core locations to suppress the neutron leakage (up until Cycle 21) to minimize neutron fluence on the reactor vessel. IFBA is used beginning in Cycle 21, along with discrete absorbers of Pyrex or B_4C-$A1_2O_3$ rodlets [20].

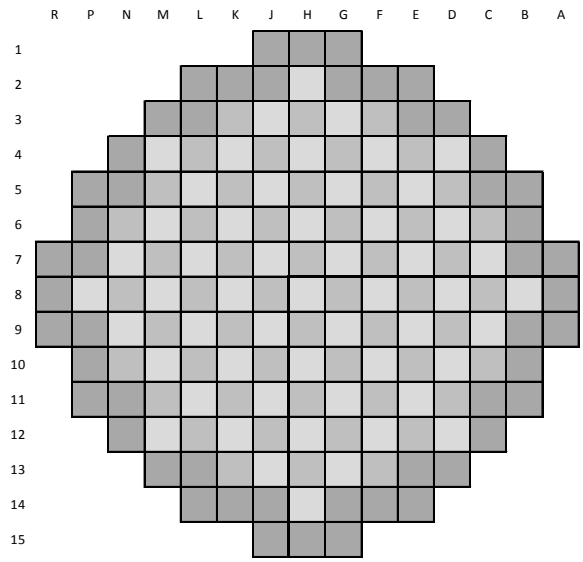


Figure 3-10 Westinghouse 3-Loop Core [18]

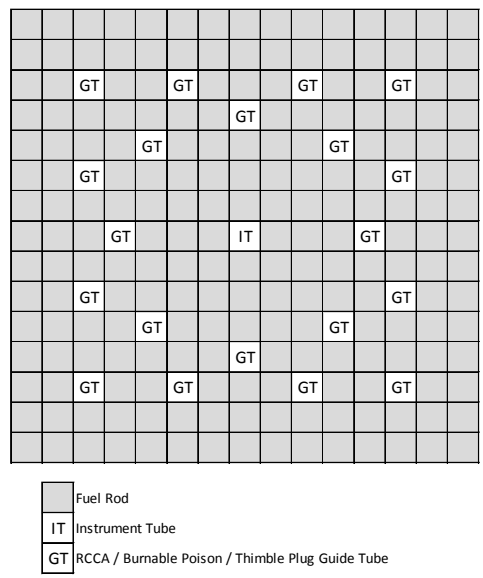


Figure 3-11 Westinghouse 15x15 Fuel [20]

It is noted here that eight high burnup fuel rods irradiated in North Anna were shipped to Studsvik Nuclear's hot cells in Sweden for a detailed post-irradiation exam (PIE). All the rods were characterized non-destructively and three rods were chosen for detailed destructive examinations, including fuel isotopic analyses. In the future North Anna may need to be considered for validation in Section 5 as more information is obtained.

H.B. Robison Unit 2 is also a Westinghouse 3-loop plant that utilizes 15x15 fuel. It may be an interesting choice for validation because PIEs were conducted on nine spent fuel rods discharged from Cycle 2. This is discussed later in Section 5.4. Though this reactor is now owned by Duke Energy, it is not known at this time if the reactor operating history and refueling shuffle maps can be obtained from forty years ago.

Either Robinson, Harris, or the Dominion plants (4 units) would be excellent sources of validation data. Like Duke, Dominion operates many nuclear reactors and brings significant experience in operation and M&S validation. Dominion is also on the CASL Industry Council and may be interested in the testing and validation of VERA-CS for their applications.

3.6 Krško Nuclear Power Plant

The Krško Nuclear Power Plant is a Westinghouse 2-loop PWR operated by Nuklearna Elektrarna Krško (NEK) in Slovenia [21]. Currently a Joint Development Project exists between Westinghouse and the Slovenian Jožef Stefan Institute (JSI) to analyze the measured plant data from Krško with the latest M&S tools, including VERA-CS. This effort is primarily being led by Westinghouse [22]. Krško is very similar to the Point Beach and Prairie Island reactors in the U.S except for the fuel assembly design used (14x14). Krško began commercial operation in 1983.

Figure 3-12 Krško Nuclear Power Plant [21]

The Krško core consists of 121 fuel assemblies arranged as shown in Figure 3-13. The fuel assembly is based on the Westinghouse 16x16 fuel lattice design, with 235 fuel locations, 20 guide thimbles and 1 instrumented thimble, shown in Figure 3-14. The fuel assembly composition is very similar to the 17x17 fuel, except the spacer grids are made of Inconel with type 304 stainless steel sleeves. The presence of moderate neutron absorbers in Inconel and stainless steel leads to a larger flux depression in grid locations compared to Zr-based grids.

The clad is Zircaloy-4 with an outside diameter (OD) of 0.374 in, and a pellet OD of 0.3225 in; the fuel pitch is 0.485 in. This results in an H/U of ~3.6 and to a lower moderated lattice than other typical designs, e.g. ~ 4.0 for Westinghouse standard 17x17 fuel. This drier lattice and the ensuing harder spectrum lead to increased ^{238}U reasonance absorptions and higher Pu production, which can introduce some challenges to the self-shielding methods adopted in typical neutronics codes [22].

The core features 33 Reactivity Control Cluster Assemblies (RCCAs) arranged in seven banks. AIC is used as the neutron absorber material. Burnable poison inserts, containing Pyrex glass with 12.5 w/o B_2O_3, were initially used for reactivity hold-down and power shaping, but in recent cycles IFBA is used. Currently the reactor is in Cycle 27.

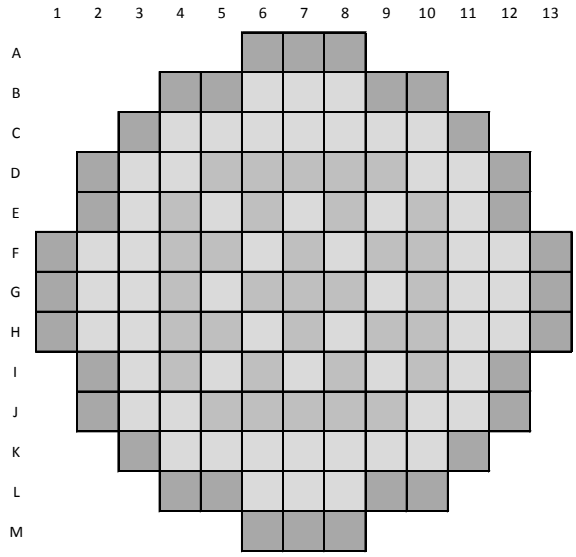


Figure 3-13 Krško 2-Loop Core [22]

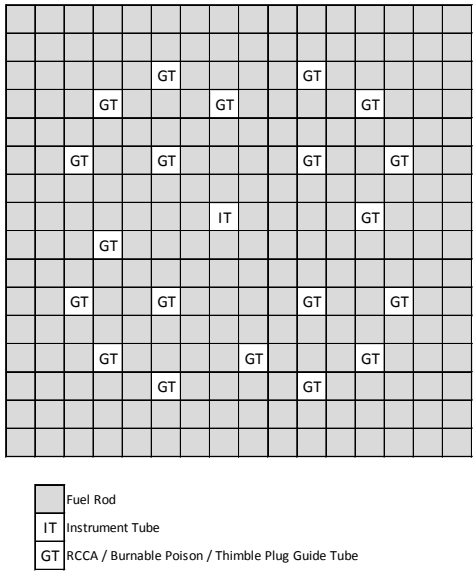


Figure 3-14 Westinghouse 16x16 Fuel [22]

The use of Krško for validation of VERA-CS is an important activity in terms of application of the code by Westinghouse, significant data availability, and mutually beneficial collaboration with engineers outside of CASL. The interest shown by NEK and JSI in VERA-CS makes this an ideal validation source, despite it being a non-U.S. reactor.

3.7 B&W-Type Nuclear Power Plant

A B&W PWR has not yet been selected for validation. Examples of such a plant are Oconee (Duke Energy), Three Mile Island (Exelon), Davis-Besse, Arkansas Unit 1, and Crystal River (now retired)[2]. These plants have 177 B&W 15x15 fuel assemblies with 208 fuel rods and 16 control rod guide tubes (Figure 3-15 and Figure 3-16)[23, 24].

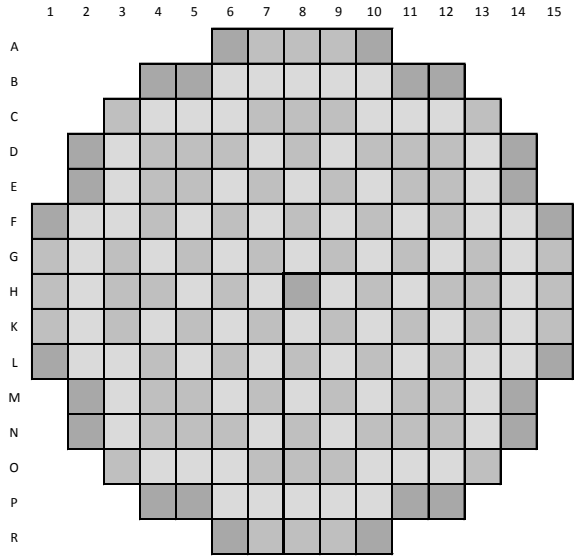


Figure 3-15 B&W Core [23]

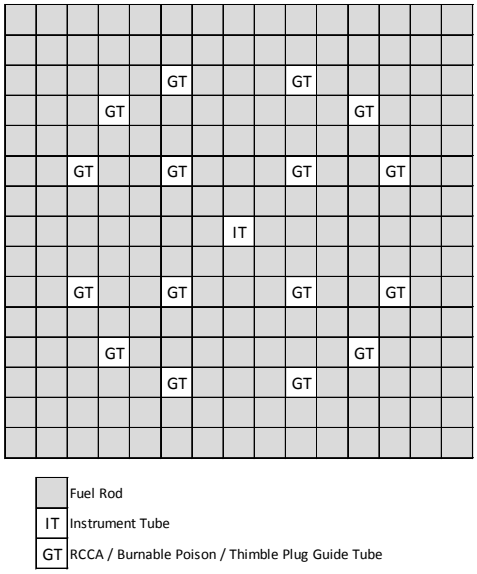


Figure 3-16 B&W 15x15 Fuel Layout [24]

For instance, Oconee Nuclear station (ONS) has three units licensed to 2568 MW_{th} power. It has been operating commercially since 1973 and is located in Seneca, SC [25]. In addition to 177 fuel assemblies, it uses 61 AIC control rod assemblies assigned to seven groups, and eight part-length axial power shaping rods (APSRs) composed of Inconel 600. The APSRs are positioned in the core approximately centered axially during normal operation. Burnable poisons have been discrete solid B_4C-Al_2O_3 rodlets, but Oconee has recently transitions to 24 month cycles and use of the integral absorber gadolinia [23].

Figure 3-17 Oconee Nuclear Station [25]

For in-core instrumentation, ONS uses a fixed system composed 52 self-powered neutron detectors (SPNDs) strings, each containing seven levels of rhodium neutron detectors. Each string also includes a thermocouple and a background "detector". The strings are inserted into the core through guide tubes in the bottom of the reactor vessel, and are left in the reactor continuously for the entire fuel cycle. Over time, the signals must be corrected for the depletion of rhodium, and the strings are replaced when the projected rhodium depletion exceeds a predetermined threshold. This instrumentation is very different than that of the Westinghouse plants and is a needed aspect of validation for VERA-CS [26]. (Note: These detectors are similar to those used in the B&W criticals, Section 4.1.)

ONS would make an ideal validation source for VERA-CS. With three units, and currently operating cycles 28, 27, and 28, there is an extremely large amount of data for comparison. Furthermore, with Duke Energy on the CASL Industry Council, and already assisting in the benchmarking of Catawba and McGuire Nuclear Stations, it would be logical to extend that collaboration to Oconee. Finally, Duke has recently performed M&S validation activities on this data for the licensing of CASMO-4/SIMULATE-3 [23], so the staff there will already have significant expertise and data collected to assist in this activity.

Additionally, Three Mile Island Nuclear Station, Unit 1 (TMI-1) would also provide a good source of validation. It is very similar to Oconee, but has long been operating 24-month cycles with gadolinia as an integral burnable absorber (supports face-to-face adjacent feed assemblies). As discussed in Section 5.2, TMI experienced several fuel rod failures in Cycle 10 which led to a post-irradiation hot cell examination of 4 fuel rods. It is thought that these failures were due to CIPS/CILC in the highest powered fresh fuel rods in the core [56]. Three Mile Island is operated by Exelon, which is also an Industry Council member.

3.8 CE-Type Nuclear Power Plant

A Combustion Engineering (CE) PWR has not yet been selected for validation. Examples of such a plant are Palo Verde (System 80), Waterford Unit 3, Arkansas Nuclear Unit 2, Millstone Power Station Unit 2 (Dominion), and St. Lucie Plant [2].

At Palo Verde [27], the three 2-loop 3990 MW_{th} reactors each have 241 fuel assemblies and 89 control elements assemblies (CEAs). Each fuel assembly contains a 16x16 square array of UO_2 fuel rods, except for five large guide tube locations (which each occupy four lattice locations). These guide tubes may contain control rods or discrete burnable poisons containing B_4C-Al_2O_3. Also, an integral burnable poison may be used such as erbia (Er_2O_3). The control elements may be full strength (B_4C) or part strength (Inconel 625) absorbers [28]. The core and fuel assembly layouts are shown in the following figures.

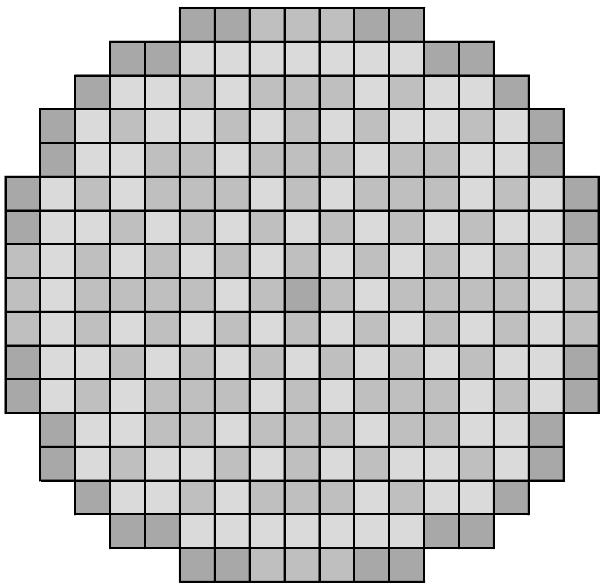


Figure 3-18 CE System 80 Core [28]

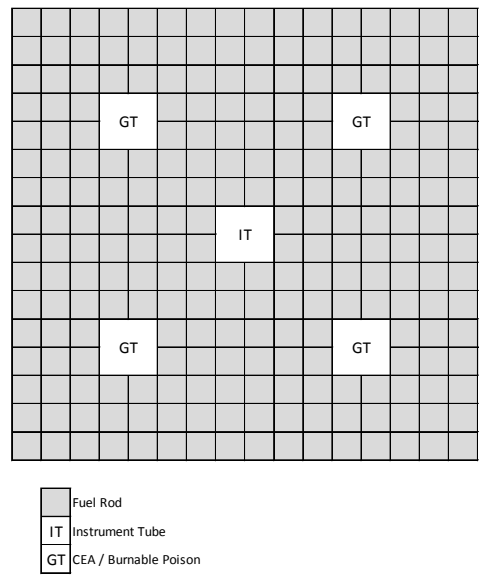


Figure 3-19 CE 16x16 Fuel Layout [27]

Millstone Unit 2 is a 2700 MW_{th} CE plant owned and operated by Dominion, another CASL Industry Council member. Millstone began commercial operation in 1975 [2]. It has 217 14x14 fuel assemblies (Figure 3-20), also with 5 large guide tubes (Figure 3-21).

Additionally, Calvert Cliffs, discussed in Section 5.5, is also a CE 2-loop plant which uses 14x14 fuel assemblies. It began commercial operations in 1974 [2]. It is included in Section 5.5 as a potential source of depletion validation [60,63]. It is not known if the full reactor operating history can be obtained for this reactor.

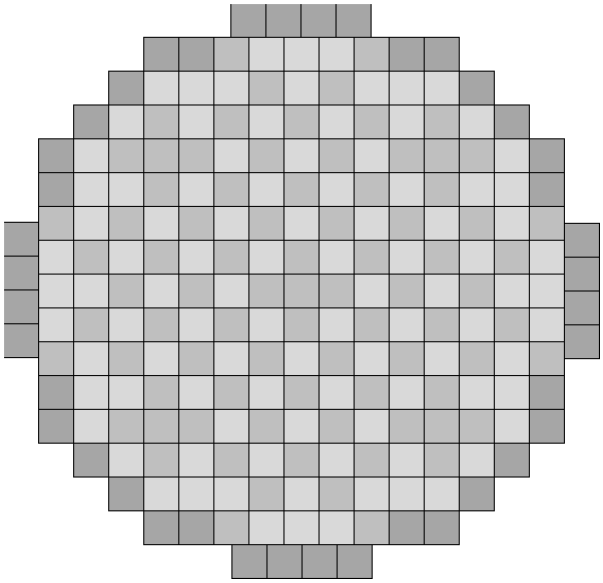


Figure 3-20 CE Core [63]

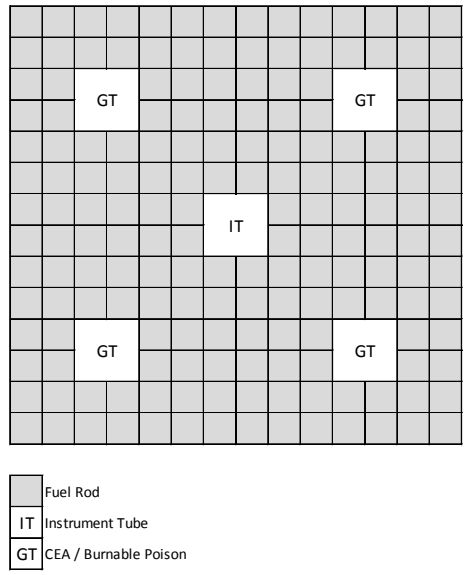


Figure 3-21 CE 14x14 Fuel Layout [63]

4. CRITICAL EXPERIMENTS

Critical experiments are small nuclear reactors typically designed to provide validation data for nuclear methods and software (M&S), particularly for materials and geometries similar to those found in operating nuclear power plants. These experiments are usually performed without power at isothermal conditions and without fuel depletion. If the reaction rate distribution is measured for the fuel rods, it may be done so indirectly through gamma scanning or activation of other materials. Finally, the experiments are usually performed in a manner that can be simulated in 2D.

Simulation of critical experiments is a common practice in the nuclear industry to establish local pin peaking factor uncertainties for the nuclear methodology being licensed, and in particular the 2D lattice physics component. In that regard, VERA-CS is not expected to perform significantly better than the industry licensed codes for many of these problems, but unlike industry lattice physics methods, the performance can be assumed to be more applicable to 3D or other non-experiment conditions.

This section contains a significant number of potential experiments to be used for validation. Not all are required. Many are documented in the International Reactor Physics Experiment Evaluation Project (IRPhEP)[29], previously listed and described for CASL in Reference 30. Others are included here as common validation sources for nuclear industry methods. In some cases the info is at a very high level and will need to include more details when the detailed experiments specifications are obtained. As the validation plan is executed, this section may be modified as new information is obtained.

4.1 Babcock & Wilcox Critical Experiments

4.1.1 1484 – Fuel Storage

The Babcock & Wilcox 1484 critical experiments were designed to provide criticality data to support the long term storage of LWR fuel in spent fuel pools. 20 critical configurations were constructed to provide measured benchmark data for validation of nuclear M&S. The report for the experiments, funded by what is now the U.S. Department of Energy (DOE), was released in 1979 [31].

The twenty critical configurations were built in a core tank with low enriched (2.46%) UO_2 fuel rods and water as the neutron moderator. The rods were clustered into nine LWR-like assemblies in a 3x3 configuration, with variable spacing in between the assemblies, as shown in Figure 4-1. In some configurations, stainless steel or borated aluminum sheets are placed in between the assemblies to simulate a spent for storage configuration. Therefore only a subset of the experiments is consistent with power plant geometries for validation.

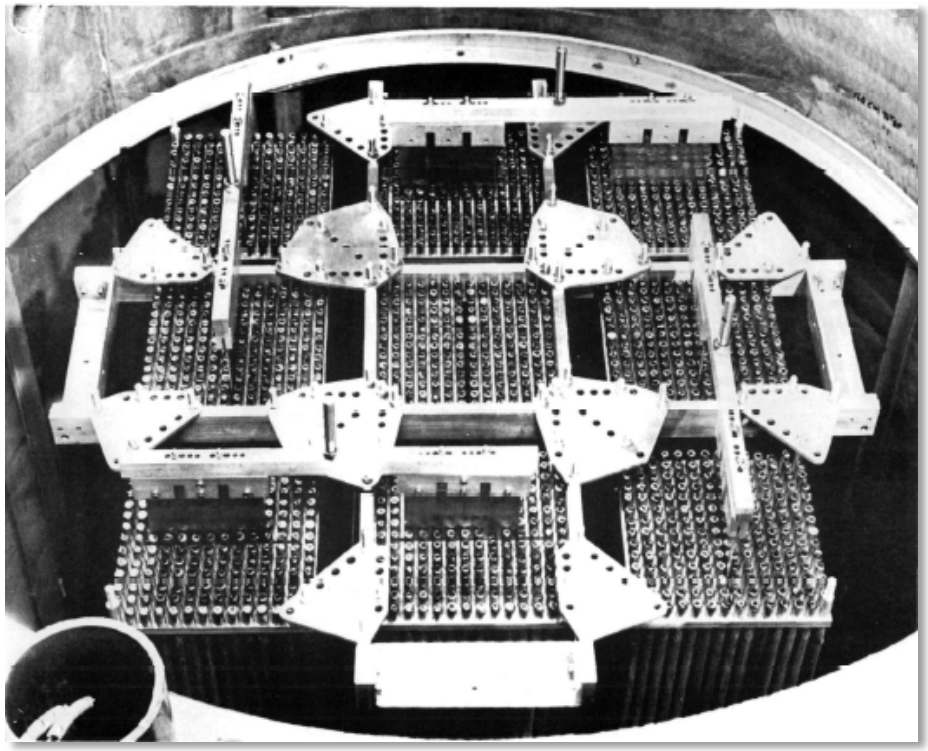


Figure 4-1 Typical B&W 1484 Core Configuration of Unit Assemblies [31]

For each core, criticality was achieved through soluble boron and/or adjustable moderator height. The reactivity worth of the moderator height was also measured so that the results could be adjusted to a uniform water level.

4.1.2 1820 – Urania-Gadolinia Benchmark

The B&W 1810 series of critical experiments were developed by B&W, Duke Power, and DOE to provide beginning-of-life (BOL) benchmark data to support the development of an advanced PWR fuel assembly for extended fuel burnup. This design employed gadolinia as the integral burnable absorber. 23 core configurations were constructed, and the following measurements were taken [32]:

a. Reactivity worths of gadolinia, control, and void rods
b. Core radial power distribution
c. Radial power profiles within a UO_2 pellet containing gadolinia
d. ^{238}U resonance integrals for solid and annual fuel pellets
e. Rhodium in-core detector signals

The experiments were performed at B&W's Lynchburg Research Center using UO_2 fuel rods with 2.46% and 4.02% ^{235}U enrichment. Both solid and annular rods containing 4.0% gadolinia are including in some of the arrangements, and Ag-In-Cd (AIC) and B_4C control rods are also used. The rods are arranged inside of a large core tank with variable moderator height as discussed in the previous section. In some cases, multiple fuel rods are removed to simulate the large water rods in the Combustion Engineering (CE) lattice design. A sample core configuration is shown in Figure 4-2.

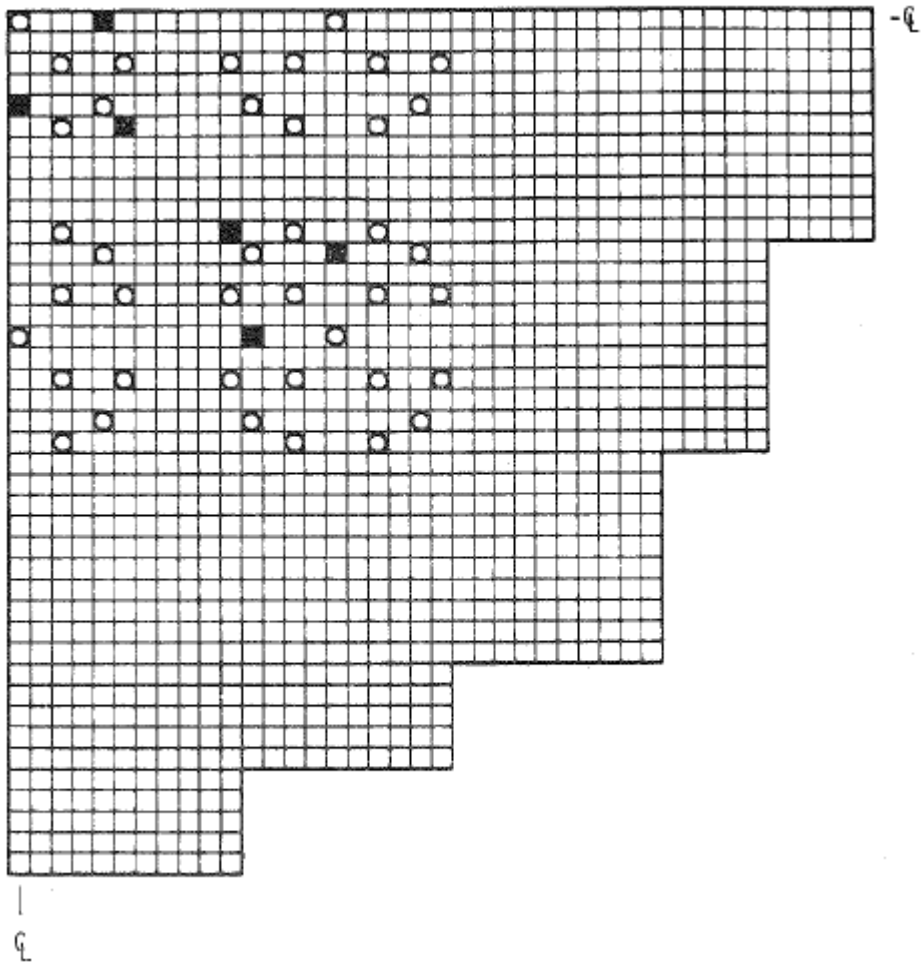


Figure 4-2 B&W 1810 Core 3 Configuration with Gadolinia Rods [32]

4.2 Helstrand Resonance Integral Experiments

The Helstrand experiments were performed in Stockholm, Sweden to determine the effective resonance integrals in uranium metal and oxide fuels in a variety of geometries [33]. The neutron capture rate in ^{238}U was measured by the 105 keV gamma decay of ^{239}Np. Metal rod diameters ranging from 4 mm to 29 mm were included, and UO_2 rod diameters of 8.4, 15.6, and 20.0 mm were also tested. Three rod array geometries were tested with empty or water-filled tubes in the lattice (similar to control rod guide tubes). Measurements were also obtained for the spatial distribution of the resonance absorption within a fuel rod.

While the Helstrand experiments provide more fundamental validation for the VERA-CS cross sections, they do not require the reactor-based modeling approach of the other critical experiments. However, due to the importance of these comparisons, they are included here for now and may be removed at a later date.

4.3 KRITZ LEU and MOX Experiments

The KRITZ reactor operated at Studsvik, Sweden, during the first half of the 1970s. It is comprised of fuel rods in square-pitched lattices in a 5m high, 1.5m diameter cylindrical pressure tank. Figure 4-3 shows the vertical cross section of the KRITZ reactor [34]. The "KRITZ experiments," performed in the period from September 1972 through February 1973, included several series of criticality experiments on light-water-moderated lattices with uranium dioxide rods, mixed-oxide

(MOX) rods, or both, at room temperature and at elevated temperatures up to 245 °C (473 °F) covering temperatures close to the range used in light water reactors. Criticality was obtained by controlling the boron content in the moderator and by adjusting the moderator level. Critical levels were measured at low power, often as low as 10 W, to minimize the activation of the fuel. Measurements of the activated copper wires and gamma scans of the fuel rods were used to determine experimentally the axial buckling. For most of the cores measured relative powers for selected rods, obtained from gamma scans, are provided. The data released describe four experiments: three with uranium rods (KRITZ-1, KRITZ-2:1 and KRITZ-2:13) and one with mixed-oxide rods (KRITZ-2:19). Because of limited data available for KRITZ-1, the IRPhEP handbook focuses on the other three configurations.

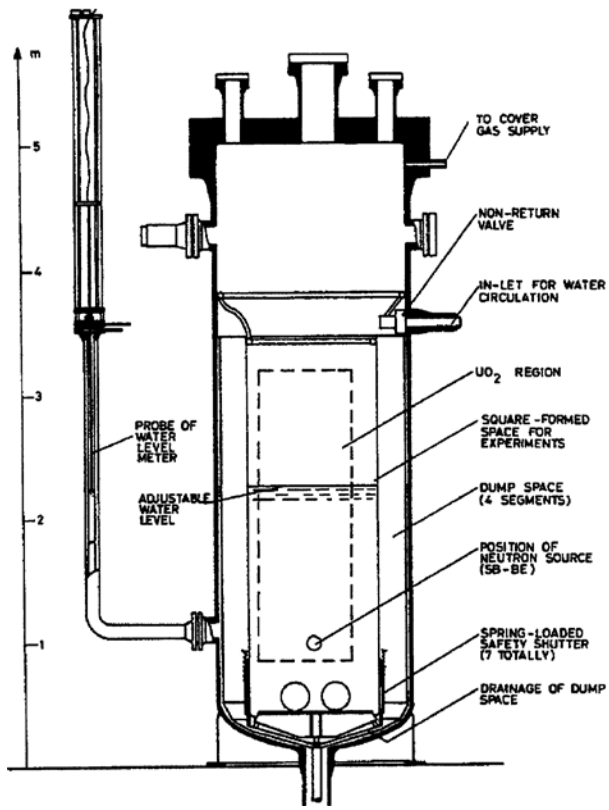


Figure 4-3 Vertical cross-section of the KRITZ reactor [34]

4.3.1 KRITZ-2:19 Experiment on Regular H_2O/Fuel Pin Lattices with MOX Fuel at Temperatures 21.1 and 235.9 °C

The objective of the KRITZ-2:19 experiments was to attain criticality of a rectangular array of mixed-oxide Zircaloy-2 clad fuel rods in light water by regulating the concentration of boron in the moderator and by adjusting the moderator level. Criticality was achieved at isothermal conditions at room temperature (21.1 °C) and at elevated temperature (235.9 °C). Besides the critical boron concentrations and moderator level, the axial buckling was also determined, and the relative rod powers (fission rate distributions) were measured for selected fuel rods. Measurements of the activated copper wires and gamma scans of the fuel rods were used to determine the axial buckling experimentally. The experiment with MOX rods at 18.00 mm pitch is considered. The experiments were performed in the KRITZ reactor in the period from September 1972 through February 1973 [35].

4.3.2 KRITZ-2:1 Experiment on Regular H_2O/Fuel Pin Lattices with LEU Fuel at Temperatures 248.5 °C

The objective of the KRITZ-2:1 experiment was to attain criticality of a rectangular array of low enriched uranium Zircaloy-2 clad fuel rods in light water by regulating the concentration of boron in the moderator and by adjusting the moderator level. Criticality was achieved at isothermal conditions at room temperature (19.7 °C) and at elevated temperature (248.5 °C). Besides the critical boron concentrations and moderator level, the axial buckling was also determined, and the relative rod fission rates were measured for selected fuel rods. Measurements of the activated copper wires and gamma scans of the fuel rods were used to determine the axial buckling experimentally [35].

4.3.3 KRITZ-2:13 Experiment on Regular H_2O/Fuel Pin Lattices with LEU Fuel at Temperature 243 °C

The objective of the KRITZ 2:13 experiment was to attain criticality of a rectangular array of mixed-oxide Zircaloy-2 clad fuel rods in light water by regulating the concentration of boron in the moderator and by adjusting the moderator level. Criticality was achieved at isothermal conditions at room temperature (22.1°C) and at elevated temperature (243.0 °C). Besides the critical boron concentrations and moderator level, the axial buckling was also determined, and the relative rod powers (fission rate distributions) were measured for selected fuel rods. Both critical configurations described in the IRPhEP handbook – one at the room temperature and one at the elevated temperature – are considered acceptable for use as criticality and reaction rate benchmark experiments [35].

4.3.4 An International Benchmark Exercise on KRITZ Critical Experiments

The KRITZ benchmark experiments have been extensively studied by an international benchmark exercise, which was launched in October 2000 in the framework of the joint activities of the OECD/NEA Working Party on the Physics of Plutonium Fuels and Innovative Fuel Cycles and the Task Force on Reactor-based Plutonium Disposition [35]. The aim of this exercise was to investigate the capabilities of the current production neutronics codes and nuclear data libraries to analyze MOX-fueled systems, and to compare the accuracy of the predictions for the MOX and UO2 fueled configurations. Institutions from seven countries participated in this exercise, providing 13 solutions. Table 4-1 summarizes the KRITZ critical configuration specifications. Figure 4-4 shows the horizontal cross section of the benchmark model. Reference 35 provides comparative analyses of calculated and measured results, as well as inter-comparisons of some of the results obtained by participants by calculation only. The computer codes used were deterministic codes such as THREEDANT and HELIOS, as well as some Monte Carlo codes. The results presented in Reference 35 include the critical configuration core calculations and the fuel pin cell calculations. The critical configuration core calculations include: 1) the configuration multiplication factors at room and elevated temperatures, and 2) the relative rod powers for the rods for which the measurements were performed. The fuel pin cell calculations include: 1) the infinite multiplication factor, and 2) absorption and fission rates for the selected nuclides (e.g. ^{234}U, ^{235}U, ^{238}U, for UO_2 fuel and ^{239}Pu, ^{230}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, etc. for MOX fuel).

Table 4-1 Summary of the KRITZ Critical Configuration Specifications [30]

Experiment	Fuel	Fuel OD (mm)	Clad OD (mm)	Rod Pitch (mm)	Num. Rods	Temp (°C)	Boron Conc. (ppm)	Water Height (mm)
KRITZ-2:1	UO_2,	10.58	12.25	14.85	44x44	19.7	217.9	652.8

	1.86% ^{235}U					248.5	26.2	1055.2
KRITZ-2:13	UO_2, 1.86% ^{235}U	10.58	12.25	16.35	40x40	22.1	451.9	961.7
						243.0	280.1	1109.6
KRITZ-2:19	MOX 1.5% PuO_2 91.41 at% ^{239}Pu	9.45	10.79	18.00	25x24	21.1	4.8	665.6
						235.9	5.2	1000.1

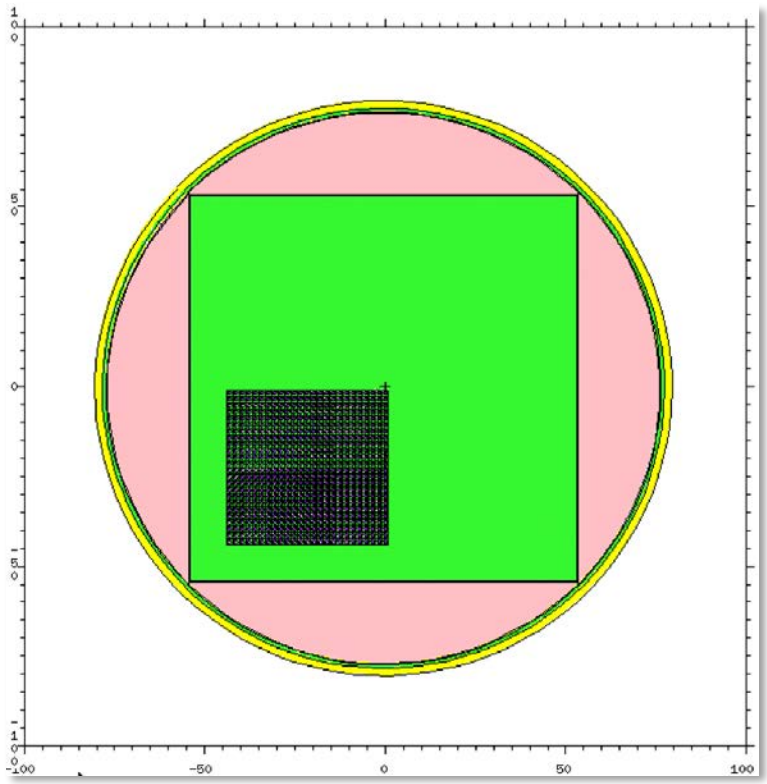


Figure 4-4 Horizontal Cross Section of the KRITZ Benchmark Model [35]

4.4 DIMPLE Low Power Reactor

The DIMPLE is a low power reactor located at the U.K.A.E.A's Winfrith site. The DIMPLE facility consisted of a large aluminum primary vessel, with 2.591m inner diameter and 4m high with a side-wall thickness of 0.65cm, surrounded by concrete shielding. Figure 4-5 [29] shows the general view of the DIMPLE reactor. The reactor core was supported inside the tank by a steel chassis. A number of 'U' shaped beams were accurately arranged within the chassis to support two sets of aluminum lattice plates which in turn supported the fuel pins. The lower lattice plates were secured to the 'U' beams by a tubular stainless steel chassis. The upper lattice plates, approximately 60cm above the lower lattice plates, were attached to the 'U' beams by two support brackets. Two experiments have been accepted in the IRPhEP handbook as benchmark experiments – DIMPLE S01 and DIMPLE S06 [36].

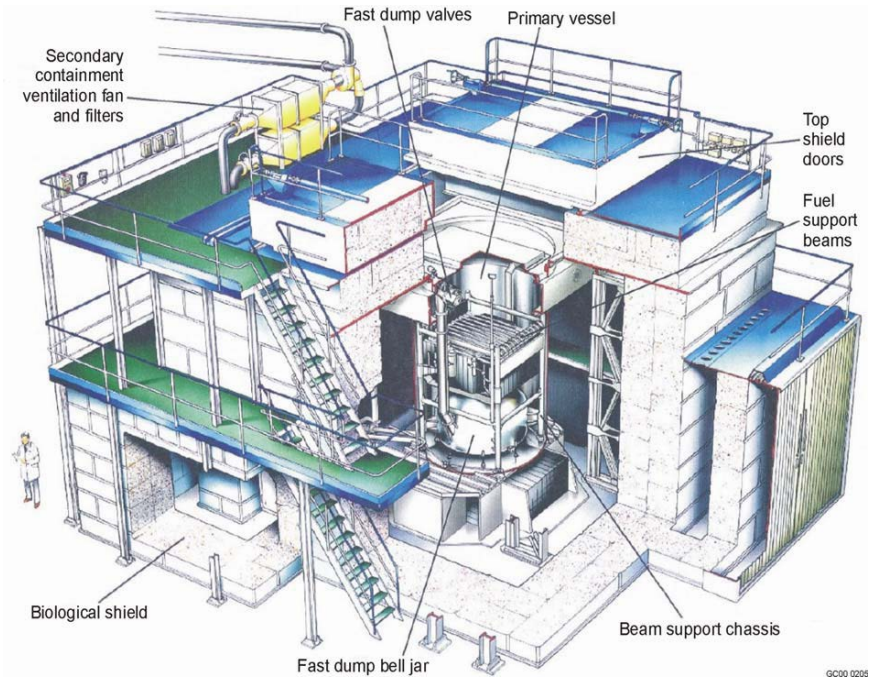


Figure 4-5 General View of the DIMPLE Reactor [29]

4.4.1 DIMPLE S01

The DIMPLE S01 experimental program considered critical experiments with low enriched uranium dioxide fuel rods containing 3.0% ^{235}U with light water moderation and reflection. These experiments were performed in the DIMPLE low power reactor at U.K.A.E.A's Winfrith site during 1983.

Comprehensive axial and radial reaction rate distributions were measured with foils in the assembly S01A and these, along with fission chamber traverses (axial only), were used to derive experimental buckling values. The experimental configuration comprised a cylindrical array of fuel rods centrally located within a large aluminum vessel (2.6 m diameter and 4 m high) containing water. The array of rods was light water moderated and fully reflected to a critical height of around 50 cm above the base of the fuel stack in the rods. The fuel rods were located on a square pitch of 1.32 cm and were supported by an upper and lower lattice plate. The lower lattice plate was situated on an aluminum fuel support assembly.

4.4.2 DIMPLE S06

The experimental program encompassed critical experiments with low enriched uranium dioxide fuel rods containing 3.0% ^{235}U with light water moderation and reflection. The experiments were performed in the DIMPLE low power reactor at U.K.A.E.A's Winfrith site during the late 1980's and early 1990's. The experimental program extended previous studies in water-reflected cylindrical systems to power reactor geometries by assembling a cruciform array of 3% enriched uranium dioxide fuel pins. The array simulated the rectangular corner configuration of a Pressurized Water Reactor (PWR) and effectively represented twelve PWR fuel assemblies. Four primary versions of the cruciform assembly were constructed, the first being water reflected, as with the cylindrical systems. The assembly was then surrounded azimuthally by a stainless steel region simulating a PWR core baffle. A third version incorporated discrete burnable poison pins and empty guide thimbles in a series of different arrays. The fourth version was an ex-core detector benchmark, which consisted of a realistic simulation of the core baffle, barrel, neutron shield pad, and pressure vessel

of a PWR. The experiments were performed in three separate phases, collectively known as the S06 series. The first phase, covering S06A and B, is the subject of this evaluation. The second phase, S06C, covering discrete burnable poison rods and empty guide thimbles, consisted of twelve configurations but is not the subject of this evaluation. The final phase, S06D, the ex-core detector benchmark study, consisted of 7 configurations. Neither S06C nor S06D are described in the IRPhEP handbook. More data will be needed to use these cases for validation [29].

4.5 VENUS Critical Facility

The VENUS critical facility is a zero power reactor located at SCK-CEN, Mol (Belgium). This facility was built in 1963-1964, as a nuclear mock-up of a projected marine reactor called VULCAIN; hence the name VENUS which means "Vulcain Experimental NUclear Study". In 1967, this facility was adapted and improved in order to study LWR core designs and to provide experimental data for nuclear code validation. Goals of the facility included flexibility, easy handling of the fuel pins, handled one by one, and results with great precision.

In 1980, additional material was purchased for studying typical 17×17 PWR fuel assemblies. Such an adaptation was easy: only new reactor grids and small devices adapted to the new fuel geometry were necessary. In 1982, special stainless steel pieces were manufactured in order to build a pressure vessel mock-up representative of a three-loop Westinghouse power plant. These first stainless steel pieces were delivered at the beginning of December 1982 and the reactor, loaded with this mock-up core, was made critical on December 20, 1982.

4.5.1 VENUS-1 PWR UO_2 Core 2D Benchmark Experiment

The VENUS-1 experiment period was from 24 January 1983 to 23 June 1983. The VENUS-1 experiments were composed of (1) gamma scans to determine the core power distribution, (2) in-core and ex-core foil activations and (3) measurement of vertical bucklings in the core and core exterior. The VENUS-1 experiment was established for the validation of 2D dosimetry analysis. In this format, the experiments were classified into three experiments: critical configuration, reaction rate distribution measurement, and power distribution measurements.

The VENUS-1 experiment was established to build a pressure vessel mock-up representative of a 3-loop Westinghouse power plant. The experiments provide the measured reaction rates at dosimeters installed in various locations, pin-wise power distribution at the axial mid-plane, absolute power level, and vertical buckling.

The evaluation of the RPV (Reactor Pressure Vessel) integrity for both pressurized thermal shock and end-of-life (EOL) considerations requires the accurate determination of neutron fluence accumulated on the RPV. The evaluations of RPV integrity is validated through various measurements of the dosimeters extracted from the capsules.

The VENUS-1 experiments provide the reaction rate data measured at both in-core and ex-core locations. The measurement data are useful for the validation of the M&S used for RPV irradiation analysis. In addition, as the VENUS-1 core configuration is similar to PWR core, the measured pin-wise power distribution is useful for the validation of fission source calculation [29].

4.5.2 VENUS-3 PWR UO2 Core 3D Benchmark Experiment

The VENUS-3 experiment was established to build a pressure vessel mock-up representative

of a 3-loop Westinghouse power plant. The experimental configuration was made to be representative of typical irradiation conditions of a modern PWR vessel. The experiment was established for 3-dimensional dosimetry analysis. The activities were measured at various locations inside the vessel. The experiments provide the measured reaction rates at dosimeters installed in various locations and 3D power distributions and absolute power level. The evaluation of the RPV (Reactor Pressure Vessel) integrity for both pressurized thermal shock and end-of-life (EOL_ considerations requires the accurate determination of neutron fluence accumulated by the RPV. The neutron influence can be validated through various measurements of the dosimeters extracted from the capsules. For some early reactors, part-length shielded assemblies (PLSAs) were loaded to reduce the neutron irradiation at the critical position on RPV. The VENUS-3 experiment was built for the mock-up of this modification. The VENUS-3 experiments provide the reaction rate data measured along the axial direction at both in-core and ex-core locations. Thus, the experimental data are useful for the validation of three-dimensional neutron transport calculation. In addition, the measured pin-wise power distribution is useful for the estimation of three-dimensional power distribution calculations. The criticality of VENUS-3 reactor was achieved to measure the reaction rate and neutron source distributions in steady state condition. The VENUS-3 experiment was classified into three measured experiments: critical configuration, reaction rate distribution measurement, and power distribution measurements.

The VENUS-3 core evaluated in the IRPhEP handbook was made critical for the first time on March 16, 1988. The experimental program started on March 29, 1988 and was planned to run until the end of December 1988. At a later date, the facility was modified with the installation of pressure vessel internals, for the purpose of providing benchmarks for the PWR pressure surveillance program, partly supported by Westinghouse Electric and the NRC. This was the VENUS-LWR-PVS (Light Water Reactor – Pressure Vessel Surveillance) program that comprised three mock-up configurations of the core periphery of a PWR. This program was carried out between 1983 and 1989. The LWR-PVS benchmark experiment in VENUS is aimed at validating the analytical methods needed to predict the azimuthal variation of the fluence in the pressure vessel [29].

4.5.3 Experimental Study of the VENUS-PRP Configurations No. 9 and 9/1

The VENUS-PRP Configurations 9 and 9/1 were designed to study boundary effects between zones with different plutonium content and the influence of perturbations at the boundary. The following nuclear parameters were measured: criticality, spectral index $\sigma_f^{239}/\sigma_f^{235}$, power distributions within fuel rods and by regions, and power sharing between LEU and MOX fuel.

The VENUS-PRP critical experiments' first aim of the program was to provide experimental data for the validation of group cross-sections and design methods for Light Water Reactors (LWRs) with UO_2-PuO_2 mixed oxide (MOX) fuel [29].

The scale of the whole VENUS-PRP program can be gauged through some relevant figures: The fuel stock for the VENUS facility consisted of 2800 fuel rods divided into several different compositions (UO_2, UO_2-PuO_2, UO_2-Gd_2O_3) and three different fabrication methods (pellets, homogeneous, and heterogeneous vibro-pack oxides). It also included an additional 6600 pellets of eight different kinds used either separately, or in fuel rods capable of being disassembled. In order to make the daily operations easier, each fuel was named by a fraction with the ^{235}U enrichment of UO_2, as the numerator and the PuO_2 content in the MOX as denominator. Both figures are expressed in wt% and are nominal or rounded to the nearest integer. Criticality was achieved in approximately 100 different core configurations, by adjusting the moderator level.

The VENUS-PRP experiments included, amongst others:

- 100 measurements of k_{eff};
- 200 irradiations;
- 4000 γ-scans of fuel rods;
- 120 measurements of reactivity;
- 500 measurements of the fine structure of the fission power distribution.

The VENUS-PRP Configuration 9 focused on the study of the power distribution across the boundary between a standard UO_2 fuel region, enriched to 4% ^{235}U (4/0 type) and a MOX fuel region made of UO_2 enriched to 3% ^{235}U with ~ 1% PuO_2 (3/1 type), simulating a once burned fuel assembly [29].

4.5.4 International Benchmark Exercise of VENUS-2 MOX Core Measurements

To better understand the behavior of MOX fuel in challenging situations such as multiple recycle and high burn-up of plutonium in PWRs, the OECD/NEA launched a blind international benchmark exercise for the prediction of power distribution in the 2D VENUS-2 MOX core experiments in 1999 [37]. This experiment-based benchmark was completed in 2000. The results showed that the calculations overestimated fission rates of MOX pins and slightly underestimated those of UO_2 pins. A 3D VENUS-2 MOX core benchmark was then launched in 2001 for a more thorough investigation of the calculation methods for MOX-fueled systems [38]. Twelve participants contributed to the 3D benchmark exercise, providing more than 20 solutions. Deterministic codes such as TORT, DANTSYS, PARCS, as well as Monte Carlo codes were used by the participants. Figure Figure 4-6 shows the VENUS-2 core geometry. In this benchmark exercise, both cell calculations and core calculations were performed. The k_∞ and reaction rates (absorption and fission) per isotope were calculated in the cell calculations for each fuel cell type. In the core calculations, the k_{eff} and normalized radial fission rate distribution at 325 fuel pin positions and normalized axial fission distribution of six fuel pins were calculated. The calculated resulted were compared with measured values.

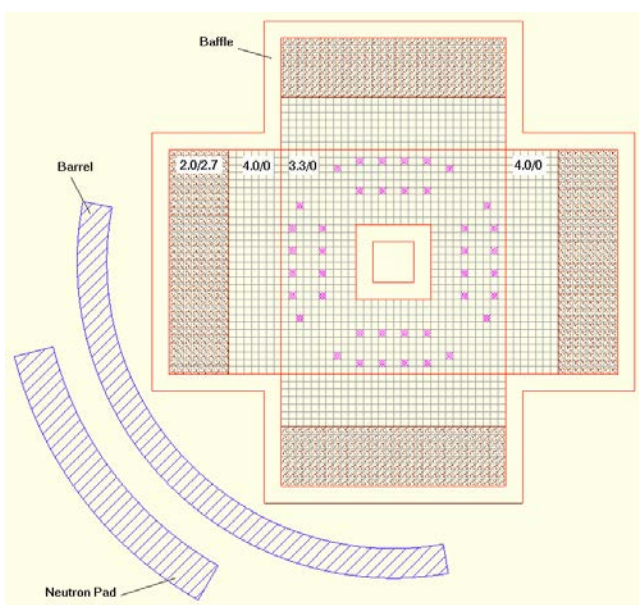


Figure 4-6 VENUS-2 Core Geometry [37]

4.6 Experiments with the IPEN/MB-01 Research Reactor

The IPEN/MB-01 research reactor is a zero power critical facility specially designed for measurement of a wide variety of reactor physics parameters to be used as benchmark experimental data for checking nuclear M&S and related nuclear data libraries commonly used in the field of reactor physics. This facility is located in the city of São Paulo, Brazil, and reached its first criticality on November 9, 1988. Since then it has been utilized for basic reactor physics research and as an instruction laboratory system.

Figure 4-7 [29] shows some details of the IPEN/MB-01 core. This facility consists of a 28x26 rectangular array of UO_2 fuel rods 4.35% enriched uranium and clad by stainless steel (SS-304) inside a light water tank. The maximum allowed power is 100 W. The control of the IPEN/MB-01 reactor is via two control banks diagonally placed. The control banks are composed of 12 Ag-In-Cd (AIC) rods and the safety banks by 12 B_4C rods. The square pitch of the IPEN/MB-01 reactor was chosen to be close to the optimum fuel-to-moderator ratio (maximum k_∞). This feature favors the thermal neutron energy region and mainly the ^{235}U events. The experiments performed at the IPEN/MB-01 reactor were the following:

- critical configurations
- buckling and extrapolation length
- spectral characteristics
- reactivity measurements
- temperature reactivity coefficient
- effective kinetic parameters
- reaction rate distributions
- power distribution

The criticality portion of the experiments has been documented under the ICSBEP [34].

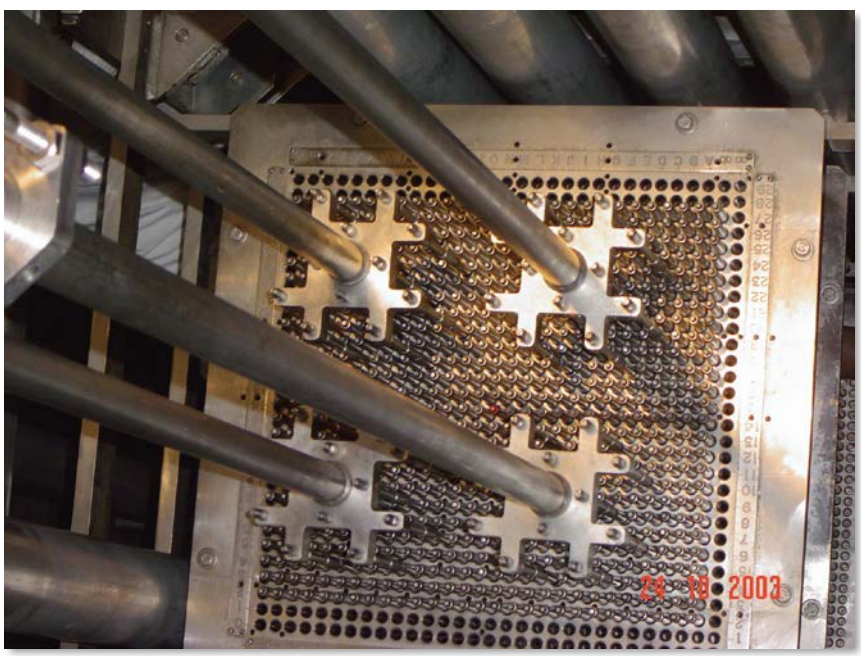


Figure 4-7 IPEN/MB-01 Core [29]

4.7 RPI Critical Experiments with Erbia

Upon the development of the erbia (Er_2O_3) integral burnable absorber for CE fuel, ABB-CE engaged in a development program with utilities to perform a serious of critical experiments in the Reactor Critical Facility (RCF) of Rensselaer Polytechnic Institute (RPI)[39]. The primary measured quantities were reactivity and local pin power distributions for various erbia-urania rod configurations, temperatures, and soluble boron concentrations. The lattice configurations used in the experiment were similar to fuel assembly designs used at CE plants such as Pale Verde [28]. The measurements were obtained by counting delayed fission product gammas, with the count rate being proportional to the fission rate. Both 16x16 and 14x14 fuel types were simulated in the experiment.

More information on these experiments is not available but may be added at a later date. These experiments are low priority as they mainly support the validation of models and methods for CE-type fuels and erbia, which are not as common as other fuel types in the U.S.

4.8 Special Power Excursion Reactor Test (SPERT) III E-Core

The SPERT project was established as part of the U.S. Atomic Energy Commission's reactor safety program in 1954, designed to investigate the kinetic behavior of nuclear reactors subjected to large positive reactivity insertions [40,41]. The SPERT III E-Core consisted of 60 5x5 fuel assemblies (BWR-like) in a small core arrangement as shown in Figure 4-8.

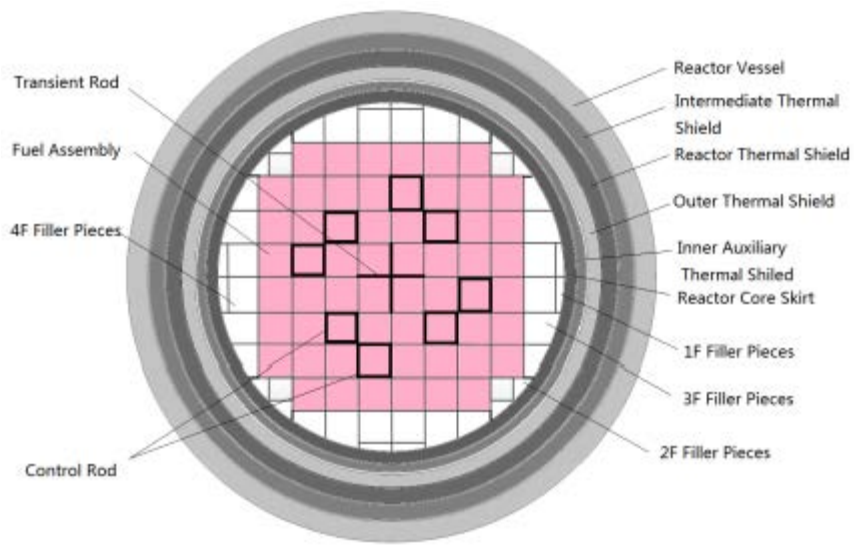


Figure 4-8 SPERT III E-Core Cross-Section [40]

Though the experiment was designed for understanding kinetic behavior, the steady-state criticality conditions at cold zero power (CZP) and HZP can also be used for validation of core reactivity. Pin-wise fission rate distributions were not measured. Because of unique characteristics such as the fuel assembly channels (or box), control rod assemblies, flux suppressors, and cruciform-shaped transient rod assembly, the SPERT geometry will not be supported by the VERA common input.

Validation activities using SPERT have already been initiated for transient capabilities, which have included these steady-state criticality cases. Additionally, a detailed KENO-V.a model has been developed and used for inferred validation of the SPERT III E-Core power distribution [40]. This comparison is consistent with those described in Section 6.

4.9 Strawbridge and Barry 101

The Strawbridge and Barry 101 criticals are uniform light water lattice critical experiments with several lattice parameters such as water to Uranium ratio, enrichment, experimental buckling, pellet diameter and boron concentration. These criticals contain 40 uranium oxide and 61 uranium metal cold clean experiments [42]. These critical experiments have historically been included in industry neutronics code qualifications since they cover a wide range of lattice parameters and therefore provide a good validation set for lattice physics codes to accurately predict reactivity accurately over a broad range of conditions.

Since the Strawbridge and Barry criticals are uniform lattices for which experimental bucklings have been reported, these criticals can be treated as single pin cells in CASL neutronics codes validation. The ranges of lattice parameters covered by these criticals are [42]:

- Enrichment (^{235}U at%) : 1.04 to 4.069
- Boron concentration (ppm): 0 to 3392
- Water to uranium ratio: 1.0 to 11.96
- Pellet diameter (cm): 0.44 to 2.35
- Lattice pitch (cm): 0.95 to 4.95
- Clad material: none, aluminum, stainless steel
- Lattice type: square, hexagonal
- Fuel density (g/cm^3): 7.5 to 18.9

4.10 Saxton Plutonium Program

The Saxton critical experiments were performed at the Westinghouse Reactor Evaluation Center (WREC) at the CRX reactor critical facility in Waltz Mill, Pennsylvania in 1965. The purpose of these critical experiments was to verify the nuclear design of Saxton partial plutonium cores while obtaining parameters of fundamental significance such as buckling, control rod worth, soluble poison worth, neutron flux, power peaking, relative pin power, and power sharing factors of MOX and UO_2 lattices [43]. There were 49 LWR-type configurations using UO_2 and MOX fueled cores, controlled by moderator level adjustments, some with AIC control rods. Single region, multiple regions, and void-effect experiments were performed. Figure 4-8 provides a top view of the WREC CRX core.

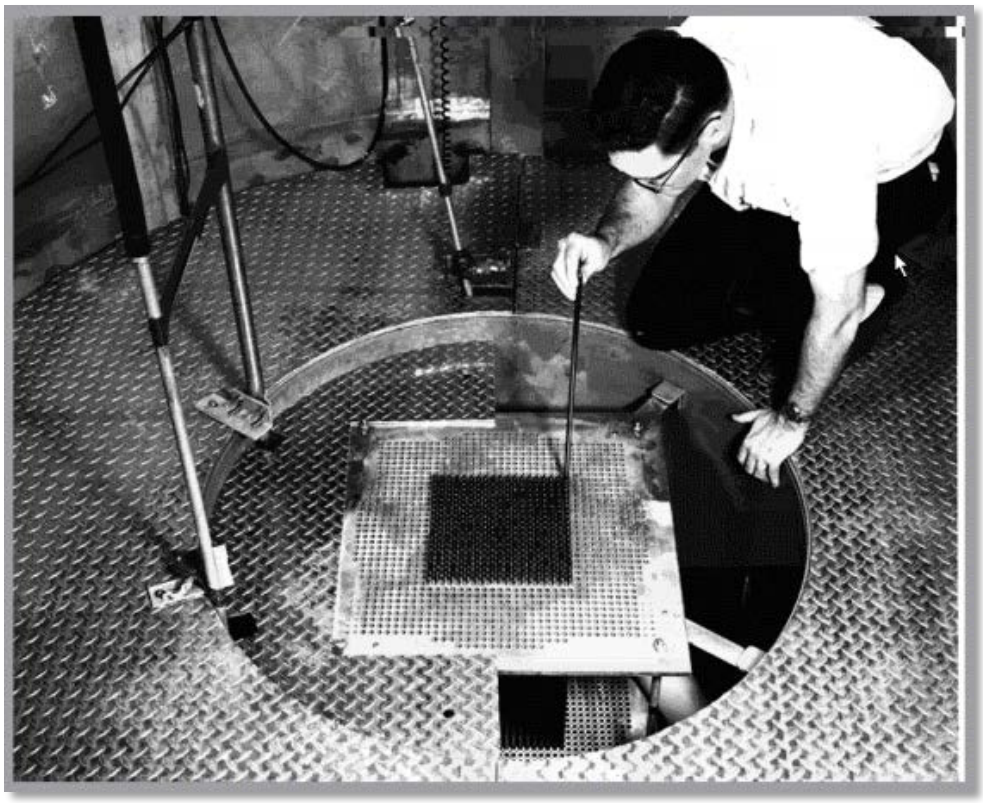


Figure 4-9 Top View of Saxton WREC-CRX Reactor Core [43]

For the experiments, a variety of pin pitches and soluble boron concentrations were used. Measurements of critical buckling, relative power, and reactivity worth were made. For more information see Reference 43.

4.11 CREOLE PWR Reactivity Temperature Coefficient Experiment

The CREOLE critical facility is a zero power reactor located at Commissariat à l'Energie Atomique (CEA) in France [29]. This facility was built in 1965 in order to study LWR core designs in support of the French large-scale program of building PWR power plants, which was launched during the middle of the seventies. The CREOLE (Coefficients of Reactivity in EOLE) experimental program was conceived to supply accurate differential information on the Reactivity Temperature Coefficient in the whole temperature range of interest in a large PWR (from room temperature up to 300 °C). The measurements were performed in the EOLE facility at CEA-Cadarache during the two last years of the seventies. The experimental facility consists of a pressurized central test loop in which it is possible to achieve operating conditions of a large PWR power reactor in terms of pressure and moderator temperature, a large vacuum-gap separation zone and a peripheral driver core of variable sizes surrounded by a water reflector.

The differential isothermal reactivity temperature coefficient (ITC) of UO_2 and UO_2-PuO_2 lattices were measured from 20 °C up to 300 °C in a central pressurized loop. Additionally, the integral temperature reactivity effect of $\Delta T \cong 280$ °C was obtained in terms of soluble boron equivalence in the central loop and driver-zone critical loading variations. The reactivity effects due to moderator temperature and density changes in the test loop were measured by the doubling-time technique during reactor divergence with all control rods extracted or through the adjustment of the driver-core

critical sizes when necessary. Radial fission-rate distributions were measured by direct gamma scanning of the fuel rods in the central-loop and driver-core zones, and axial flux maps were achieved using fission chambers.

The measurements were carried out in four basic experimental configurations of the central test loop with 200 fuel rod locations with a pitch of 1.26 cm, which is typical for the 17x17 PWR fuel assemblies. The basic core configuration is displayed in Figure 4-9. The configurations were:

1) UO_2 clean lattice (200 fuel rods of 3.1% ^{235}U enrichment).
2) UO_2 poisoned lattice with 1166 ppm of boron in water (200 fuel rods of 3.1% ^{235}U enrichment).
3) UO_2-PuO_2 clean lattice (80 fuel rods with 3.2% fissile Pu and 120 fuel rods with 2% fissile Pu).
4) UO_2-PuO_2 clean lattice with water channels (72 fuel rods with 3.2% fissile Pu, 108 fuel rods with 2% fissile Pu and 20 water channels).

Five additional configurations were obtained from the previous ones by using aluminum over-claddings to simulate moderator density changes. In summary, the following measurements were performed in the CREOLE experiment [44]:

- Operating parameters: temperature, pressure, and boron content
- Basic geometry and material compositions
- Critical sizes of different experimental configurations at room temperature
- Temperature reactivity coefficients
- Soluble-boron reactivity worth and equivalence with the temperature reactivity effect
- Effects of moderator density variations (using over-claddings)
- Reaction rate distribution

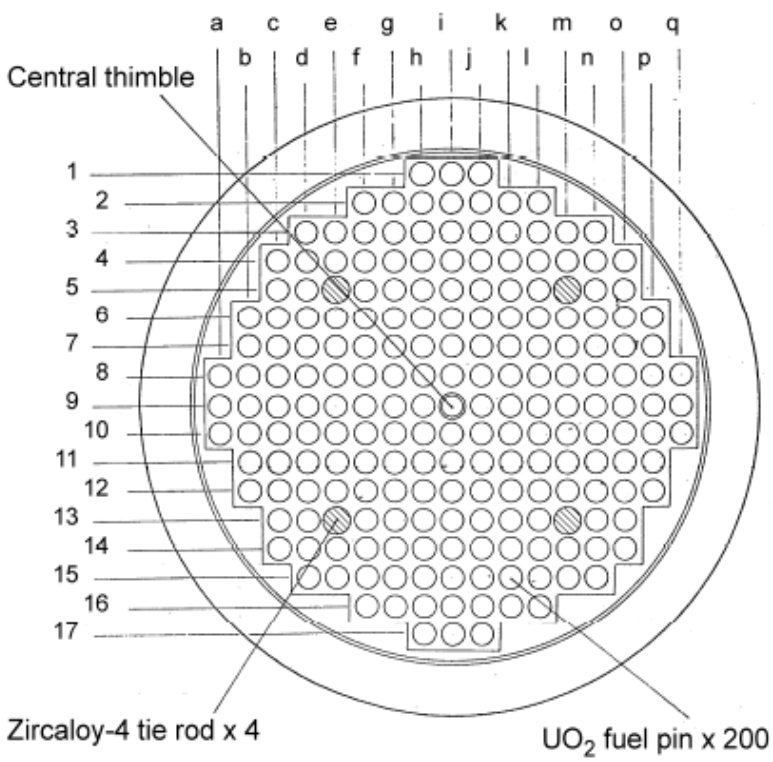


Figure 4-10 UO_2 Configuration of the CREOLE Reactor [44]

4.12 EPICURE

The EPICURE experiments, performed by Commissariat à l'Energie Atomique (CEA) in France, provided benchmark data for validating nuclear M&S simulating UO_2 and MOX fueled PWRs. Both reactivity and fission rate distributions were measured [45]. Current information is limited about this experiment and will be provided as it becomes available. Assistance may be obtained from Areva, a CASL Industry Council member.

The UH1.2 core of the EPICURE experiments is shown in Figure 4-10. It contains ~1400 fuel rods with 3.7% enriched UO_2 and ~600 ppm soluble boron concentration [45]. These experiments were used to measure the effects of different absorber materials [46].

The MH1.2 core of the EPICURE experiments is shown in Figure 4-11. It contains ~2350 fuel rods with 3.7 enriched UO_2, 7% MOX, and ~220 ppm soluble boron concentration [45].

Reference 46 also includes some information on the MISTRAL cores from the EOLE zero-power research facility that may also be useful in the future.

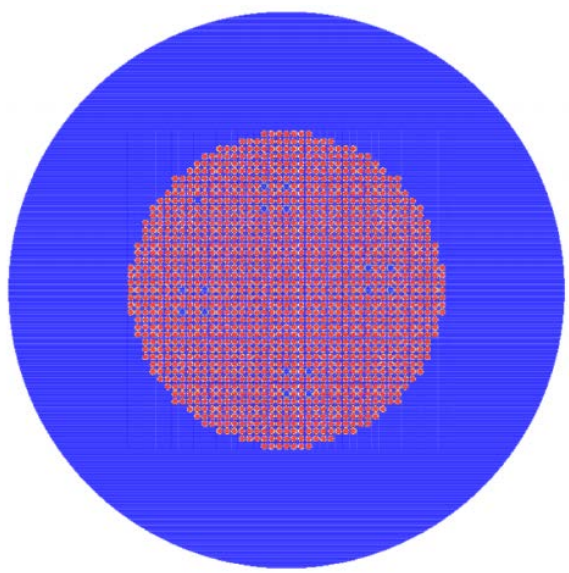


Figure 4-11 EPICURE UH1.2 Radial Cross Section [46]

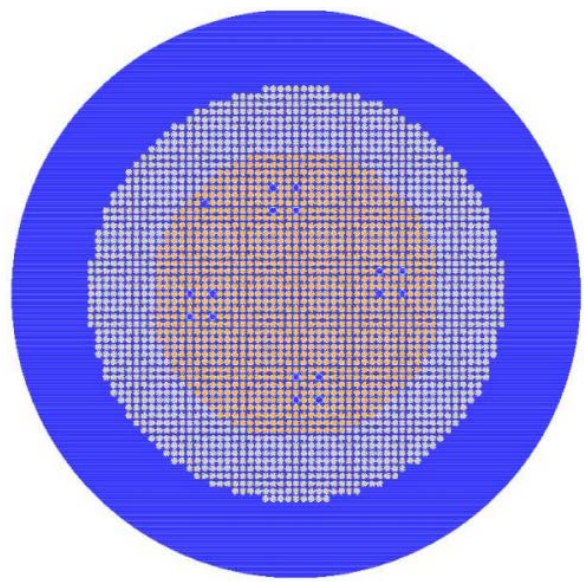


Figure 4-12 EPICURE MH1.2 Radial Cross Section [46]

4.13 CAMELEON

Like EPICURE, the CAMELEON experiments were carried out in the 1980's by CEA to provide validation for M&S for UO_2 fuel with several absorbers [45, 47]. They were performed at CEA's EOLE experimental core at Cadarache. Current information is limited about this experiment and will be provided as it becomes available.

4.14 CROCUS Reactor

The CROCUS reactor, operated by the Swiss Federal Institute of Technology, Lausanne, is a simple two-zone uranium-fueled, H_2O-moderated critical research facility. Figure 4-12 [29, 48] gives a view of the facility, a so-called zero-power reactor, with a maximum allowed power of 100 W. The core is approximately cylindrical in shape with a diameter of about 60 cm and a height of 100 cm. In 1995, a configuration with a central zone of 1.8% enriched UO_2 rods and an outer zone of 0.95% enriched uranium metal rods was made critical by raising the moderator level. Later, the reactor was used for different experiments. Some configurations are evaluated in the IRPhEP handbook [29] and are approved as benchmark configurations for the reactivity difference between the critical moderator level and some supercritical conditions.

CROCUS-LWR-RESR-001 is a benchmark on kinetics parameters in CROCUS. Two different kinds of measurements were carried out in the CROCUS reactor:

1) Variation of the water moderator level. For this type of configuration, four different water levels were measured, one for the critical state (i.e. when the inverse reactor period is zero) and three higher water levels for supercritical states. These experiments were performed in 1996.

2) Insertion of an absorber rod (B_4C), adjustment of the water level to the new critical height, and then removal of the absorber rod. For this second set of experiments, a total of four configurations were measured, two for the critical states with the absorber rod inserted and two supercritical states obtained after withdrawing the absorber rod. These experiments were performed in 1997.

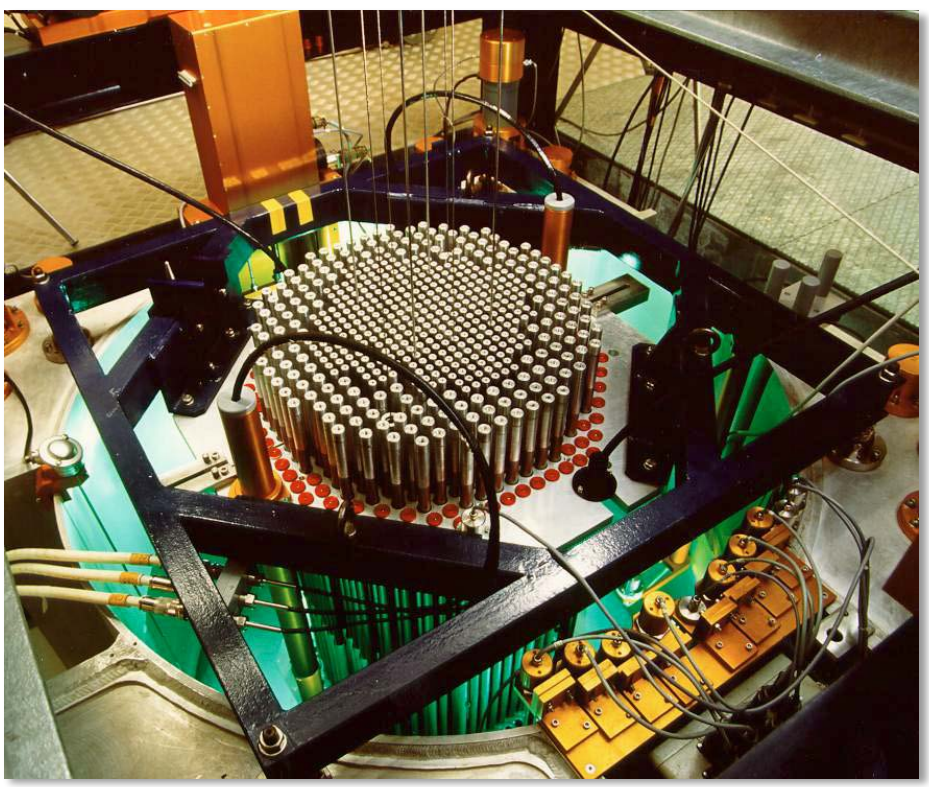


Figure 4-13 The CROCUS Reactor [29]

4.15 JAERI TCA Temperature Effects on Reactivity in LWR UO_2 Cores

From the point of view of the nuclear criticality safety of the dissolvers in reprocessing plants, the temperature coefficient of reactivity is one of the most important neutronics quantities to evaluate the criticality safety margin. Soluble neutron poisons such as boric acid H_3BO_3 and gadolinium nitrate $Gd(NO_3)_3$ are useful to enlarge the capacity of the fuel dissolvers under the condition that the nuclear safety is assured. The operating conditions of the dissolvers vary widely with temperature, and with water-fuel volume ratio. Therefore, the accumulation of experimental data on the temperature coefficients of reactivity in these heterogeneous systems with soluble poisons is important for the advanced nuclear criticality safety design [30, 49].

Experimental and computational studies on the temperature coefficient of reactivity have been performed using light-water moderated and reflected UO_2 cores with soluble poisons such as boron and gadolinium. The experiments were carried out with the Tank-type Critical Assembly (TCA) in Japan Atomic Energy Research Institute (JAERI) from April 1988 to January 1989 [49]. The temperature coefficients of reactivity in the cores with soluble poisons were measured by changing temperature of the moderator and the reflector from the room temperature up to about 60°C. The geometries of the core configurations were simple. The experiments were well documented and carefully performed. There were no serious omissions of data. All experimental data regarding the temperature effect of reactivity are acceptable benchmarks. Light water moderated and reflected rectangular cores were constructed in the TCA to simulate the reactivity effect and temperature coefficient in a dissolver loaded with low enriched UO_2 rods. The experiments were performed in order to obtain benchmark data of temperature effects on reactivity in simple cores with soluble poisons, and the dependence of temperature coefficients on the core configuration and the concentration of soluble poison in the moderator and reflector were mainly studied. The experiments were performed with respect to three kinds of cores, i.e., cores without poisons (named A-cores), ones with H_3BO_3 (B-cores), and ones with $Gd(NO_3)_3$ (C-cores).

4.16 International Criticality Safety Benchmark Evaluation Project (ICSBEP)

The International Criticality Safety Benchmark Evaluation Project (ICSBEP)[34] was initiated in 1992 by the Department of Energy Defense Programs Systems Engineering Division and is managed through Idaho National Laboratory (INL). It became an official activity of the Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) in 1995. The purpose of the ICSBEP is to:

a. Identify a comprehensive set of critical benchmark data and, to the extent possible, verify the data by reviewing original and subsequently revised documentation, and by talking with the experimenters or individuals who are familiar with the experimenters or the experimental facility.
b. Evaluate the data and quantify overall uncertainties through various types of sensitivity analysis.
c. Compile the data into a standardized format.
d. Perform calculations of each experiment with standard criticality safety codes.
e. Formally document the work into a single source of verified benchmark critical data. The work of the ICSBEP is documented as an International Handbook of Evaluated Criticality Safety Benchmark Experiments.

Currently, the handbook spans nearly 55,000 pages and contains 516 evaluations representing 4405 critical, near-critical, or subcritical configurations, 24 criticality alarm placement/shielding configurations with multiple dose points for each, and 200 configurations that have been categorized as fundamental physics measurements that are relevant to criticality safety applications. The handbook is intended for use by criticality safety analysts to perform necessary validations of their calculation techniques and is expected to be a valuable tool for decades to come. The ICSBEP Handbook is available both on DVD and the Internet. A DVD may be obtained by completing the DVD Request Form through http://icsbep.inel.gov [30,34].

The SCALE project at ORNL maintains a large number of the ICSBEP criticality benchmark experiments within ORNL's Verified, Archived Library of Inputs and Data (VALID) database. A total of more than 350 cases are available in the VALID library, which contains the input, output, and other associated files [50]. The critical experiments available cover a broad array of different categories of systems. These systems use a range of fissile materials including a range of uranium enrichments, various plutonium isotopic vectors, and mixed uranium-plutonium oxides. The physical form of the fissile material also varies and is represented as metal, solutions, or arrays of rods or plates in a water moderator. The neutron energy spectra of the systems also vary and cover both fast and thermal spectra.

There are a total of 118 cases available in VALID that are applicable to validation of VERA-CS. Table 4-2 summarizes the listing of benchmark critical experiments which are application to validation of PWR criticality calculations. These criticality experiments are additional validation but are not likely required in addition to the other detailed critical experiments discussed previously.

Table 4-2 Critical Benchmark Experiments in VALID library Applicable to VERA-CS [30]

Evaluation	Cases	Fissile material	Experiment description
LEU-COMP-THERM-001	8	UO_2 2% ^{235}U	Water-moderated UO_2 fuel rods in 2.032 cm square-pitched arrays
LEU-COMP-THERM-002	5	UO_2 4% ^{235}U	Water-moderated UO_2 fuel rods in 2.54 cm square-pitched arrays
LEU-COMP-THERM-010	30	UO_2 4% ^{235}U	Water-moderated UO_2 fuel rods reflected by two lead, uranium, or steel walls
LEU-COMP-THERM-017	29	UO_2 2% ^{235}U	Water-moderated UO_2 fuel rods reflected by two lead, uranium, or steel walls
LEU-COMP-THERM-042	7	UO_2 2% ^{235}U	Water-moderated rectangular clusters of UO_2 fuel rods (1.684 cm Pitch) separated by steel, boral, boraflex, cadmium, or copper Plates with steel reflecting walls
LEU-COMP-THERM-050	18	UO_2 5% ^{235}U	^{149}Sm solution tank in the middle of water-moderated 4.738% enriched UO_2 fuel rod arrays
MIX-COMP-THERM-001	4	Pu & Nat. UO_2	Water-reflected mixed plutonium-uranium oxide (20 wt % Pu) fuel rods
MIX-COMP-THERM-002	6	Pu & Nat. UO_2	Rectangular arrays of water-moderated UO_2 with 2% PuO_2 (8% ^{240}Pu) fuel rods
MIX-COMP-THERM-004	11	Pu & Nat. UO_2	Critical arrays of mixed plutonium-uranium fuel rods with water-to-fuel volume ratios ranging from 2.4 to 5.6

5. POST-IRRADIATION EXAMINATIONS

The purpose of this portion of the validation plan is to demonstrate the accuracy of the isotopic depletion and decay calculations in VERA-CS using whole core calculations as is possible. The reactivity effects of depletion are also addressed by the power plant benchmarks in Section 3 and the Monte Carlo comparisons in Section 6, but these only indirectly support the isotopic concentrations of the fuel, particularly those with large neutron cross sections. VERA-CS will enable more detailed comparisons to measured data than are typically performed due to the capabilities 3D pin-wise-powers, isotopic depletion, and decay. Such comparisons include the traditional radiochemical assay characterizations used to benchmark pin cell depletions or lattice physics codes, but also include axial gamma scans, radial pellet gamma scans, and inferred burnup distributions. VERA-CS can also more accurately capture the fuel rod power and spectral history as well as better predict the performance of rods near the periphery of assemblies or adjacent to large absorbers.

The most limiting factor in the performance of these analyses is obtaining an accurate power and fuel shuffle history for each reactor. Traditionally power histories and T/H conditions are provided for the examined samples based on industrial methods or simple assumptions in 2D models. The goal of this portion of the validation is to perform these calculations with as much reactor benchmarking data as possible for a fully integrated application of VERA-CS. To achieve this, preference should be given to benchmarks where significant data is available or to benchmarks where relationships exist such that CASL may obtain any additional needed data.

5.1 Catawba MOX Lead Test Assembly Program

In support of the U.S. Department of Energy's (DOE) program to dispose of a significant quantity of the nation's surplus weapons-grade plutonium, four mixed oxide (MOX) Lead Test Assemblies (LTAs) containing weapons-grade plutonium were irradiated in two cycles of Catawba Nuclear Station, Unit 1 (CNS1). Unlike reactor-grade MOX, which is commonly used for nuclear power in Europe, the use of weapons-grade MOX is new and has never been licensed in the U.S. The primary purpose of the LTA program was to provide operational experience to demonstrate the acceptability of the MOX fuel design [14, 51, 52].

The four MOX LTAs were irradiated in Catawba from June 2005 to May 2008, in Catawba 1 Cycles 16 and 17. As described in Section 3.3, Catawba 1 is a standard Westinghouse 4-loop PWRs with ice condenser containment operated by Duke Energy [2]. It began commercial operation in 1985 and is licensed to 3411 MW_{th} power. The fuel design is the typical 17x17 PWR layout using either Westinghouse fuel with IFBA/WABA or AREVA fuel with lumped B_4C-Al_2O_3. The LTAs were loaded in high power (but non-limiting), instrumented core locations (Figure 5-1). In order to reduce the pin power peaking factors near the assembly periphery, which is due to high thermal flux from the adjacent LEU fuel assemblies, the LTAs were designed with a three region radially-zoned rod layout using different plutonium loadings (as is standard practice in European pressurized water reactor MOX fuel assemblies). The fuel rod layout is shown in Figure 5-2 [52, 53, 54, 55].

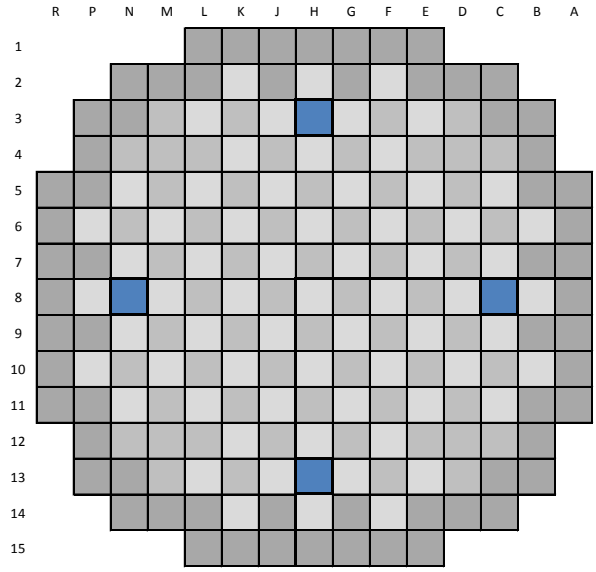


Figure 5-1 Catawba 1 MOX LTA Cycle 16 Core Locations [55]

Following irradiation, five fuel rods with rod burnups of 39.7 to 47.3 GWd/MT were removed from one assembly (NJ13GG) and shipped to Oak Ridge National Laboratory (ORNL) for hot cell examination [51]. Of the five rods, all were examined extensively with nondestructive testing, including gamma scanning, and four were selected for destructive testing (rod B-14 survived). The location of these five rods is shown in Figure 5-2.

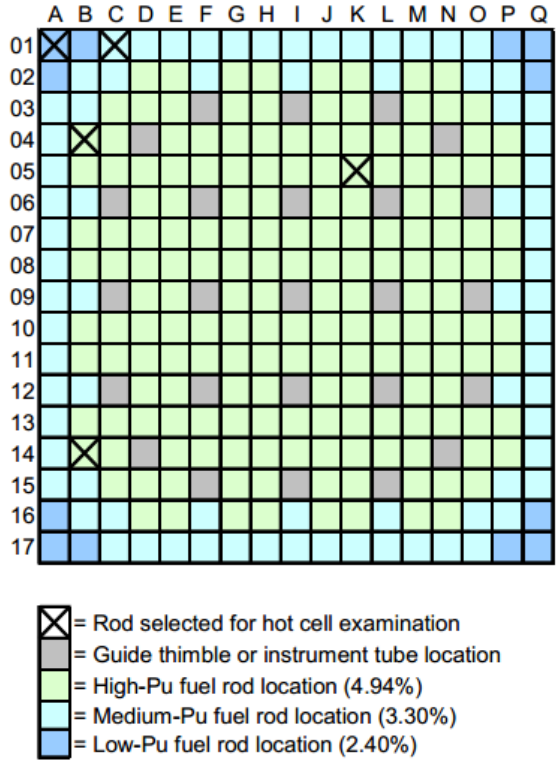


Figure 5-2 Catawba 1 MOX LTA Design and Examined Locations [51]

The extent of the hot cell examinations is shown in Table 5-1. This examination was performed at ORNL and the data should be available to CASL for validation.

Table 5-1 Catawba MOX LTA Extent of Hot Cell Examination [54]

Inspection or Test	Extent of Examination
Visual examination	5 rods
Fuel rod length	5 rods
Gamma scanning	5 rods
Eddy current testing	5 rods
Fuel rod profilometry	5 rods
Gas pressure, void volume, and gas analysis	4 rods
Optical microscopy of fuel and cladding	9 sections, with cladding metallography and pellet ceramography on each
Transmission electron microscopy of cladding	5 irradiated samples plus 1 archival sample
Scanning electron microscopy of fuel and cladding	5 samples
Radial burnup profile	5 samples
Burnup determination	11 samples
Isotopic analysis of fuel	11 samples
Gallium analysis of archival pellets	None
Gallium analysis of irradiated pellets	3 samples
Gallium analysis of archival cladding	2 samples
Gallium analysis of irradiated cladding	5 samples
Cladding hydrogen analysis	7 irradiated samples plus 2 archival samples
Expanding plug testing of cladding (room and elevated temperature)	(Tests are still in progress)
Axial tensile testing of cladding (room and elevated temperature)	(Tests are still in progress)
Pellet density	5 samples
Inspection of wear marks	1 rod survey plus 1 sample
Cladding surface microscopy	1 sample

The gamma scans were one-dimensional using two energy ranges. The low range (400 to 950 keV) was used to detect the decay of fission products that are proportional to the axial burnup profile. The absolute burnup of the rods cannot be determined by gamma scans because of the heterogeneity in the LTA lattice design. The scans were crisply defined, with even pellet dishes and chamfers and pellet-to-pellet gaps being observable. Figure 5-3 provides an example of the gamma scan results for one of the five rods [51]. This could be directly compared to ^{137}Cs concentrations calculated by VERA-CS.

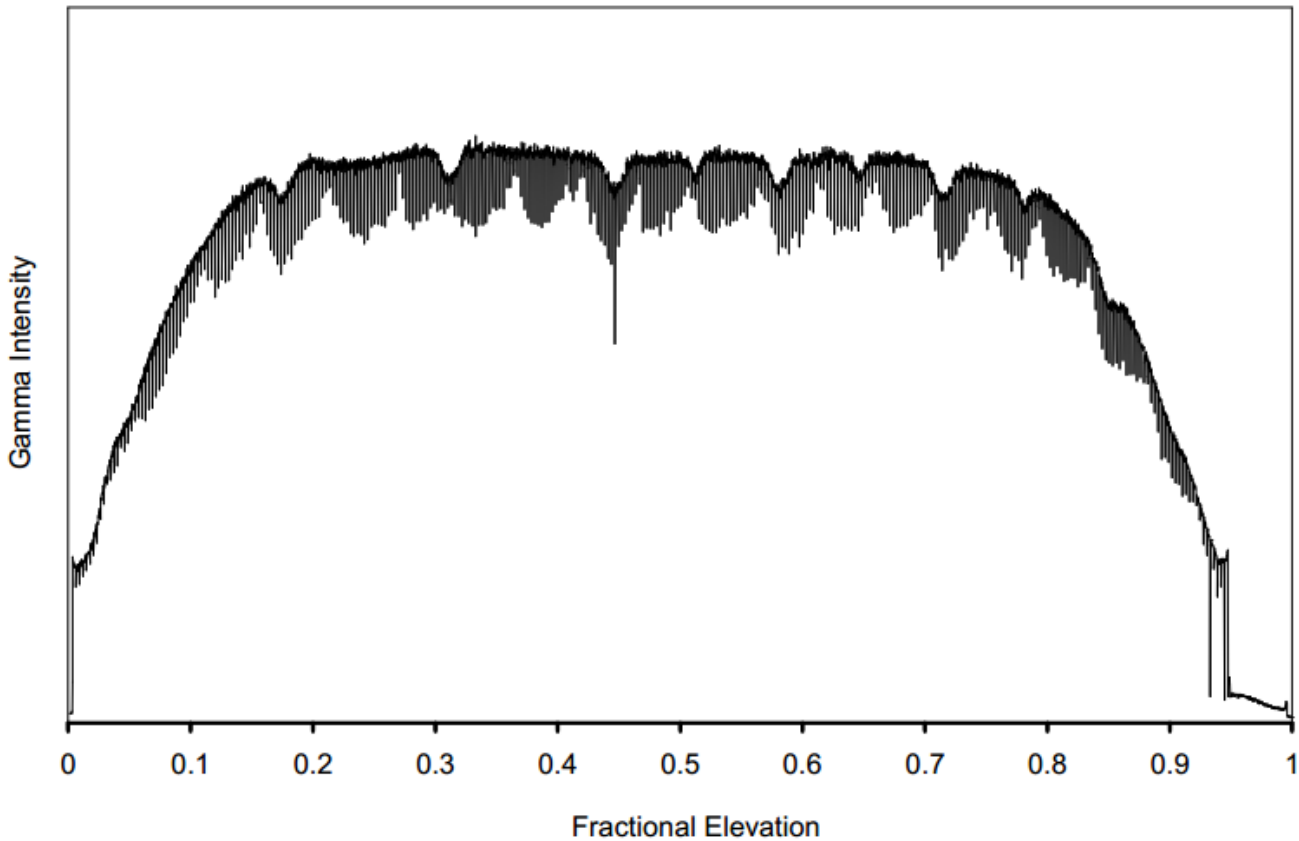


Figure 5-3 Catawba 1 MOX LTA Low Energy Gamma Scan for Rod A-01 [51]

Burnup was determined by the ^{148}Nd method for 11 samples. In Reference 54, the predicted burnup of rod C-01 is significantly different than the measured value. Because this rod is on the assembly periphery, the accuracy of the prediction comes into question (nodal methods with pin power reconstruction). This could be an opportunity for VERA-CS to demonstrate the improvements gained by higher fidelity M&S. Reference 54 states that it "is expected that more advanced neutronic methods will reduce the prediction uncertainty for peripheral rods".

Detailed radiochemical analyses were also performed on the 11 samples of fuel, including chemical analyses of Cs, Ce, Nd, Sm, Eu, Gd, U, and Pu, as well as their isotopic constituents. Again, the predictions in Reference 54 are deficient and use of a higher fidelity tool like VERA-CS would be a significant improvement. The decay times were also manually accounted for in post-processing, while VERA-CS will enable shutdown decay calculations directly through the code.

The Catawba MOX LTA is a desirable validation activity for several reasons:

a. Very highly characterized spent fuel with gamma scans and radiochemical assays
b. Test rods located at ORNL and experiments performed by ORNL staff
c. Difficult physics required at the LEU/MOX interface is a good opportunity to demonstrate the application of an advanced core simulator.
d. Duke Energy is on the CASL Industry Council and is supportive of VERA-CS validation.
e. Catawba is already part of the validation plan in Section 3.3 and detailed core power and fuel shuffle history will be available.

f. Catawba utilized both Westinghouse and Areva fuel. Westinghouse is a CASL core partner and Areva is on the CASL industry Council. ORNL is in optimum position to manage sensitive data as a neutral third party.
g. ORNL has already been funded to perform some of the MOX LTA analysis through additional programs and already has access to some of the data.

One drawbacks to this activity are that characterization of MOX currently has little application in the U.S. since MOX is currently not being used. Furthermore, it may be difficult to get proprietary data agreements in place for all the concerned parties (ORNL, Duke, Westinghouse, Areva, Shaw, etc.).

5.2 Three Mile Island

During Cycle 10 of Three Mile Island Nuclear Station - Unit 1 (TMI-1), some of the fresh fuel assemblies experienced localized cladding corrosion damage. A root cause assessment was initiated, which eventually led to hot cell examinations of four fuel rods discharged after Cycle 10 in 1995. The exams included axial gamma scans, micro gamma scans across a fuel pellet, pellet and axial burnup measurements, and other detailed tests [56].

Two independent facilities performed the radiochemical analyses: Argonne National Laboratory (ANL) and General Electric-Vallecitos (GE-VNC). Of 19 total samples, 11 were from rod H-6 of assembly NJ05YU, which had an initial enrichment of 4.013% ^{235}U and had local sample burnups of 45-56 GWd/MT over two cycles (9 and 10). Eight of the samples were from rods O1, O12, and O13 of assembly NJ070G, which had an initial enrichment of 4.657% ^{235}U and achieved burnups between 22-30 GWd/MT in one cycle (Cycle 10) [57].

Three Mile Island is a single unit B&W-type PWR as described in Section 3.7. It is operated by Exelon (on the CASL Industry Council) in Middletown, PA and is licensed to a rated power level of 2568 MW_{th} [2]. The reactor core is comprised of 177 15x15 fuel assemblies as shown in Figure 3-16. Cycle 10 was designed to a length of 661 Effective Full Power Days (EFPDs), and included 80 fresh fuel assemblies with both gadolinia and discrete B_4C-Al_2O_3 rods as burnable poisons. For in-core instrumentation, TMI-1 uses the seven level fixed SPND system described in Section 3.7. The Cycle 10 core loading pattern in shown in Figure 5-4 in octant symmetry [56].

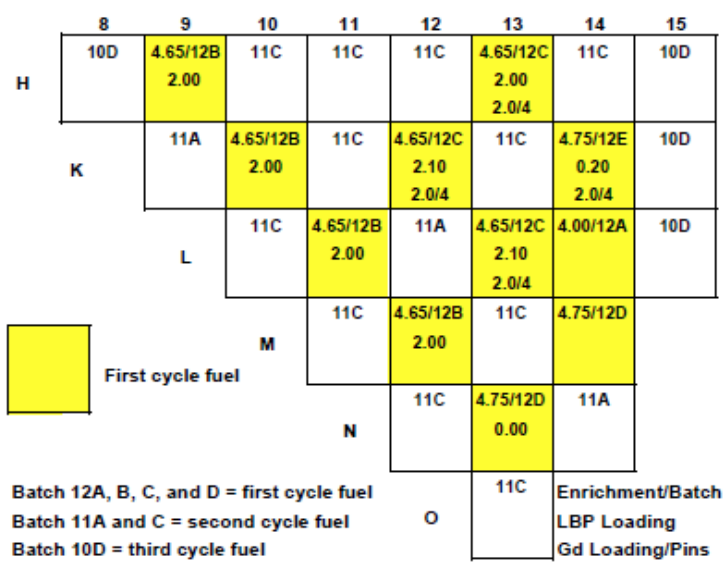


Figure 5-4 TMI-1 Cycle 10 Core Loading Pattern [56]

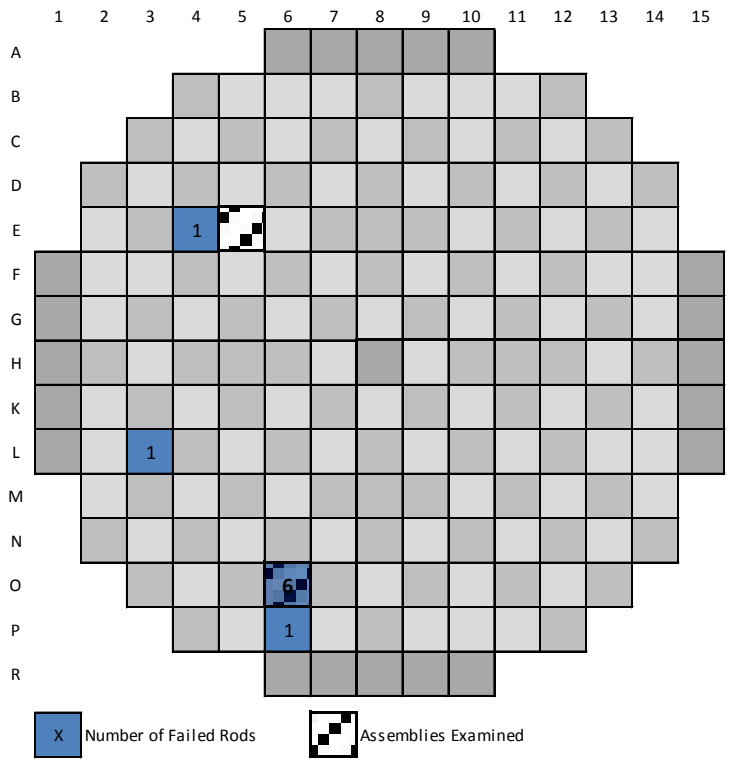


Figure 5-5 TMI-1 Cycle 10 Failed Rod Locations and Examined Assemblies [56]

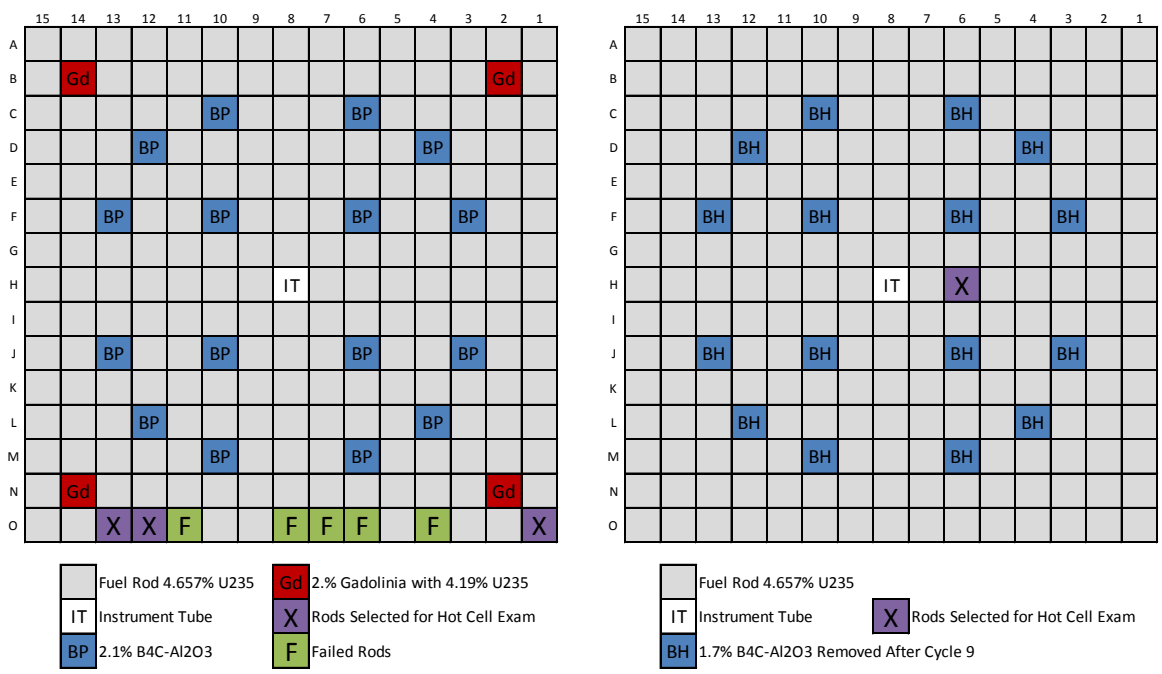


Figure 5-6 TMI-1 Cycle 10 Examined Rod Locations in Assemblies NJ070G and NJ05YU [56,57]

Figure 5-5 above displays the locations of failed and examined rods. It is noted that the rods in assembly NJ070G are predominated on the assembly periphery, face-to-face adjacent with another fresh fuel assembly (in the 'T' of the ring-of-fire pattern). Two of the rods are diagonally adjacent to gadolinia-bearing fuel rods, and there are 16 solid burnable absorber rods in the assembly guide tubes. One of these rods, likely the most limiting one in terms of power and CRUD deposition, is located on the corner of the assembly. Precisely determining the pin powers of the edge and corner

fuel rods in this configuration is a challenge for industrial nodal methods (using pin power reconstruction). The heavy solid absorbers depress the thermal flux near the center of the assembly, forcing more power to the peripheral rods. The burnup of the gadolinia is also sensitive to local conditions and can create localized power peaking when the gadolinia is significantly depleted. VERA-CS should be able to provide a more accurate prediction of the pin power history for these rods, assuming the data can be obtained for the adjacent assemblies (and preferably the entire core).

The hot cell examinations for TMI-1 began in 1997 and included the following [56]:

- Visual inspections
- Fission gas release measurements
- Gamma scans (axially)
- Micro gamma scans (radially)
- Neodymium pellet burnup measurements
- Metallography (cladding)
- Ceramography (fuel)
- Cladding hydrogen content

The measured burnup for the 19 samples are provided in Table 5-2 [57, 58]. The axial burnup profile for two rods, O1 and O12, are shown in Figure 5-10 along with previous calculations from industry methods.

Table 5-2 Measured Burnup from TMI Samples [57,58]

Assembly	Sample (Rod-Sample ID)	Initial Enrichment (wt% ^{235}U)	Axial Location from Tip of Bottom End Plug (cm)	Measured Burnup (GWd/MT)	Rod Average Burnup (GWd/MT)
NJ05YU	H6-A2	4.013	74.676	50.6	----
	H6-B2		115.062	50.1	
	H6-C1		235.458	50.2	
	H6-C3		156.21	51.3	
	H6-D2		322.072	44.8	
	H6-A1B		38.735	44.8	
	H6-B1B		155.956	54.5	
	H6-B3J		77.013	53.0	
	H6-C2B		194.615	52.6	
	H6-D1A2		261.899	55.7	
	H6-D1A4		292.379	50.5	
NJ070G	O1-S1	4.657	39.37	25.8	29.5
	O1-S2		197.104	29.9	
	O1-S3		278.13	26.7	
	O12-S4		39.37	23.7	25.9
	O12-S5		197.104	26.5	
	O12-S6		278.13	24.0	
	O13-S7		39.37	22.8	~27.6
	O13-S8		197.104	26.3	

Gamma scanning was performed on each of the fuel rods from assembly NJ070G. The scanning is recorded by integrating counts in the 0.5 to 3 MeV range, to discriminate on the 0.6617 MeV gammas released by the decay of ^{137}Cs. The measured activity of rods O12 and O13 were comparable, but the results from rod O1 were approximately 15% higher. The axial shapes had a normal appearance until the upper elevations were reached, when the activity appeared to taper off. Figure 5-7 and Figure 5-8 display the gamma scan results for fuel rods O1 and O11, respectfully.

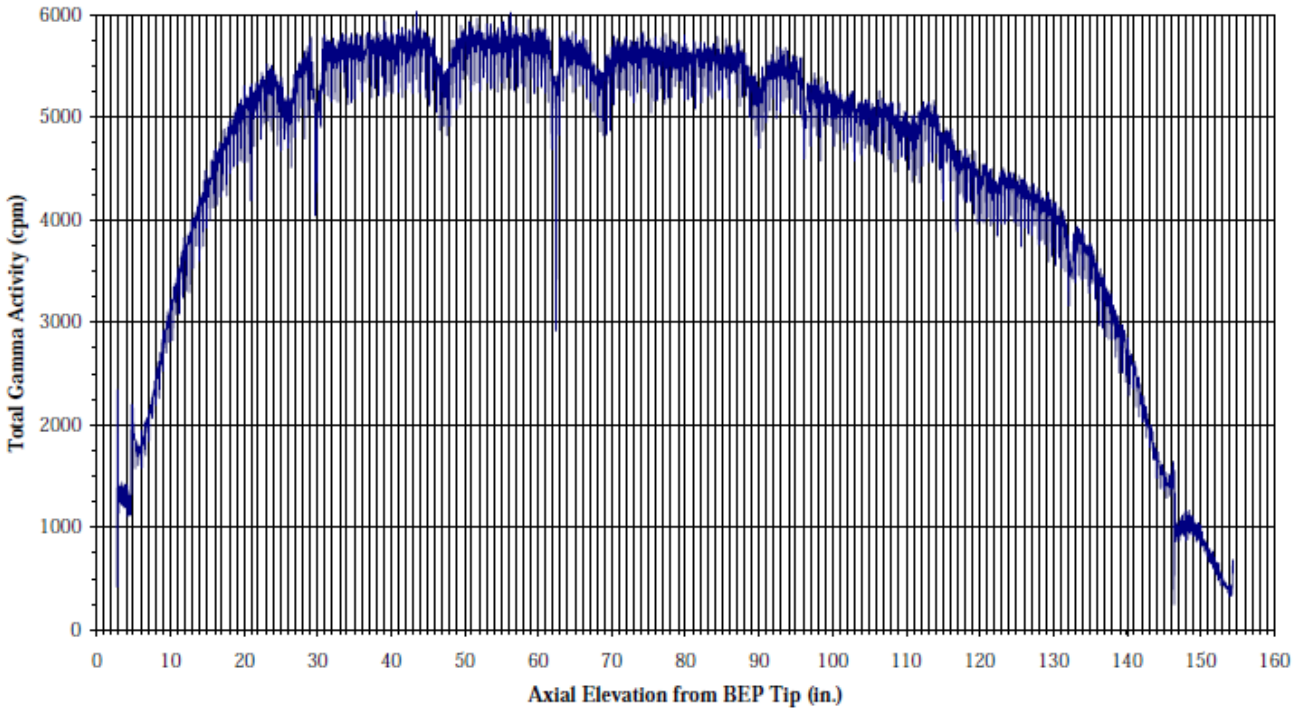


Figure 5-7 TMI-1 Axial Gamma Scan of Corner Fuel Rod O1 in Assembly NJ070G [56]

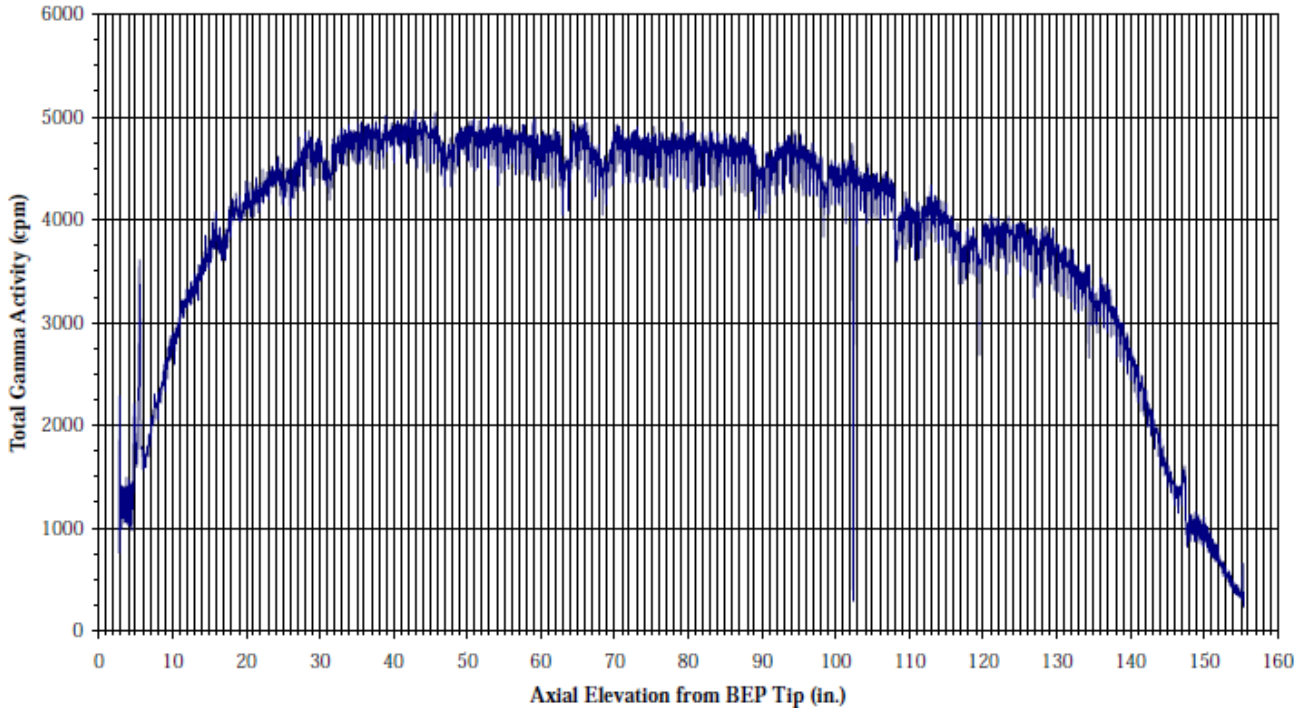


Figure 5-8 TMI-1 Axial Gamma Scan of Fuel Rod O11 in Assembly NJ070G [56]

In addition to axial gamma scanning, diametral gamma scans were performed at different axial locations using cross-sectional slices of fuel pellets, in order to determine if the rods were subjected to a large power gradient. Measurements were taken for activities of ^{134}Cs, ^{137}Cs, and ^{106}Ru from samples from rods O1 and O12, at axial locations of 80 and 119 inches (from the tip of the bottom end plug (BEP)). In the results in Figure 5-9, a slight burnup asymmetry is evident across the samples from rod O1.

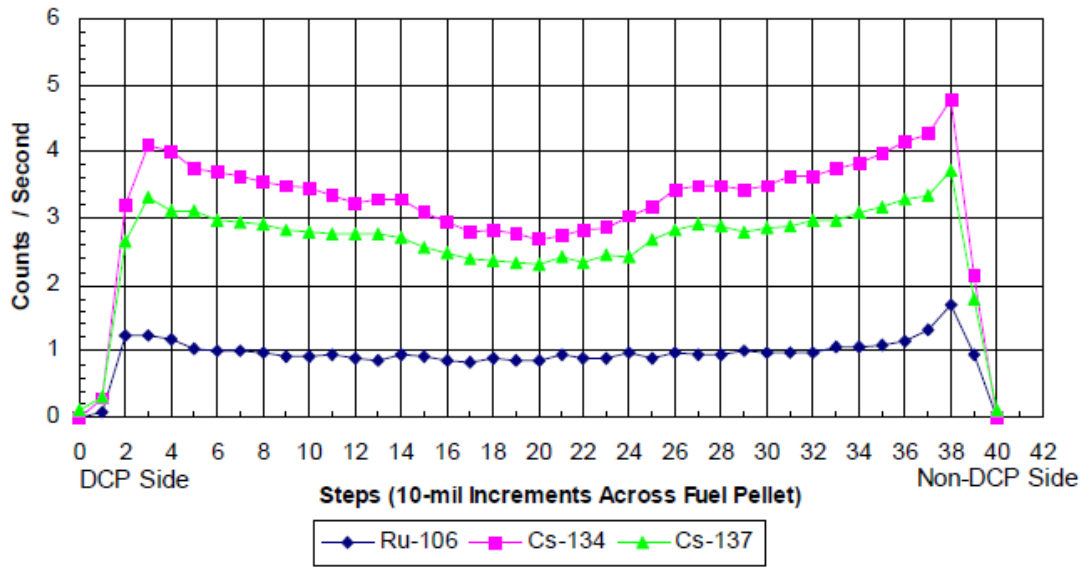


Figure 5-9 TMI-1 Diametral Gamma Scan at 120" of Fuel Rod O1 in Assembly NJ070G [56]

Fuel rod burnup was determined by chemical separation and mass spectrometric analysis of U, Pu, and Nd isotopes. The axial burnups are based on axial gamma scans, neodymium pellet burnup measurements, and the pellet micro gamma scans. Average normalization factors for converting ^{137}Cs activities to comparable ^{148}Nd benchmark burnups where determined to be 1.036 for rod O1 and 1.027 for rod O12. Using these results the inferred axial burnup profiles are shown in provided in Figure 5-10. It is noted that the noticeable depression at the top of the rods has been attributed to Crud Induced Power Shift (CIPS).

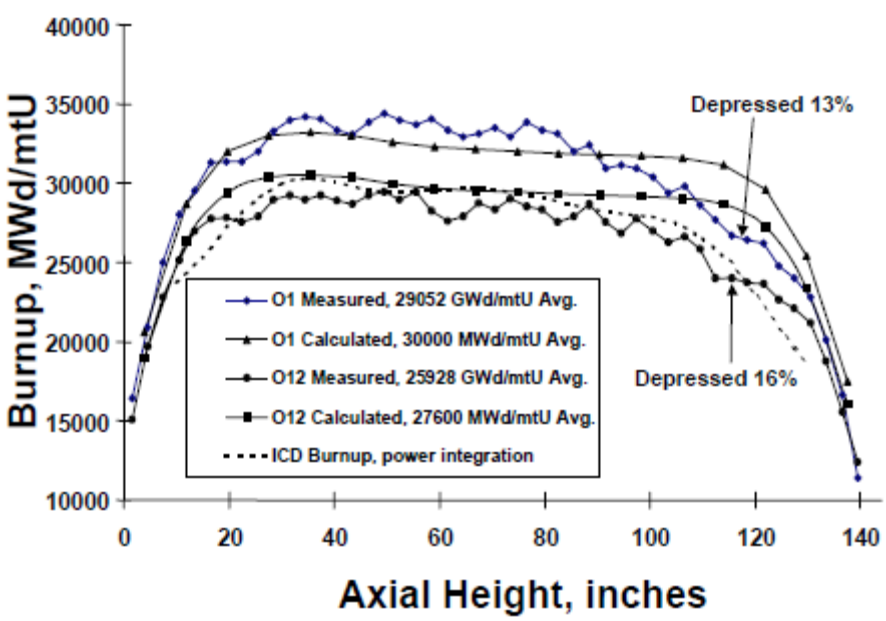


Figure 5-10 TMI-1 Measured Axial Burnups for rods O1 and O12, MWd/MT [56]

Isotopic measurements are available for the 19 samples of TMI-1 fuel for the major actinides (U and Pu isotopes), as well as ^{245}Cm, ^{244}Cm, ^{243}Cm, ^{242}Cm, ^{243}Am, ^{242m}Am, ^{241}Am, ^{237}Np, ^{150}Nd, ^{148}Nd, ^{146}Nd, ^{145}Nd, ^{143}Nd, ^{155}Gd, ^{153}Eu, ^{151}Eu, ^{152}Sm, ^{151}Sm, ^{150}Sm, ^{149}Sm, ^{147}Sm, ^{137}Cs, and ^{134}Cs. It is noted in Reference 57 that some of the largest errors between the measurements and previous predictions occurred in the corner rod O1.

The Three Mile Island hot cell examinations provide a good source for VERA-CS isotopic validation. With EPRI as a core partner, and Exelon and Areva as Industry Council members, it is likely that CASL could obtain all the data needed for the mutli-cycle full-core simulations. If the detailed reactor operating history can be obtained, at least starting in Cycles 7 or 8, then an integrated validation with VERA-CS can be performed using the radiochemical assay and gamma scan results for the corner and peripheral pins. The VERA-CS results should be much better than any previously used methods, assuming the uncertainty from the fuel failure (CIPS/CILC) does not alter the expected power history too significantly. This activity could be mutually beneficial to EPRI, Areva, Exelon, and other operators of B&W plants. This would also be an interesting application for the CRUD capabilities being developed by CASL.

5.3 MALIBU High Burnup Program

The MALIBU experimental programme performed post-irradiation examinations (PIE) on three samples of PWR fuel irradiated for four cycles in the GÖSGEN PWR [59]. The data currently provided includes only sample and core power histories, but no core loading maps and no assembly power histories. ORNL will attempt to obtain this data, but in the event it cannot, this validation problem will likely be removed from the plan.

5.4 Robinson

Radiochemical analyses of four fuel samples from H. B. Robinson Unit 2 were performed at the Materials Characterization Center (MCC) at Pacific Northwest Laboratory (PNL) as part of the Approved Testing Materials (ATM) program. The fuel sample material, designated ATM-101, was obtained from assembly BO-5 that was irradiated for the first two reactor operating cycles. The fuel rods were cut into segments at the Idaho Nuclear Engineering Laboratory and transported to the PNL hot cells at Hanford Engineering and Development Laboratory (HEDL) for destructive radiochemical analyses of the samples [60, 61].

Robinson Unit 2 is a Westinghouse 3-loop PWR design which began operations in 1970. It is currently rated at 2339 MW_{th}. It is operated by Duke Energy [2]. At the time of sample irradiation, the plant was rated at 2192 MW_{th} and was operated by the Carolina Power and Light Company. The core contains 157 15x15 fuel assemblies manufactured by Westinghouse. ATM-101 was initially 2.561% [60] enriched UO_2 and was irradiated over the first two fuel cycles to approximately 30 GWd/MT. The position of assembly BO-5 for the first two cycles is shown in Figure 5-11.

Nine fuel rods where extracted from assembly BO-5 for examination. The rod locations are provided in Figure 5-12. The power history for only the assembly is provided in Reference 61, and is shown in Figure 5-13. The core operating history is not provided, but may possibly be obtained from Duke Energy.

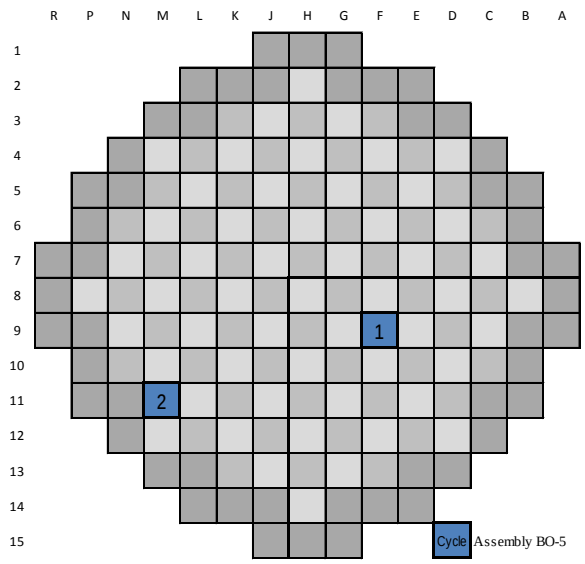


Figure 5-11 Robinson Assembly BO-5 Shuffle History [61]

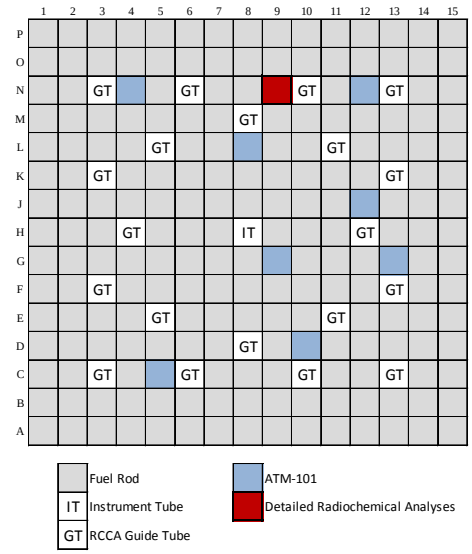


Figure 5-12 Robinson Assembly BO-5 [61]

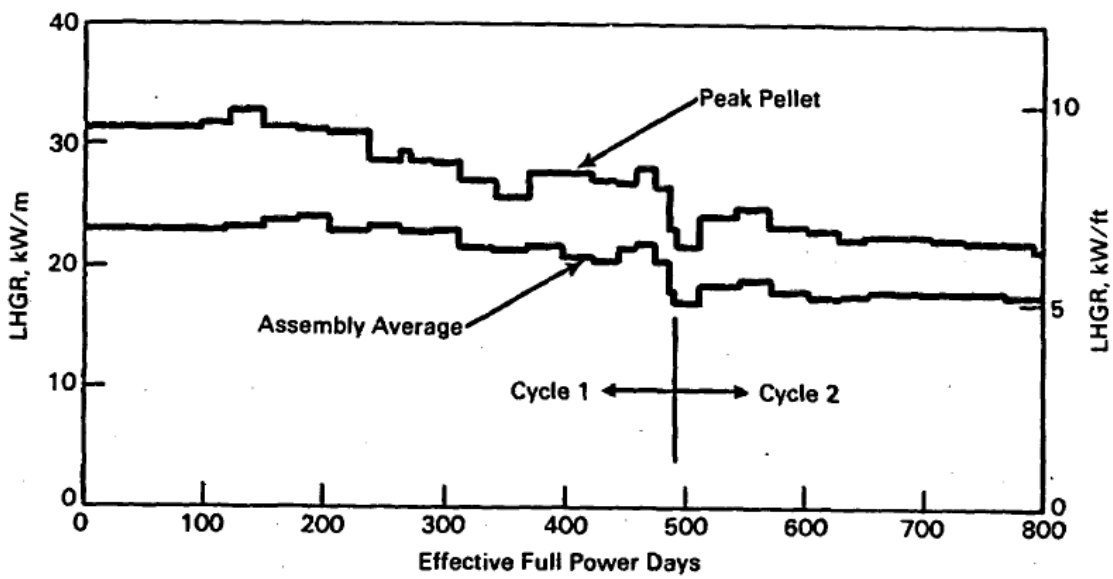


Figure 5-13 Robinson Assembly BO-5 Power History [61]

Characterization of the fuel rods obtained from Robinson included the following:

1) fission gas release measurements
2) ceramography and metallography examinations
3) fuel burnup measurements and correlations
4) gamma scanning
5) radionuclide inventory measurements

Four samples from rod N-9 were used for burnup analyses based on the ^{148}Nd concentration (Table 5-3). These were used to correlate the rod burnups with the ^{137}Cs activity measured for each rod axially during gamma scanning. Rod average burnups were then determined for each of the nine rods. As example of the gamma scan results for rod N-9 in shown in Figure 5-14.

Table 5-3 Summary of Robinson Samples [60,61]

Assembly	Sample	Initial Enrichment (wt% ^{235}U)	Axial Location from Bottom of Fuel Stack (cm)	Measured Burnup (GWd/MT)	Rod Average Burnup (GWd/MT)
BO-5	N-9B-S	2.561	11	16.02	28.4
	N-9B-N		26	23.81	
	N-9C-J		199	28.47	
	N-9C-D		226	31.66	

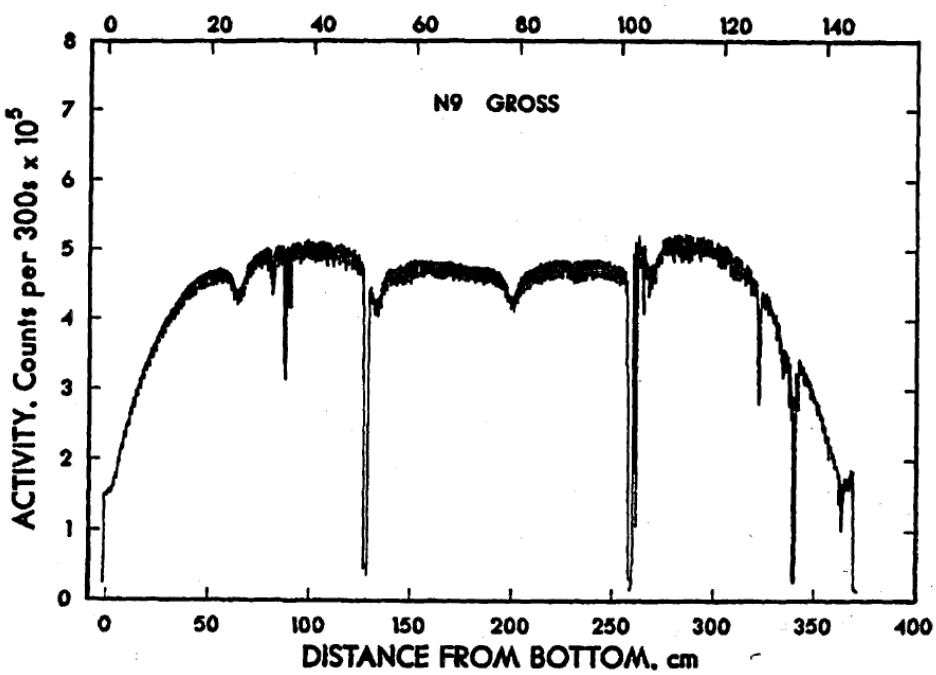


Figure 5-14 Robinson Gamma Scan for Assembly BO-5 Rod N-9 [61]

Radiochemical Analyses (RCAs) included measured isotopes of uranium, plutonium, neodymium, ^{237}Np, ^{99}Tc, and ^{137}Cs. The decay time for the samples was approximately ten years [60].

5.5 Calvert Cliffs

As part of the U.S. Department of Civilian Radioactive Waste Management Program, spent fuel from Calvert Cliffs Nuclear Power Plant, Unit 1, was extensively examined by the Materials Characterization Center (MCC) at Pacific Northwest Laboratory (PNL). Results included radionuclide inventory, cladding characteristics, and fission product redistribution. The spent fuel was obtained from three assemblies classified as Approved Testing Material 103 (ATM-103), ATM-104, and ATM-106 [60, 62, 63, 64].

Calvert Cliffs is a Combustion Engineering (CE)-designed PWR currently owned by Constellation Energy and located in Lusby, MD [2]. The rated power (at the time of irradiation) was 2560 MW_{th}. The fuel rods examined were from standard CE 14x14 fuel assemblies D047, D101, and BT03 (Figure 5-11). The assemblies were loaded as fresh fuel in Cycles 1 (BT03) and 2 (D047,D101) in 1977 and were irradiated for three (D101) or four (BT03,D047) cycles, including shuffling during each refueling outage (Figure 5-12). The fuel rods were fabricated using uniform UO_2 pellets and

Zircolay-4 cladding. Assembly BT03 contained twelve B_4C-Al_2O_3 burnable absorber rods, while the prior two were unpoisoned [60]. Decay times for each of the assemblies were 6.5, 5.1, and 6.7 years, respectively.

One rod from each assembly was used for performing detailed destructive examinations. The enrichment and measured burnups of these rods are provided in Table 5-4. The exams included:

1) gamma scanning
2) fission gas analyses
3) ceramography of the fuel
4) metallography of the cladding
5) electron probe microanalyses
6) analytical transmission electron microscopy
7) fuel burnup analysis (based on ^{148}Nd)
8) radiochemical analyses of the fuel and cladding

Table 5-4 Summary of Calvert Cliffs Samples [60,62,63,64]

Assembly	Sample	Initial Enrichment (wt% ^{235}U)	Axial Location from Bottom of Fuel Stack (cm)	Measured Burnup (GWd/MT)	Rod Average Burnup (GWd/MT)
D101	103-MLA098-JJ	2.72	11.68	18.68	~30
	103-MLA098-BB		27.09	26.62	
	103-MLA098-P		164.53	33.17	
D047	104-MKP109-LL	3.038	12.70	27.35	39.6
	104-MKP109-CC		27.41	37.12	
	104-MKP109-P		164.09	44.34	
BT03	106-NBD107-MM	2.453	14.06	31.40	~43
	106-NBD107-GG		22.70	37.27	
	106-NBD107-Q		163.89	46.46	

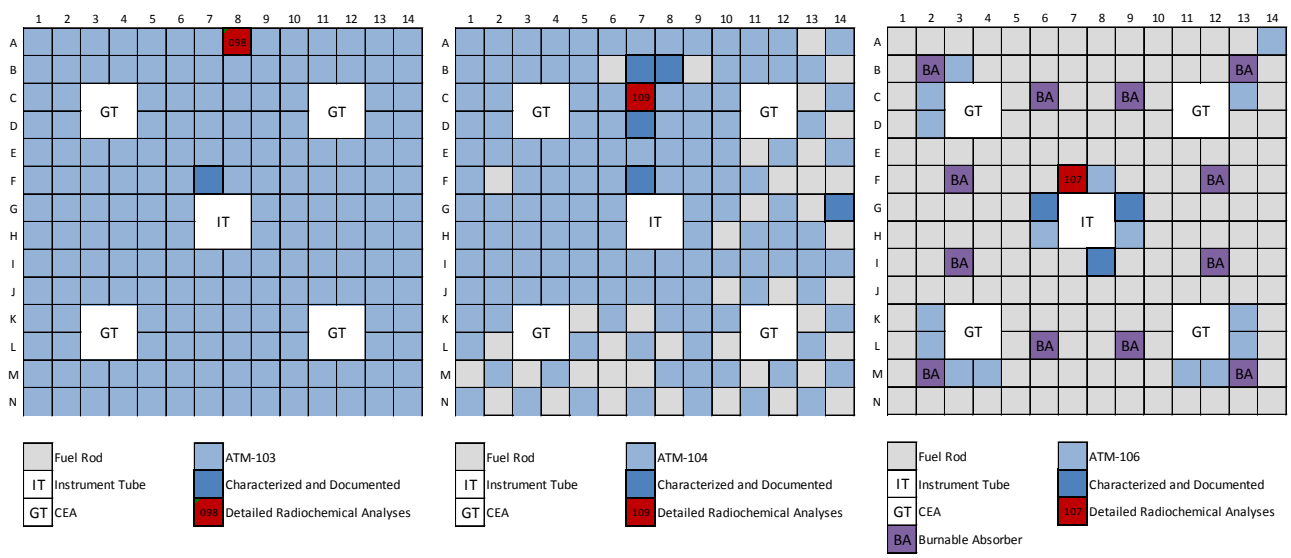


Figure 5-15 Calvert Cliffs Assemblies D101, D047, and BT01 [60,62,63,64]

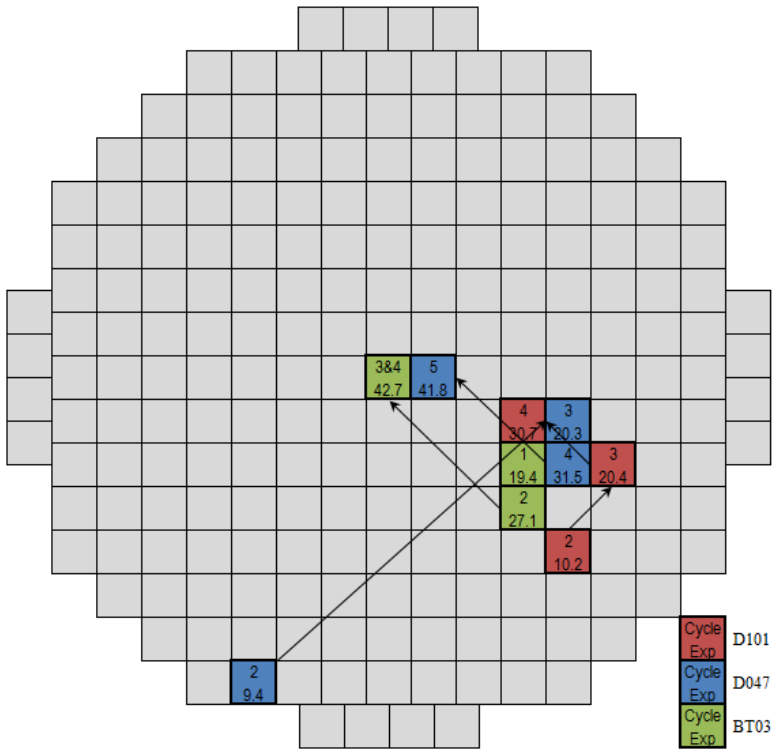


Figure 5-16 Calvert Cliffs Assembly Shuffle Histories and EOC Exposures (GWd/MT) [60,62,63,64]

The references do not contain the detailed operating history for Cycles 1-5, nor the full core shuffle data. Ideally, VERA-CS could be used to model the entirely of these cycles, including reactivity and power distribution comparisons, if this data could be obtained, and provide a direct prediction of the isotopic inventory. If lieu of this, the estimated power history for each assembly is provided based on CE M&S results at the time. The histories for the three rods are shown in Figure 5-13, and was extracted from data reported by CE [62,63,64].

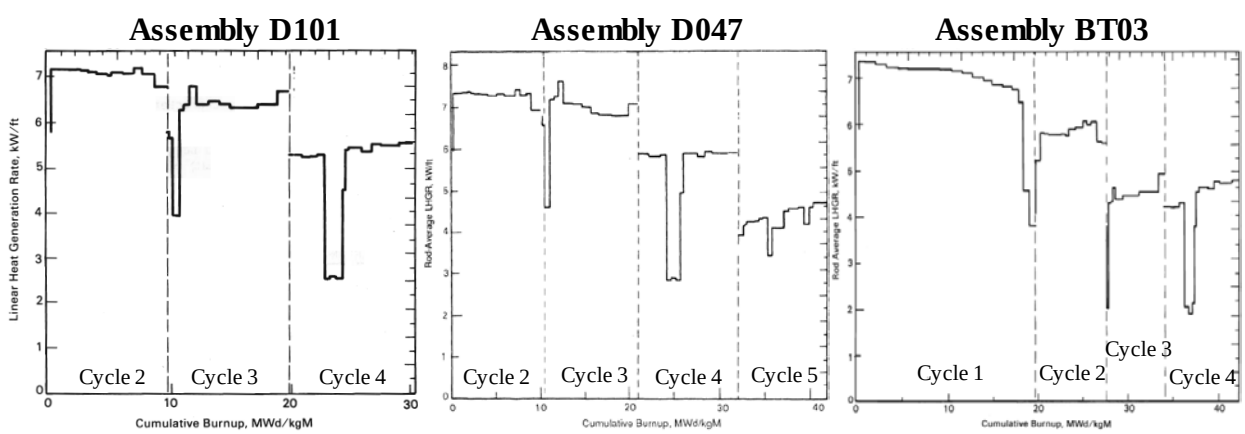


Figure 5-17 Calvert Cliffs Assembly Power Histories [62,63,64]

Radiochemical assays (RCAs) were performed at three axial locations on each rod to determine the bulk concentrations of radionuclides of interest to spent fuel disposal. These isotopes were: ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am, $^{243+244}$Cm, ^{79}Se, ^{90}Sr, ^{99}Tc, ^{126}Sn, ^{135}Cs, and ^{137}Cs [63]. Later measurements were performed to provide additional data for fission products with large neutron cross sections important to nuclear criticality safety, including isotopes of cesium, and the lanthanides samarium, europium, and gadolinium [60]. Additionally, the

calculated results from ORIGEN2 are also provided for comparison. More recent calculated results are also available from Reference 60, where a 2D NEWT/TRITON model was used for the depletion.

In addition to the measured radionuclide activities and burnup of each sample, the ATM reports (References 61-64) also provide gamma scan results for several of the fuel rods in each experiment group, including the rod selected for detailed radiochemical assay. These detailed axial scans are based on ^{137}Cs activity, and could be used to compare directly to the axial pin-wise distribution of ^{137}Cs calculated by VERA-CS. A sample gamma scan is provided in Figure 5-14. In addition to the shape of the ^{137}Cs activity (normalized to the radiochemical assay result for burnup), the references demonstrate a very linear dependence of the ^{137}Cs activity on fuel burnup, allowing for the determination of the fuel burnup in any of the gamma scanned rods. In addition to the rods undergoing destructive testing, there is one additional rod in assembly D101, four additional rods in assembly D047, and three additional rods in assembly BT03, in some cases also including the measured activity of ^{134}Cs (0.6 and 0.8 MeV).

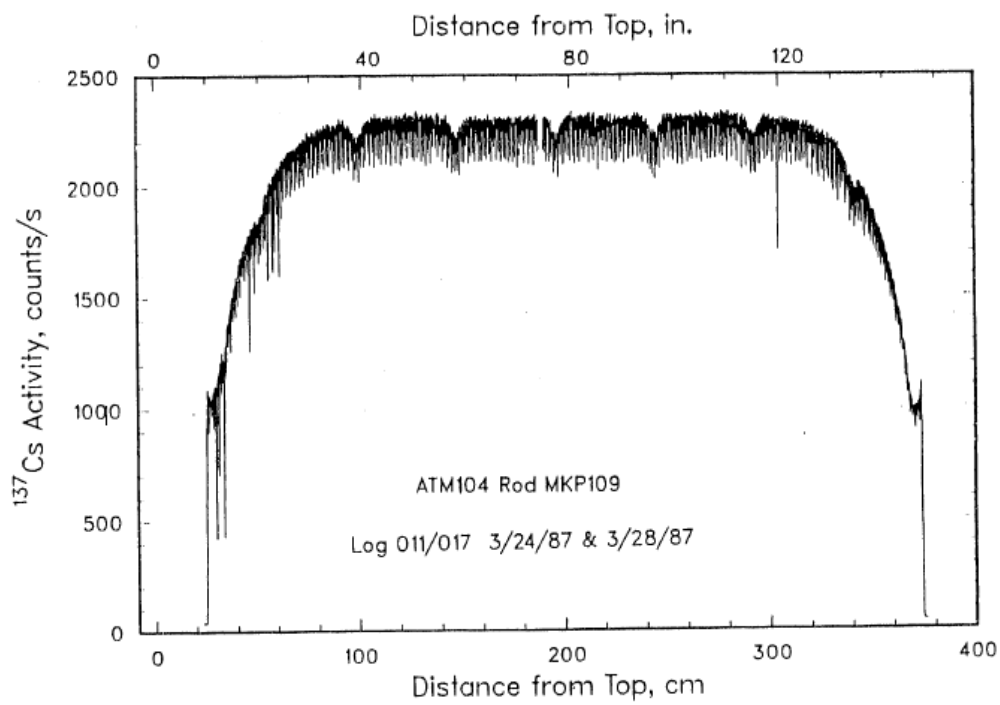


Figure 5-18 Calvert Cliffs Gamma Scan for Assembly D047, Rod MKP109 [63]

Finally, electron probe microanalysis and autoradiography were used on ATM-104 to measure the radial distribution of some fission products and actinides in the fuel pellets. The distribution occurs because of the high thermal neutron flux near the pellet edge due to the proximity of the moderator. Deeper inside the pellet, the thermal flux is relatively constant. This increase in flux results in more fissions and absorptions, and thus more fission products and actinides near the surface of the pellet. The radial burnup distribution is inferred from the radial neodymium concentration and results from the three radiochemical assays. This data may provide a good validation source for the intra-pin distributions calculated by VERA-CS. Figure 5-15 displays some of the results from Reference 63.

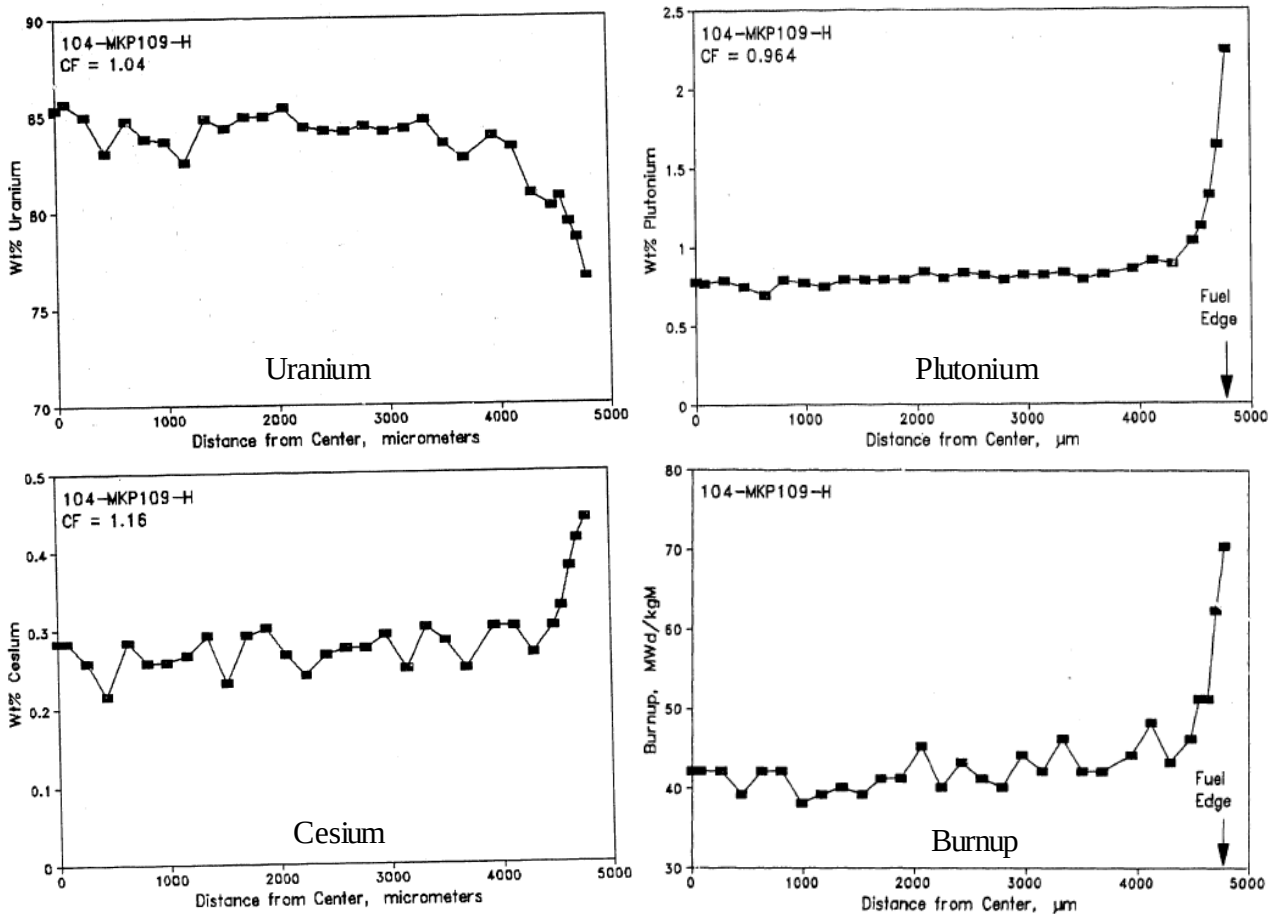


Figure 5-19 Calvert Cliffs Radial Pellet Distributions in Rod MKP109 [63]

6. CONTINUOUS ENERGY MONTE CARLO BENCHMARKS

This portion of the validation plan uses high-fidelity reactor simulations to augment the measured data discussed in the previous three sections. The use of continuous energy (CE) cross sections and Monte Carlo physics for particle transport represents the highest level of accuracy achievable by M&S tools, given enough particle histories to reduce the uncertainty in the results to an acceptable level. Monte Carlo simulations have increasing become more prominent because of faster computers and more efficient parallel algorithms [10], as well as improved methods for Doppler broadening and thermal scattering interpolation, providing the most reliable numerical reference solution for problems with deterministic solutions. For the purposes of VERA-CS validation, Monte Carlo tools will be used for two main purposes:

1) Monte Carlo transport solutions will be used for reference when measured data does not exist. For instance, this could include:
    a. Pin powers for full power reactor geometries, including simulation of the localized effects of spacer grids, control rod tips, or radial reflector/structure
    b. Pin powers for operating conditions requiring thermal-hydraulic feedback
    c. Pin powers for depleted fuel
    d. Intra-pin powers for fuel pellets with a radial temperature distribution
    e. Intra-pin powers for fuel pellets subjected to an azimuthally asymmetric flux gradient, such as a fuel rod adjacent to a large water rod or gadolinia rod
    f. In-core instrumentation response for cases where moveable detectors are not inserted, or for more axial detail in a fixed in-core system
    g. Effects due to thick (also known as heavy) steel shrouds
    h. Gamma energy deposition
    i. Ex-core detector responses

2) Monte Carlo may be used as an intermediate validation step for VERA-CS for experiments that cannot be easily simulated with the VERA-CS inputs and models. In this two-step approach, the Monte Carlo code is validated first against the measurement, and then VERA-CS is validated against the Monte Carlo code for the same or very similar problem. If CASL is validating a Monte Carlo code in addition to VERA-CS, then this mode may be available either way. The drawback is that the uncertainty in the VERA-CS results must be calculated as a statistical combination of the uncertainty in each of the two sub-steps. This means that it is likely simpler and will result in better results to model the experiments directly with VERA-CS whenever possible.

Use of a Monte Carlo code for validation of VERA-CS presumes that code already has a reasonable pedigree of validation for similar applications. This could mean direct benchmarking against another similar data source/problem, or it could mean validation of more coarse results of the same problem. For instance, validation of pin powers during a flux map could require comparison of both the Monte Carlo and VERA-CS instrument responses to measured flux traces, then comparison of VERA-CS pin powers to Monte Carlo pin powers. The pin power uncertainty would then be presumed to be the statistical combination of the uncertainty on the Monte Carlo code instrument responses (compared to the measured) and the uncertainty in the VERA-CS pin powers (compared to Monte Carlo). In some way, the confidence in the Monte Carlo code must be established before use as a benchmark is acceptable.

Several Monte Carlo codes are currently in use or under development that can provide a reliable high-fidelity reference solution for VERA-CS. These are listed below, in no particular order.

- KENO-VI, developed at ORNL, is a 3D generalized geometry Monte Carlo particle transport program for shielding and criticality safety analysis with a long history of validation in the SCALE code system [65]. CASL has extensively tested KENO-VI and applied it to hundreds of PWR reactor physics problems, recently benchmarking it against startup physics data from Watts Bar Unit 1 Cycle 1 [66, 67]. 2D lattice comparisons with MCNP have shown excellent agreement [74]. KENO-VI is moderately parallel and scales to 200-300 computing cores. Full core models with low pin power uncertainty are possible but difficult, with 100e9 particles requiring nearly a month of runtime. The latest development version of KENO-VI has Doppler broadening and thermal scattering interpolation for the CE cross sections, as well as Doppler upscatter treatments.

- MCNP, developed at LANL, is a general purpose Monte Carlo transport code used for a variety of nuclear engineering applications (criticality, shielding, reactor physics, etc.). MCNP has historically been the standard reference solution for neutronics code validation. MCNP is also limited in parallel scalability so determination of fission rate distributions for large problems with low uncertainties is possible but can be difficult [68].

- Shift is a new Monte Carlo capability being developed at ORNL through CASL and other programs. PWR geometries supported by the VERA common input can be easily modeled, and a python-based scripting front end is also available for more complicated geometries. The normalized fission rate distribution in the fuel is automatically produced consistently with VERA output specifications. Shift is massively parallel and has demonstrated excellent scalability on the Oak Ridge Leadership Computing Facility (OLCF) supercomputer Titan [69]. It has been utilized by CASL on up to 250,000 cores (15,625 nodes) using two trillion particle histories [22]. Quarter-core pin-by-pin comparisons between KENO-VI and Shift show good agreement with 0.56% RMS and 2.8% maximum difference. These differences are influenced by a minor 1% radial tilt in the power distributions between the codes, and are also partially an effect of the uncertainties in the KENO-VI solution, which after nearly a month of runtime still used only $1/10^{th}$ of the particle histories as Shift [70]. In the future, Shift will inherit the CE cross section treatment capabilities of KENO-VI (Doppler broadening, thermal scattering, and Doppler upscatter).

- A new Monte Carlo code called OpenMC is currently under development at the Massachusetts Institute of Technology (MIT) as a tool for simulation on high-performance computing platforms. Given that many legacy codes do not scale well on existing and future parallel computer architectures, OpenMC has been developed from scratch with a focus on high performance scalable algorithms as well as modern software design practices [71]. OpenMC has been used to setup and deplete the realistic BEAVRS benchmark problem, including thermal-hydraulic feedback, depletion, and in-core instrument responses [7]. MIT is a core partner in the CASL consortium and collaborates extensively with the Shift development team in the Radiation Transport Methods Focus Area.

- MC21 is a continuous energy Monte Carlo neutron and photon transport code under joint development by Bechtel Marine Propulsion Corporation at the Bettis Atomic Power Laboratory and the Knolls Atomic Power Laboratory (KAPL). MC21 is the Monte Carlo transport kernel of a system of codes that provides an automated, computer-aided modeling

and post-processing environment. MC21 is designed with reactor analysis calculations in mind, and includes in-line reactor feedback effects including depletion, thermal/hydraulics, xenon, eigenvalue search, and neutron and photon heating [72]. KAPL is on the CASL Industry Council and is collaborated with the CASL team on the BEAVRS and VERA benchmark problems.

- McCARD is a Monte Carlo neutron and photon transport simulation code developed by Seoul National University (SNU). It has been developed exclusively for the neutronics design of nuclear reactors and fuel systems. It is capable of performing whole-core neutronics calculations with temperature feedback, reactor fuel burnup, the few group diffusion constant generation, sensitivity and uncertainty (S/U) analyses, and uncertainty propagation analysis. It has some special features such as the anterior convergence diagnostics, real variance estimation, B1 theory-augmented few group constants generation, kinetics parameter generation and S/U analysis based on the use of adjoint flux [73]. Though it is being developed in Korea, planned collaboration with SNU may lead to future benchmarking opportunities with McCARD and/or alternate confirmatory solutions for difficult Monte Carlo problems.

Note that use of KENO-VI and Shift for VERA-CS Validation excludes base cross section data as a source of disagreement. All of these methods will use AMPX processed ENDF data, though the data treatment may be different for different methods. In particular, the KENO-VI and Shift data methods will be fairly similar.

6.1 Pin-by-Pin Fission Rates

Continuous energy Monte Carlo solutions will be used as a validation source for 2D and 3D pin-by-pin fission rate distributions for operating power plant conditions. Detailed Monte Carlo models have been developed for several power plant reactor startup configurations with fairly fine axial meshing (40 to 50 axial levels) and detailed fuel rod, burnable poisons, and assembly and reactor structural representations [3, 74]. Representations of spacer grids, end plugs, thimble plugs, and the core baffle are included. In most cases, the instrument thimble tubes are removed from the models to take advantage of $1/8^{th}$ core symmetry for the reaction rate tallies.

Benchmark cases with measured plant validation of coarser results are preferred, such as flux maps, startup critical conditions, etc. In this way the accuracy of the Monte Carlo code's coarser predictions (reactivity, assembly power shape) can be confirmed, in turn providing more confidence in the predicted pin power distributions. The validity of the 2D Monte Carlo results (without partial control rods) may be included to obtain lower estimated uncertainties in the fission distribution, while the validity of the 2D results is inferred from the performance of the 3D cases.

Both HZP and HFP cases will be considered. HZP cases have already been generated for several plants, located in the subsequent sections. HFP flux map cases will be included in the future when either 1) Shift has the capabilities to generate fission rates at operating conditions (with T/H feedback and depletion, or from VERA-CS restart files), or 2) the solutions are obtained from one of the other codes.

Note that in the results presented below that Shift is still under development and it is believed that the reported estimated uncertainty in power distribution is likely 2-4 times too low.

6.1.1 Watts Bar Nuclear Unit 1

CASL has generated several large CE Monte Carlo solutions for the initial criticality of Watts Bar 1 Cycle 1 at HZP isothermal conditions. Watts Bar 1 is described in Section 3.1.1. For this case, the regulating Bank D was at 167 steps withdrawn and the soluble boron concentration was 1293 ppm.

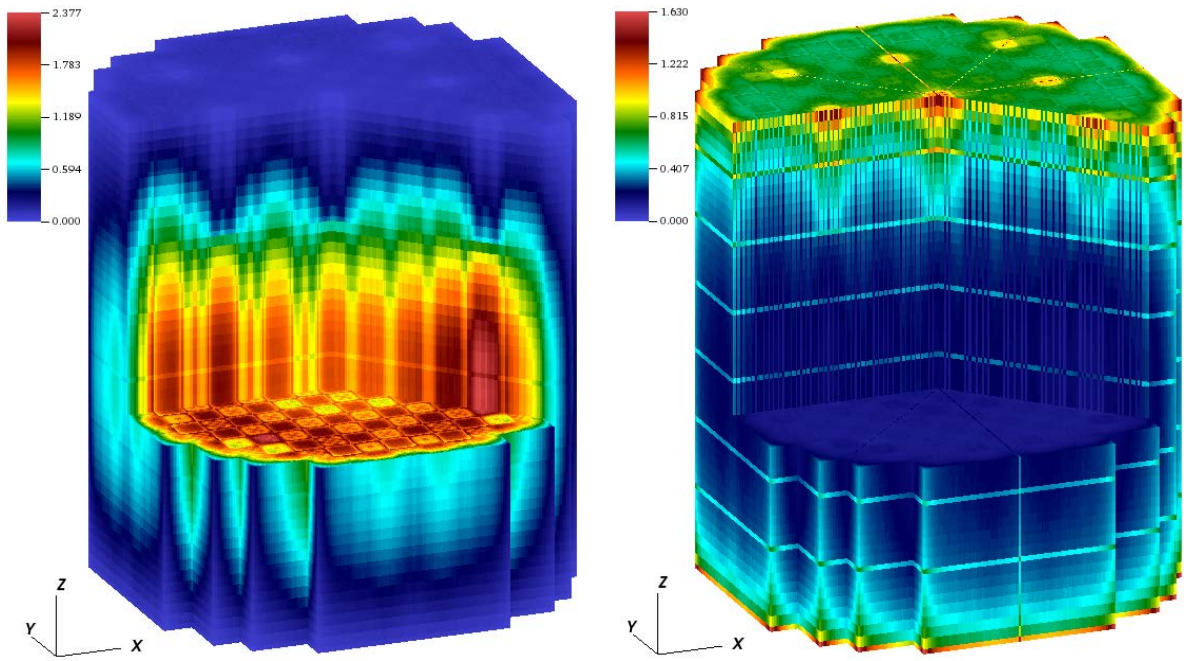
Reference 3 (and presented in Reference 67) generated the fission rate distributions from a very large quarter-core KENO-VI model. The KENO-VI model included detailed radial core structure (baffle, barrel, neutron pads, and vessel). The estimated uncertainties in the 3D results are larger than desired, so comparable 2D models (uncontrolled and controlled) using a variety of radial reflector models were also generated and contrasted.

Additionally, Reference 70 generated the fission rate distribution from Shift for the exact same conditions. With Shift, the radial core structure only includes the thin core baffle. One trillion particle histories were used to significantly reduce the estimated uncertainties. 2D results were also generated.

For these cases, the predicted fission rates agree between KENO-VI and Shift with a 0.56% RMS difference. There is a minor ~1% deviation in radial power prediction, with Shift predicting higher in the center of the core. As shown in the references, both codes match measured data well in terms of criticality and control rod worth. The largest difference in control bank worth between KENO-VI and Shift is 1.3% for a small worth bank. For this case, no measured power distribution data is available. Table 6-1, Figure 6-1, and Figure 6-2 provide results from these cases.

Table 6-1 Watts Bar 1 Representative Monte Carlo Parameters and Uncertainties [3,70]

	2D Core		3D Core	
	KENO-VI	**Shift**	**KENO-VI**	**Shift**
Total # Particles	25e9	100e9	100e9	1e12
# Particles/Generation	5e6	50e6	10e6	500e6
# Generations	5,000	2,000	10,000	2,000
# Skipped Generations	250	250	500	500
# Cores	300	36,640	180	240,000
Memory/Core	< 4 GB	< 2 GB	10.7 GB	< 2 GB
Runtime	~6 days	1.8 hours	29 days	3.2 hours
Eigenvalue Uncertainty	± 0.8 pcm	± 3.2 pcm	± 0.25 pcm	± 2.8 pcm
Average Pin Power Uncertainty	± 0.06%	± 0.011%	± 0.21%	± 0.02%
Maximum Pin Power Uncertainty (by Power)	P < 1.0: ± 0.14% P > 1.0 ± 0.09%	P < 1.0: ± 0.03% P > 1.0 ± 0.03%	P < 1.0: ± 1.63% P > 1.0: ± 0.41%	P < 1.0: ± 0.09% P > 1.0 ± 0.05%



Normalized Fission Rates

Relative Uncertainty (%)

Figure 6-1 CE KENO-VI 3D Normalized Fission Rates for WBN1 Initial Criticality [3]

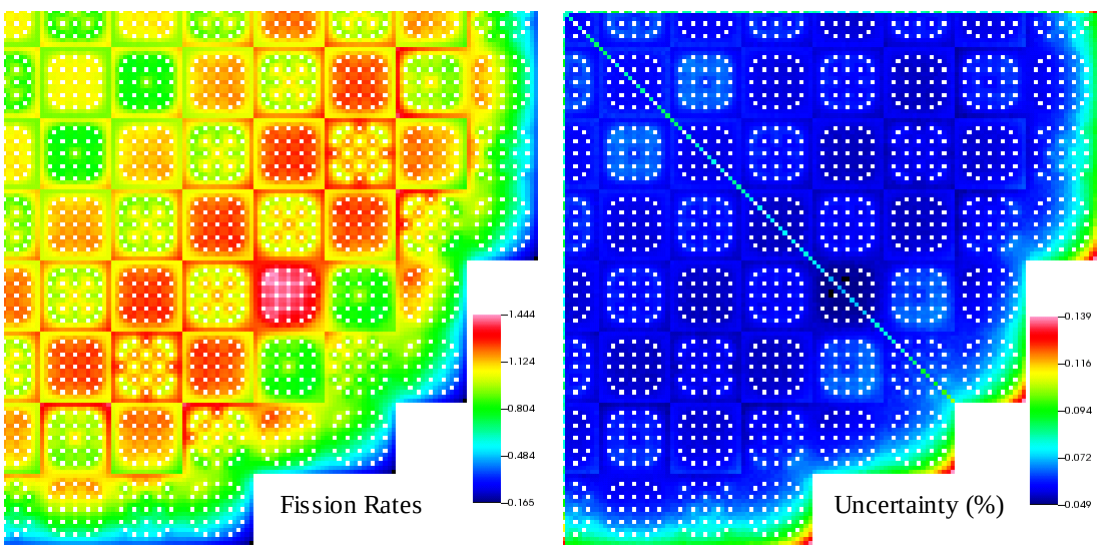


Figure 6-2 CE KENO-VI 2D Normalized Fission Rates for WBN1 HZP ARO [3]

As Shift gains capability, it will be used to generate full core pin-by-pin normalized fission rate distributions for selected HFP flux map cases in the Cycle 1 depletion. The VERA-CS pin powers can then be compared to Shift results while its instrument response is compared to measured data. Alternately, the OpenMC or MC21 results for this case may also be available in the future.

6.1.2 BEAVRS

The MIT Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS)[7] is becoming a standard reactor benchmark calculation across the nuclear industry. The BEAVRS reactor is described in Section 3.2. Currently the following codes have been used to generate 3D solutions to a subset of the benchmark cases.

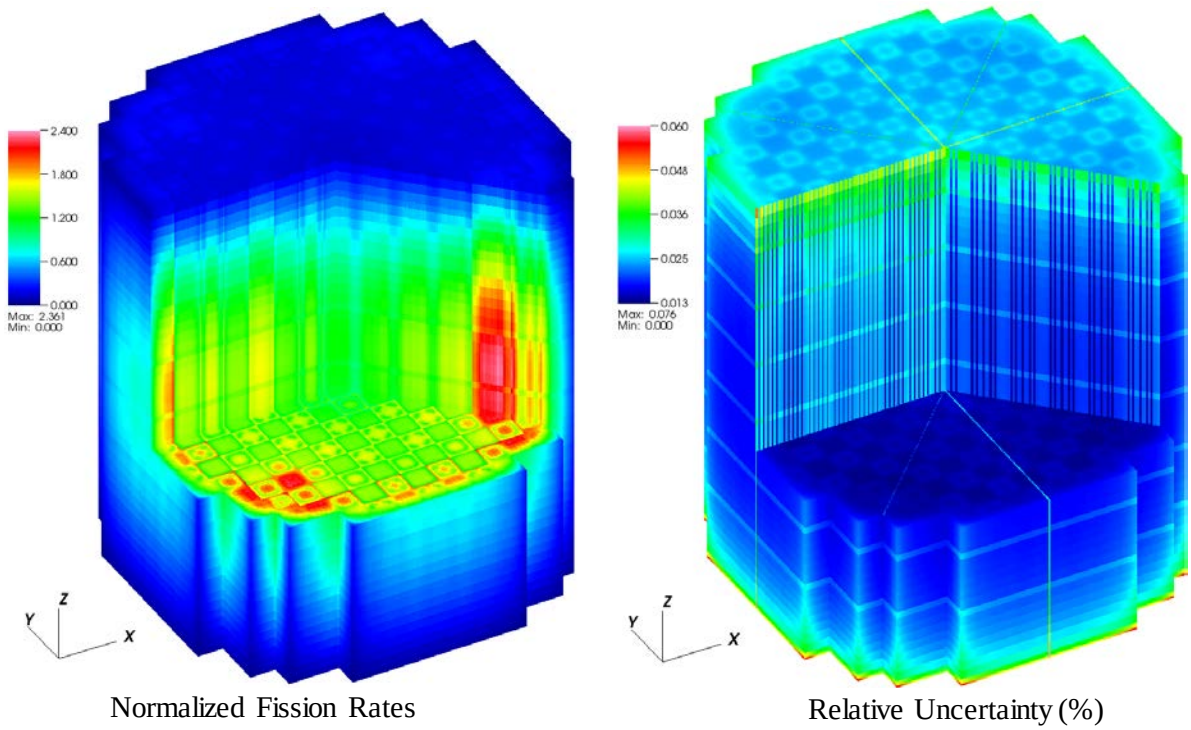
- OpenMC: Cycle 1 ARO criticality, ZPPT tests, and depletion flux maps [8,10]
- MC21: Cycle 1 ARO criticality, ZPPT tests, and depletion flux maps [9,10]
- Shift: Cycle 1 ARO criticality and ZPPT tests (undocumented)

At this time the Monte Carlo parameters and pin fission rate uncertainties for the OpenMC and MC21 results are unknown. They could be requested, if needed, along with the final pin power distributions for certain cases (including HFP cases with depletion). Currently the only known pin power solution with very low estimated uncertainties has been produced by Shift, with parameters shown below. There is currently no documentation for this result. Table 6-2 and Figure 6-3 provide some results from this case.

Note that all three codes have shown good agreement with the Cycle 1 ARO criticality measurements of the BEAVRS reactor, and early flux map results from OpenMC and MC21 are also showing good comparisons [10]. Shift cannot be used yet for flux maps because it does not have T/H feedback or Doppler broadening at this time.

Table 6-2 BEAVRS Representative Monte Carlo Parameters and Uncertainties

	Shift
Total # Particles	1e12
# Particles/Generation	500e6
# Generations	2,000
# Skipped Generations	500
# Cores	200,000
Memory/Core	< 2 GB
Runtime	3.5 hours
Eigenvalue Uncertainty	± 2.7 pcm
Average Pin Power Uncertainty	± 0.02%
Maximum Pin Power Uncertainty (by Power)	P < 1.0: ± 0.09% P > 1.0 ± 0.04%



Normalized Fission Rates

Relative Uncertainty (%)

Figure 6-3 CE Shift 3D Normalized Fission Rates for BEAVRS ARO Criticality

As Shift gains capability, it will be used to generate full core pin-by-pin normalized fission rate distributions for selected HFP flux map cases in the Cycle 1 depletion. The VERA-CS pin powers can then be compared to Shift results while its instrument response is compared to measured data. Alternately, the OpenMC or MC21 results for this case may also be available.

6.1.3 AP1000®

In support of the Westinghouse Test Stand for AP1000® startup analyses [74], CASL generated KENO-VI reference solutions for the initial 2D and 3D cores at BOC HZP isothermal conditions. Subsequently, comparable Shift results (with baffle only) were created under the OLCF "Early Science" allocation award [75, 76]. The AP1000® is Westinghouse's new advanced PWR reactor that is in the construction phase at several sites around the world. There is no measured data from this plant. The core design is challenging, with a five region, low leakage initial core loading with various combinations of IFBA and WABA absorbers and some locations with natural uranium assemblies [74].

KENO-VI results from a very large number of particle histories were generated for rodded and ARO conditions with and without the full radial reflector structures. For Shift, 14 one trillion particle cases were executed using the Early Science allocation, requiring nearly 25 million core-hours. These cases resulted in detailed Monte Carlo fission rate distributions for ARO, reduced boron (boron worth), and each control bank insertion. These parameters are summarized in Table 6-3.

The rodded case simulated by both KENO-VI and Shift used rod positions which represented realistic load follow configurations and produced a nearly neutral axial offset. The control bank positions are AO at 226 steps, MA fully inserted, and MB at 88 steps withdrawn. The KENO-VI

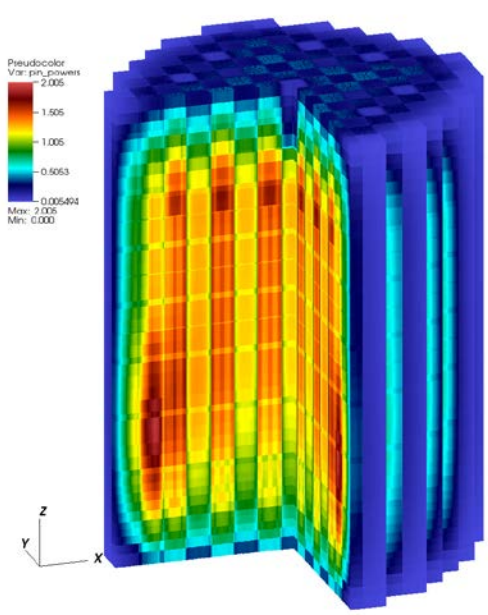
estimated uncertainties in the 3D results are larger than desired, so comparable 2D models using a variety of radial reflector models were also generated.

For these cases, the predicted fission rates agree between KENO-VI and Shift with a 0.46% RMS difference in 3D and 0.2% RMS in 2D. There is a minor ~1% deviation in radial power prediction, with Shift predicting higher in the center of the core. The largest difference in control bank worth between KENO-VI and Shift is 2.3% for the tungsten banks, and 0.9% for the AIC banks. Table 6-3, Figure 6-4, and Figure 6-5 provide results from these cases. The reactivity agreement between Shift and KENO-VI is approximately 50 pcm.

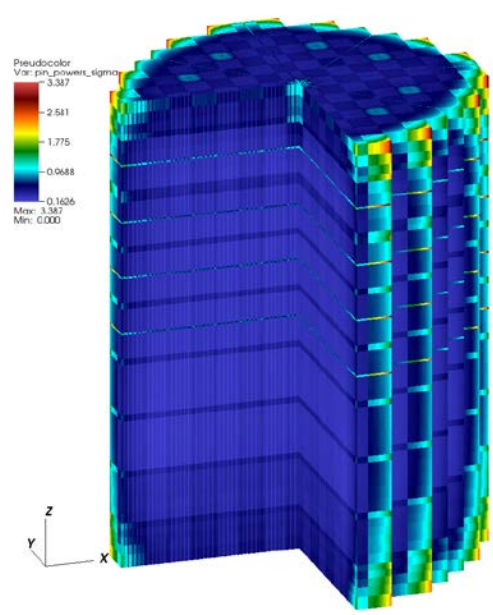
At this time, no measured data is available to validate any AP1000® calculations.

Table 6-3 AP1000® Representative Monte Carlo Parameters and Uncertainties [74,75,76]

	2D Core		3D Core	
	KENO-VI	**Shift**	**KENO-VI**	**Shift**
Total # Particles	25e9	100e9	100e9	1e12
# Particles/Generation	5e6	50e6	10e6	500e6
# Generations	5,000	2,000	10,000	2,000
# Skipped Generations	500	300	500	500
# Cores	300	56,000	240	240,000
Memory/Core	5.3 GB	< 2 GB	8.3 GB	< 2 GB
Runtime	6 days	1.6 hours	22 days	3 hours
Eigenvalue Uncertainty	± 0.5 pcm	± 3.5 pcm	± 0.3 pcm	± 2.7 pcm
Average Pin Power Uncertainty	± 0.06%	± 0.01%	± 0.19%	± 0.02%
Maximum Pin Power Uncertainty (by Power)	P < 1.0: ± 0.28% P > 1.0 ± 0.08%	P < 1.0: ± 0.03% P > 1.0 ± 0.02%	P < 1.0: ± 2.36% P > 1.0: ± 0.61%	P < 1.0: ± 0.06% P > 1.0 ± 0.05%



Normalized Fission Rates



Relative Uncertainty (%)

Figure 6-4 CE KENO-VI 3D Normalized Fission Rates for AP1000® BOC HZP Rodded Case [74]

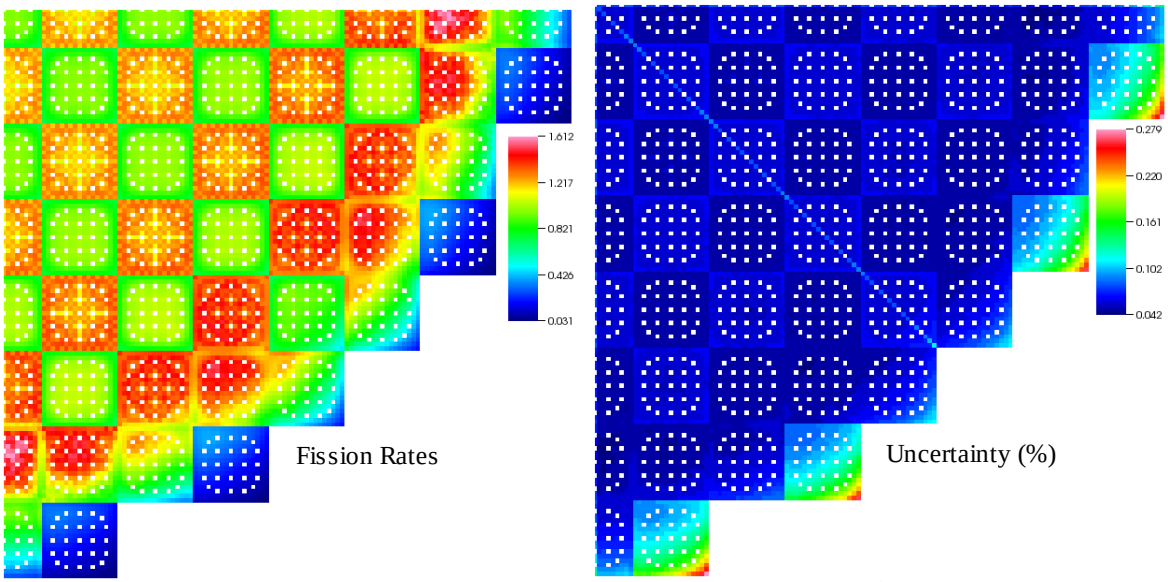


Figure 6-5 CE KENO-VI 2D Normalized Fission Rates for AP1000® BOC HZP ARO [74]

6.1.4 Krško Nuclear Power Plant

Westinghouse has generated CE Monte Carlo solutions with very large numbers of particle histories for the initial criticality and startup testing of Krško Nuclear Power Plant Cycle 1 at HZP isothermal conditions [22]. Krško is described in Section 3.6. The largest solution has Banks C and D fully inserted, while another case for ARO was executed with less particle histories. Because Krško lacks quarter-core symmetry, this is the only full-core benchmark that has been executed in full symmetry.

The Krško results were generated with Shift for a CASL milestone resulting in Reference 22. For the radial core structure, the Shift model only includes the core baffle. Comparable 2D core results were also generated for more direct radial comparisons. For the 3D results, an ARO case was executed with 500 billion particles, and a case with Banks C&D inserted (for bank worth measurement) was run with 2 trillion particles. This case was selected because of the challenging flux gradients produced by the inserted control rods. Both of these cases compared well to measured plant reactivity and control bank worth. Table 6-4, Figure 6-6, and Figure 6-7 provide results from these cases. Note that the reported estimated uncertainty from Shift is believed to be 2-4 times too low.

Table 6-4 Krško Representative Monte Carlo Parameters and Uncertainties [22]

	2D Core	3D Core	
	Shift	**ARO**	**C & D Inserted**
Total # Particles	100e9	500e9	2e12
# Particles/Generation	50e6	250e6	1e9
# Generations	2,000	2,000	2,000
# Skipped Generations	250	500	500
# Cores	50,000	124,992	250,000
Memory/Core	< 2 GB	< 2 GB	< 2 GB
Runtime	1.2 hours	2.7 hours	5.1 hours
Eigenvalue Uncertainty	± 3.7 pcm	± 3 pcm	± 2.2 pcm

Average Pin Power Uncertainty	± 0.02%	± 0.03%	± 0.02%
Maximum Pin Power Uncertainty (by Power)	P < 1.0: ± 0.03% P > 1.0 ± 0.02%	P < 1.0: ± 0.11% P > 1.0: ± 0.04%	P < 1.0: ± 0.07% P > 1.0 ± 0.03%

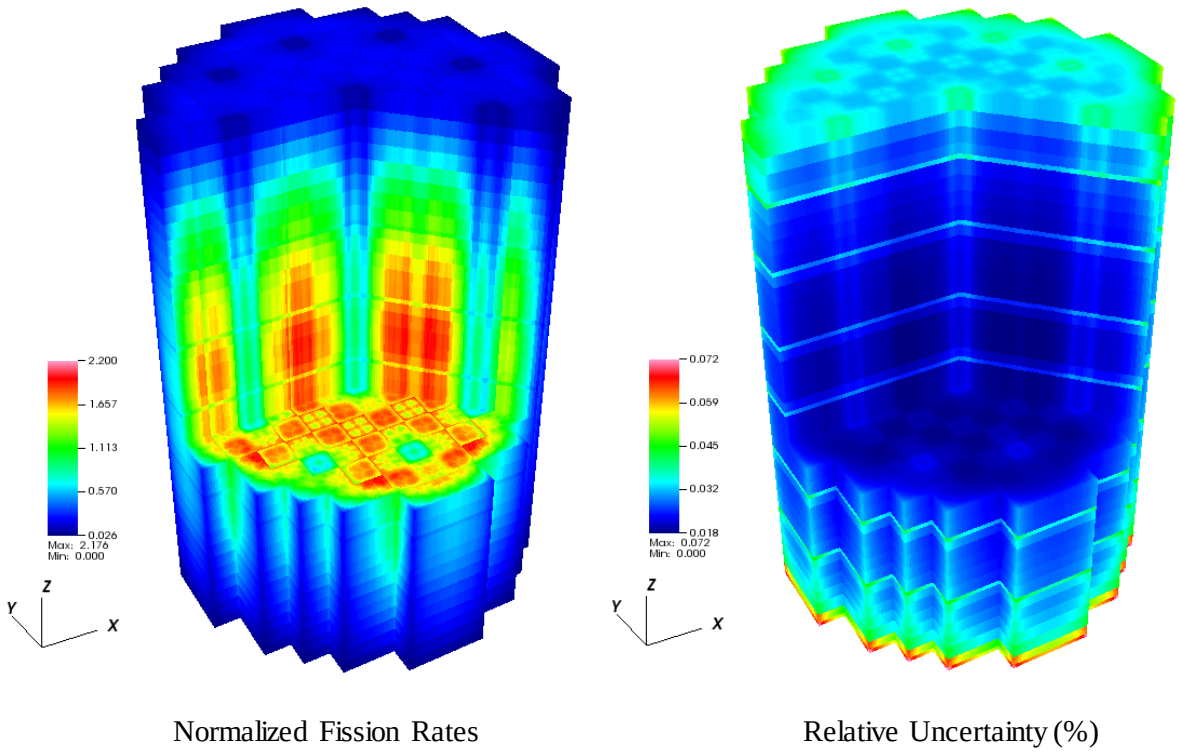


Figure 6-6 CE Shift 3D Normalized Fission Rates for Krško Banks C and D Inserted [22]

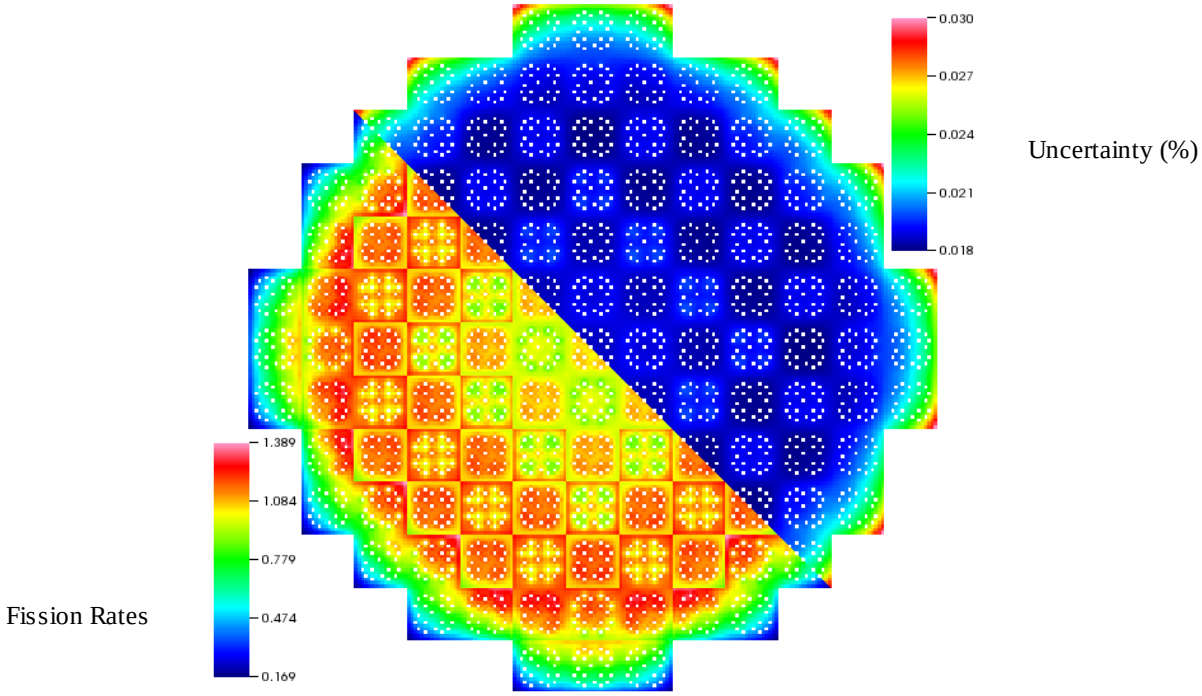


Figure 6-7 CE Shift 2D Normalized Fission Rates for Krško HZP ARO [22]

As Shift gains capability, it will be used to generate full core pin-by-pin normalized fission rate distributions for selected HFP flux map cases in the Cycle 1 depletion. The VERA-CS pin powers can then be compared to Shift results while its instrument response is compared to measured data.

6.2 Intra-Pin Distribution Benchmarks

The distribution of temperature, fissions, fission products, and burnup within a fuel pellet are important quantities for accurate fuel rod reactivity, power distribution, and thermo-mechanical performance calculations. Unfortunately there are almost no validation data available to benchmark the VERA-CS predictions of these distributions, other than a few rough measurements of discharged fuel rods like those described in Section 5. In order to provide confidence in the code performance at this scale, continuous energy Monte Carlo methods will be used to provide reference solutions of 2D pin cells and lattices with radially (and potentially azimuthal) variations in the fuel pellet conditions.

Continuous energy Monte Carlo codes with proper continuous temperature feedback from the cross sections (i.e. Doppler broadening) do not require the multi-group resonance self-shielding approximation needed by deterministic codes. Furthermore, the spatial discretization can be much finer with Monte Carlo so long as enough particle histories are tracked to provide low uncertainties in the reaction rates. CE KENO-VI will be used to generate reference solutions for a single pin problem similar to the work performed in Reference 77. The pellet is divided into ten radial rings and a non-uniform (quadratic) fuel temperature is imposed as shown in Figure 6-8. Results are generated for different magnitudes of temperature and for comparable uniform distributions.

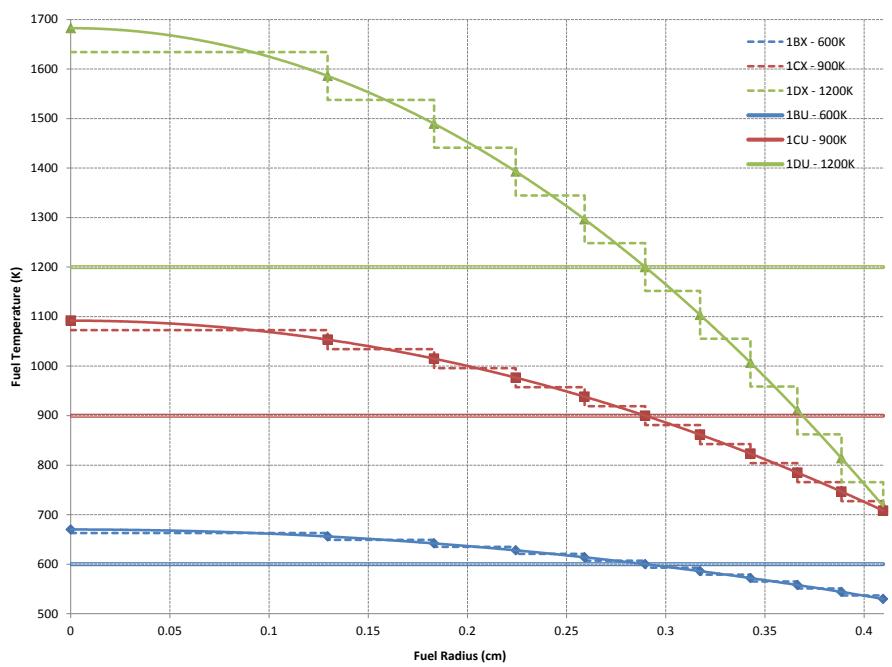


Figure 6-8 Sample Intra-pin Temperature Distributions in UO_2 Fuel

The reaction rate tallies can be performed on a small or large number of energy bins. Nominally results will be generated in five energy groups (thermal, epithermal, resolved resonance, etc.), but results can also be obtained on the exact VERA-CS group structure (current 47 groups). Reaction

rate tallies will also be performed by isotope, such as ^{238}U absorptions. This detail is very useful for validation and cannot be obtained with any other approach.

Figure 6-9 and Figure 6-10 provide sample results from KENO-VI for the fission and energy-dependent capture rate radial distributions inside of a single fuel pellet for a variety of temperatures.

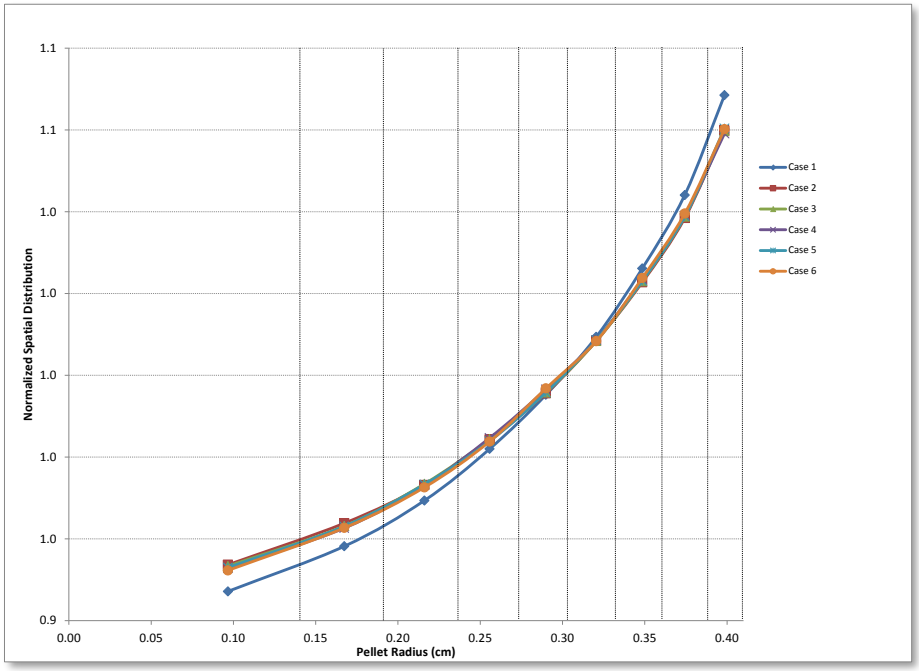


Figure 6-9 Sample KENO-VI Calculated Fission Rate Distributions in UO_2 Fuel

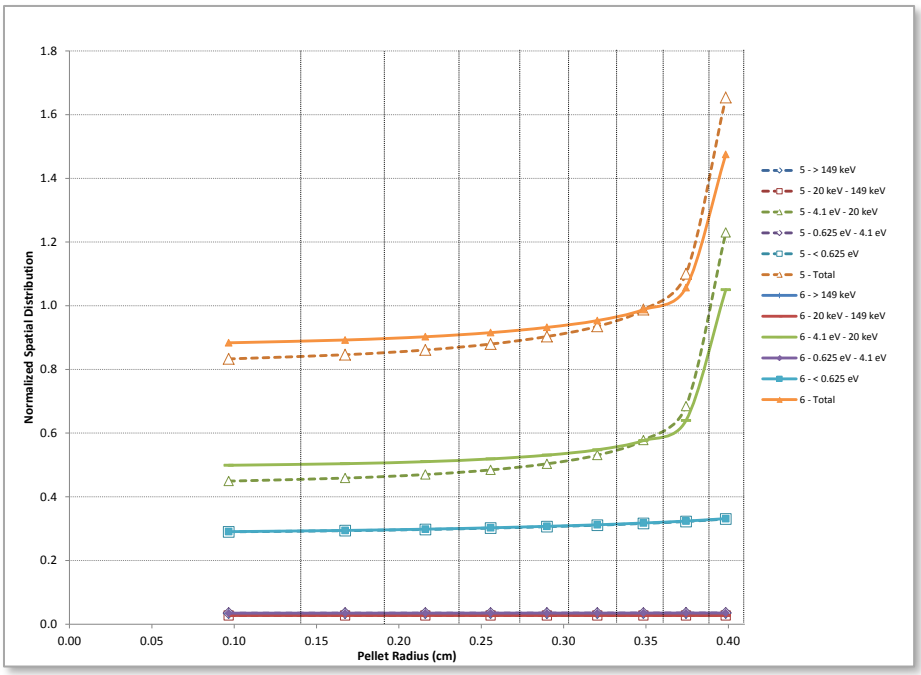


Figure 6-10 Sample KENO-VI Calculated Capture Rate Distributions by Energy in UO_2 Fuel

In addition, these pin cells can also be depleted to benchmark the intra-pin power distribution over the burnup of the rod, or to benchmark the effects on isotopic distributions within the rod. Further validation could include extension of this problems to 2D lattices with large heterogeneities (such as control rods or burnable poisons) to better validate the effects of azimuthal variations of neutron flux on the fuel. In all cases, Monte Carlo is required as the reference due to the lack of measured data. However, models chosen for analysis should be consistent with those from plants and experiments for which the Monte Carlo code has performed well against measurements.

6.3 Depleted Isotopics Benchmarks

While significant data is available for validation of reactivity for fresh fuel (See Section 4), there are no experiments which provide validation data for depleted, or burned, fuel. The power plant models do provide some confidence in this capability indirectly, as the core reactivity verses core average (or cycle) burnup will be available for comparison. However, the measurement of core reactivity is a very coarse quantity and could be significantly affected by error cancellation. Likewise, the measure flux maps assure that the normalized distribution of power is well predicted, which again indirectly presumes the accurate accumulation of the major actinides and fission products more locally. But since the in-core instruments are only located in the central tube of the assemblies, their responses are highly driven by the fission rates in the nearest four fuel rods, and don't provide much confidence in the pin-by-pin isotopics or power distribution for the entire lattice. The burned isotopics are somewhat validated through the post-irradiation exams discussed in Section 5, but these are typically only a few high burnup cases after fuel discharge and can have substantial measurement and power history uncertainties.

To provide more confidence in the pin power and reactivity results from VERA-CS for burned fuel, continuous energy Monte Carlo will be used as a reference solution for PWR geometries containing burned isotopics. These geometries will be similar to those in Reference 3, basic PWR geometries like those in WBN1, or they will come directly from some of the full core calculations discussed in Section 3. For the same geometric and isotopic distributions, VERA-CS will be compared to CE Monte Carlo over a variety of burnups.

In the most ideal case, the 3D isotopics from burned fuel in instrumented core locations will be used as a reference. If the Monte Carlo can predict the normalized instrument response well from a measured case, that (crudely) connects the Monte Carlo reference to measured data, and serves to validated the VERA-CS pin power distribution for depleted isotopics on a fine scale. However, smaller cases need to be executed initially to help isolate any potential problems in the cross section treatments for burned fuel.

Note that the critical experiments that contain MOX, as well as the Catawba model, both serve to improve the confidence in the reactivity predictions for fuels containing plutonium, which is another indirect confirmation for the likely performance for burned fuels.

Note that this activity does (necessarily) include isotopic comparisons to CE Monte Carlo code depletion. This activity is code comparisons given the same isotopics and distributions, which also somewhat overlaps with the HFP discussion in Section 6.1.

7. SUMMARY

Enclosed is presented a four-part approach for the validation of VERA-CS, based on combining common industrial methods for benchmarking and licensing of lattice physics codes, 3D nodal core simulators, and spent fuel criticality and storage methods. The fully integrated and high-fidelity nature of VERA-CS permits comparison to all of these forms of measured data with the same integrated product, utilizing the VERA common input and output in most cases. Furthermore, 3D comparisons to PIE data and 3D pin-by-pin fission rate comparisons go beyond the validation methods typically used in the industry today and will help demonstrate the advancements made by CASL.

The included plan contains enough validation to provide confidence to CASL stakeholders that VERA-CS is a capable and accurate core simulator tool for pseudo-steady-state PWR operations. Each capability and feature listed on the left side of the validation assessment matrix in Section 2 is addressed to some degree by one or more analyses against measured data. Though CASL does not intend to license VERA-CS for any particular application with the NRC, the results of the problems listed in this plan should clearly demonstrate that VERA-CS is ready for industrial application.

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