INL/EXT-17-42461

Light Water Reactor Sustainability Program

Industry Application ECCS / LOCA Integrated Cladding/Emergency Core Cooling System Performance: Demonstration of LOTUS-Baseline Coupled Analysis of the South Texas Plant Model

Hongbin Zhang, Ronaldo Szilard, Aaron Epiney, Carlo Parisi, Rodolfo Vaghetto, Alessandro Vanni, Kaleb Neptune

June 2017

U.S. Department of Energy Office of Nuclear Energy

INL/EXT-17-42461

Light Water Reactor Sustainability Program

Industry Application ECCS / LOCA Integrated Cladding/Emergency Core Cooling System Performance: Demonstration of LOTUS-Baseline Coupled Analysis of the South Texas Plant Model

Hongbin Zhang[1], Ronaldo Szilard[1], Aaron Epiney[1], Carlo Parisi[1],
Rodolfo Vaghetto[2], Alessandro Vanni[2], Kaleb Neptune[2]

[1] Idaho National Laboratory, Idaho Falls, Idaho 83415

[2] Department of Nuclear Engineering
Texas A&M University, College Station, TX 77843

June 2017

Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

EXECUTIVE SUMMARY

Under the auspices of the DOE LWRS Program RISMC Industry Application ECCS/LOCA, INL has engaged staff from both South Texas Project (STP) and the Texas A&M University (TAMU) to produce a generic pressurized water reactor (PWR) model including reactor core, clad/fuel design and systems thermal hydraulics based on the South Texas Project (STP) nuclear power plant, a 4-Loop Westinghouse PWR. A RISMC toolkit, named LOCA Toolkit for the U.S. (LOTUS), has been developed for use in this generic PWR plant model to assess safety margins for the proposed NRC 10 CFR 50.46c rule, Emergency Core Cooling System (ECCS) performance during LOCA.

This demonstration includes coupled analysis of core design, fuel design, thermal-hydraulics and systems analysis, using advanced risk analysis tools and methods to investigate a wide range of results. Within this context, a multi-physics best estimate plus uncertainty (MP-BEPU) methodology framework is proposed.

The set of modeled results shows that peak clad temperature (PCT) and fuel cladding oxidation responses (measured by equivalent cladding reacted (ECR)) are well characterized by performance based modeling under large break LOCA conditions. Both PCT and ECR comply with a proposed acceptance criteria with sufficient margins available. Furthermore, these demonstration calculations indicate the importance of safety margin management and planning for future operating cycles. Since nuclear fuel stays in a reactor for multiple operating cycles, planning of loading and operating strategies needs to be well thought of.

The intrinsic value of successful research and development for the proposed LOTUS framework is expected to be significant. LOTUS has the potential of becoming a powerful safety margin management tool for industry stakeholders to address the challenges imposed by the proposed 10 CFR 50.46c rulemaking and other emerging issues such as plant equipment upgrades to support the implementation of FLEX, additional new passive cooling systems, improved operational control, accident-tolerant instrumentation, and to gain potential benefits from 50.69 safety significance reclassification by relying on a more rigorous mathematical and physics-based apparatus to address model and data uncertainty in safety analysis.

The importance of the LOTUS framework also extends to current and future nuclear fuels development and applications. The progress shown on the Industry Application ECCS/LOCA is a step forward towards modeling and simulation predictive capability, and it can provide useful tools for the development and characterization of accident tolerant fuels (ATF), a joint program pursued by DOE and the nuclear industry. ATF could have "game changing" attributes to transform the nuclear industry. Having an integrated multi-physics toolkit that is fuel/clad-, fuel cycle-, and scenario-centric provides a ready platform for the analysis of novel fuel and cladding systems. The coupled multi-physics, multi-scale LOTUS analysis framework allows plant system configuration variations to be studied with speed and precision, including detailed assessment of introducing ATF into current LWR plants for design enhancements. This detailed evaluation approach becomes important when analyzing magnitude range and timeline of system response for relevant sequence of events.

Another added benefit of LOTUS is the ability to analyze inverse problem configurations, which are not easily done with traditional sequential processes. This is an important attribute in systems analysis, where various plant economics and safety metrics can be studied to provide a full spectrum of information for decision making.

The advancements proposed with LOTUS can potentially outweigh some of the costs associated with the proposed 50.46c rule rollout and implementation, hence contributing to the U.S. fleet competitiveness with other sources of energy. LOTUS has the potential to offer nuclear plant owners/operators a multi-physics, multi-scale systems analysis capability that was not available before. A rightly informed risk and safety analysis, and margin management, can potentially reduce extensive (and costly) iterations between licensees and regulators when dealing with rule compliance and operational issues. Ultimately, studying and understanding the available data in a risk-informed manner will yield a higher degree of safety and cost efficiency.

CONTENTS

EXECUTIVE SUMMARY iii
FIGURES vi
TABLES viii
ACRONYMS ix
1. INTRODUCTION 12
1.1 The NRC Proposed 10 CFR 50.46c Rule and its Implications 13
1.2 RISMC INDUSTRY APPLICATION – ECCS/LOCA 14
2. LOTUS: A RISK-INFORMED SAFETY MARGIN MANAGEMENT TOOLKIT FOR INTEGRATED CLADDING/ECCS PERFORMANCE ANALYSIS 18
2.1 Introduction of LOTUS 18
2.2 Description of LOTUS 21
2.3 LOTUS Software Development 29
3. INTRODUCTION OF THE SOUTH TEXAS PROJECT (STP) PLANT 34
4. LOTUS APPLICATION ON STP 37
4.1 Core Design Automation 37
4.1.1 LOTUS CD-A computer codes 37
4.1.2 LOTUS CD-A methodology 38
4.2 Fuels Performance 40
4.3 Systems Analysis 42
4.4 Risk Assessment 46
4.4.1 Uncertainty propagation and risk assessment 46
4.4.2 Sensitivity Analysis 48
5. STP ANALYSIS RESULTS 51
5.1 Core Design Automation 51
5.1.1 Input data for core design calculations 51
5.1.2 Cross section library calculation model 55
5.1.3 PHISICS core model 57
5.1.4 Coupled PHISICS/RELAP5-3D calculation model 58
5.1.5 Transient Power Maneuvers model 59
5.1.6 Core design: STP results 61
5.1.7 Transient power maneuvers for STP core designs 67
5.2 Fuels Performance 69
5.3 Systems Analysis 71
5.4 Uncertainty Quantification, Risk Assessment and Sensitivity Analysis 72
5.4.1 Uncertainty quantification and risk assessment 72
5.4.2 Sensitivity analysis 75
6. CONCLUSIONS, FUTURE WORK AND THE PATH FORWARD 77
6.1 Results Conclusions 77
6.2 Industry Application ECCS/LOCA Future Work 77
6.3 Path Forward 78
7. REFERENCES 80

FIGURES

Figure 1. Analytical Generic Limit Proposed by the NRC for Existing Fuel, ECR & PCT versus Hydrogen Content. [1] 14

Figure 2. Flow Chart of the RIMM Integrated Evaluation Model. [5] 17

Figure 3. RISMC Margin Quantification and Risk Assessment Paradigm. 17

Figure 4. Schematic Illustration of LOTUS. 19

Figure 5. Schematic Illustration of Current BEPU Process for LOCA Analysis. 24

Figure 6. Paradigm Shift with LOTUS Multi-Physics BEPU. 26

Figure 7. Illustration of LOTUS Multi-Physics BEPU (MP-BEPU) Safety Analysis Framework. 27

Figure 8. LOTUS Data Stream. 29

Figure 9. Schematic Illustration of LOTUS Steady-State Analysis Manager (LOTUS SS Manager). 31

Figure 10. Schematic Illustration of LOTUS Transient Analysis Manager (LOTUS Transient Manager). 32

Figure 11. Illustration of LOTUS Managers. 32

Figure 12. STPEGS Units. 36

Figure 13. Industry Application ECCS/LOCA Demonstration of a PWR Design Strategy. 39

Figure 14. Schematic Illustration of the Mapping between the Core Design Analysis and the RELAP5-3D Analysis Core Model for the Generic PWR Model Based on STP. 43

Figure 15. Schematic Illustration of the Heat Structure Mapping for the Hot Assembly and Its Hot Rod with the Hot Channel (One for Each Group of Assemblies). 44

Figure 16. Schematic Illustration of the Heat Structure Mapping for Average Assemblies and their Respective Hot Rods with the Average Flow Channel. 44

Figure 17. Schematic of Double Ended Guillotine Break. 45

Figure 18. STP Core: 17x17 Pin Assembly. Shown are 64, 104 and 128 IFBA Rods (Circles) and 25 Guide Tubes (Black). 54

Figure 19. Left) STP Fuel Rod Schematic (This is Figure 4.2-3 in [9]); Right) Axial Fuel Pin Design: High Enriched Center Part with Top and Bottom Blankets 2.6% Enriched. 55

Figure 20. HELIOS-2 Model for STP. 56

Figure 21. Assumed STP Core: Equilibrium Cycle Loading Pattern. 58

Figure 22. RELAP5-3D Core Nodalisation Used for the Core Simulation. 59

Figure 23. Load Following Maneuver Power History. 60

Figure 24. Control Rod Positions. 61

Figure 25. RAVEN Samples the LOCA Start Times and Runs RELAP5 in Multi-Deck Mode. 61

Figure 26. STP Equilibrium Cycle: Reloading Pattern, Fresh Fuel Enrichment and Number of Burnable Absorber (BA) Pins in the Fresh Fuel Assemblies. 63

Figure 27. STP Equilibrium Cycle: Boron Letdown Curve. 64

Figure 28. STP Equilibrium Cycle: Pbar, Fdh, Fq and Burnup for Each Assembly at BOC (Top) and EOC (Bottom). 65

Figure 29. STP Equilibrium Cycle: Core Averaged Axial Power Distribution at BOC, MOC and EOC Compared to Plant Data. 66

Figure 30. STP Equilibrium Cycle: Maximum Pin Peaking Factors for Each Assembly at BOC and EOC.66

Figure 31. STP Equilibrium Cycle: Skewed Power Shapes at the End of the Maneuver at BOC (Top), at 300 Days (Middle) and at EOC (Bottom). Shown are Core Average Axial Power Distributions for Fresh (0B), Once Burned (1B) and Twice Burned (2B) Fuel Assemblies. 68

Figure 32. Power History for the Hot Rod in a Twice Burned Fuel Assembly. 70

Figure 33. Cladding Hydrogen Content versus Rod Average Burnup. 70

Figure 34. Comparison of PCT in LB-LOCA Transients with Maneuvered Power Shapes versus Cosine Power Shapes. 71

Figure 35. PDF and CDF for PCTR at EOC. 73

Figure 36. PDF and CDF for ECRR at EOC. 73

Figure 37. PCT versus Pre-Transient Cladding Hydrogen Content. 74

Figure 38. ECR versus Pre-Transient Cladding Hydrogen Content. 75

Figure 39. Comparison of Sensitivity Measures for PCTR at 300 Days. 76

Figure 40. Comparison of Sensitivity Measures for ECRR at 300 Days. 76

Figure 41. Schematic Illustration of Core Design Optimization Development for LOTUS. 78

Figure 42. Illustration of “Game Changers” in Delivering Nuclear Promises (Reproduced from Scot Greenlee’s Presentation at 2016 American Nuclear Society Utility Working Conference). [23] 79

TABLES

Table 1. Common Data from Fuel Rod Design for Different Physics in LOCA Analysis. 41

Table 2. Distribution of Parameter Uncertainties. 47

Table 3. STP Core Characteristics. 52

Table 4. STP Cycle Characteristics. 52

Table 5. STP Fuel Assembly Characteristics. 52

Table 6. Fuel Rod Characteristics. 53

Table 7. Reactor Coolant System. 53

Table 8. Collapsed Energy Structure. 57

Table 9. Cross Section Library Tabulation Points. 57

Table 10. Summary of the 95/95 Estimators for PCT and ECR for the STP Core Design. 74

ACRONYMS

10 CFR	Title 10, Code of Federal Regulations
2D	Two Dimensional
3D	Three Dimensional
ANS	American National Standard
AOR	Analysis of Record
ATF	Accident Tolerant Fuel
BEAVRS	Benchmark for Evaluation And Validation of Reactor Simulations
BEPU	Best Estimate Plus Uncertainty
BOC	Beginning of Cycle
BOL	Beginning of Life
CASL	Consortium for the Advanced Simulation of Light Water Reactors
CCFL	Counter Current Flow Limitation
CD-A	Core Design Automation
CD-O	Core Design Optimization
CDF	Cumulative Distribution Function
CFR	Code of Federal Regulation
CHF	Critical Heat Flux
COBRA-TF	Coolant Boiling in Rod Arrays – Two Fluid
CR	Control Rod
CTF	COBRA-TF
CWO	Core Wide Oxidation
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
DOE	U.S. Department of Energy
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECR	Equivalent Cladding Reacted
ECRR	ECR ratio
EOC	End of Cycle
EPRI	Electric Power Research Institute
FA	Fuel Assembly
FLEX	Diverse and Flexible Coping Strategies

FOM	Figure of Merit
FP	Fuel Performance
FY	Fiscal Year
GWd	Giga Watt days
HPC	High Performance Computing
IA	Industry Application
IEM	Integrated Evaluation Model
IFBA	Integral Fuel Burnable Absorber
INL	Idaho National Laboratory
INSTANT	Intelligent Nodal and Semi-structured Treatment for Advanced Neutron Transport
LB-LOCA	Large Break LOCA
LOCA	Loss of Coolant Accident
LOTUS	LOCA Analysis Toolkit for the U.S.
LOTUS-A	LOTUS Advanced
LOTUS-B	LOTUS Baseline
LWR	Light Water Reactor
LWRS	Light Water Reactor Sustainability
MLO	Maximum Local Oxidation
MOC	Middle of Cycle
MOOSE	Multi-Physics Object-Oriented Simulation Environment
MPACT	Michigan Parallel Characteristics based Transport
MP-BEPU	Multi-Physics Best Estimate Plus Uncertainty
MRTAU	Multi-Reactor Transmutation Analysis Utility
MWd	Mega Watt days
MWth	Mega Watt thermal
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
ODE	Ordinary Differential Equation
PCT	Peak Clad Temperature
PCTR	PCT ratio
PDE	Partial Differential Equation
PDF	Probability Density Function
PHISICS	Parallel and Highly Innovative Simulation for INL Code System
PIRT	Phenomena Identification and Ranking Table

PPM	Parts per Million
PWR	Pressurized Water Reactor
R&D	Research and Development
RA	Risk Assessment
RAVEN	Risk Analysis Virtual Environment
RCCA	Rod Cluster Control Assemblies
RCP	Reactor Coolant Pumps
RCS	Reactor Coolant System
RELAP5	Reactor Excursion and Leak Analysis Program 5
RELAP-7	Reactor Excursion and Leak Analysis Program 7
RFA	Robust Fuel Assembly
RIMM	Risk Informed Margin Management
RISMC	Risk Informed Safety Margin Characterization
ROM	Reduced Order Model
SA	System Analysis
SDD	Software Design Description
SPH	Super Homogenization
SRS	Software Specifications Requirements
STP	South Texas Project
TAMU	Texas A&M University
TH	Thermal Hydraulics
tHM	(metric) ton of Heavy Metal
UFSAR	Updated Final Safety Analysis Report
UQ	Uncertainty Quantification
VERA	Virtual Environment for Reactor Applications
VERA-CS	VERA Core Simulator
WEC	Westinghouse Electric Company
ZrB2	Zirconium Diboride

Industry Application ECCS / LOCA Integrated Cladding/Emergency Core Cooling System Performance: Demonstration of LOTUS-Baseline Coupled Analysis of the South Texas Plant Model

1. INTRODUCTION

The existing fleet of nuclear power plants in the U.S. faces a variety of challenges such as lower natural gas prices and increasing deployment in renewable energy, as well as the extra costs associated with security and safety upgrades post Fukushima accidents. In order to maintain economic competitiveness, the nuclear engineering research community and the industry are developing higher burnup fuel and accident tolerant fuel, and other innovative technologies in hopes of lowering fuel cycle and plant operation costs. To help facilitate this shift, greater predictive methods are required, which entails a clear representation of the multi-physics phenomena as well as the uncertainties.

Additionally, the existing fleet also faces potential regulatory hurdles. The U.S. Nuclear Regulatory Commission (NRC) is currently proposing rulemaking 10 CFR 50.46c to revise the loss-of-coolant-accident (LOCA) and emergency core cooling system (ECCS) acceptance criteria to include the effects of higher burnup on cladding performance [1]. The key implications of this proposition are that the core, fuels, and cladding performance cannot be evaluated in isolation anymore. Both cladding and ECCS performance need to be considered in a coupled manner and the safety analyses have to be carried out in a multi-physics framework. This may also suggest that models for cladding performance as well as LOCA methodologies need to be updated. Given the acceptance criteria levied by the proposed rule, a question is raised: How can we best configure the core and operate the plant while still satisfying the proposed regulatory acceptance criteria?

In 2015, the Risk-Informed Safety Margin Characterization (RISMC) Pathway, as part of the DOE Light Water Reactor Sustainability (LWRS) research and development program initialed a set of demonstration activities to support the industry in the transition to the proposed 10 CFR 50.46c rule and to offer potential solutions to LOCA/ECCS analysis. This is the subject of this report, the Industry Application ECCS/LOCA. Its purpose is to provide to the plant owner/operator a vehicle to inform decisions to manage margins related to compliance with the proposed 10 CFR 50.46c rule. This is the driver behind the RISMC Industry Application ECCS/LOCA, and motivates a risk-informed margin management (RIMM) project. In this project, margin is measured relative to the 10 CFR 50.46c proposed rule. The industry will need to comply with the proposed rule within seven years of the proposed change (the timeline for implementation is still being discussed among the NRC, fuel vendors, and licensees, and will depend on many factors, such as methodology changes, amount of work to be submitted for regulatory approval, and regulatory reviews).

The RISMC toolkit offers advanced LOCA analysis tools that provide the plant owner/operator a vehicle to manage margins and inform decisions if compliance with the proposed 10 CFR 50.46 is challenged by changes in the operational envelope. Industry Application ECCS/LOCA's goal is to develop an Integrated Evaluation Model (IEM) to understand how uncertainties are propagated across the physical disciplines and data involved, as well as how risks are evaluated in postulated LOCA events under 10 CFR 50.46c. This IEM is called LOCA Toolkit for the U.S. (LOTUS) and it connects five major disciplines involved in LOCA analysis, namely core design, fuels performance, system analysis, risk assessment and finally core optimization. The focus of LOTUS is to establish the automation interfaces among the five disciplines. LOTUS will utilize, in a first step, current state-of-the-art computer codes. The risk-informed margins management approach for Industry Application ECCS/LOCA will provide a means of quantifying the impact on the key LOCA analysis figures-of-merit like peak cladding temperature (PCT), equivalent cladding reacted (ECR), etc. of a change in LOCA analysis inputs. The information that the risk analysis and associated tools provide can then be used for decision-making and margin management.

An Industry Application ECCS/LOCA initial report in 2015 [2] focused on presenting the proposed methodology and showing an early demonstration using reduced order models. A follow-on work [3] in 2016 presented a LOTUS demonstration including coupled disciplines in core design, fuels performance, system analysis and risk assessment for a generic PWR model. The current work presents LOTUS coupled calculations for a generic PWR model, built in collaboration with the Texas A&M University, based on the South Texas Project plant. The core optimization discipline is left for future work. This report presents a first-of-its-kind application of a margins management toolkit for core, fuel design, and safety analysis coupling various physics disciplines and multiple levels of fidelity using models resemble a real operating nuclear power plant.

1.1 The NRC Proposed 10 CFR 50.46c Rule and its Implications

As mentioned, the U.S. NRC is considering a rulemaking change that would revise the requirements in 10 CFR 50.46. In the proposed rulemaking, designated as 10 CFR 50.46c, the NRC proposed a fuel performance-based equivalent cladding reacted criterion as a function of cladding hydrogen content before the accident (pre-transient) in order to include the effects of higher burnup on cladding performance as well as to address other technical issues. The pre-transient cladding hydrogen content, in turn, is a function of the fuel burnup and cladding materials. The proposed rule would apply to all light water reactors and to all zirconium based cladding types. The key points of the proposed rule are as follows:

- Cladding performance cannot be evaluated in isolation. Cladding performance and ECCS performance need to be considered in a coupled way, which examines the interactions across the disciplines involved.
- Models for cladding performance even within the design basis will need to be updated for regulatory purposes.
- Effort needs to be expended in searching regulatory issue space for the limiting case ("ECCS performance must be demonstrated for a range of postulated loss-of-coolant accidents of different sizes, locations, and other properties, sufficient to provide assurance that the most

severe postulated loss-of-coolant accidents have been identified. ECCS performance must be demonstrated for the accident, and the post-accident recovery and recirculation period".)

A characteristic of the proposed rulemaking, as illustrated in Figure 1, imposes more restrictive and fuel rod-dependent cladding embrittlement criteria. Therefore, a thorough characterization of the reactor core is required in large break LOCA (LB-LOCA) analyses in order to identify the limiting case and limiting rods.

The rule implementation process is expected to take approximately seven years following the rule effective date. A loss of operational margin may result due to the more restrictive cladding embrittlement criteria. Initial and future compliance with the rule may significantly increase vendor workload and licensee costs, as a spectrum of fuel rod initial burnup states may need to be analyzed to demonstrate compliance.

The total costs for the industry to accommodate the proposed rule can be in excess of $500 million. If plants have to operate at more restrictive conditions than currently allowed, the indirect cost could be even larger. Consequently, there will be an increased focus on licensee decision making related to LOCA analysis to minimize cost and impact, and to manage margin.

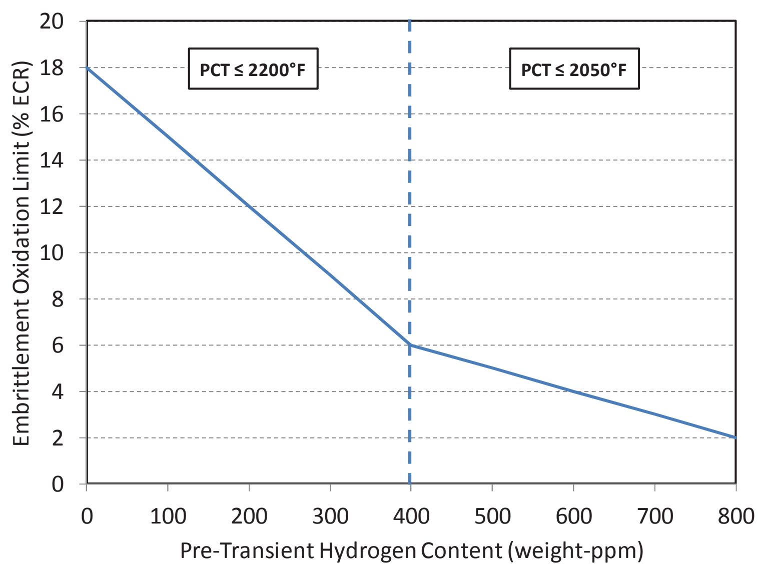


Figure 1. Analytical Generic Limit Proposed by the NRC for Existing Fuel, ECR & PCT versus Hydrogen Content. [1]

1.2 RISMC INDUSTRY APPLICATION – ECCS/LOCA

As mentioned previously, in 2015 INL initiated the Industry Application ECCS/LOCA within the Risk-Informed Safety Margin Characterization (RISMC) Pathway within the DOE's LWRS Program to develop analytical capabilities to support the industry in the transition to the proposed LOCA acceptance rule [2, 3, 4, 5].

RISMC will develop and provide methodologies and tools to plant operators/owners to support plant decisions for risk-informed margins management. These include improved economics, reliability and sustain safety of current nuclear power plants. The RISMC R&D Pathway includes the RISMC Toolkit development and the Risk-Informed Margin Management (RIMM) Applications. The corresponding R&D activities are separated in “Tools”, “Data” and “Methods”. The “Tools” development focuses on MOOSE and its attached advanced codes. The “Data” part includes the Verification, Validation and Uncertainty part for all levels of the analysis, i.e. from the component to the facility scale. Finally, the Industry Applications form the “Methods” part of RISMC. The Industry Applications seek a close collaboration with the industry either through EPRI or directly with a plant owner, operator or vendor. During FY-2017, INL has established a close collaboration with the South Texas Project on Industry Application ECCS/LOCA. The safety analysis guidelines resulting from the Industry Applications will be made available to the industry.

Nuclear installation designers, vendors and licensees (plant owner/operators) operate in a regulated environment. Traditionally, the economics of the industry prevent large deviations from well-established procedures within the licensing basis of the evaluation models, which are already in place. For example, the complex multi-physics LOCA problem is solved via operator splitting where various engineering disciplines are interfaced with well-set rules, which have been developed over the years consistently with specific acceptance criteria and regulatory requirements. Further, the propagation of uncertainties across the various functional groups is addressed by defining bounding assumptions at the interfaces which limit the possibility that the impact of an issue in a specific discipline (error discovered, design change or other) to cross-over to other physics in an efficient fashion.

Such traditional processes and interfaces do not easily adapt to new integrated methods and cannot fully leverage the progress that has been made in computation and numerical algorithms. Also there is a difficulty in absorbing new knowledge in the processes, which is now recognized by regulators and the industry as a whole. In other words, the methods are limited in their responsiveness. Even state-of-the-art, best estimate plus uncertainty methods provide little information on the actual margin available in the plants. Most margins reside in engineering judgment and conservative assumptions, which were built to deal with the imperfect knowledge.

Moving forward, the industry is expected to develop better-standardized databases and improved interfaces across the various engineering disciplines as more automation is implemented in the processes. This will enable consideration of new paradigms to manage the uncertainties across the various disciplines with a truly multi-physics approach to the LOCA problem.

The proposed RIMM Industry Application ECCS/LOCA methodology and tool will provide a means of quantifying the impact on the key LOCA analysis figures of merit PCT, ECR, and core-wide oxidation (CWO) of a change in LOCA analysis inputs. This information would be obtained without the resource requirement, cost, and schedule, of an actual LOCA reanalysis using the integrated LOCA evaluation model. The information that the tool provides can then be used for decision-making and margin management. The project is expected to create value by anticipating the trends towards integrated multi-physics models and focusing on developing a methodology that effectively addresses the limitations of traditional LOCA methods as presented above. The primary goal is to explore an integrated approach for knowledge and uncertainty management, as illustrated in Figure 2.

The global vision for the RIMM IEM LOTUS is summarized in the following propositions:

- Provide a responsive toolkit for the plant operator, which enables rapid decisions on considered changes within the LOCA issue space (as regulated under the proposed 10 CFR 50.46c). The goal is to greatly reduce the response cycle.
- Enable current knowledge to be factored into the process to enhance safety and operation optimization.
- Quantify currently not quantified uncertainties (to the extent practical) and trends to a realistic representation of the LOCA, which provides insights on the design. This includes the combination of risk and physics simulations as illustrated in Figure 3.
- Foster an approach that can lead to new knowledge and understanding of the LOCA scenarios, which could be “locked” in the engineering assumption of licensing calculations. Enable a more effective “exploration” of the issue space in order to improve core design.
- Eliminate issues associated with the Wilks’ approach (including variability in the estimator, risk of under-prediction of or over-prediction of FOM, lack of knowledge in what is limiting in the design, incapacity to perform sensitivity studies, etc.)
- A “plug-and-play” design of the multi-physics tool, which enables plant owner/operators and vendors to consider and further develop the RIMM Framework for use with their established codes and methods.

Note that LOTUS is not intended to replace licensing Analyses of Record (AORs) but rather to replace or aid the engineering judgment applied in managing those AORs. In other words, LOTUS is a margin management and optimization tool. This objective is achieved by representing the plant realistically, but in a way that makes it feasible to explore the issue space thoroughly, with all the uncertainties included and by considering and managing the entire body of knowledge.

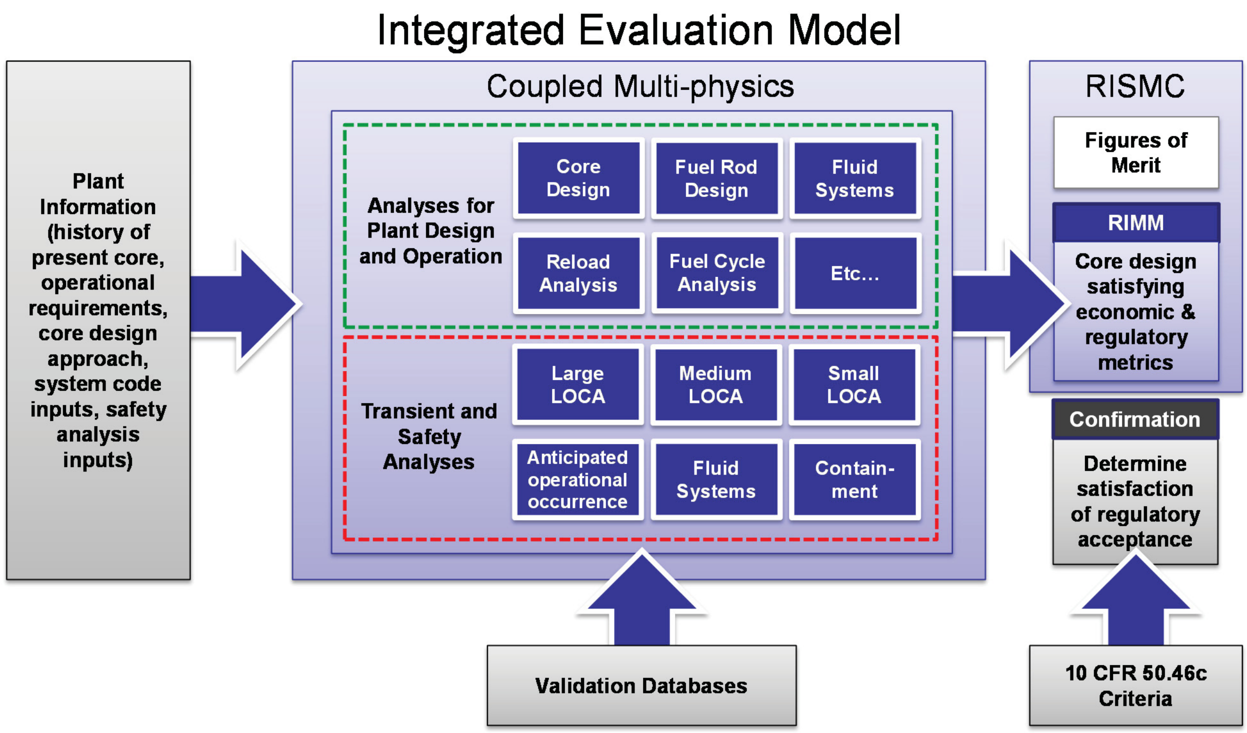


Figure 2. Flow Chart of the RIMM Integrated Evaluation Model. [5]

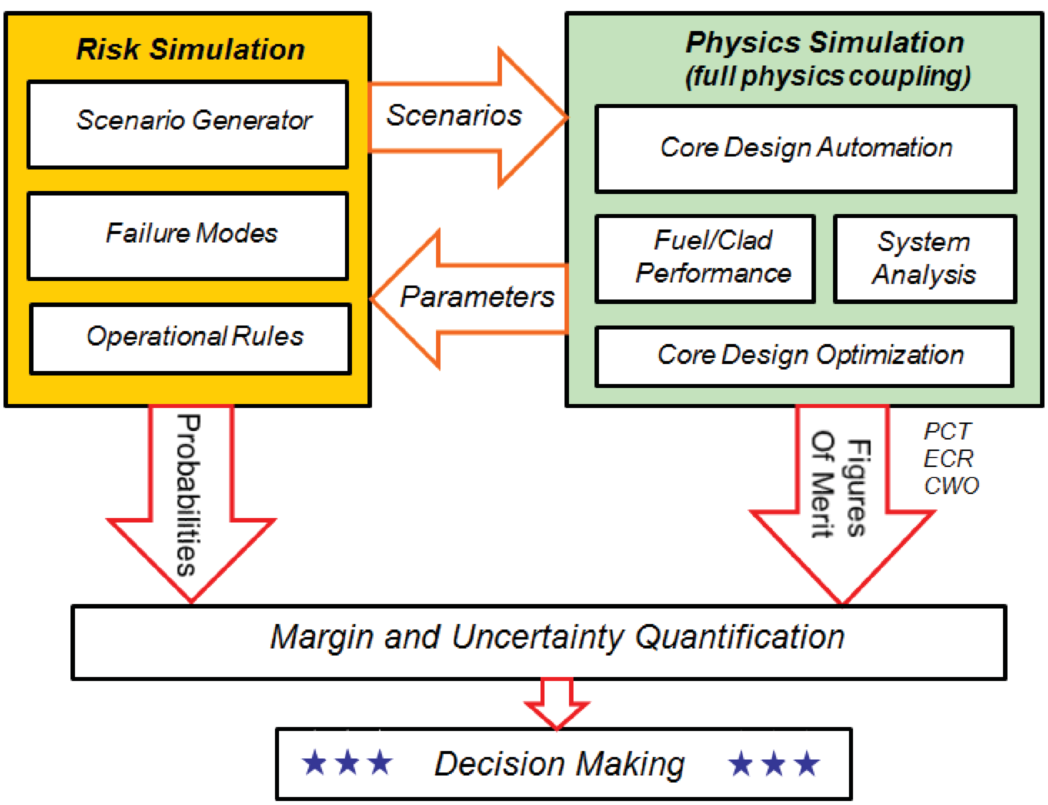


Figure 3. RISMC Margin Quantification and Risk Assessment Paradigm.

2. LOTUS: A RISK-INFORMED SAFETY MARGIN MANAGEMENT TOOLKIT FOR INTEGRATED CLADDING/ECCS PERFORMANCE ANALYSIS

This section provides more in-depth discussions of the LOTUS framework and its software structure.

2.1 Introduction of LOTUS

As mentioned, the general idea behind the Industry Application ECCS/LOCA is the development of an Integrated Evaluation Model (IEM). The motivation is to revisit how uncertainties are propagated across the stream of physical disciplines and data involved, as well as well as how risks are evaluated in a LOCA safety analysis as regulated under 10 CFR 50.46c. The use of an integrated approach in managing the data stream is the most important aspect of LOTUS. This also is well suited with current trends in industry to enhance automation and develop integrated databases across their organizations. As mentioned in the introduction, this IEM is called LOTUS, which stands for LOCA Toolkit for the U.S., and it represents the LWRS Program's response to the stated problem.

A LOCA safety analysis involves several disciplines, which are computationally loosely (externally) coupled to facilitate the process and maintenance of legacy codes and methods. A review of a few examples of analyses performed by vendors such as AREVA and Westinghouse Electric Company (WEC) is instructive to define the state-of-the-art in the industry. The key disciplines involved in a LOCA analysis are:

- Core physics;
- Fuel rod thermo-mechanics;
- Clad corrosion;
- LOCA thermal-hydraulics;
- Containment behavior.

The focus of LOTUS is to establish the automation interfaces among the five disciplines as depicted in Figure 4. These five disciplines include:

1. Core Design Automation (CD-A), which focuses on automating the cross section generation, core design and power maneuvering process.
2. Fuel Performance (FP), which focuses on automating the interface between core design and fuel performance calculations and the interface between fuel performance and system analysis.
3. System Analysis (SA), which focuses on automating the process required to setup large number of system analysis codes runs needed to facilitate RISMC applications on LOCA.
4. Uncertainty Quantification and Risk Assessment (RA), which focuses on uncertainty quantification, sensitivity analysis as well as establishing the interfaces to enable combined deterministic and probabilistic analysis.

5. Core Design Optimization (CD-O), which focuses on developing core design optimization tool that can perform in-core and out-of-core design optimization.

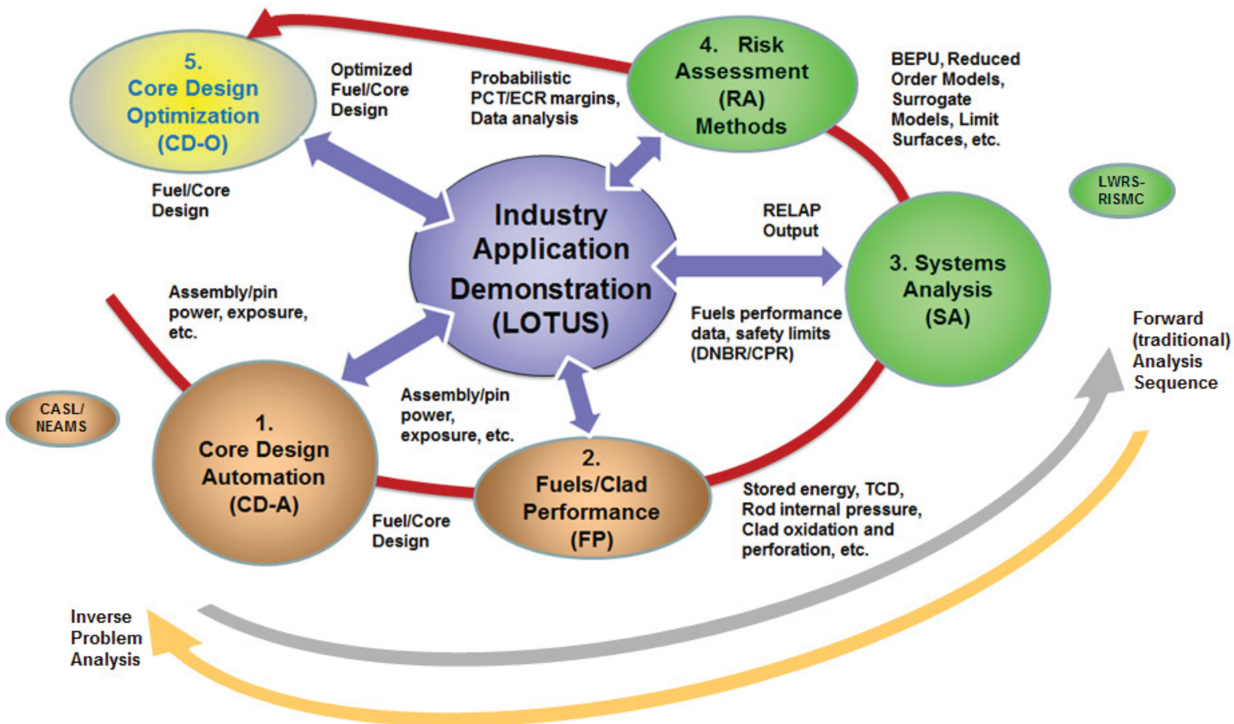


Figure 4. Schematic Illustration of LOTUS.

LOTUS utilizes existing and mature computer codes as well as advanced computer codes still being developed under various DOE programs to provide feedback and guide development of advanced tools. Regardless of the specific codes used to model the physics involved, the methodology proposed here is a paradigm shift in managing the uncertainties and assessing risks. The primary characteristic of LOTUS is to be an integrated multi-physics tool. This is a model in sharp contrast with the "operator split" or "divide-and-conquer" approach currently adopted in the industry, where every physics is resolved independently and coupling is addressed by complex interface procedures. There are significant assumptions and engineering judgment in setting up these procedures which make the propagation of uncertainties across the disciplines complex, prone to errors, and more importantly, current methods retain analytical margin which cannot be exploited.

The value proposition of LOTUS for industry stakeholders can be summarized in the following objectives:

- Provide a multi-physics simulation toolkit such that more plant "realism" would be available to the plant owner and operators.
- Provide a first-of-a-kind safety analysis capability that is efficient and affordable to the plant owners and operators to provide quantitative estimates of design or operational margin loss or gain associated with various combinations of changes in the plant.

- Allow/inform plant modification and equipment upgrade related safety analysis by better informing licensees in their decision process.
- Provide studies in response to customer inquiries and requests.
- Respond to regulatory inquiries and requests for additional information.

Currently there are no nuclear plant owner-operators in the U.S. that perform LOCA analysis for determining compliance with 10 CFR 50.46. In the U.S., Analysis of Records (AORs) is generated by nuclear vendors under contract by the plant operators. The vendor is responsible of the development of codes and methods while seeking generic approval of the methodology over a target class of plants. The vendor then performs the plant specific analysis for the licensee to demonstrate compliance with the 10 CFR 50.46 criteria. The plant operator manages the analysis inputs and maintains the AORs. The vendor is responsible of managing the analysis process and assessing impact of input errors, which may be found after the AOR is in place.

A limited number of owner-operators perform LOCA analysis for other purposes such as pipe break mass and energy release or training simulator validation. For an owner/operator, the LOTUS methodology and tool has two distinct types of potential applications. The more likely type of potential applications is for LOCA analysis related work contracted to the fuel vendor, which could include the following potential uses:

- Obtain quantitative estimates of design or operational margin loss or gain associated with various combinations of changes in LOCA analysis inputs.
- Obtain quantitative estimates of impact on the LOCA analysis figures-of-merit due to changes in LOCA analysis inputs (including reporting of LOCA analysis ΔPCT and ΔECR due to LOCA analysis input changes that are required by 10 CFR 50.46).

Another possible type of potential application is to use the LOTUS methodology and tools as an independent owner-operator LOCA analysis capability; especially LOTUS requires minimum infrastructure and training for its usage. This capability could be used to perform vendor-independent scoping or audit calculations that would facilitate decision making related to the impact of plant and fuel design changes, as well as provide an enhanced vendor oversight capability. An owner-operator could develop this capability with in-house staff or by outsourcing to an engineering services or consulting entity. In principle, a reload engineer that has trained LOTUS to analyze a given core design can re-analyze such reference design in much faster time than using a traditional reload design analysis process.

The 5th area shown in Figure 4, the core design optimization focuses on developing core design optimization tool that can perform in-core and out-of-core design optimization. This area is a possibility to be investigated in the future. Eventually, we will be able to incorporate optimization schemes into LOTUS that can quickly reshape a desired parameter envelope (in this case ECR) as an optimization feature of a core design process. In practice, such a step will require additional changes of today's design process, in order to incorporate LOCA analysis as an integrated element of the reload analysis process.

2.2 Description of LOTUS

As a multi-physics analytical framework, LOTUS is not intended to replace licensing Analysis of Records (AORs), but rather to replace or aid the "engineering judgment" which is typically applied in the management and maintenance of those AORs. The goal is an analytical and computational device that can represent a power plant realistically with all the uncertainties included and that considers all physical disciplines involved in an integrated fashion.

The first step in obtaining the desired technical capability to perform the type of analyses that address the challenges presented earlier is to revisit how uncertainties are propagated across the stream of physical disciplines involved. Regardless the specific codes used to model the physics involved, the methodology presented here is really a different strategy in managing the uncertainties. In the LOTUS framework uncertainties are propagated directly from all the uncertain design and model parameters. The interactions between the various model parameters are directly solved within the LOTUS framework. The development of the LOTUS framework follows the guideline specified in the Code Scaling, Applicability and Uncertainty (CSAU) methodology [6] which was approved by the NRC in 1996 after an extensive review.

When the existing NPPs were first built, there were plenty of safety margins available for them. The NPPs were able to demonstrate safety compliance with conservative safety analysis approaches such as the Appendix K approach. Over the years a number of technological innovations have been introduced into the plant operations to bolster the economic performance of the NPPs. These include: 1) longer operating fuel cycle – the cycle length has gone from annual cycle to eighteen months or twenty four months cycle, 2) higher enrichment of the fuel – the fuel enrichment nowadays is close to the license limit of 5%, 3) higher discharge burnup of the fuel, 4) power uprates - total extra power generated from power uprates is equivalent to that of building six new 1,000-MWe nuclear power plants.

All these innovations have resulted in remarkable performance of the existing NPPs. However they have also eroded the available safety margins of these plants. The plant aging would add more loss of margin to these plants. The nuclear industry is able to recover the safety margin by developing best estimate plus uncertainty (BEPU) modeling and simulation methodologies. The 1988 amendment of the 10 CFR 50.46 rule allowed the use of realistic models to analyze loss-of-coolant accident, and consequently triggered significant interests in the development of computer codes and methodologies based on best estimate plus uncertainty. A group of experts or technical program group under the sponsorship of the US NRC took an effort to demonstrate that practical methods could be developed which would be acceptable under the new regulations. Shortly after its completion, the CSAU methodology and its demonstration were described in a series of papers appeared in Nuclear Engineering and Design [7]. Since then, different variants of best-estimate plus uncertainty methodologies have been developed and are now widely employed in the nuclear industry. Since any realistic calculation requires the assessment of uncertainties, an essential step in a BEPU is the assessment of uncertainties associated with physical models and data, and plant initial and boundary condition variability. As uncertainties are incorporated into the safety analysis process, a procedure is developed where results from a number of calculations are collected to develop a statement whether compliance with prescriptive rules or acceptance criteria is demonstrated.

The CSAU process is divided into three main elements. Element 1 includes: 1) specify scenario, 2) select NPP, 3) develop an Phenomena Identification and Ranking Table (PIRT), 4) Select frozen code, 5) Provide complete documentation, and 6) determine code applicability. PIRT is a critical element of CSAU-based methodologies. This element is designed to focus the prioritization of code assessment and facilitate the decisions on physical model and methodology development.

Element 2 is the assessment of the code. This element includes: 1) Establish assessment matrix, 2) Code validation with separate effect tests (SET) data and integral effect test (IET) data, 3) Determine code and experiment accuracy, 4) Determine effect of scale. A key output from this element is the establishment of probability distributions and biases for the contributors identified in Element 1. In addition to the generation of probability distributions, this element required a thorough assessment of the code's ability to correctly predict all the dominant physical processes during the transient.

Element 3 is the actual implementation stage of the methodology with sensitivity and uncertainty analysis performed here. This element includes: 1) Determine effect of reactor input parameters and state, 2) Perform NPP sensitivity calculations, 3) Combine biases and uncertainties, 4) Total uncertainty to calculate specific scenario in a specific NPP.

The CSAU methodology was structured, traceable, and practical and therefore it is ideally suited for application in the regulatory and design arenas. This has been demonstrated by several successful implementations of the CSAU-based methodologies by the fuel vendors (Westinghouse, AREVA, GE-Hitachi) currently licensed and applied to safety analysis in the industry.

The convolution of the many LB-LOCA uncertainty contributors to the figures of merit (i.e. PCT) is an inherently statistical approach. The two commonly used approaches are generally classified as either parametric or nonparametric. The response surface method, a parametric method, was the approach demonstrated in the CSAU demonstration problem. The objective of that method is the development of a response surface describing peak clad temperature sensitivity to the dominant LB-LOCA uncertainty contributors. The number of calculations required for that approach is dependent on the number of LB-LOCA uncertainty contributors considered. Since the original rollout of the CSAU method, the number of phenomena considered important has increased. Practical constraints limit the number of uncertainty parameters with parametric methods.

The fuel vendors, Westinghouse, Areva and GE-Hitachi, chose to apply a nonparametric approach originally recommended in the German Gesellschaft fur Anlagen und Reaktorsicherheit (GRS) methodology. This statistical method is often referred to as Wilks' method [8]. The nonparametric approach decouples the association between the number of uncertainty parameters and the number of required calculations. The desired quantification of PCT uncertainty is the identification of a specific result that represents coverage of the results domain at or above 95% with a 95% confidence. The 95/95 coverage/confidence has been recognized by the U.S. NRC as having sufficient conservatism for LB-LOCA analyses. Non-parametric methods allow an unlimited number of uncertainty parameters.

The main advantage of nonparametric statistical methods is that the number of treatable uncertainty contributors is independent of the number of plant calculations. This characteristic provides flexibility during the development process to explicitly address as many or as few analysis contributors as necessary to resolve the outcome of the phenomena identification and ranking table. As this is a product of engineering judgment, the uncertainty associated with this exercise can be reduced by explicitly addressing additional analysis contributors. In addition, this methodology characteristic provides the opportunity to incorporate customer requests for the explicit treatment of plant process uncertainty.

It is worth to note that the Wilks' approach has a number of issues associated with it. These include: 1) variability in the estimator, i.e. risk of under-prediction of or over-prediction of figures of merit, 2) lack of knowledge in what's truly limiting in the design, 3) incapacity to perform sensitivity studies and impact assessment etc. Despite its widespread adoption by the industry (AREVA, GE-Hitachi and Westinghouse), the use of small sample sizes to infer statement of compliance to the 10 CFR 50.46 rule, has been a cause of unrealized operational margin in today's best estimate plus uncertainty methods. Moreover, the debate on the proper interpretation of the Wilks' theorem in the context of safety analyses is not fully resolved yet more than a decade after its introduction in the frame of safety analyses in the nuclear industry. This represents both a regulatory and applicant risk in rolling out new methods.

The proposed 10 CFR 50.46c rule added another layer of complexity for the demonstration of compliance. Under the current rule, PCT, Maximum Local Oxidation (MLO), and Core Wide Oxidation limits are set to specific values (2200 F, 17% and 1% respectively). Using Wilks' method, compliance is easily demonstrated by ranking the corresponding values obtained from the simulations in the sample and ensuring that the rank representing the 95/95 estimates from a small sample is below those limits. Considering there are three outcomes (PCT, MLO and CWO), the highest ranked set from a sample of 124 can be chosen. With the proposed rule in 10 CFR 50.46c, the limit is a curve, more specifically both PCT and MLO (maximum Equivalent Clad Reacted (ECR) in this case) limits are functions of cladding hydrogen content, which varies from rod to rod in the core. Applying Wilks' method would require to define new figures of merit that synthetize this relationship. Additionally, if the analyst is ultimately interested in tracking the margin in each core region that would not be possible unless a much larger sample size is used.

Another important aspect to note that the existing BEPU methodologies have been primarily focused on thermal hydraulic systems analysis codes when they were developed. The LOCA problem is truly a complex multi-physics problem involving core design, fuel rod performance, and systems analysis. However due to the limitation of the computing power and computer codes as well as limited appreciation of advanced statistical methods, the standard industry practice is to break the problem into more manageable set of disciplines with simplified computer models. Each discipline requires experienced developers to develop and maintain the computer codes and experienced users to perform the analyses, hence the analyses are typically done in silos via operator split approach. The analysis results would be passed from one discipline to another with complex interfaces setup between the disciplines. These interfaces have been developed over the years consistently with specific acceptance criteria and regulatory requirements. This process tends to be: 1) error-prone, 2) inherently inconsistent between displaces, 3) complex interface between disciplines, 4) inefficient. The existing BEPU process is schematically illustrated in Figure 5.

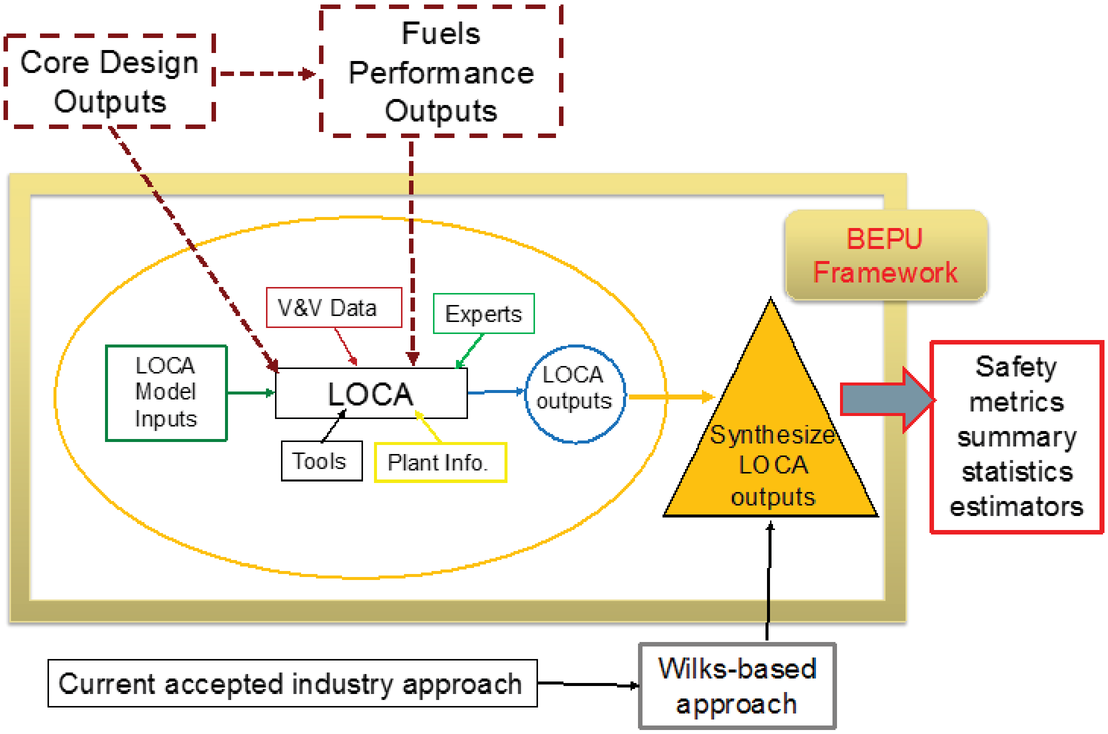


Figure 5. Schematic Illustration of Current BEPU Process for LOCA Analysis.

Because of the high cost to keep the safety analysis capability in an organization, nowadays only fuel vendors such as Westinghouse, Areva and GE-Hitachi have the capability (codes and analysts) to perform safety analysis. A few large size utilities have limited capability to perform some transient analyses but not the design basis accident (DBA) type of analysis such as LOCA. It is also noted that the current BEPU method still contains a high degree of conservatism, mostly to cover a lack of knowledge in some phenomena and to make easy to obtain the licensing and implementation. Further, the propagation of uncertainties across the various disciplines is addressed by defining bounding assumptions at the interfaces which limit the possibility that the impact of an issue in a specific discipline (error discovered, design change or other) to cross-over to other physics in an efficient fashion.

Such traditional processes and interfaces do not easily adapt to new integrated methods and cannot fully leverage the progress that has been made in computation and numerical algorithms. Also there is a difficulty in absorbing new knowledge in the processes, which is now recognized by regulators and the industry as a whole. In other words the methods are limited in their responsiveness to incorporate new data, models and methods. The existing best estimate plus uncertainty methods provide little information on the actual margin available in the plants. Most margins reside in engineering judgment and conservative assumptions, which were built to deal with the imperfect knowledge.

With the performance based regulation on the horizon (e.g. 10 CFR 50.46c) and the industry's push to introduce accident tolerant fuel and adapting flexible operating strategy, multi-physics simulations are mandatory. Computational constraints to analyze highly complex systems with many variables to be considered have kept us in the past from executing multi-physics types of schemes. Moving forward, the industry is expected to develop better-standardized databases and improved interfaces across the various engineering disciplines as more automation is implemented in the processes. This will enable consideration of new paradigms to manage the uncertainties across the various disciplines with a truly multi-physics approach to the safety analysis problem. Fortunately, with the impressive advancements of the computing power over the past few decades, the multi-physics simulations are now becoming practical. Today, with the development of the LOTUS Toolkit built in a state-of-the-art computational environment we have the potential to implement complex multi-physics approaches solving fully coupled systems problems in acceptable time. It is recognized that in the multi-physics simulation environment, the management of uncertainty is much more complex than the industry practice and consequently warrants thorough evaluation. To make the multiphysics simulations applicable to solve real world reactor safety problems, it is imperative to develop Multi-Physics Best Estimate Plus Uncertainty (MP-BEPU) methodology such that uncertainties can be propagated consistently in multi-scale and multi-physics environment to fully realize the benefits of multi-physics simulations.

Figure 6 shows the comparison between the current BEPU approach and the MP-BEPU approach to be developed in the LOTUS framework. The column on the right in Figure 6 is the "ideal final solution" in a situation of "perfect knowledge". In that situation, LOTUS should be able to predict the "true" best-estimate or "nominal" state of the device (given plant, scenario, etc.) and then account for all the uncertainties which can be combined in what is called the "true/theoretical value of total uncertainty". Compliance with the rule is demonstrated by showing that the MP-BEPU value is below the regulatory limit which is designed by regulators to be below the physical limit.

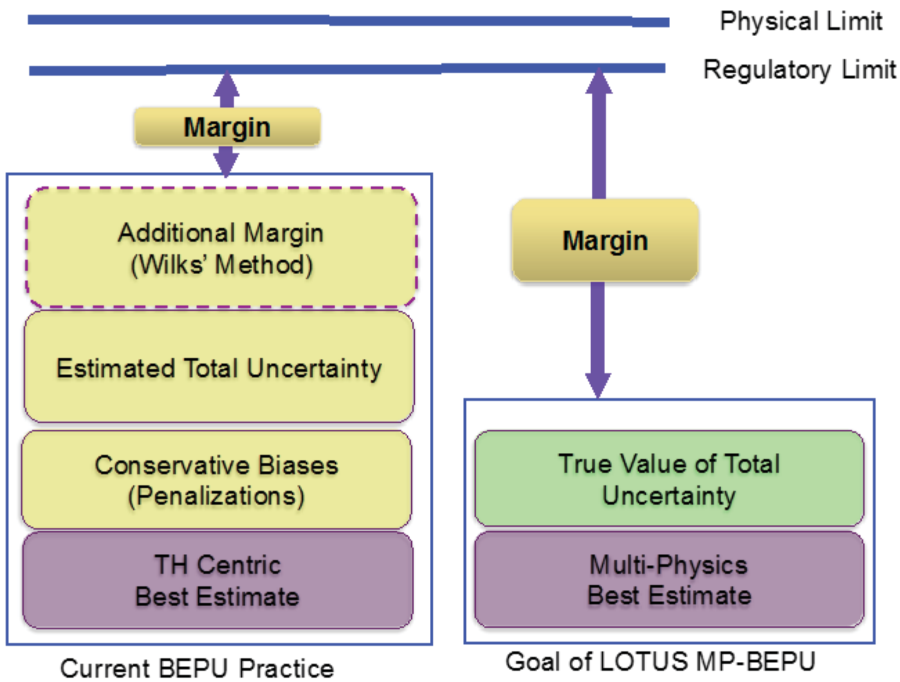


Figure 6. Paradigm Shift with LOTUS Multi-Physics BEPU.

The general principles and analysis steps laid out in the CSAU methodology will be followed in the LOTUS MB-BEPU methodology development. LOTUS, as illustrated in Figure 7, is envisioned as a virtual environment that is composed of many different computer codes such as VERA-CS and PHISICS for core design automation, FRAPCON/FRAPTRAN and BISON for fuels performance and RELAP5-3D and RELAP-7 for systems analysis. Each code has its own input. In order to reduce the burden on the users to use these codes, LOTUS will be developed to prepare the input files necessary to run these various computer codes from a common input file. One added benefit to start the multi-physics analyses from one common input file is that the chance of making errors can be greatly reduced due to the inconsistencies between the inputs for different computer codes.

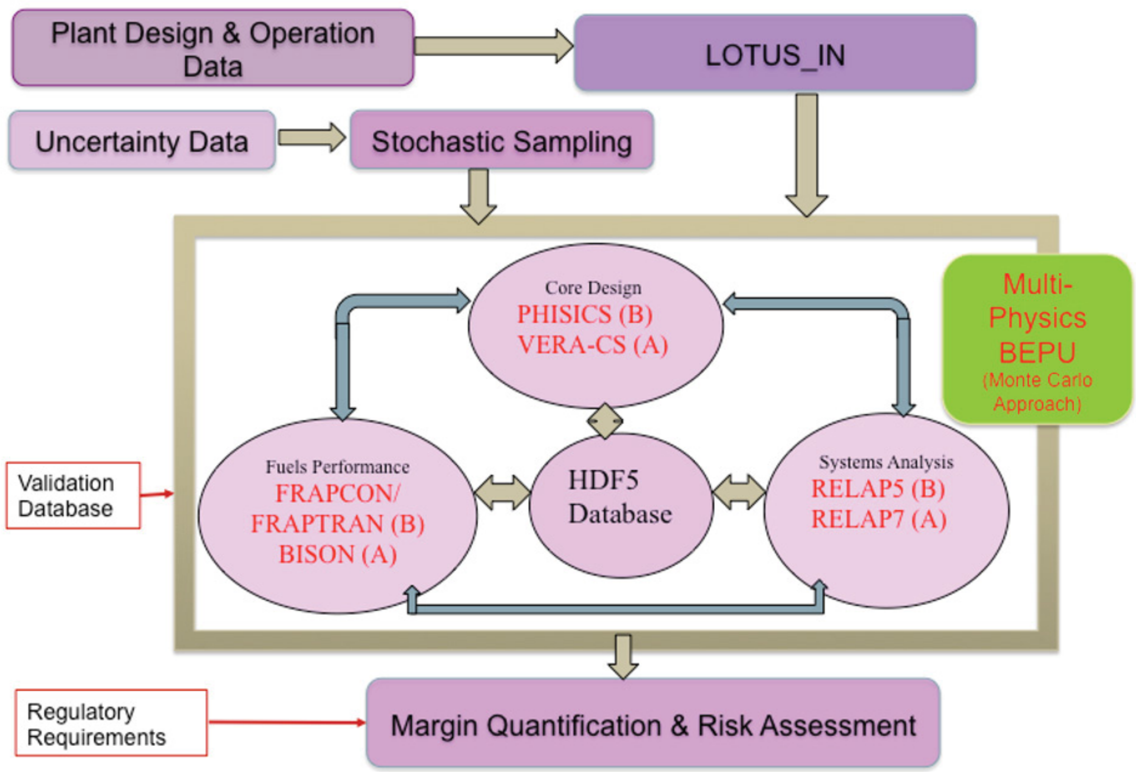


Figure 7. Illustration of LOTUS Multi-Physics BEPU (MP-BEPU) Safety Analysis Framework.

The LOTUS development is subdivided into two phases:

Phase I – LOTUS-B (Baseline)

In this phase, four elements (core design automation, fuels performance, systems analysis and risk assessment) of the LOTUS-B toolkit are exercised. Each element implements a set of existing, well-established code(s) into the LOTUS-B toolkit, as illustrated in Figure 7 by (B).

Phase II – LOTUS-A (Advanced)

In conjunction with the Baseline development, the advanced phase of the LOTUS project will be executed. The duration and timeline associated with the LOTUS-A phase is in part dependent on the execution and lessons learned of the Baseline phase, and availability and maturity of the advanced tools in development today. An example of tools and methods to be implemented during the advanced phase is shown in Figure 7 by the (A) symbol.

It is noted that one distinguishing feature with LOTUS MP-BEPU is that full Monte Carlo simulations will be used when it comes to managing uncertainties. It has to be acknowledged that the sample size needed to reduce the confidence interval on the estimate (standard error) to the magnitude desired may require sample sizes in excess of 1,000-10,000 cases. However the benefit is that full Monte Carlo simulations allow sensitivity analyses to be performed such that the impact and significance of input parameter changes can be assessed with high confidence.

Another important feature to stress on in the LOTUS development is that the uncertainties are propagated directly from all the uncertain design and model parameters. The interactions between the various model parameters are directly solved within the LOTUS framework. This interaction not only facilitates the automation of the process, but it is also mathematically more robust because the advanced procedure considered to propagate uncertainties and/or perform global sensitivity and risk studies requires inputs sampled to be independent. This requirement is hard to achieve following the traditional “divide-and-conquer” or operator split approach.

Conventional methods are strongly “code-oriented.” The analyst has to be familiar with the details of the codes utilized, in particular with respect to their input and output structures. This represents a significant barrier for widespread use beside the small pool of experts within the specific organization or even groups within the organization that develop such codes. It becomes apparent how difficult is to make changes and accelerate progress under such paradigm, especially in heavily regulated environment where even a minor line changes in a code carries a heavy cost of bookkeeping and regulatory actions.

The LOTUS vision is to move toward to a “plug and play” or “task oriented” approach where the codes are simply modules ‘under the hood’ that provides the input-output relationship for a specific discipline. The focus shifts on managing the data stream at a system level, as depicted in Figure 8. LOTUS is essentially a workflow engine with capability to drive physics simulators, model complex systems and provide risk assessment.

A “plug-and-play” approach will enable plant owners and vendors to consider and further customize the LOTUS framework for use within their established codes and methods. Therefore, it could potentially become the engine for license-grade methodologies. In other words, it is possible that LOTUS technology could be advanced in the future to a level of fidelity and maturity that it could be used for some licensing or regulatory situations. An example would be the reporting of LOCA analysis ΔPCT and ΔECR related to LOCA analysis input changes that are required by 10 CFR 50.46c.

As mentioned earlier, the ultimate goal is then to incorporate optimization schemes in LOTUS that can quickly reshape a desired parameter envelope (for example ECR) as an optimization feature of a core design process. This step will require additional changes to today’s design process, in order to incorporate LOCA analysis as an integrated element of the reload analysis process.

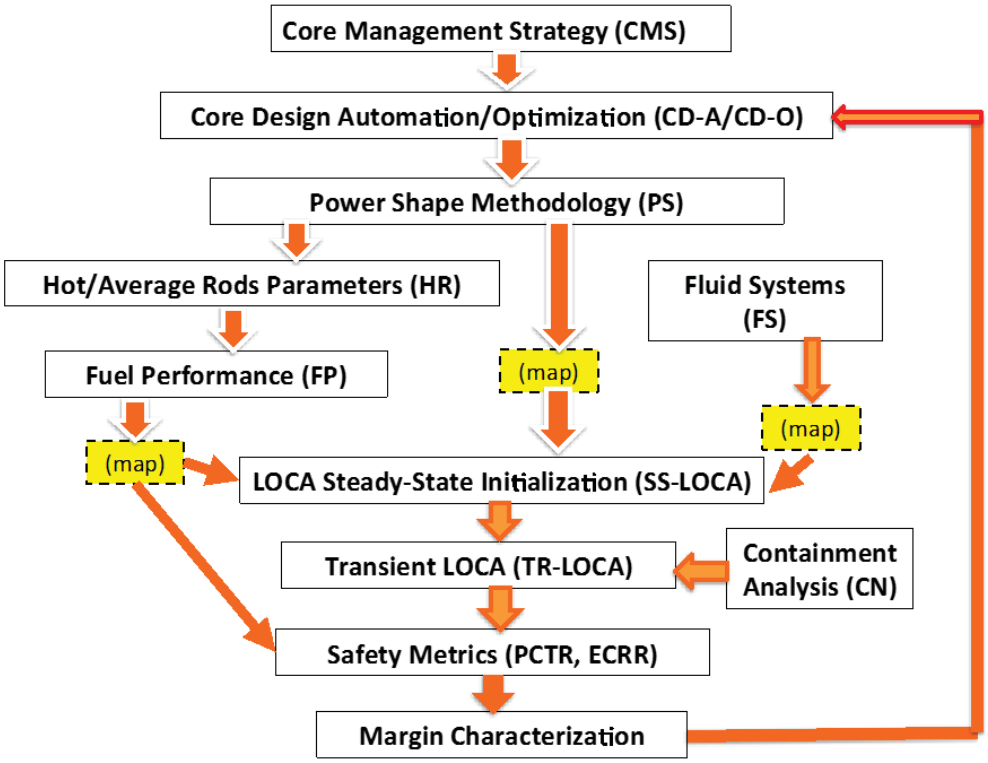


Figure 8. LOTUS Data Stream.

2.3 LOTUS Software Development

As illustrated in Figure 7, the software structure of LOTUS will compose of three main components: 1) LOTUS-IN, 2) Multi-Physics BEPU, and 3) Margin Quantification and Risk Assessment.

The **first component** is called LOTUS-IN which is a common input processor that will be developed such that the input files for the different physics codes will be generated from a single common input file. The single common input file would contain the input syntax that is easily apprehended by the users. LOTUS_IN will convert the common input file into the input files readable by the various computer codes such as VERA-CS, FRAPCON/FRATRAN, RELAP5-3D, etc. LOTUS_IN also prepares the scripts that would drive the execution all of the different physics codes.

As an example, the following is a block from the common input file to build the heat structure model for an assembly labeled as E-1:

Assembly coordinate	E-1
Fuel pellet radius	0.01344
Clad inner radius	0.01371
Clad outer radius	0.01558
Number of axial heat structures	6
Number of radial mesh points	9
Fuel rod geometry	Cylindrical

Temperature steady-state initialization	1
Left boundary coordinate	0.0
Reflood condition	1
Boundary volume indicator	1
Maximum number of axial intervals per heat structure	4
Heat structure number	700
Initial gap internal pressure	1000
Gap conductance reference volume	20660000
Initial oxide thickness on cladding's outer surface	0.0
Cladding material density	6500.0
Activation Energy (cal/mole)	35890.0
Reaction rate constant (variable K) (m2/s)	2.252E-06
Reaction heat release (J/kg-mole)	5.940E+08
Cladding material molecular weight (kg/kg-mole)	91.22
Molecular weight of reaction product divided by Word 6	0.0442
Fuel surface roughness	3.28E-06
Clad surface roughness	6.56E-06
Radial displacement due to fission gas-induced fuel swelling and densification	0.0

LOTUS_IN would convert the above input block into the following input for the RELAP5-3D model:

```
* fuel assembly E-1
*  crdno  no  axial  no  radial  cyl  stdy-st  left  bound  reflood
17000000  6  9  2  1  0.0  1  1  4
*  crdno  P gap  P vol
17000001  1000.0  20060000
*  crdno  oxide thicknessdensity  activation E  rate constant  heat release  mol. wt.  mol. wt. ratio
17000003  0.0  6500.0  35890.0  2.252E-06  5.940E+08  91.22  0.0442
*  crdno  Floss flag
17000004  1
*  crdno  fuel rough  clad rough  rad. Displ. Fuelrad. Displ. Clano
17000011  3.28E-06  6.56E-06  0.0  0.0  6
*  crdno  mesh  flag  mesh  spec
17000100  0  1
*  crdno  no  int  right  bound
17000101  5  0.01344
17000102  1  0.01371
17000103  2  0.01558
```

The **second component** is where all the multi-physics best estimate plus uncertainty simulations will be performed. This component performs the coupled codes calculations. The so-called LOTUS analysis manager will be developed to handle the following tasks: 1) stochastic sampling of the PIRT table, 2) data mapping between disciplines, 3) preparation of the large number of input files with the perturbed model parameters generated from the stochastic sampling, 4) execution of the large number of simulations.

Reactor safety analysis calculations are normally done in two sequential steps. Step 1 is the steady-state initialization and step 2 is the transient calculations. In the steady-state initialization, the calculated parameters would match those of the plant conditions. The transient calculations predict the plant accident behavior. The coupling between computer codes for steady-state initialization is done through Python. It is an interpreted, object-oriented, high-level programming language with dynamic semantics. Its high-level built in data structures, combined with dynamic typing and dynamic binding, make it very attractive for rapid application development, as well as for use as a scripting or glue language to connect existing components together. For instance, the Python scripts developed in LOTUS extracted the fuel rod power history from Core Design Automation and mapped into the FRAPCON input files. For transient calculations, the tightly coupled calculations between computer codes would be necessary. For instance, the coupling between computer codes for transient calculations (e.g. LOCA) will be carried out through tightly coupled simulations, i.e. coupled RELAP5-3D/FRAPTRAN or coupled RELAP5-3D/BISON runs, under LOCA conditions.

There are two types of LOTUS managers to be developed: 1) LOTUS SS Manager for steady-state initialization analysis and 2) LOTUS Transient Manager for the transient analysis. Figure 9 shows a schematic illustration of the LOTUS SS Manager, with which all the data mapping between disciplines is to be carried out through a central database in HDF5 format. Figure 10 shows a schematic illustration of the LOTUS Transient Manager, with which the transient scenarios will be generated and the strongly coupled calculations between computer codes will be carried out. Figure 11 illustrates how LOTUS SS Manager and LOTUS Transient Manager work in sequence to carry out the calculations for safety analyses.

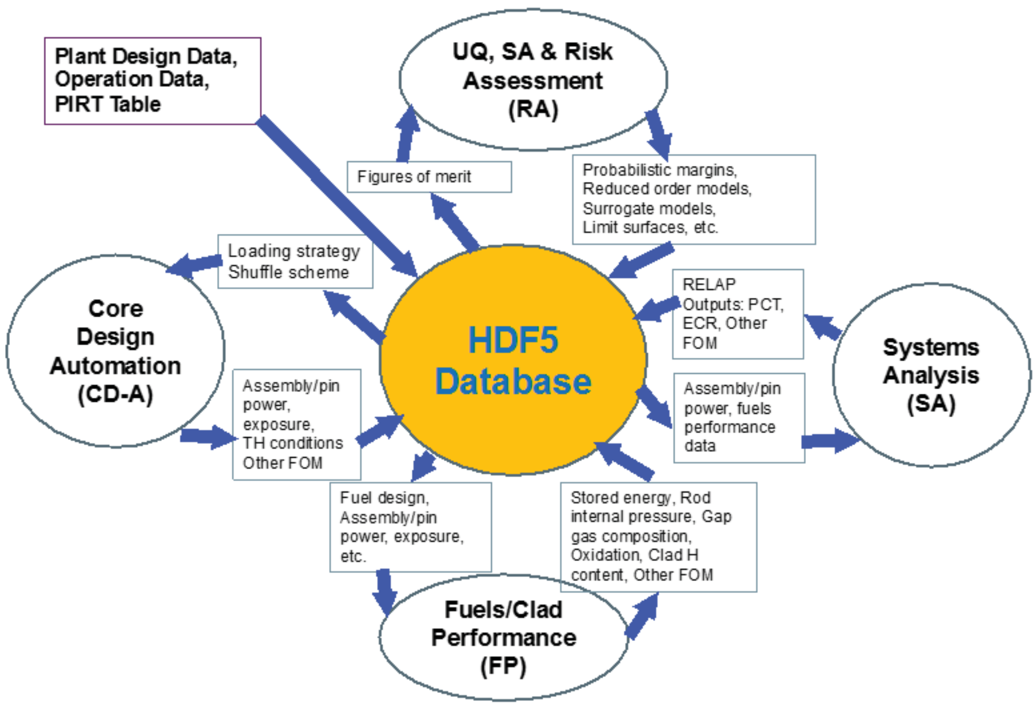


Figure 9. Schematic Illustration of LOTUS Steady-State Analysis Manager (LOTUS SS Manager).

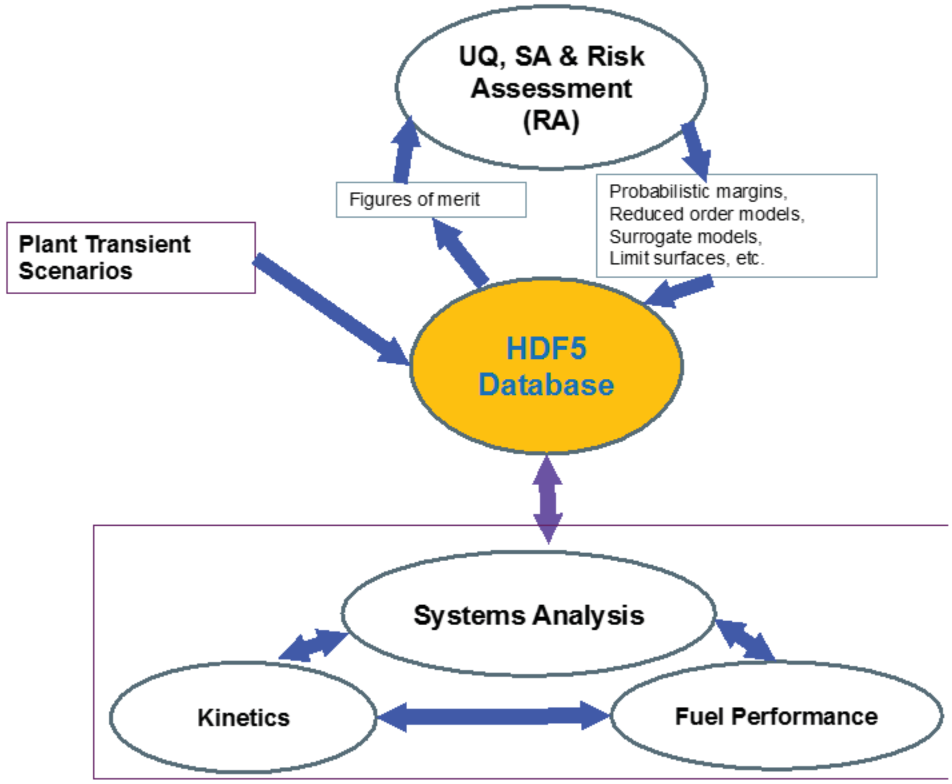


Figure 10. Schematic Illustration of LOTUS Transient Analysis Manager (LOTUS Transient Manager).

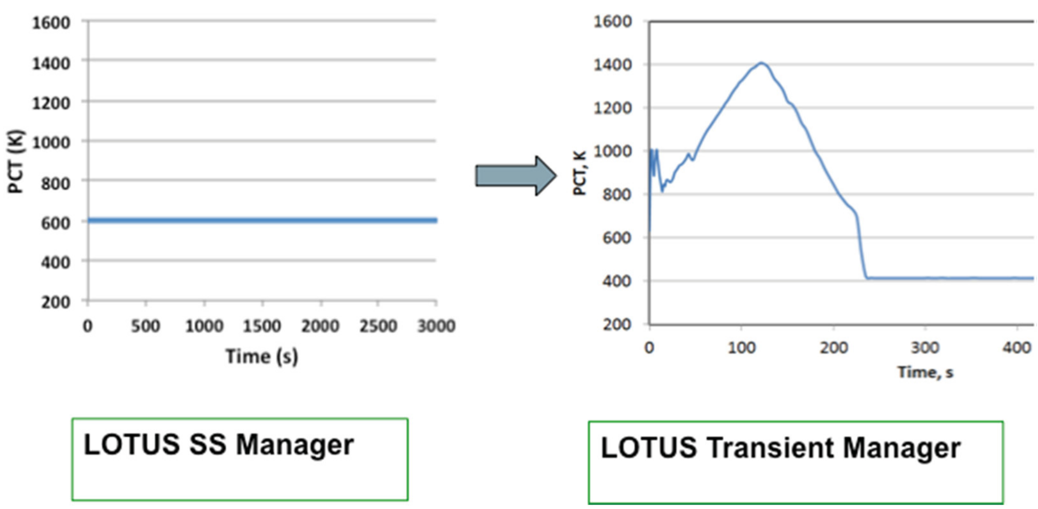


Figure 11. Illustration of LOTUS Managers.

The **third component** is essentially the post-processing of LOTUS with which margin quantification and risk assessment will be performed. All the outputs for figures of merit will be extracted along with the perturbed parameters from the PIRT table. The Monte Carlo based nonparametric statistical analysis approach will be used to perform uncertainty quantification and sensitivity analysis. Uncertainty quantification establishes confidence intervals for outputs of interest while sensitivity analysis quantifies the amount of output variance attributable to specific inputs. Uncertainty quantification and sensitivity analysis play important roles in the margin quantification and risk assessment. This component will establish the 95/95 upper tolerance limit for figures of merit to provide the risk assessment capability. The LOTUS framework will be developed to have easy interface with the probabilistic risk assessment tool such that probabilistic safety margin can be quantified and the plant risk can be assessed.

3. INTRODUCTION OF THE SOUTH TEXAS PROJECT (STP) PLANT

The South Texas Project Electric Generating Station (STPEGS) is one of the newest and largest nuclear power facilities in the nation. All the information presented in this section is available in Ref. [9]. The site is located in south-central Matagorda County west of the Colorado River, 8 miles north-northwest of the town of Matagorda and about 89 miles southwest of Houston. It consists of approximately 12,220 acres of land and includes areas being used for a plant, a railroad, and a cooling reservoir. STPEGS has two nuclear power units – units 1 & 2. STP's two units produce 2,700 megawatts of carbon-free electricity - providing clean energy to two million Texas homes. Both units are cooled by a 7000-acre Main Cooling Reservoir (MCR), which eliminates the need for cooling towers. The MCR is fully enclosed with an embankment, baffle dikes direct the flow of water. The station is located at the north end of the MCR with condenser cooling water being discharged into the western half of the MCR and returned to the power plant intake through the eastern half of the MCR. Figure 12 show a picture of STP Units 1 and 2.

The station is composed of two identical pressurized water reactor (PWR) nuclear steam supply system (NSSS) and turbine generator. Unit 1 reached initial criticality on March 8, 1998 and went into commercial operation on August 25, 1988. Unit 2 reached initial criticality on March 12, 1989 and went into commercial operation on June 19, 1989. Each unit utilizes a four-loop, PWR Nuclear Steam Supply System and supporting auxiliary systems designed by Westinghouse Electric Corporation. The rated core thermal power of each unit is 3853 MWt plus 21 MWt reactor coolant pump (RCP) energy. Each unit was designed for a net electrical output of 1380 MWe.

The reactor has a multi-region-cycled core. The fuel rods are Zircaloy tubes containing slightly enriched uranium dioxide fuel. The fuel assembly is of the canless type basically consisting of guide thimbles attached to top and bottom grids, and top and bottom nozzles. The fuel rods are held by spring clip grids which provide very stiff support. The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas releases, and excessive cladding stresses and strains. Rod cluster control assemblies (RCCAs) are inserted into guide thimbles of certain fuel assemblies for reactor control. The control rods use hafnium or silver-indium-cadmium as the neutron absorber. Above the core, each cluster of absorber rods is attached to a spider connector and drive shaft that is raised and lowered by a drive mechanism mounted on the reactor vessel head. Upon reactor trip, the RCCAs are inserted into the core by gravity. The control rods are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. A soluble neutron absorber is utilized for long-term reactivity control and refueling operations.

The reactor vessel and internals contain and support the fuel. The vessel has a low-alloy carbon steel hemispherical head and bottom, and is clad inside with stainless steel.

High-pressure light water serves as the coolant, neutron moderator, reflector, and solvent for the neutron absorber. The Reactor Coolant System (RCS), comprised of four parallel loops (each with a reactor coolant pump (RCP) and a steam generator (SG)), is used to transfer the heat generated in the core to the SGs using RCPs to circulate the water. RCS pressure is maintained by means of a pressurizer attached to the hot leg of one of the loops. The RCS is designed to circulate borated demineralized water at temperatures, pressures and flow rates consistent with the design thermal and hydraulic performance of the NSSS.

The emergency core cooling system (ECCS) [9] injects borated water into the RCS following a LOCA to limit core damage, metal/water reaction, and fission product release, and to provide, in conjunction with the control rods, sufficient negative reactivity to assure safe shutdown of the reactor core. Borated water is injected from the accumulators and the refueling water storage tank (RWST). The ECCS also provides long-term, post-accident cooling of the core by recirculating borated water from the containment sump to the core. The system consists of three independent trains, each one capable of providing 100 percent of the required flow to the core in the unlikely event of a LOCA. Each train consists of one high-head safety injection pump and one low-head safety injection pump. Heat is removed from the system during recirculation by the residual heat removal heat exchanger (low-head pump only). The piping and valving associated with each of the three subsystems are identical. In the event of a steam pipe rupture, the ECCS provides adequate shutdown capability.

The containment structure is a post-tensioned concrete cylinder with steel liner plates, hemispherical top, and flat bottom. It provides a virtually leak tight barrier to prevent escape of fission products to the environment in the unlikely event of a LOCA. It is designed to withstand the internal pressure and coincident temperature resulting from the mass and energy release of a LOCA.

The Essential Cooling Water System supplies cooling for those loads which are necessary for the safe shutdown of the reactor and to mitigate the consequences of postulated accidents. It also supplies cooling water to various systems during normal operation and shutdown. Heat rejection to the Essential Cooling Water System during either normal operation, normal shutdown, or design basis accident conditions is accomplished by three redundant cooling water loops, each having its own pump and motor, piping, valves, and instrumentation. Each loop cools one set of diesel generator heat exchangers, component cooling water heat exchanger, one essential chiller and the component cooling water pump supplementary cooler. The required cooling water is taken from the essential cooling pond which is also the ultimate heat sink.

The Reactor Trip System (RTS) automatically prevents operation of the reactor in an unsafe region by shutting down the reactor whenever preset limits are approached. The safe operating region is defined by several considerations, such as mechanical/hydraulic limitations on equipment and heat transfer phenomena. Therefore, the reactor trip system keeps surveillance on process variables which are directly related to equipment mechanical limitations, such as pressure, pressurizer water level, and also on variables which directly affect the heat transfer capability of the reactor, e.g., flow and reactor coolant temperatures. Still other parameters utilized in the reactor trip system are calculated from various process variables. In any event, whenever a direct process or calculated variable exceeds a set point, the reactor will be shut down in order to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the containment.

Figure 12. STPEGS Units.

4. LOTUS APPLICATION ON STP

From the nuclear power perspective, we have engaged staff from both South Texas Project (STP) and the Texas A&M University (TAMU) in the research for constructing LOTUS tailored to an existing nuclear power plant. The TAMU researchers offer collaborative expertise for design, modeling, and simulation of the pressurized water reactor. Toward that end, TAMU is assisting INL on the development and application of LOTUS for STP by constructing the associated thermal-hydraulics model that was used for the large break LOCA demonstration in this study. While the thermal-hydraulics model is built using existing plant information, to the extent possible, plant and fuel proprietary information is being replaced by generic and/or publicly available information in order to facilitate the sharing of information to all interested stakeholders.

Further, with the assistance of STP and TAMU, INL has started the construction of a database of information that will assist in the core design automation, the fuel/clad modeling, and the thermal-hydraulics systems analysis. These items are deemed essential to conduct future modeling and simulation and safety analyses activities using the LOTUS framework.

This chapter presents a demonstration of the application of the above-presented LOTUS methodology to a generic PWR model built based on the STP plant. The demonstration includes all aspects of LOTUS except the core optimization part, which is planned to be added in the future. The demonstration is carried out for the generic reference PWR model and a reference LOCA transient presented in the subsequent sections. Furthermore, the presented demonstration uses existing computer codes. This is called LOTUS-B, "B" for baseline. It is planned to replace these existing computer codes as the new, advanced codes currently under development become available. This will be called LOTUS-A, where "A" stands for advanced. A timeline for the transition from LOTUS-B to LOTUS-A is shown later in Ref. [4].

4.1 Core Design Automation

The modular approach for LOTUS will enable plant owners/vendors to further customize the LOTUS framework for use within their established codes and methods. One of the LOTUS modules deals with the core design automation (see Figure 4) (CD-A). The purpose of this module is to supply the subsequent LOCA analysis with initial conditions, i.e. assembly/pin power histories, power shapes, etc. to be employed by the Fuels/Clad Performance (FP) and System Analysis (SA) modules.

For the STP demonstration calculation presented in this report, the LOTUS tools and methodology introduced in the September 2016 Milestone [3], the LOTUS early demonstration, are used here. The following sections will give a short summary of these codes and methods used in LOTUS for the CD-A. The reader is referred to [3], for more details.

4.1.1 LOTUS CD-A computer codes

The codes used in the core design calculations for the STP power pant are RELAP5-3D, PHISICS and HELIOS-2.

RELAP5-3D [10] is a simulation tool that allows users to model the behavior of the reactor coolant system and the core for various operational transients and postulated accidents that might occur in a nuclear reactor. RELAP5-3D (Reactor Excursion and Leak Analysis Program) can be used for reactor safety analysis, reactor design, simulator training of operators, and as an educational tool by universities. RELAP5-3D is developed and maintained at the Idaho National Laboratory. It is able to model the behavior of the plant system (heat exchangers, steam generators, pumps, valves, etc.) and the thermal-hydraulics of the reactor core. The code was specifically designed for simulations of light water reactor transients such as loss of coolant (LOCA), anticipated transients without scram, and operational transients such as loss of feed-water, etc.

The PHISICS (Parallel and Highly Innovative Simulation for the INL Code System) code toolkit is being developed at the Idaho National Laboratory [11,12]. This package is intended to provide a modern analysis tool for reactor physics investigations. It is designed to maximize the accuracy for a given availability of computational resources and to give state-of-the-art tools to the nuclear engineer. Several different algorithms and meshing approaches are implemented among which the user can choose in order to optimize his computational resources and accuracy needs. The software is completely modular in order to simplify independent development of modules and maintenance by different teams. The different modules currently available in the PHISICS package are a nodal and semi-structured transport core solver (INSTANT), a depletion module (MRTAU), a time-dependent solver (TimeIntegrator), a cross section interpolation and manipulation framework (MIXER), a criticality search module (CRITICALITY) and a fuel management and shuffling component (SHUFFLE). PHISICS can be run in parallel to take advantage of multiple computer cores (10 to 100 cores). The package is coupled to the system safety analysis code RELAP5-3D and can be activated as an alternative to the NESTLE code [13] that is also integrated in RELAP5-3D.

The software HELIOS-2 (Studsvik ScandPower) is known to be one of the most popular lattice codes [14]. Lattice codes are used to calculate the neutron flux distribution over a user-defined 2D region of the reactor. This region can range from a fraction of an assembly to the full core. The lattice geometry can be input with great detail, i.e. fuel pins including gaps and cladding, control rods, burnable absorbers, etc. can be modeled explicitly. The resulting 2D transport solution of the flux is usually used to generate cross section libraries for the use in subsequent 3D core calculations. These cross section libraries can be generated with the desired number of energy groups, as well as for different combination of rector state variables (e.g. fuel temperature, moderator density, control rod insertion, burn-up, etc.) in which the subsequent 3D core simulator can interpolate.

4.1.2 LOTUS CD-A methodology

The LOTUS methodology for the CD-A is shown in Figure 13 [15]. The methodology as shown in the picture focuses on analyzing the equilibrium cycle of a given plant, but if specific cycle information is available, the methodology can be used to analyze any given cycle in the plant. The calculations to provide initial data for the LOCA analysis are done with the above-described PHISICS, RELAP5-3D and HELIOS-2 codes, including cross section generation.

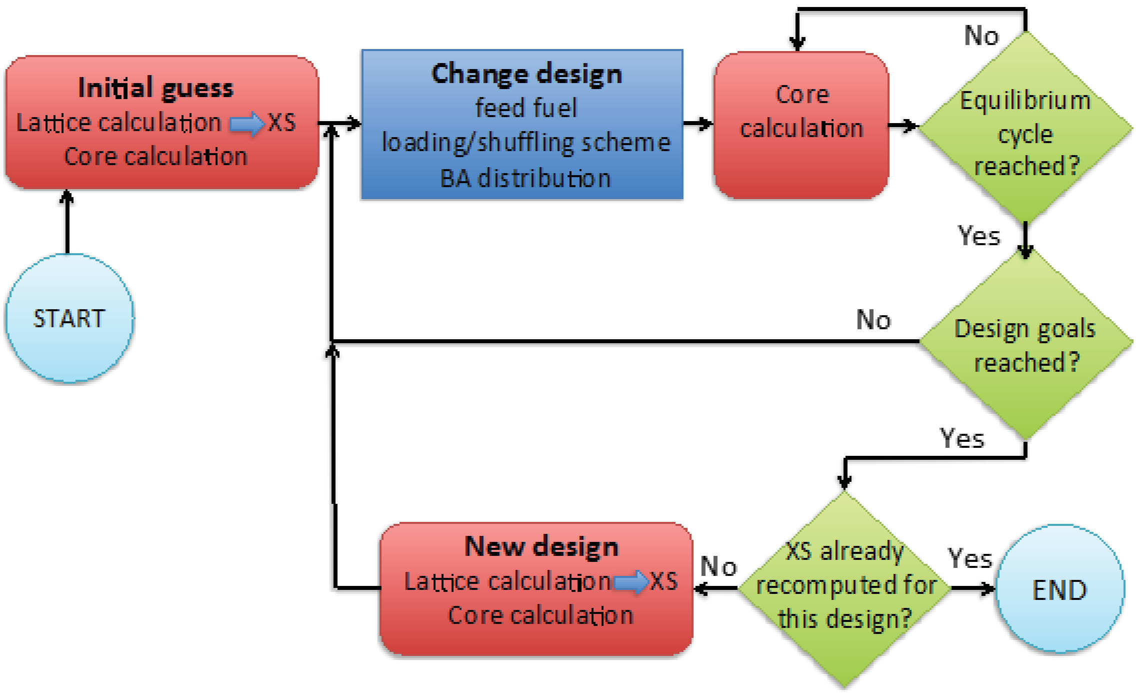


Figure 13. Industry Application ECCS/LOCA Demonstration of a PWR Design Strategy.

The first step of the strategy is to generate homogenized neutron cross sections. HELIOS-2 computes the cross sections for different geometrical conditions and different reactor states in the core. In this manner, a cross section library is generated that captures effects like control rods, burnable poisons, etc. as well as different fuel temperatures, moderator densities, boron concentrations and burnup levels for a given reactor design. Traditionally, the two steps in the core analysis, i.e. cross section generation and subsequent 3D core calculation are performed independently. This means that cross sections for an a-priori known core design are calculated and then passed to the reactor calculation. During the reactor calculation studies, the cross section libraries are not changed anymore, because the geometry and fuel compositions normally do not change anymore. In the case of core design studies, where the core design is not known a-priory, cross section generation and core calculation need to be iterated. This is also true if only an incomplete set of design values is available for example because part of the design is proprietary. Every time during the design process when the core/assembly geometry or fuel are changed, a new cross section library needs to be calculated. A suite of tools has been developed to assist the user to automatize the generation of cross sections to be used with PHISICS/RELAP5-3D as well as to pre- and post-process results. In order to automatically re-compute cross sections when needed, for example after a fuel reloading simulated by PHISICS, a generic interface for lattice codes has been developed for the PHISICS code package [3]. These tools include:

- a "material density feedback to the lattice code tool" that transfers all needed information from PHISCS to HELIOS-2,
- a "library assembly tool" that post processes cross sections from HELIOS-2 to be used with PHISICS,

- a “SPH tool” that implements the super-homogenization method (SPH) to assure consistency between the lattice code and PHISICS calculation since the PHISICS code does not support discontinuity factors yet,
- and a “pin power reconstruction tool” that collects pin power distributions from the lattice calculation. These can then be used to reconstruct pin powers in the 3D PHISICS core calculation results.

The PHISICS reactor physics package coupled to the thermal-hydraulic system code RELAP5-3D is used in the second step, in order to compute 3D assembly power distributions, burnups, etc. needed as initial conditions for the subsequent LOCA analysis. Depending on the available data base to initiate the calculation, (core and fuel geometry description, burnup maps, reloading pattern, power distributions, etc.), the PHISICS package can, in addition to solve the 3D core, also burn the core to the desired burnup level, shuffle and reload the core and search for critical control rod positions or boron concentrations.

At least 8 cycles are computed until the equilibrium cycle is reached. The equilibrium cycle is analyzed in terms of desired cycle length, maximum assembly burnup as well as radial and axial power distributions. Especially if not all design information of the core is available, the design is adjusted to meet the design goals and the equilibrium cycle is recomputed. Once the design goals are reached, cross sections are recomputed with HELIOS-2 for the new design. The equilibrium cycle is then recomputed with the new cross sections and is reanalyzed. Since the equilibrium cycle characteristics may have changed due to the updated cross sections, further optimization and changes in the core design may become necessary. These “inner” (change core design and recomputed equilibrium cycle) and “outer” (recomputed cross sections) loops as shown in Figure 13 [15] are repeated until convergence is reached.

As an example, assume that the core designer wants to find the fuel enrichment needed to obtain a certain cycle length. The cycle length is his design goal and the fuel enrichment is the parameter he can change in his core design. The designer has to iterate the schema in Figure 13 until convergence, i.e. he will search for the fuel enrichment (leading to the desired cycle length) using a certain cross section library. Once he finds the fuel enrichment that gives him the desired cycle length, he will re-compute the cross sections for this fuel enrichment and redo the core calculation. Using the new cross sections, he adjusts the enrichment again until he finds the desired cycle length. When the core calculations with the new computed cross sections and the cross section from the previous iteration lead to the same cycle length, the iteration process ends. The final cross section library including all dependencies like fuel temperature, moderator density, boron concentration, control rod insertion, burnup etc. can then be used in subsequent core analyses for the particular design.

4.2 Fuels Performance

Fuel rod design information is necessary for reactor core design, fuel performance, and reactor system analysis codes as input parameters. To ensure data consistency, a common set of fuel rod design input data should be shared among all the codes. Table 1 shows several common fuel rod design input data needed for different codes. Note that these fuel rod data may vary with time and have to be updated under different conditions.

For baseline calculation purpose, we choose FRAPCON [16] code for our fuel performance steady state calculations. Power history and axial power profile from the output of core design codes such as PHISICS are required as FRAPCON inputs. The linear heat generation rate (kW/m) at each time step of FRAPCON simulation should also be provided.

Table 1. Common Data from Fuel Rod Design for Different Physics in LOCA Analysis.

Fuel Rod Data	**Fuel Performance**	**Core Design**	**System Code**
Rod geometry information such as cladding outer diameter, cladding thickness, fabricated gap, active fuel length, and plenum length	☑	☑	☑
Spring dimensions such as outer diameter of plenum spring, diameter of the plenum spring wire, and number of turns in the plenum spring	☑	☑	☐
Pellet shape such as height (length) of each pellet, height (depth) of pellet dish, pellet end-dish shoulder width, Chamfer height and width	☑	☐	☐
Pellet isotopics such as fuel pellet U-235 enrichment, oxygen-to-metal atomic ratio, weight fraction of gadolinia in urania-gadolinia fuel pellets, Boron-10 enrichment in ZrB2, parts per million by weight of moisture in the as-fabricated pellets, and parts per million by weight of nitrogen in the as-fabricated pellets	☑	☑	☐
Pellet fabrication such as as-fabricated apparent fuel density, open porosity fraction for pellets, the fuel pellet surface arithmetic mean roughness, etc.	☑	☑	☑
Cladding fabrication such as cladding type, the cladding surface arithmetic mean roughness, as-fabricated hydrogen in cladding, etc.	☑	☑	☑
Rod fill conditions such as initial fill gas pressure, Initial fill gas type and their mole fractions	☑	☑	☑
Fuel assembly geometry such as pitch	☑	☑	☑

System code RELAP5-3D already has some simple fuel performance models such as the rupture model and ballooning model, but it could not provide detailed analysis of fuel rods' behaviors such as the fission gas released, rod internal pressure, and fuel-cladding mechanical interaction, etc., which requires the simulation from fuel performance codes.

The most important issue when coupling the system code and fuel performance code is to make sure that the stored energy in the fuel pin for the RELAP5 steady state result equals to the stored energy calculated by the fuel performance code. The stored energy in the fuel rod is calculated by summing the energy of each pellet ring calculated at the ring temperature. The expression for stored energy is

$$E_s = \frac{\sum_{i=1}^{I} m_i \int_{298K}^{T_i} C_p(T)dT}{m} \tag{1}$$

where m_i is mass of ring segment i, T_i is temperature of ring segment i, $C_p(T)$ is specific heat evaluated at temperature T, m is total mass of the axial node, I is the number of annular rings. The stored energy is calculated for each axial node.

The fuel performance codes were developed only for single fuel rod calculations so that they are not capable of capturing the detailed TH conditions due to the impact from neighboring fuel rods and assemblies. System codes have best-estimate two phase flow and heat transfer models which can provide local steady state TH data at different depletion cycle points as input for FRAPCON simulations. By using axial dependent cladding surface temperature and coolant pressure for each fuel rod of interest, more accurate results can be obtained from fuel performance codes. In the meantime, part of the outputs from FRAPCON simulations are used to prepare for the steady state run of a system LOCA model, ensuring correct stored energy and initial conditions for the transient fuel performance models.

4.3 Systems Analysis

System analysis normally starts with building a plant model with a reactor system analysis code. Texas A&M University has assisted INL to build a typical four-loop pressurized water reactor (PWR) plant model, based on the STP plant with 3850 MW rated thermal power, for analysis with RELAP5-3D. The accident scenario selected is a LB-LOCA with a double-ended guillotine break in a cold leg. The plant model is documented in INL/LTD-17-41482.

The safety criteria are the generic acceptance criteria for the peak clad temperature and the maximum oxidation rate (as shown in Figure 1) proposed in the rulemaking. Since both PCT and ECR limits are burnup-dependent, this added complexity requires defining new safety metrics that would synthesize PCT and ECR with fuel rod dependent cladding pre-transient hydrogen content. The safety metrics are defined as the ratios of the calculated PCT over PCT limits for each fuel rod, as well as the ratios of the calculated ECR over ECR limits for each fuel rod and are expressed in the following:

$$PCTR = \frac{PCT^{Calculated}}{PCT^{Limit}} \tag{2}$$

$$ECRR = \frac{ECR^{Calculated}}{ECR^{Limit}} \tag{3}$$

If we define PCTRmax and ECRRmax as the maximum value of PCTR and the maximum value of ECRR, respectively, the acceptance criteria for the safety metrics are the following:

1) $PCTR_{max} < 1.0$

and

2) $ECRR_{max} < 1.0$

Using the above criteria, the limiting fuel rods can be identified as the fuel rods with PCTRmax or the ECRRmax.

The reactor core modeling in RELAP5-3D used different homogenization approaches for thermal fluid dynamics calculations than for the heat conduction and clad oxidation calculations in the fuel rods. A multiple channel approach was used for the thermal fluid dynamics calculation, as illustrated in Figure 14. Specifically, the assemblies in the core were grouped into various regions based on their burnup history. The assemblies with fresh fuel, once burned fuel and twice burned fuel were grouped together respectively. Two flow channels – one average channel and one hot channel – were built to represent each group of assemblies. Hence there are a total of six flow channels in this study. The flow channels are connected in the lateral direction to allow crossflow to be calculated. Crossflow is modeled at each axial elevation in the core between the three average core channels. It is also modeled at each axial elevation between the hot channels and the adjacent average channels. This allows flow to be redistributed around a blockage caused by cladding ballooning or rupture. The crossflow area is based on the minimum gap between the fuel rods along one side of a fuel assembly and the number of fuel assembly sides at the interface between the three average core channels; for example, for the hot assembly in each region, there are four sides at the interface. Loss coefficients are approximated based on flow across in-line and staggered rows of tubes, with the average distance of travel estimated to be about half an assembly width.

For heat conduction and clad oxidation calculations, it is computationally prohibitive to consider all the fuel rods in the reactor core. Instead a homogenization technique is used to reduce the number of fuel rods to be simulated. Two sets of heat structures were used for each assembly – one set represents the highest power rod or the hot rod in the assembly and the other set represents the average of the remaining fuel rods in the assembly. This is a reasonable approximation given that the fuel rod burnup normally does not vary too much within a PWR assembly and the hot rod in an assembly would be the limiting rod for that assembly.

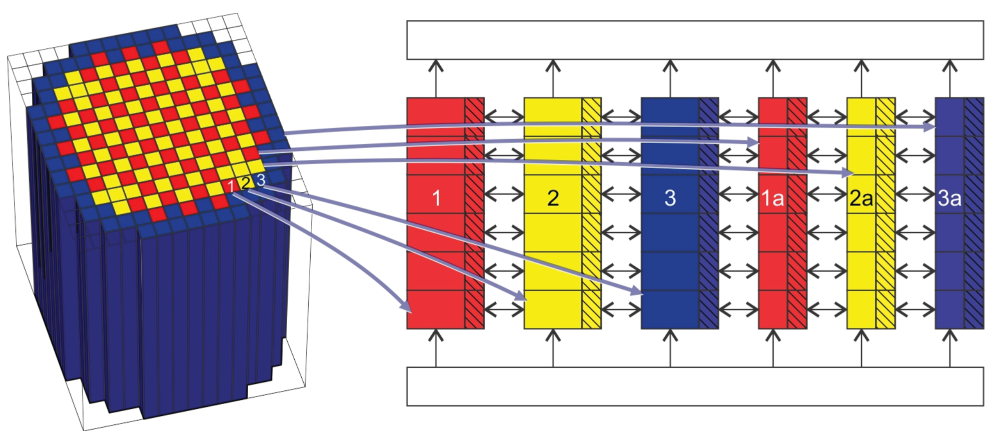


Figure 14. Schematic Illustration of the Mapping between the Core Design Analysis and the RELAP5-3D Analysis Core Model for the Generic PWR Model Based on STP.

As a result, heat structures for the highest power assembly (hot assembly) and its hot rod in each group of assemblies were built and attached to the hot channel, as shown schematically in Figure 15 such that the PCT and ECR in the average rods and hot rod can be calculated. Analogously, the heat structures for the other assemblies and their respective hot rods were built and connected to the average channel, as shown in Figure 16, such that the PCT and ECR can be calculated for the average rods and hot rod in each assembly. Therefore, there are a total of 386 sets of heat structures for the fuel in this study (193 for assemblies plus 193 for hot rods). It is noted that the hot rod power has been subtracted from each assembly to yield the correct power for the average fuel rods in each assembly such that the reactor total power is conserved.

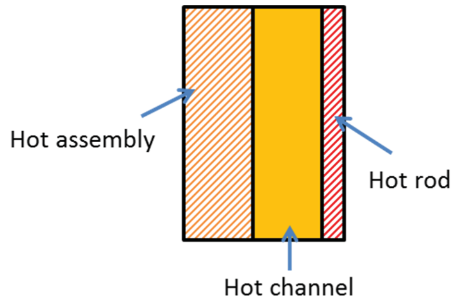


Figure 15. Schematic Illustration of the Heat Structure Mapping for the Hot Assembly and Its Hot Rod with the Hot Channel (One for Each Group of Assemblies).

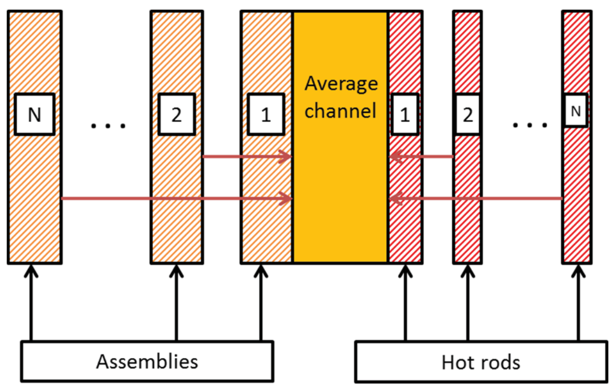


Figure 16. Schematic Illustration of the Heat Structure Mapping for Average Assemblies and their Respective Hot Rods with the Average Flow Channel.

The plant model built based on the STP plant is the reference system to be simulated in this work. All the major flow paths for both primary and secondary systems are described, including the main steam and feed systems. The emergency core cooling system (ECCS) was included in the modeling of the primary side, and the auxiliary feedwater system was included in the secondary side modeling.

The Large Break LOCA scenario considered in this analysis is initiated by a large break of one of the cold legs (Figure 17). The cold leg is typically considered as the most limiting location as it limits the ECCS injection in the cold leg, it promotes flow stagnation in the core, and it cause ECCS injection bypass. The transient is characterized by three distinct periods: blow-down, refill, and reflood. The scenario is described for PWRs equipped with U-tube steam generators.

The blow-down period extends from the initiation of the break until the primary side depressurizes sufficiently that emergency core cooling (ECC) water can start to penetrate the downcomer (20-30 seconds into the transient). The flow out of the break is large, but limited by critical flow phenomena. No control rod insertion is credited in the event. The boiling and flashing, which occurs in the core as result of the rapid depressurization, is sufficient to shut down the fission process due to negative reactivity feedbacks.

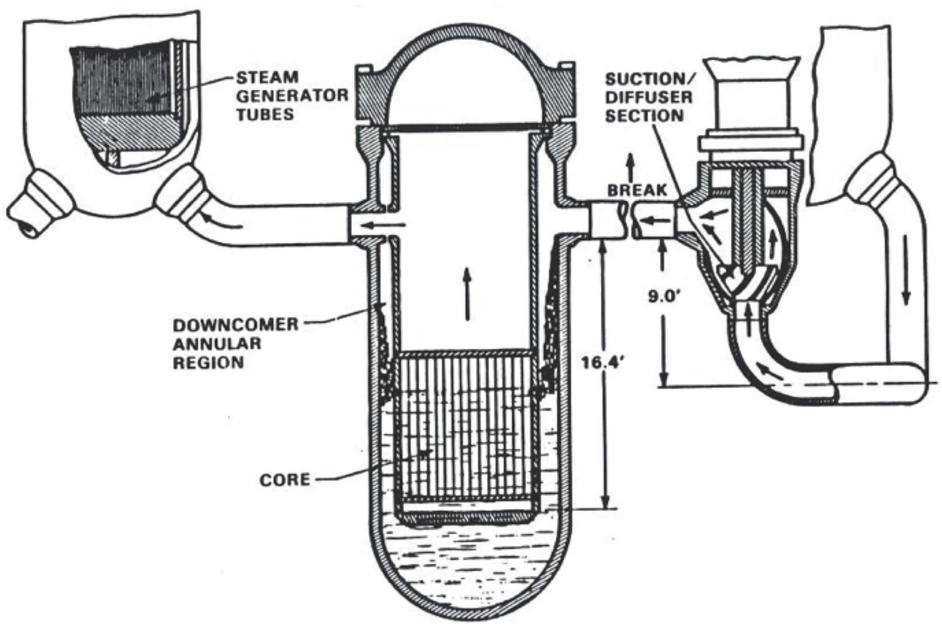


Figure 17. Schematic of Double Ended Guillotine Break.

The reactor coolant pumps (RCPs) are assumed to be either functioning or coasting down depending on the off-site power availability assumption. Even if the RCPs are running, the performance will be degraded by the void. The break flow will eventually reverse the flow in the core and a stagnation situation can be reached in the core. The hot fuel rods quickly exceed the critical heat flux, as the core flow reverses, resulting in a sharp reduction in heat transfer to the coolant. As the pressure decreases, the reversed flow induced by the break diminishes and positive core flow can be reestablished.

During the blow-down phase, the cladding temperature first rises rapidly as the initial stored energy in the fuel pellets is transferred to the cladding. After the initial heat-up, the cladding temperature will decrease due to the down flow of high velocity steam through the core. The lower power regions in the core may even quench during this blow-down cooling phase.

Between 10 to 20 seconds after the break, the RCS pressure decreases below the accumulator pressure. The accumulators begin injecting cold water into the cold legs, but the initial injection is swept out of the vessel and into the broken cold leg by the continuing high flow of steam from the core. This is called ECCS bypass.

Approximately 20 to 30 seconds after the break, the RCS pressure approaches the containment pressure and break flow becomes un-choked. This initiates the refill phase. The ECCS water from the accumulators and the pumped safety injection refill the lower plenum and establishes a water level in the downcomer. As the coolant enters the core, the reflooding process begins.

The reflood of the core is gravity driven. The water head in the downcomer and the backpressure in the upper plenum determine the core-flooding rate. The flow into the core is initially oscillatory, as cold water rewets the hot fuel rods, generating steam, which in turn creates a local pressurization in the core. This, in turn creates a feedback mechanism and a manometric effect between the level in the downcomer and the level in the core.

As the core gradually quenches, steam is generated as the fuel rod dump their stored energy in the liquid and droplets entrained in the steam. This steam, and the water it entrains flows through the vessel upper plenum, the broken loop hot leg, the steam generator, and the pump before it can be vented out the break. Water that condensed in the steam generators cannot flow back into the core due to the counter currant flow limitation (CCFL) that exist in different places in the system, especially the top of the core and the entrance to the steam generators.

As the accumulators empty nitrogen gas is discharged in the system, the loop and the downcomer steam condensation is suppressed. This creates a temporary pressurization in the downcomer and core liquid in-surge during reflood.

Core reflood is a relatively slow process. As the bottom elevations are quenched, the top elevations continue to gradually heat up and eventually turnaround once the cooling rate exceeds the decay heat rate. The cooling mechanism is controlled by dispersed flow film boiling where droplets act as the ultimate heat sink to de-superheat steam at higher elevation. The reflood transient may last for several minutes.

4.4 Risk Assessment

In this subsection the uncertainty propagation and risk assessment techniques employed by LOTUS to compare statistical analysis results with regulatory limits are considered.

4.4.1 Uncertainty propagation and risk assessment

Traditional approaches to NPP operation analysis and safety margin involve strong conservatism and sometimes unphysical assumptions. BEPU analysis seeks to refine that analysis to return margin to the operator and better characterize plant operation. Key to the BEPU approach in margin characterization is propagation of input uncertainties to statistics of the figures of merit. Traditional approaches to quantifying uncertainty for NPP include the Wilks method and the more costly but arguably more reliable Monte Carlo method. Studies have suggested that while Wilks's method requires relatively few samples to produce statistics, the fluctuation of those statistics is significant; Monte Carlo, on the other hand, requires many samples to produce reliable statistics, but does so in a consistent and reliable manner [17].

Ultimately, uncertainty propagation produces the likely values and dispersion of response figures of merit. Likely value is given by the expectation value (or mean) of a sample set, and dispersion is demonstrated by statistics such as the standard deviation (or sigma), variance, and 5^{th} or 95^{th} percentiles. In NPP margin characterization, the 95^{th} percentile is used to compare to operational limits such as those for PCT and ECR.

In order to perform BEPU analysis, the important phenomena affecting the progression of the LB-LOCA accident are first determined by the phenomena identification and ranking table process. A large number of studies have been done previously to identify the important phenomena. A PIRT analysis has been conducted in this work. For demonstration purposes, a reduced set of parameters with high importance to LB-LOCA has been selected and is shown in Table 2.

Table 2. Distribution of Parameter Uncertainties.

Parameter	**PDF type**	**Min**	**Max**	**Comments**
Reactor thermal power	Uniform	1.0	1.02	Multiplier
Reactor decay heat power multiplier	Uniform	0.94	1.06	Multiplier
Accumulator pressure (psia)	Normal	0.9	1.1	Multiplier
Accumulator liquid volume (ft^3/m^3)	Uniform	-10/-0.28	10/0.28	Additive
Accumulator temperature (F/K)	Uniform	-20/-11.11	30/16.67	Additive
Subcooled multiplier for discharge critical flow	Uniform	0.8	1.2	Multiplier
Two-phase multiplier for discharge critical flow	Uniform	0.8	1.2	Multiplier
Superheated vapor multiplier for discharge critical flow	Uniform	0.8	1.2	Multiplier
Fuel thermal conductivity	Normal	0.93	1.07	Multiplier
Average core coolant temperature (F/K)	Normal	-3/-1.67	3/1.67	Additive
Turbulent forced convection heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Nucleate boiling heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Multiplier on Critical Heat Flux (CHF)	Uniform	0.7	1.3	Multiplier
Multiplier on transition boiling heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Film boiling heat transfer coefficient	Uniform	0.7	1.3	Multiplier
Fuel rod gap width	Uniform	0.2	0.8	Multiplier

The uncertainty quantification of the LB-LOCA analysis is carried out using the Monte Carlo approach to determine the 95/95 upper tolerance limits.

Since a LOCA event is equal-probable in time, the time in a cycle is an additional random variable whose uncertainty is propagated through the analysis in a typical LB-LOCA analysis conducted in the current practice. Including the time in a cycle as a random variable would require LOCA calculations to be carried out on a very large number of exposure points, which is not practical. As a result, in our demonstration calculations we selected a few specific exposure points in a cycle and then propagated all the uncertainties for each exposure point.

The exposure points selected for the LB-LOCA calculations cover the entire range of the cycle length. The selected exposure points are at the beginning of cycle (BOC), 100 days, 200 days, 300 days, and end of cycle (EOC). This way, the dynamic response of the plant with regards to LB-LOCA transients would be fully characterized at different core conditions during the entire cycle.

A set of 1000 RELAP5-3D input files has been prepared respectively at each of the five selected exposure points by randomly perturbing the input parameters using their associated probability density functions defined in Table 2. All the RELAP5-3D cases were run to steady state first. Large break LOCA cases were then initiated by assuming a double-ended guillotine break. Following the initiation of the LB-LOCA, a fast depressurization of the primary system ensues. The ECCS is activated to provide emergency cooling water to the core. The entire process lasts about 5 minutes. To be conservative, in our RELAP5-3D plant model simulations, the shutdown of the reactor following the initiation of LB-LOCA is achieved through the negative reactivity feedback, rather than through the scram of the reactor. In our LB-LOCA runs, it is assumed that only one out of the three ECCS systems are functioning and able to inject water into the reactor core. However, as passive components, it is assumed that all three accumulators in the intact loops are functioning and able to inject water into the reactor core.

The LOTUS toolkit automatically samples each uncertain parameter shown in Table 2 from its distribution. For a uniform distribution, the minimum and maximum values are the boundaries of the sampling. For a normal distribution, the sampling boundaries were truncated at the minimum and maximum values, which is effectively a truncated normal distribution. No dependencies between parameters were considered in the sampling. The LOTUS toolkit then modifies the RELAP5-3D input files according to the perturbed values. It automatically drives the desired number of RELAP5-3D runs on Idaho National Laboratory's high performance computers (HPC). The toolkit also performs the postprocessing of the RELAP5-3D output files and presents the *PCTRmax* and *ECRRmax* values according to the Monte Carlo approach.

4.4.2 Sensitivity analysis

One obstacle to some uncertainty propagation techniques is the dimensionality of the uncertain input space. As the number of uncertain inputs grows, the number of samples required to represent that space accurately grows exponentially. To help alleviate this problem, global sensitivity analysis can be employed. Global sensitivity analysis methods explore the whole input parameter space by sampling chosen input parameters simultaneously rather than performing perturbations of input parameters one-at-a-time. Global sensitivity analysis has the advantage of being able to identify nonlinear uncertainty structures over the global admissible input parameter space. The non-influential parameters in nonlinearly parameterized models can be fixed for subsequent model calibration or uncertainty propagation. In global sensitivity analysis, the effect of perturbing an input on the moments of a response is quantified. Often, a response is much more sensitive to some inputs than others. In some cases, no responses are sensitive to perturbations of a particular input. If this is discovered, the uncertainty in that parameter can be ignored without negatively impacting the BEPU analysis. There exist numerous sensitivity analysis methods that should be carefully chosen based on the complexity and specific model to be evaluated. In this work, a Monte Carlo, or sampling based, approach is used to evaluate those parameters that most profoundly affect the figures of merits. The same set of LB-LOCA simulations used for uncertainty quantification is also used for sensitivity analysis. In this report, we focus on the variance based methods of Pearson and Sobol, and density based Delta Moment Independent measures [18].

4.4.2.1 Pearson correlation coefficients

Pearson correlation coefficients may be thought of as a normalized covariance of an output and input, and are found by Equation 4,

$$R_i^2 = \frac{cov(X_i,Y)^2}{var(X_i)var(Y)} \tag{4}$$

The sampling is inherently simpler than higher order methods, with no need to isolate variables or partition the data as is required for both Sobol and Delta measures [19]. However, for nonlinear relations, Pearson measures may indicate no correlation between variables that are strongly interconnected. For instance the simple model $Y = X^2$, with a random sampling in $X \in (-1,1)$, has a Pearson correlation coefficient value that approaches zero as the sample size approaches infinity.

The Pearson measures do allow for an assessment of the linearity of the system. For a purely linear system, the sum of all Pearson values ($\sum_{i=1}^{N} R_i^2$) is unity, with increases in nonlinearity resulting in a sum closer to zero.

4.4.2.2 Sobol indices

The Sobol variance decomposition method entails comparing the contribution of one input to the variance of an output. Sobol indices differ from Pearson correlation coefficients in that Pearson measures are based upon linear regression, while Sobol indices capture more complex interactions. Here only the first order terms are presented. Sobol indices are expressed mathematically in Equation 5,

$$S_i = \frac{var\left(E_{X_i}\left(Y_{Y|X_i}\right)\right)}{var(Y)} \tag{5}$$

where S_i is the Sobol indices, and $E_{X_i}()$ is the expected value operator, expanded below in Equation 6,

$$E_{X_i}\left(var\left(Y_{Y|X_i}\right)\right) = \int f_{X_i}(x_i)\, var\left(Y_{Y|X_i}\right) dx_i \tag{6}$$

Sobol indices are computationally expense. The sum of Sobol indices lies between zero (non-additive) and unity (additive). It is important to note that a sum of unity does not necessarily indicate linearity. If the definitions in Equations 5 and 6 are strictly adhered to, a double loop Monte Carlo method is required, as opposed to the more random sampling typically used for Pearson measures. Plinkes method allows for the partitioning of a typical random distribution into approximately equally spaced partitions based upon the rankings of a given input [19]. Estimators for the Sobol indices can be recast as Equation 7,

$$\widehat{S_i} = \frac{\frac{1}{N}\sum_{j=1}^{M} N_j var\left(Y_{Y|x_i \in X_j}\right)}{var(Y)} \tag{7}$$

where X_j represents a partition of the sample with a population of N_j, M is the number of partitions, and N the total sample size. In order to reduce bias, the boot strapping method is used with 10 resamples with replacement. Sobol indices were calculated using the SA Library in Python [20].

4.4.2.3 Delta moment independent measures

The delta moment independent measures are a recent metric established by Borogonovo [18]. The delta measure is based upon the expected L[1] norm differences between conditional and unconditional probability density functions for a given input. This is expressed mathematically in Equations 8 and 9,

$$s(X_i) = \int \left| f_Y(y) - f_{Y|X_i}(y) \right| dy \tag{8}$$

$$\delta_i = \frac{1}{2} E_{X_i}\big(s(X_i)\big) = \frac{1}{2} \int f_{X_i}(x_i) s(x_i)\, dx_i \tag{9}$$

where $s(X_i)$ is the L[1] norm between the unconditional density $f_Y(y)$ and the conditional density $f_{Y|X_i}(y)$, $E_{X_i}()$ is the expected value operator, and δ_i is the delta moment independent measure. The delta measure is advantageous in that it is density based. As a result, complex relations that effect distribution but not necessarily variance, are captured [18]. The summation of all delta indices is between zero and unity. Sums close to unity indicate that the contributions from a group of inputs on an output are separable from each other, while lower sums indicate the shifts in distribution are inseparable.

As with the Sobol method, Delta measures ideally use a double loop Monte Carlo method. The partitioning strategy [18] presented in the above Sobol Indices is used to recast Equations 8 and 9 as Equations 10 and 11,

$$s\big(x_i \in X_j\big) = \int \left| f_Y(y) - f_{Y|x_i \in X_j}(y) \right| dy \tag{10}$$

$$\widehat{\delta_i} = \frac{1}{2N} \sum_{j=1}^{m} N_j s\big(x_i \in X_j\big) \tag{11}$$

The integral used in Equation 10 is estimated via kernel density estimators. While a variety of kernels are available, this work uses the more common Gaussian kernels, with future work planned to implement the more recent diffusion kernels. Bootstrapping methods are used for delta indices as well, with 10 resamples with replacement. Delta measures were calculated using the SA library in Python [20].

5. STP ANALYSIS RESULTS

This section presents the simulation results performed by using LOTUS for the generic PWR model based on STP.

5.1 Core Design Automation

The LOTUS core design methodology and the lattice code interface between RELAP5-3D/PHISICS and HELIOS-2 have been applied to the STP core design. The PHISICS/RELAP5-3D core simulator has been used to compute assembly-homogenized quantities (burnup maps, assembly power peaking maps, power history, etc.) together with pin power reconstruction (from HELIOS-2) for STP. This is an acceptable current state-of-the-art methodology to provide input data for safety analysis, like LOCAs.

5.1.1 Input data for core design calculations

To develop the different input models (HELIOS, PHISICS and RELAP5-3D) for the STP core design, publicly available data from the STP FSAR rev 18 [9] has been used. Data that is not publicly available in the FSAR, has either been taken from another public source [3] or assumed according to best engineering judgment. Table 3 through Table 7 show the core characteristics, the cycle characteristics, the fuel assembly characteristics, the fuel rod characteristics as well as the reactor coolant system characteristics assembled from the STP FSAR. The third row of these tables shows where in the STP FSAR the data can be found.

The assembly design is shown in Figure 18. It's a 17 x 17 assembly with 264 fuel rods and 25 non-fuel locations. All non-fuel locations are guide or instrument tubes. The IFBA rods are normal fuel rods that have a boron absorber coating sprayed on the cladding. Three assemblies are shown in Figure 18, one with 64, 104 and 128 IFBA pins. Figure 19 shows an axial schematic of a fuel rod. The fuel rods contain a low enriched zone at the top and bottom. From the fuel assembly characteristics in Table 5, it can be seen that there is no geometrical data, nor enrichment available for the low enriched annular pins that form the blanket at the top and bottom of the fuel pin. It has been assumed, that the annular pellets are solid. The enrichment has been taken to be 2.6% as indicated in [3]. The IFBA coating does not extend to the blanket regions, i.e. the burnable absorber is only sprayed on the high-enriched center part of the fuel rods.

Table 3. STP Core Characteristics.

Core characteristics	**STP UFSAR**	**REF**
Thermal Power	3853 MWth	Table 4.1-1
Heat deposited in Fuel	97.40%	Table 4.1-1
Number of assemblies in core	193	Table 4.1-1
Active core height	168 in (426.72 cm)	Table 4.1-1

Table 4. STP Cycle Characteristics.

Cycle characteristics	**STP UFSAR**	**REF**
Cycle length	18 month	Chap 4.3-7
Initial enrichment	3.8-4.4w% (5.0 max)	Table 4.1-1
Burn-up per cycle	17000 MWD/MTU	Chap 4.3-7

Table 5. STP Fuel Assembly Characteristics.

Fuel assembly characteristics	**STP UFSAR**	**REF**
Design	XL Robust Fuel Assembly (RFA)	Chap 4.1-1
Active fuel length	14' (426.72 cm)	Chap 4.1-1
Number of rods	264	Chap 4.1-1
Rod array	17x17	Table 4.1-1
Rod pitch	0.496in (1.25984 cm)	Table 4.1-1
Grids per assembly	11	Table 4.1-1
Axial blankets	Top/bottom using annular pellets	Chap 4.3-6
Annular pellet length in assembly	7 in (17.78 cm) top/bottom	Fig 4.2-3
Assembly pitch	8.466 in (21.50364 cm)	Fig 4.2-1
Lattice cell pitch	8.426 in (21.40204 cm)	Fig 4.2-1
Assembly water gap	0.04 in (0.1016 cm)	Fig 4.2-1

Table 6. Fuel Rod Characteristics.

Fuel rod characteristics	**STP UFSAR**	**REF**
Rod diameter	0.374 in (0.94996 cm)	Table 4.1-1
Rod diametral gap (2xgap size)	0.0065 in (0.01651 cm)	Table 4.1-1
Clad thickness	0.0225 in (0.05715 cm)	Table 4.1-1
Pellet diameter	0.3225 in (0.81915 cm)	Table 4.1-1
Helium pressure level BOL	1000 psia (68.9476 bar)	Chap 4.2-21
IFBA loading	1.57 mg/inch B10	Chap 4.3-30
Cladding material	ZIRLO/Optimized ZIRLO	Chap 4.1-1
Instrument tube wall thickness	0.02 in (0.508 mm)	Chap 4.1-1
Guide tube wall thickness	0.0185 in (0.04699 cm)	Table 4.1-1
Guide tube inner diameter	0.442 in (1.12268 cm)	Fig 4.2-7B
Guide tube outer diameter	0.482 in (1.22428 cm)	Fig 4.2-7B
Control rod diameter	0.366 in (0.92964 cm)	Table 4.3-1
Control rod cladding thickness	0.0185 in (0.04699 cm)	Table 4.3-1
Control rod active length	158.9 in (403.606 cm)	Amendment page 5-6

Table 7. Reactor Coolant System.

Reactor coolant system	**STP UFSAR**	**REF**
Pressure (nominal)	2250 psia (155.132 bar)	Table 4.1-1
Pressure (minimum)	2220 psia (153.064 bar)	Table 4.1-1
Total flow rate	145200000 lbm/hr (18.2949 t/s)	Table 4.1-1
Effective flow rate	132900000 lbm/hr (16.745t/s) 8.5% bypass	Table 4.1-1
Nominal inlet temp	549.8-560.3 F (561-566.7 K)	Table 4.1-1
Av. rise in core	69.7-71.1 F	Table 4.1-1
Core pressure drop	37.3+-4 psi (2.571 bar +-0.3)	Table 4.1-1

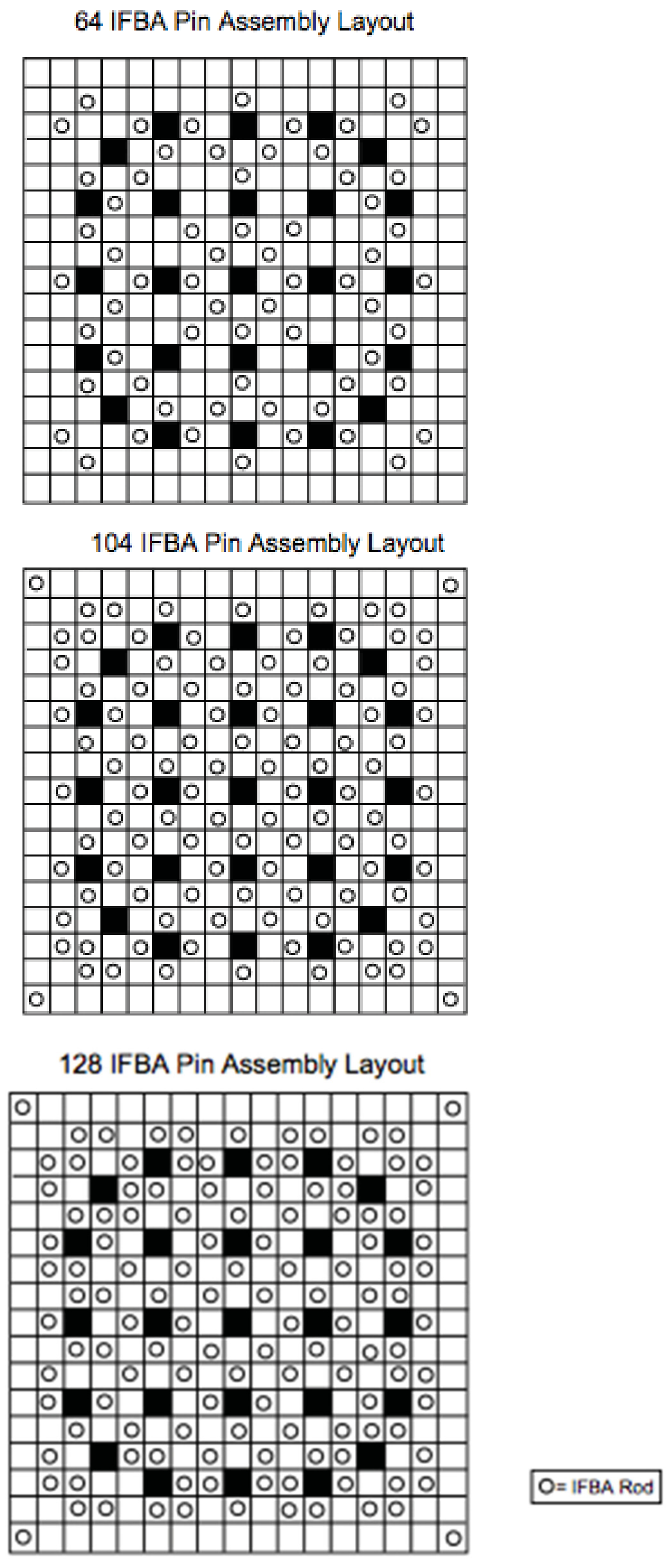


Figure 18. STP Core: 17x17 Pin Assembly. Shown are 64, 104 and 128 IFBA Rods (Circles) and 25 Guide Tubes (Black).

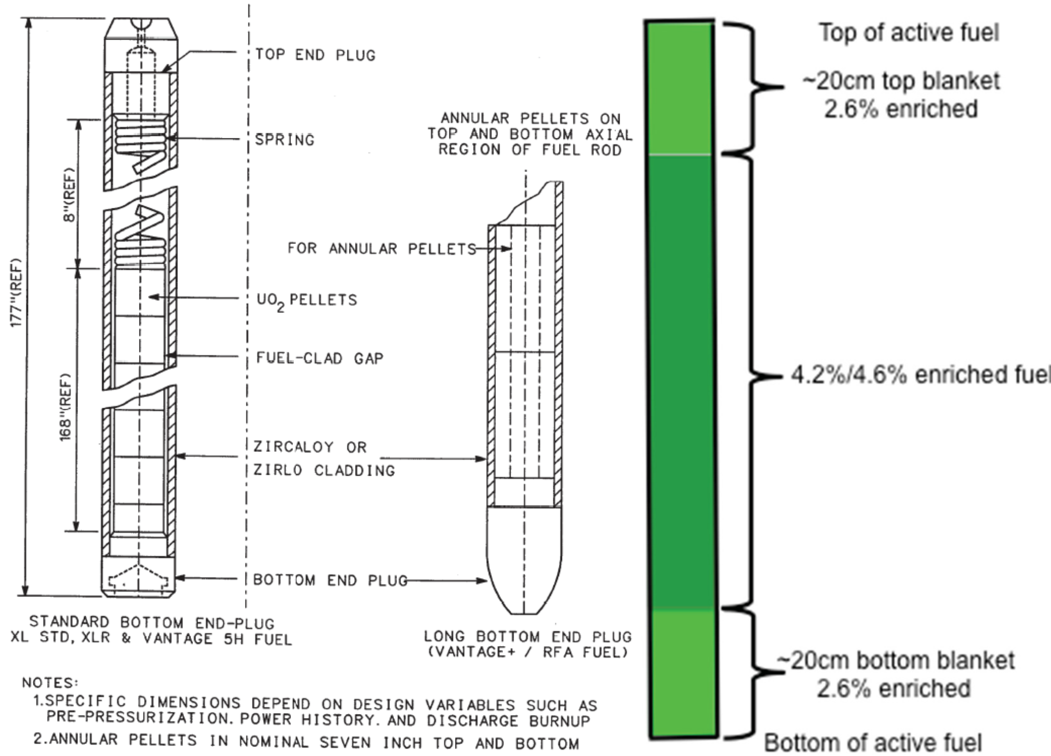


Figure 19. Left) STP Fuel Rod Schematic (This is Figure 4.2-3 in [9]); Right) Axial Fuel Pin Design: High Enriched Center Part with Top and Bottom Blankets 2.6% Enriched.

5.1.2 Cross section library calculation model

As mentioned, for the calculation of the homogenized cross sections, the lattice code HELIOS-2 has been used. A detailed 2D representation of 1/8 of the core has been modeled with HELIOS-2. Figure 20 shows the full HELIOS-2 model (including a zoom on one assembly). The IFBA coating has been modeled explicitly using a 0.15mm (minimum distance allowed in HELIOS) coating containing the 1.57mgB10/inch boron density.

HELIOS calculations, i.e. cross-section libraries at two different axial heights in the core have been generated: one in the fuel region and one in the blanket region. In addition, a library for the bottom reflector and a library for the top reflector have been generated. This leads to a total of 34 libraries (29 fuel assemblies + 1 blanket + 2 radial reflectors, one for the fuel and one for the blanket region + 2 top and bottom reflectors). The lattice calculations are generally started from pre-collapsed multi-group neutron energy structures. For the computation of the different cross sections sets, lattice calculations have been performed starting from a 44-energy group structure, then collapsed into an 8-group structure in the homogenization procedure [21]. The used energy boundaries are reported in Table 8. The reactor calculation involves the simulation of the reactor during several operational cycles and during transient/maneuver events. In order to exchange feedback between the core design tools (PHISICS) and the thermal-hydraulic code (RELAP5-3D), the microscopic cross sections sets for all isotopes except the moderator are tabulated with respect to several field parameters for each library. The cross section for the moderator regions have been tabulated as macroscopic cross sections. This allows treating the boron that is in solution in the moderator (this is a tabulation dimension) and the boron in the burnable absorbers (tabulated microscopic cross sections) separately. The cross section library tabulation dimensions and associated tabulation points are given in Table 9.

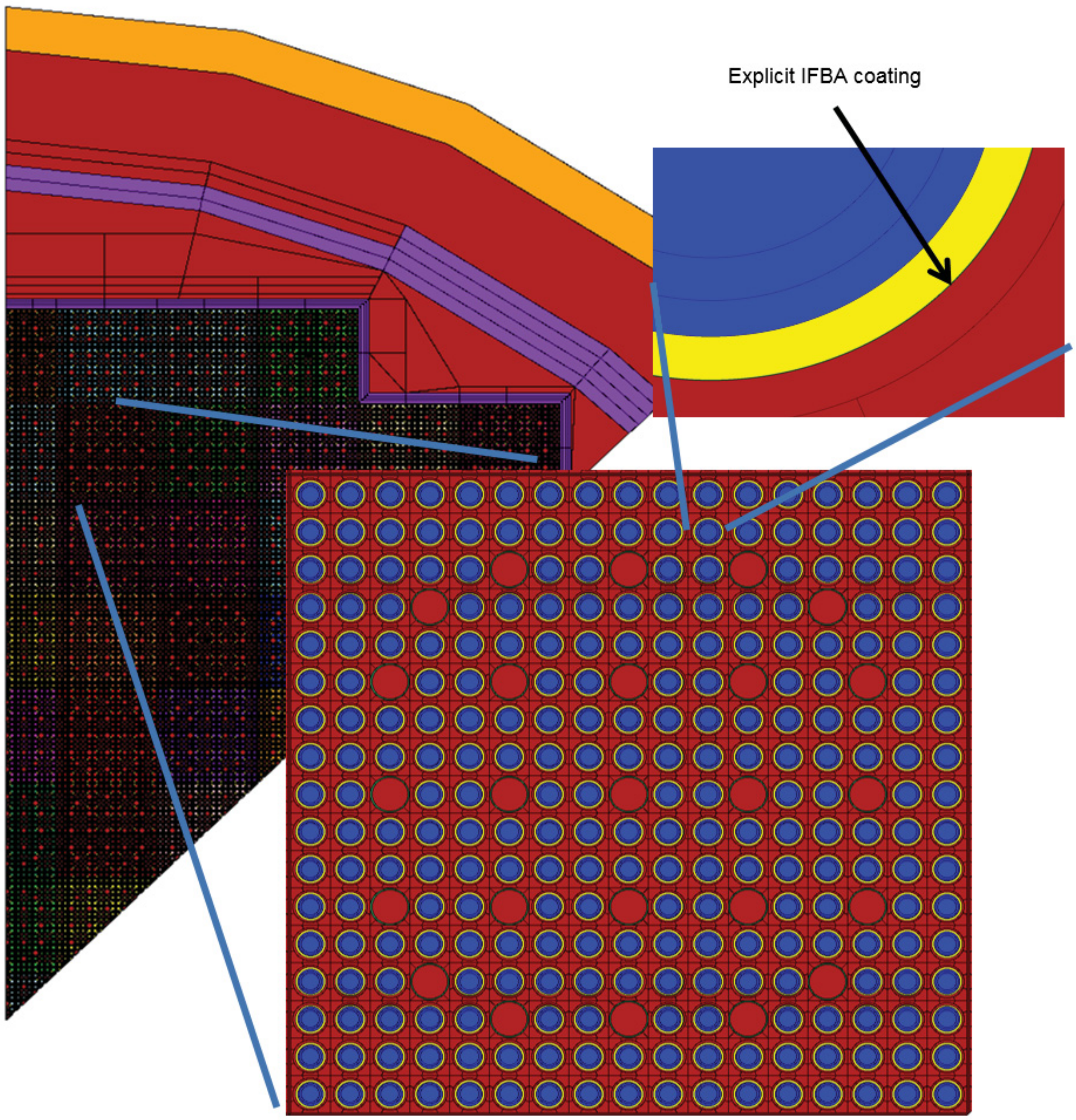


Figure 20. HELIOS-2 Model for STP.

The tabulation dimensions lead to the construction of a complete N-Dimensional (4-Dimensional in this case) grid that is characterized by 144 tabulation points in total. It should be mentioned here, that the burn-up points are for each cycle since cross section libraries are computed for fresh, once burned and twice burned assemblies. These libraries are then assembled into a library for the fuel assembly's whole life in the reactor core. The actual maximum burn-up in the combined libraries for one assembly is then the burn-up at for a twice-burned assembly at the beginning of cycle plus 25 GWd/tHM. The cross sections have been prepared burning at average core power during the cycle.

Table 8. Collapsed Energy Structure.

Group	Upper Energy bound (eV)
1	2.00E+7
2	2.23E+6
3	8.21E+5
4	9.12E+3
5	1.30E+2
6	3.93
7	6.25E-1
8	1.46E-1

Table 9. Cross Section Library Tabulation Points.

Boron concentration in H_2O (ppm):	0.0	1000	1900	
Moderator density (kg/m^3):	640.8	833.0	945.2	1000
Fuel temperature (K):	573.2	1073.2	1273.2	
Relative Burn-up (GWd/tHM):	0.0	0.152	15	25

5.1.3 PHISICS core model

The PHISICS model of this calculation is set-up as the full core, i.e. all the 193 assemblies are modeled explicitly. The materials are assembly homogenized. One ring of assemblies containing a water/steel mixture has been placed around the active core to represent the reflector. The 3D PHISICS calculation uses 18 axial layers. Each assembly is associated with its corresponding cross section library. Axially, the libraries prepared for the blanket region are associated to the axial layers containing the blanket and the "normal" fuel libraries are associated to the other axial layers. The libraries prepared to the top, bottom and radial reflectors are associated with the corresponding reflector assemblies.

Since the STP loading map was not available, a modified, publicly available, real, recent PWR plant-loading pattern has been used as shown in Figure 21. This loading pattern leads to an optimized high energy, low leakage core. (HE-LL) [3].

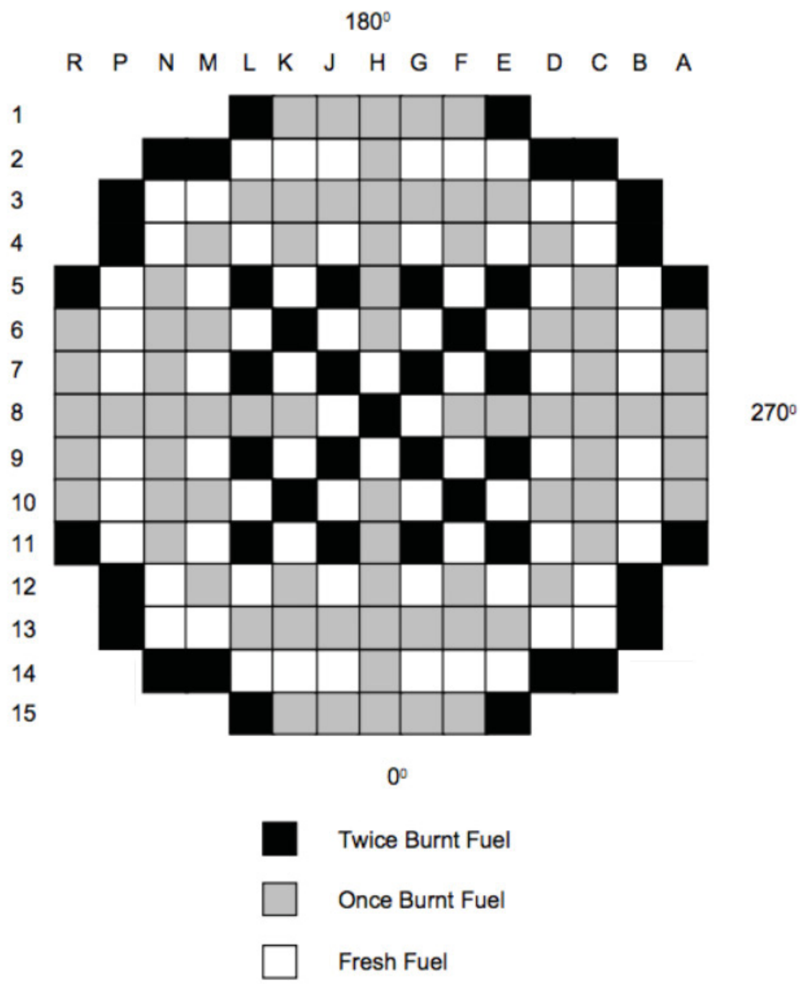


Figure 21. Assumed STP Core: Equilibrium Cycle Loading Pattern.

5.1.4 Coupled PHISICS/RELAP5-3D calculation model

The PHISICS code is coupled with the thermal-hydraulic code RELAP5-3D. For the search of the equilibrium cycle, the thermal-hydraulic model of the reactor has been set-up considering the reactor core only (without a primary or a secondary system). The approach to consider only the core region without the primary system can be taken for base irradiation calculations (like the search for the equilibrium cycle for a given core configuration), since the system does not influence the core during normal operation. For this reason, the primary system is modeled only considering the upper and lower plenum of the core. In order to be as accurate as possible for the determination of the initial conditions for the subsequent LOCA analysis, the core is modeled using one core channel per fuel assembly (193 in total). The radial reflector is modeled as a bypass channel. Figure 22 shows the RELAP5-3D nodalization used in the core design automation studies. It should be noted that this RELAP5-3D model is different that the one used in the subsequent LOCA analysis.

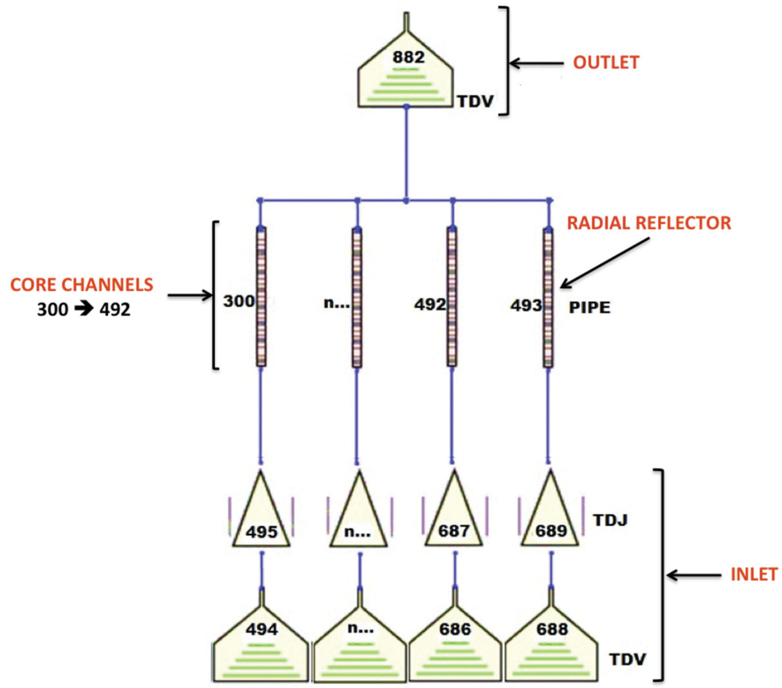


Figure 22. RELAP5-3D Core Nodalisation Used for the Core Simulation.

5.1.5 Transient Power Maneuvers model

It is well known; the core status at BOC, MOC and EOC does not impose challenging conditions for the LOCA analysis. Therefore, the LOCA scenarios for the assessment of the safety margins are generally performed considering the reactor right after a maneuver, for example a xenon transient. The goal is to skew the axial power shapes in order to get bottom peaked, cosine and top peaked power shapes. For the scope of this work, one load-following maneuver has been considered. The power history is shown in Figure 23. At the end of the maneuver, the LOCA analysis is initiated, having as boundary conditions the current status of the plant (burnup, power shape, etc.).

For this type of maneuver the power history is an input of the simulation, hence the reactivity insertion due to the cooling down of the reactor when the power decreases (due to the Doppler effect), is automatically compensated by PHISICS, determining through its Criticality Search module the critical insertion of the control rods. Figure 24 shows the control rod bank positions in the core. The plant has four banks. For the load following transient, only the control rod bank D has been considered. The bank D is the first bank that gets inserted in the core.

Since the maneuver can happen at any time during the equilibrium cycle and the LOCA accident can happen any time during the maneuver, these two variables can be treated stochastically. The RAVEN code has been used to sample different maneuver start times during the equilibrium cycle as well as different LOCA start times during the maneuver (see Figure 25). RAVEN runs then RELAP5-3D in "multi-deck" mode, i.e. it runs the equilibrium cycle base irradiation to the desired time and then starts the maneuver transient in one run.

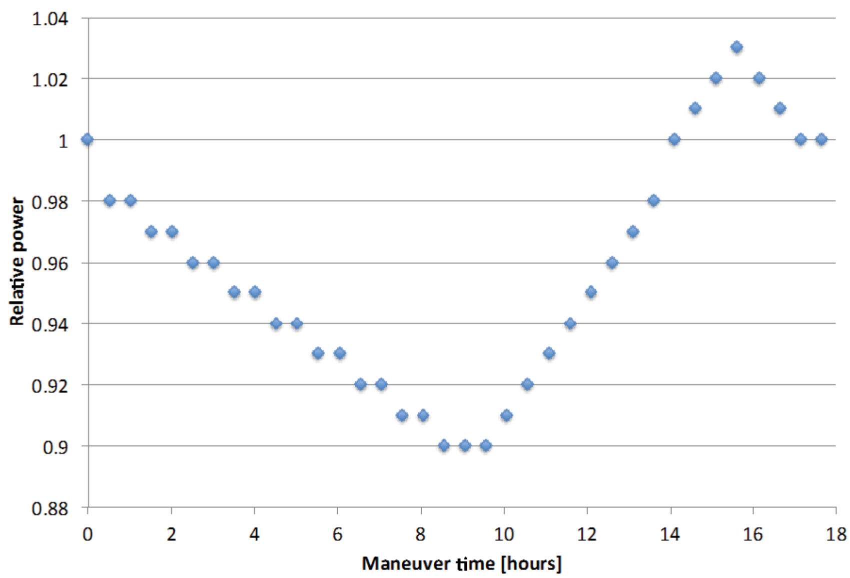
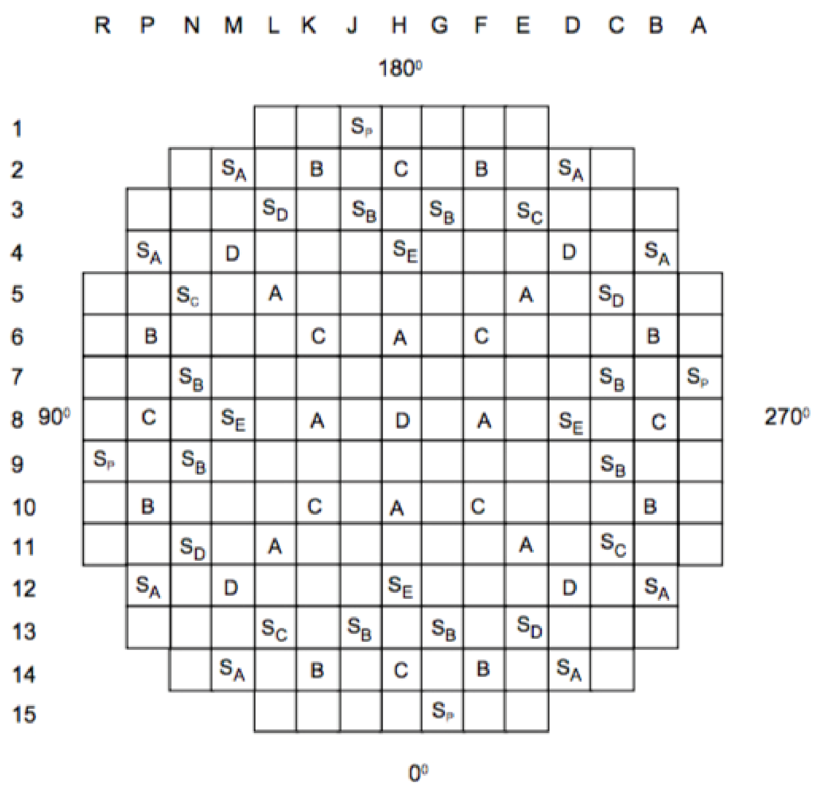


Figure 23. Load Following Maneuver Power History.



Control Bank	Number of Rods
A	8
B	8
C	8
D	5
Total	29

S_P - Spare Locations

Shutdown Bank	Number of Rods
SA	8
SB	8
SC	4
SD	4
SE	4
Total	28

Figure 24. Control Rod Positions.

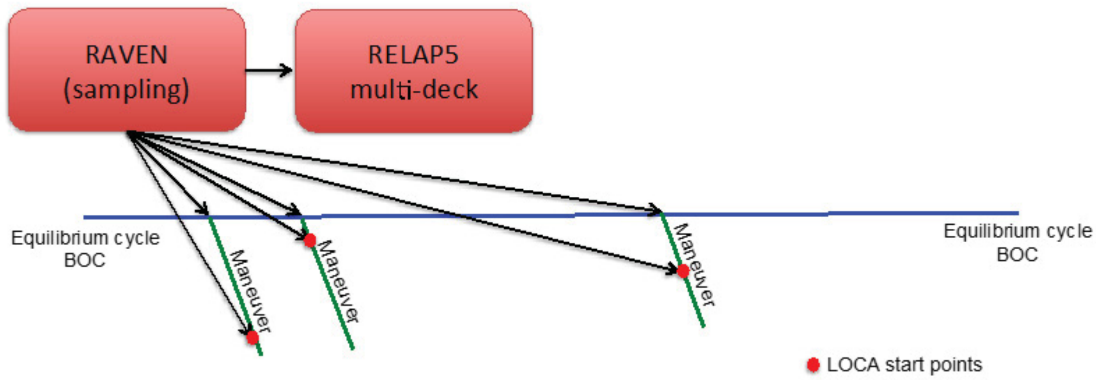


Figure 25. RAVEN Samples the LOCA Start Times and Runs RELAP5 in Multi-Deck Mode.

5.1.6 Core design: STP results

As mentioned in the input data description above, not all the design data was available for the STP core. In addition to the above-mentioned assumptions made in the core design, other degrees of freedom, since exact STP data is not available, are the shuffling scheme, the fuel enrichments for the two fresh fuel batches (since they are given as ranges only) and the IFBA rod distribution. Starting from an initial guess, the feed fuel enrichment and shuffling pattern have been changed by leaving the loading pattern untouched. The fuel enrichments have been searched to obtain an 18-month cycle. A "hand-optimization" of the core has been performed where the shuffling scheme, and the burnable absorber distribution (IFBAs) in the core has been changed in order to minimize the needed enrichment to reach the 18-month cycle as well as to minimize certain thermal-hydraulic characteristics, such as FΔh and Fq. For each step in the optimization, at least 8 cycles have been computed, in order to reach the equilibrium cycle for this configuration. When this optimization has been finished, new cross section libraries for the adjusted core design have been computed and the SPH correction has been applied to them. The "hand-optimization" has then been repeated.

As mentioned, HELIOS-2 calculations, i.e. cross-section libraries at two different axial heights have been generated: one in the fuel region and one in the blanket region. In addition, a library for the bottom reflector and a library for the top reflector have been generated. This leads to a total of 34 libraries (29 fuel assemblies + 1 blanket + 2 radial reflectors, one for the fuel and one for the blanket region + 2 top and bottom reflectors). Axially, the libraries prepared for the fuel regions are associated to the axial layers containing the corresponding fuel and the one blanket library is associated to all blanket regions in the core. The libraries prepared to the top, bottom and radial reflectors are associated with the corresponding reflector assemblies.

The boundary conditions for the thermal-hydraulics have been taken from the STP FSAR [9] (see Table 7). Therefore, the inlet temperature has been set to 561K and the upper plenum pressure has been set to 156bars in the RELAP5-3D model. The mass flow has been set to 18.3t/s with 8.5% bypass.

The two enrichments found in order to reach, at the equilibrium, a cycle length of 18 months are 4.2% and 4.6%. Figure 26 shows the core configuration including loading map and number of burnable absorbers (IFBA) in the fresh fuel assemblies.

In order to keep the reactor critical during the cycle, the Criticality Search module of PHISICS has been used to adjust the boron concentration in the moderator. As mentioned, the moderator cross sections are tabulated as macro cross sections for different boron concentrations. Therefore, the Criticality Search module is interpolating in the cross sections to find the proper boron concentration to keep the reactor critical. The cycle ends when the boron concentration in the moderator falls below 10 ppm. Figure 27 shows the boron letdown curve found for the equilibrium cycle. One can see that before the xenon has been built up (and reached its equilibrium) in the core, about ~1500ppm of diluted boron is required to compensate the excess reactivity. After a couple of days, when the xenon reached its equilibrium, the boron concentration needed to keep the reactor critical drops to ~900ppm. The boron worth being ~8pcm/ppm means that the xenon worth in the reactor is ~7000pcm. Furthermore, one can see that after the xenon drop, the boron concentration goes up again (due to the plutonium build-up in the core) to about 1100 ppm before it decreases until the end of cycle at ~540 days. Although, no data from STP about the boron let-down curve is available that this result could be compared to, this computed boron curve compares well to PWRs similar to STP for which data is available [3].

Figure 28 shows the radial assembly power peaking factors (Pbar), the thermal-hydraulic quantities $F\Delta h$ and Fq and the average burnup for each assembly at BOC end EOC. Furthermore, the figure shows the core wide maximums for these quantities. For the readers' convenience, the assemblies are colored the same way as in Figure 26, i.e. fresh fuel in yellow (green is not used), once burned in orange and twice burned in blue.

The STP FSAR includes core average axial power distributions at full power where the control rod bank D is 10% inserted. Data at BOC and EOC is available. Since no control rod cross sections have been generated, the computed results shown are without any control rods inserted. The core average axial power distributions at BOC, MOC and EOC are shown in Figure 29 and compared to the plant data. One can see that despite the assumptions made in the core and cycle design indicated above as well as the fact that no control rods are inserted, the axial power distribution compares very well to the STP data. This is an indication that the design assumptions made are close to the real one in the plant. Figure 30 shows the maximum pin peaking for each assembly. These values are computed with HELIOS-2 and are shown for BOC and EOC.

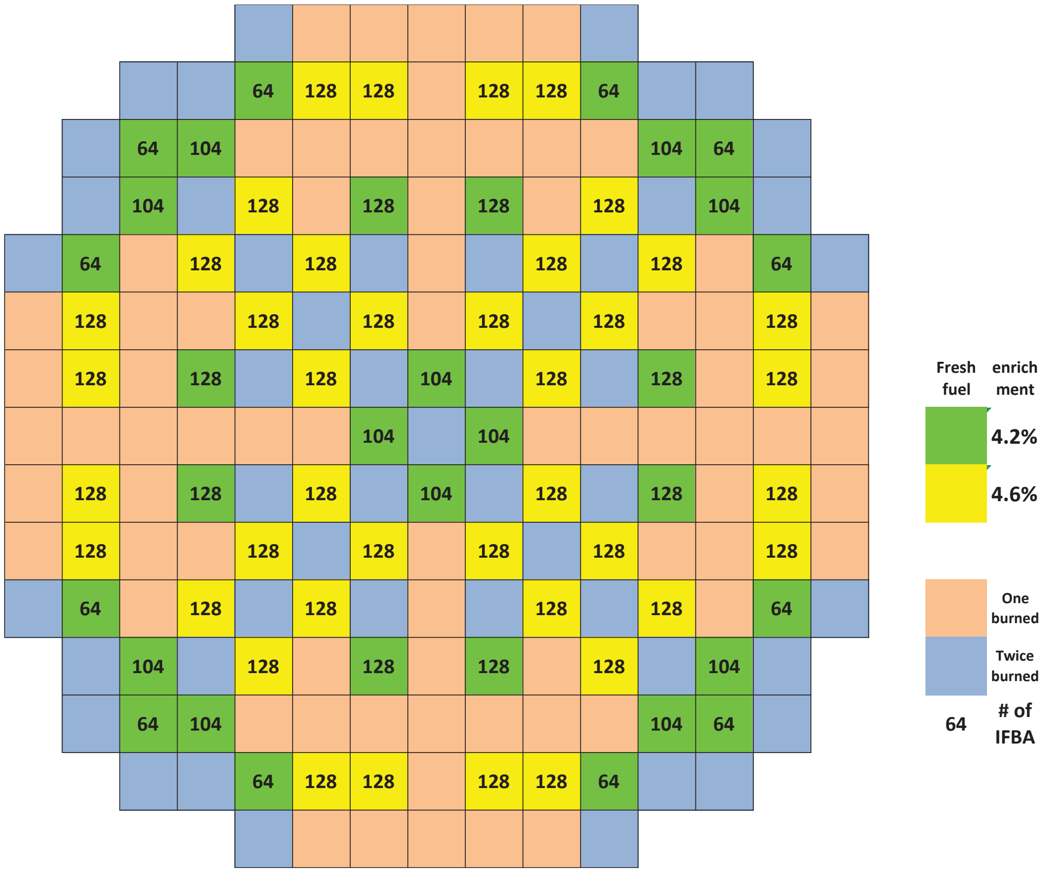


Figure 26. STP Equilibrium Cycle: Reloading Pattern, Fresh Fuel Enrichment and Number of Burnable Absorber (BA) Pins in the Fresh Fuel Assemblies.

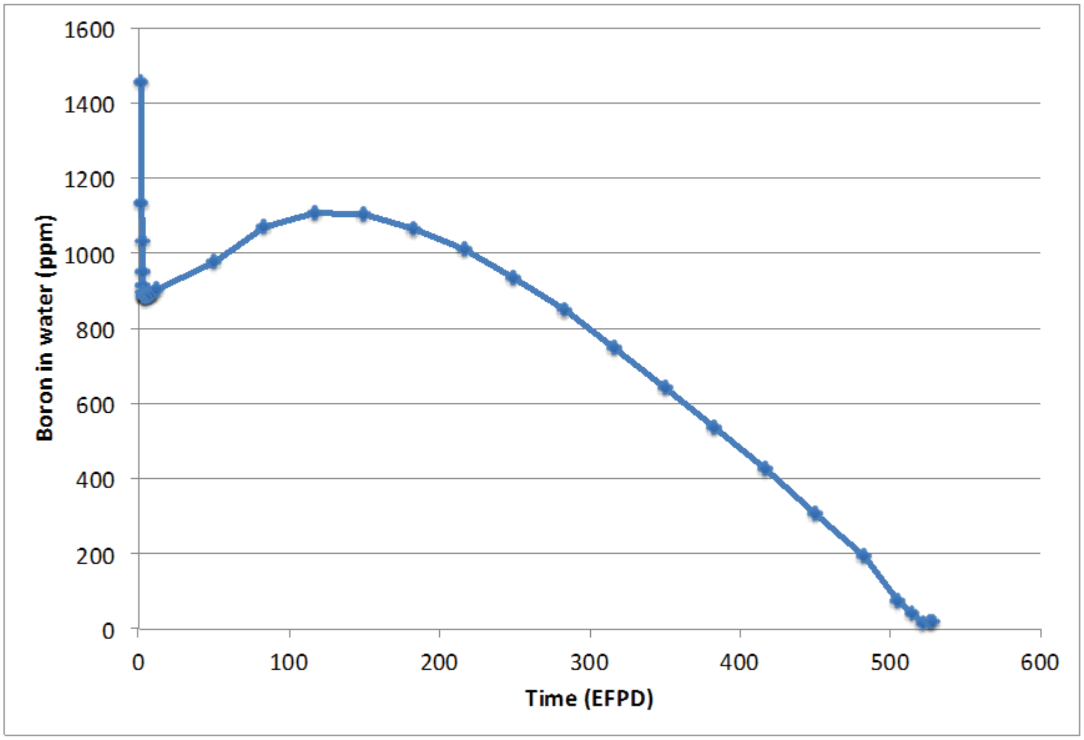


Figure 27. STP Equilibrium Cycle: Boron Letdown Curve.

1.01 1.07 1.28 36.39	1.37 1.46 1.79 0.00	1.28 1.37 1.64 20.47	1.23 1.30 1.55 21.70	1.34 1.42 1.69 17.76	1.19 1.25 1.49 25.87	0.98 1.04 1.24 25.62	0.56 0.60 0.71 25.92
1.37 1.46 1.79 0.00	1.07 1.13 1.35 38.22	1.37 1.45 1.76 0.00	1.08 1.14 1.35 38.13	1.42 1.50 1.82 0.00	1.29 1.37 1.65 24.33	1.21 1.28 1.56 0.00	0.60 0.64 0.76 26.45
1.28 1.37 1.64 20.46	1.36 1.43 1.75 0.00	1.02 1.08 1.29 38.31	1.37 1.45 1.77 0.00	1.31 1.39 1.68 20.34	1.27 1.34 1.62 24.90	1.20 1.27 1.55 0.00	0.56 0.59 0.70 26.84
1.23 1.30 1.55 21.69	1.02 1.08 1.29 36.39	1.35 1.42 1.73 0.00	1.02 1.08 1.30 38.09	1.31 1.38 1.68 0.00	1.10 1.16 1.40 26.53	1.00 1.06 1.29 0.00	0.30 0.32 0.38 43.10
1.34 1.41 1.69 17.76	1.39 1.47 1.79 0.00	1.26 1.35 1.62 21.61	1.29 1.36 1.66 0.00	0.79 0.84 0.99 43.23	0.94 1.01 1.23 0.00	0.41 0.43 0.52 42.51	
1.19 1.25 1.49 25.85	1.28 1.35 1.63 24.30	1.25 1.32 1.59 24.84	1.09 1.15 1.38 26.40	0.93 1.00 1.22 0.00	0.73 0.77 0.93 0.00	0.22 0.23 0.28 41.86	
0.98 1.04 1.24 25.62	1.20 1.27 1.55 0.00	1.18 1.25 1.53 0.00	0.98 1.04 1.27 0.00	0.39 0.42 0.50 45.39	0.22 0.24 0.28 39.56		
0.56 0.60 0.71 25.92	0.60 0.64 0.76 26.34	0.55 0.58 0.70 26.84	0.29 0.31 0.37 43.81				

	max	
Pbar	1.42	Fresh
FDH	1.50	Once burned
Fq	1.82	Twice burned
Burnup	45.40	

0.89 0.93 1.13 54.93	1.21 1.24 1.47 25.82	1.02 1.07 1.28 42.50	0.95 0.98 1.18 41.93	1.02 1.07 1.26 39.63	0.96 1.00 1.16 45.37	0.95 1.00 1.13 43.18	0.65 0.70 0.80 36.56
1.21 1.25 1.48 25.82	0.93 0.98 1.18 57.72	1.26 1.30 1.51 26.33	0.91 0.96 1.15 57.08	1.23 1.28 1.48 25.73	1.06 1.10 1.27 46.12	1.31 1.38 1.51 24.19	0.72 0.76 0.87 38.40
1.02 1.07 1.28 42.50	1.26 1.30 1.51 26.23	0.95 0.98 1.18 57.63	1.30 1.34 1.55 26.93	1.08 1.12 1.32 43.08	1.08 1.12 1.30 46.93	1.34 1.42 1.56 24.76	0.69 0.73 0.84 38.48
0.95 0.98 1.18 41.93	0.89 0.94 1.12 54.66	1.30 1.34 1.55 26.71	0.98 1.03 1.22 57.80	1.33 1.38 1.56 26.39	1.07 1.11 1.27 47.12	1.13 1.22 1.34 20.34	0.41 0.43 0.50 49.44
1.03 1.07 1.26 39.63	1.24 1.28 1.47 25.49	1.07 1.11 1.31 43.88	1.33 1.38 1.56 26.25	0.93 0.98 1.11 59.88	1.25 1.33 1.47 21.56	0.57 0.61 0.70 51.52	
0.96 1.00 1.16 45.36	1.06 1.10 1.27 46.02	1.09 1.12 1.30 46.76	1.07 1.12 1.28 46.91	1.25 1.33 1.47 21.48	1.10 1.18 1.31 17.66	0.38 0.40 0.46 47.20	
0.95 1.00 1.13 43.19	1.32 1.39 1.52 24.16	1.35 1.43 1.57 24.70	1.13 1.22 1.33 20.21	0.56 0.60 0.68 54.14	0.39 0.41 0.47 45.04		
0.65 0.70 0.80 36.56	0.72 0.76 0.87 38.30	0.70 0.73 0.84 38.26	0.40 0.42 0.50 50.06				

	max	
Pbar	1.36	Fresh
FDH	1.43	Once burned
Fq	1.63	Twice burned
Burnup	59.88	

Figure 28. STP Equilibrium Cycle: Pbar, Fdh, Fq and Burnup for Each Assembly at BOC (Top) and EOC (Bottom).

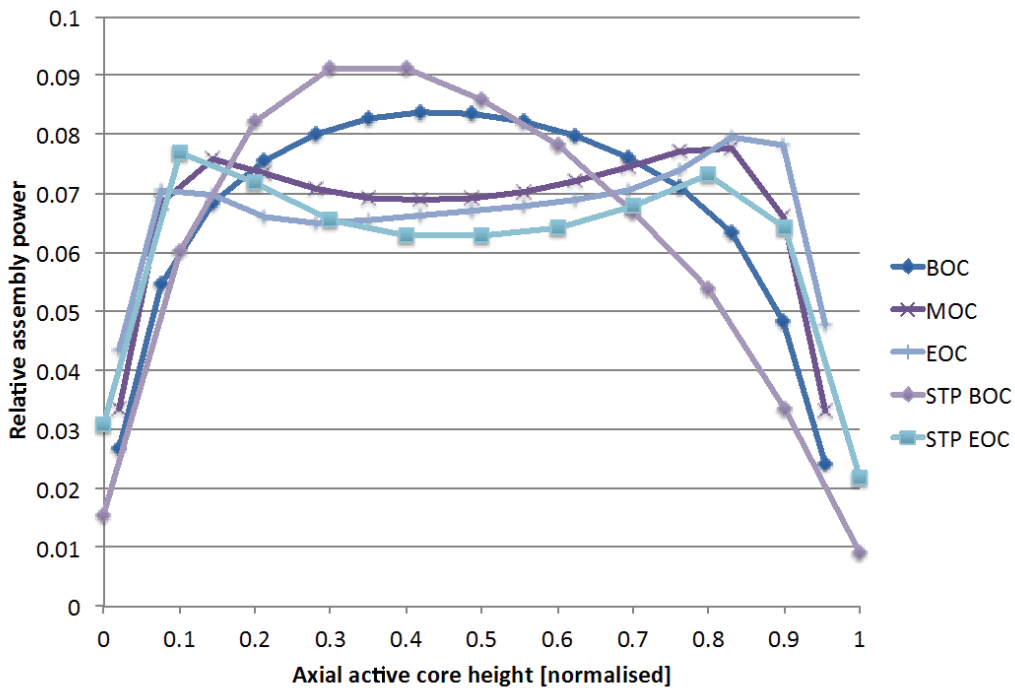


Figure 29. STP Equilibrium Cycle: Core Averaged Axial Power Distribution at BOC, MOC and EOC Compared to Plant Data.

1.05 1.04	1.06 1.03	1.06 1.05	1.06 1.04	1.06 1.04	1.06 1.04	1.06 1.05	1.08 1.08
1.06 1.03	1.06 1.04	1.05 1.03	1.06 1.05	1.06 1.04	1.06 1.04	1.06 1.05	1.06 1.05
1.06 1.05	1.05 1.03	1.06 1.04	1.05 1.03	1.06 1.05	1.06 1.04	1.06 1.06	1.06 1.05
1.06 1.04	1.06 1.05	1.05 1.03	1.06 1.04	1.06 1.03	1.05 1.04	1.06 1.07	1.06 1.05
1.06 1.04	1.06 1.04	1.06 1.05	1.06 1.03	1.06 1.05	1.07 1.06	1.06 1.07	
1.06 1.04	1.06 1.04	1.06 1.04	1.05 1.04	1.07 1.06	1.06 1.07	1.07 1.07	
1.06 1.05	1.06 1.05	1.06 1.06	1.06 1.07	1.06 1.07	1.07 1.07		
1.08 1.08	1.06 1.05	1.06 1.05	1.06 1.05				

	max	Fresh
BOC	1.08	Once burned
EOC	1.08	Twice burned

Figure 30. STP Equilibrium Cycle: Maximum Pin Peaking Factors for Each Assembly at BOC and EOC.

5.1.7 Transient power maneuvers for STP core designs

As mentioned, power distributions during normal operation do not impose challenging conditions for the LOCA analysis. Therefore, the LOCA scenarios for the assessment of the safety margins are performed considering the reactor right after a maneuver. The above-described load following maneuver was used for the STP core. For the STP case, RAVEN was instructed to sample the LOCA start times on a grid, but any other distribution can be envisaged. The power shapes passed to the subsequent LOCA analysis are at the end of the maneuver for BOC, 100 days, 200 days, 300 days, and EOC. As an example, Figure 31 shows the skewed axial power shapes at the end of the load following maneuver for BOC, 300 days and EOC.

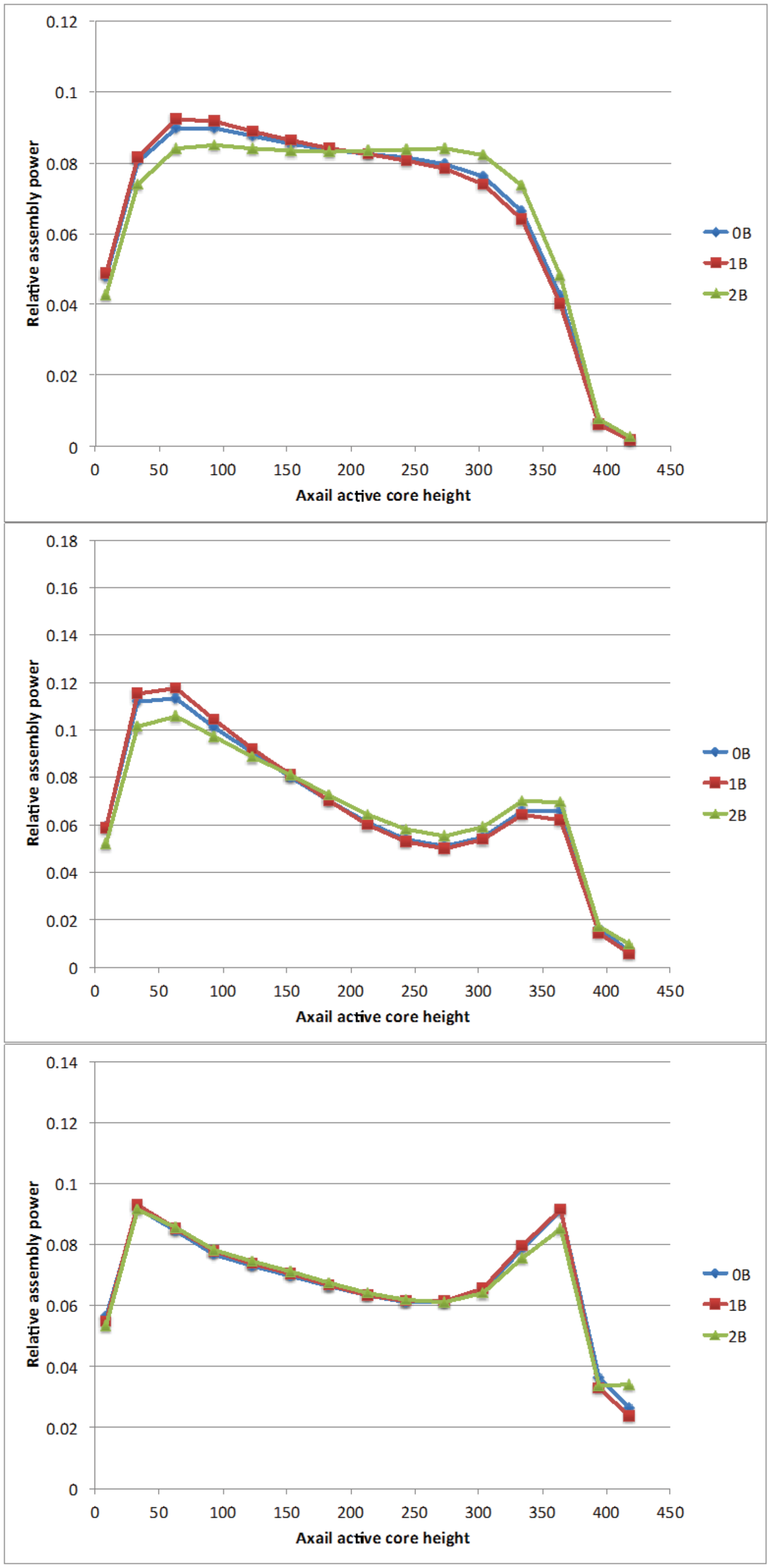


Figure 31. STP Equilibrium Cycle: Skewed Power Shapes at the End of the Maneuver at BOC (Top), at 300 Days (Middle) and at EOC (Bottom). Shown are Core Average Axial Power Distributions for Fresh (0B), Once Burned (1B) and Twice Burned (2B) Fuel Assemblies.

5.2 Fuels Performance

The proposed 10 CFR 50.46c rulemaking implies that all the fuel rods (more than fifty thousands) have to be analyzed in order to find the rods with the limiting PCTR and ECRR values. However, analyzing every fuel rod in a core is not practical to perform LB-LOCA analyses, we built two fuel rod models for each assembly, one for the hot rod and the other one for the average rods. In this way, the total number of simulated fuel rods in LB-LOCA simulations is reduced to a manageable number of 386. The hot rod is defined as the highest power rod in each assembly. The remaining fuel rods in an assembly is lumped together and represented by one FRAPCON model to simulate the behavior of the average rods. The cladding material is ZIRLO^{TM}. The NRC's fuel performance code FRAPCON is the code of choice to perform fuel performance calculations in this work. The FRAPCON input file preparation and code execution were carried out automatically by LOTUS. The power histories required in the FRAPCON calculations were automatically retrieved from the core design results. The FRAPCON calculations were done at the selected cycle exposures of BOC, 100 days, 200 days, 300 days, and EOC. The parameters required to provide the correct steady-state initialization of the RELAP5-3D simulations are subsequently obtained from the FRAPCON output files by LOTUS and mapped into the RELAP5-3D input models. These parameters include the fuel rod internal pressure, gap gas mole fraction, etc.

In this subsection, selected results from the FRAPCON runs are presented to demonstrate the automation capability of LOTUS with respect to fuel performance calculations for the STP core design. Figure 32 shows the power history used for the hot rod in a twice burned fuel assembly. The power history data is automatically retrieved by LOTUS from the core design results and included in the FRAPCON input file prepared by LOTUS. Figure 33 shows the maximum hydrogen contents calculated by FRAPCON versus fuel rod averaged burnup for the STP core design.

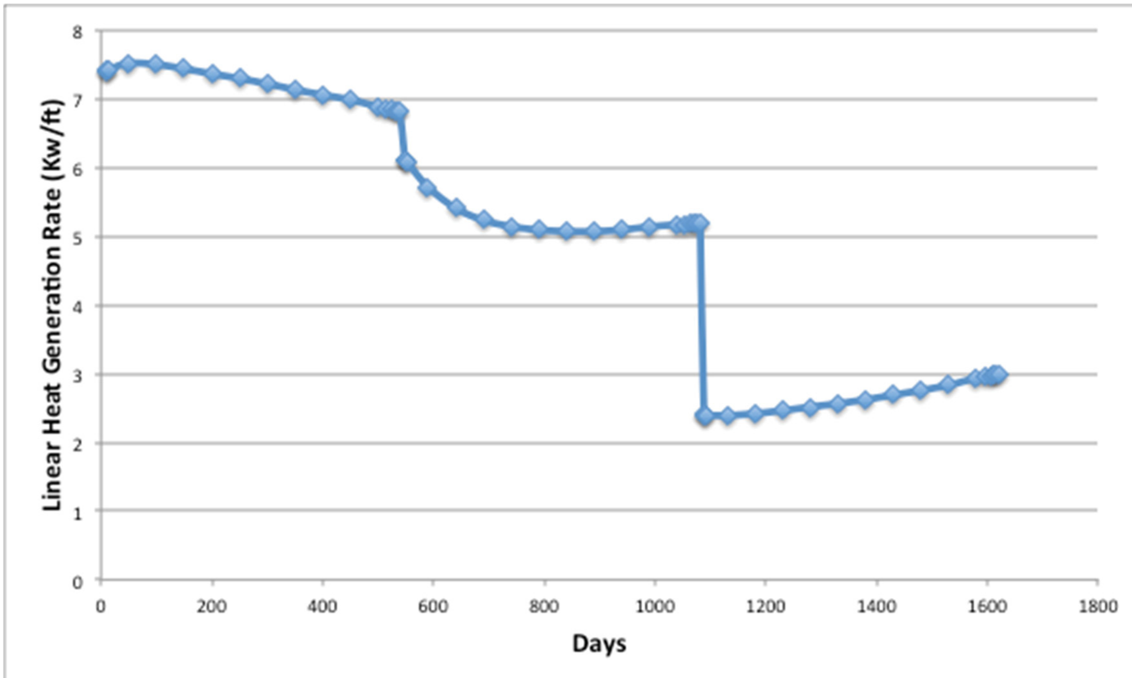


Figure 32. Power History for the Hot Rod in a Twice Burned Fuel Assembly.

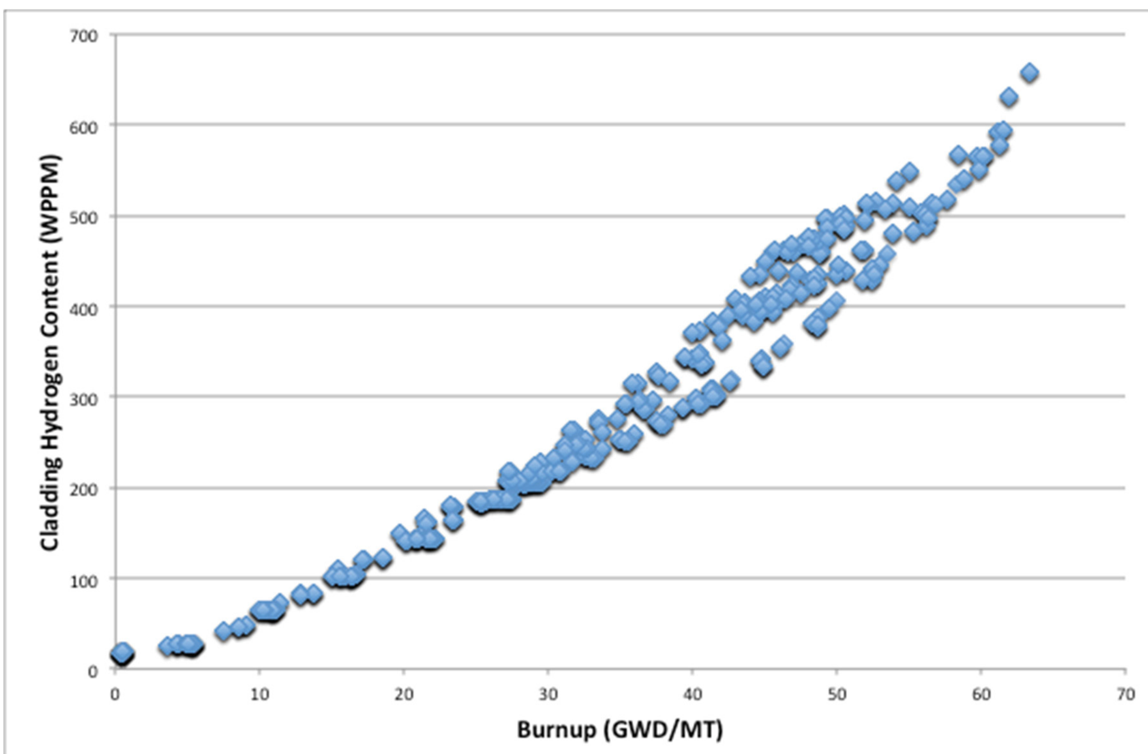


Figure 33. Cladding Hydrogen Content versus Rod Average Burnup.

5.3 Systems Analysis

RELAP5-3D is the code of choice to perform LB-LOCA analyses. The LOTUS-Baseline automation process starts with the RELAP5-3D plant system model that was built in collaboration between INL and Texas A&M University and automatically mapped in certain required parameters from fuel performance and core design calculations. To be consistent with fuel performance calculations, two sets of heat structures were built for each fuel assembly – one for the hot rod and the other for the average rods. From the fuel performance calculations, the required parameters such as rod internal pressure, gap gas mole fraction, etc. are automatically obtained from the FRAPCON output files and mapped into the respective fuel rod models in the RELAP5-3D input files. From the core design calculations, the power shapes from the power maneuvering calculations are automatically obtained from the PHISICS calculations and mapped into the RELAP5-3D input files.

Power shape sensitivity studies have been performed between using the maneuvered power shapes versus chopped cosine power shapes at EOC and it was found that using the chopped cosine power shapes gave more limiting response for peak clad temperatures, as shown in Figure 34. It was determined that the chopped cosine power shapes were to be used in the BEPU analysis and the results are presented in the following subsection.

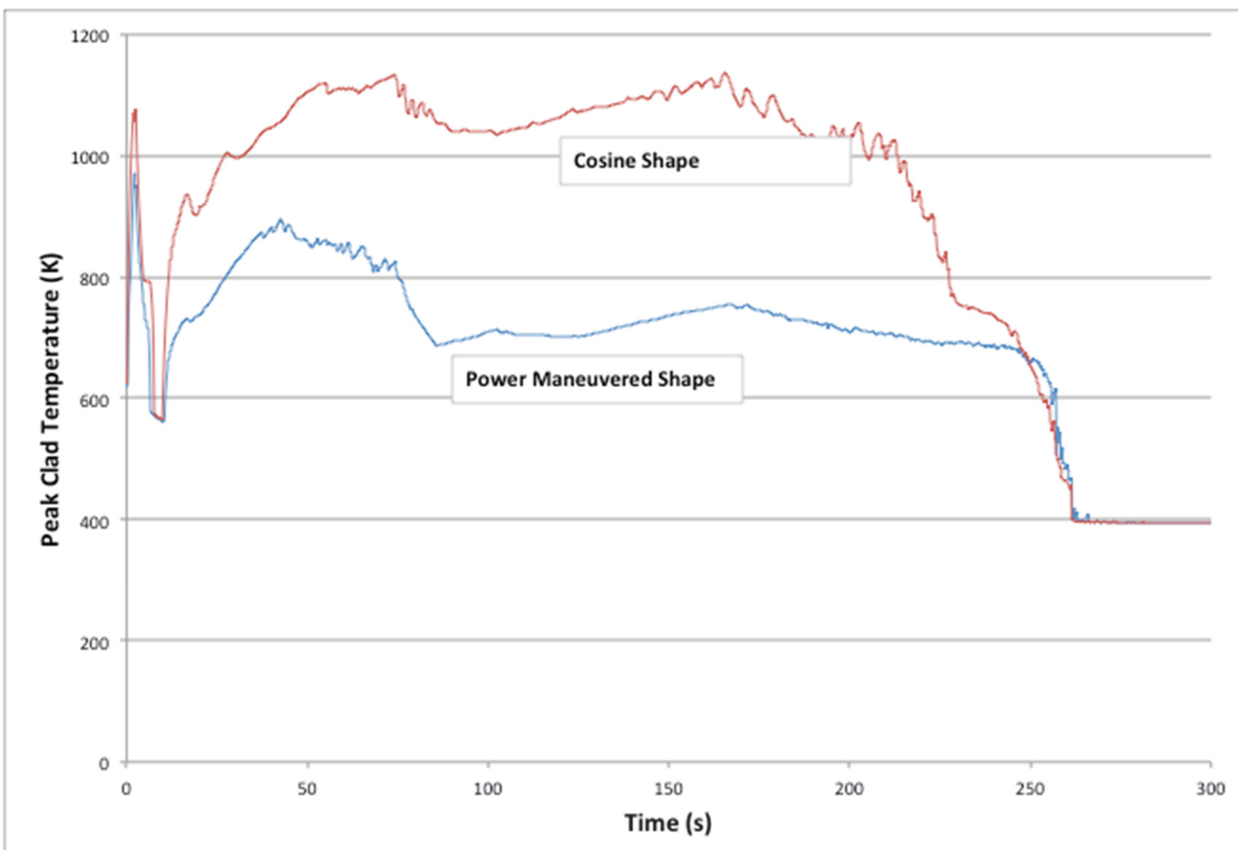


Figure 34. Comparison of PCT in LB-LOCA Transients with Maneuvered Power Shapes versus Cosine Power Shapes.

5.4 Uncertainty Quantification, Risk Assessment and Sensitivity Analysis

The results for uncertainty quantification, risk assessment and sensitivity analysis are presented in this subsection.

5.4.1 Uncertainty quantification and risk assessment

The best-estimate plus uncertainty (BEPU) analyses for LB-LOCA were carried out with the Monte Carlo method. A set of sixteen uncertain input parameters, such as reactor power, decay heat, accumulator conditions, fuel thermal conductivity, heat transfer coefficients, etc., as shown in Table 2, were randomly sampled in the BEPU analyses. The cladding pre-transient hydrogen up-take contents were also obtained from the FRAPCON output files. These are used in the determination of PCTR and ECRR calculations post RELAP5 LB-LOCA calculations. The direct Monte Carlo simulations were carried out. One thousand LB-LOCA cases with RELAP5-3D have been run on Idaho National Laboratory's high performance computers (HPC), respectively, at five selected cycle exposure points at BOC, 100 days, 200 days, 300 days, and EOC. The *PCTRmax* and *ECRRmax* values from each RELAP5-3D output file were obtained and sorted by LOTUS among the 1000 runs, respectively, at the selected cycle exposure points. The probability distribution function (PDF) and the cumulative distribution function (CDF) of the figures of merit (PCTR and ECRR) were subsequently obtained. From the CDF and PCTR and ECRR, the 95 percentile values of PCTR and ECRR, as well as their corresponding PCT and ECR values, are obtained and their associated 95% confidence intervals are subsequently calculated to construct the estimators of the 95/95 upper tolerance limits for PCT and ECR. The 95/95 estimators are then compared to the proposed 10 CFR 50.46c rule to demonstrate compliance. The 95% limit values with 95% confidence interval can be expressed as [22]:

$$Y_{95/95} = \mu_{95\%} \pm 1.96 * SE_{Q95\%} \quad (12)$$

$$SE_{Q95\%} = 2.11 * SE_M \quad (13)$$

where $Y_{95/95}$ is the 95/95 confidence interval, $\mu_{95\%}$ is the 95th percentile value, and $SE_{Q95\%}$ and SE_M are standard error of $\mu_{95\%}$and the mean respectively.

The results for the RELAP5-3D LB-LOCA simulations for the STP core design are summarized in this subsection. For illustrative purpose, the PDF and CDF for PCTR at EOC is shown in Figure 35 and the PDF and CDF for ECRR at EOC is shown in Figure 36. The 95% percentile value with 95% confidence interval are calculated following the 1000 RELAP5-3D LB-LOCA simulations at the selected cycle exposure point and the results are summarized in Table 10. The limiting cases are identified from the LB-LOCA simulations and the PCT and ECR values for the hot rod in each assembly in the limiting cases are obtained by LOTUS and shown in Figure 37 for PCT and Figure 38 for ECR.

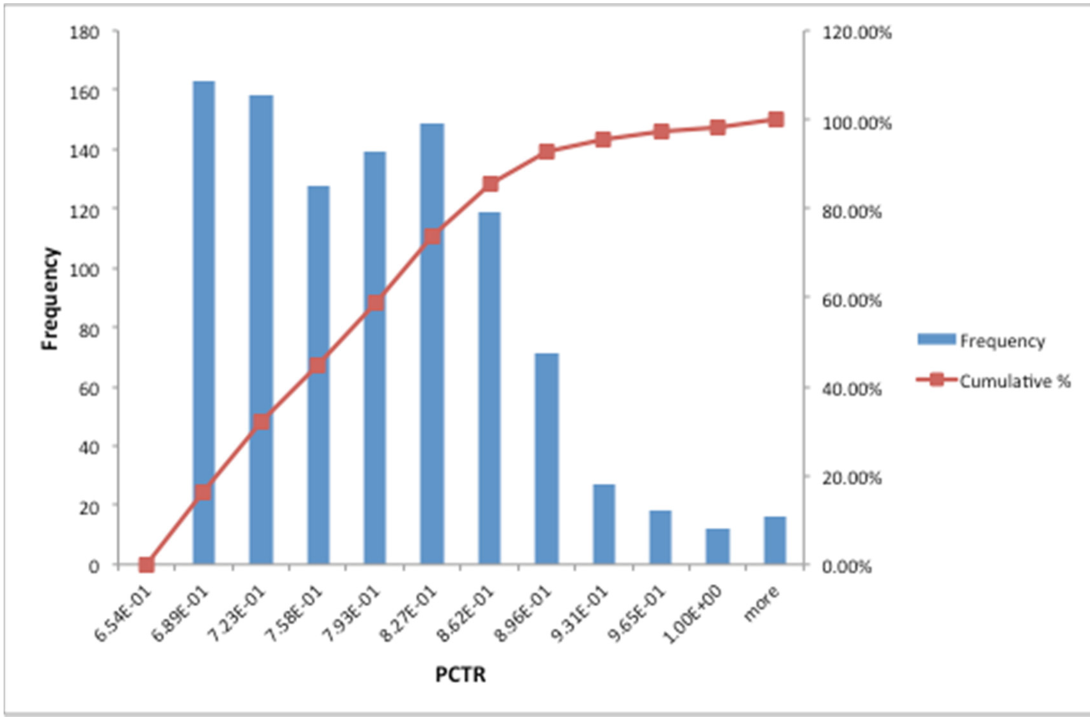


Figure 35. PDF and CDF for PCTR at EOC.

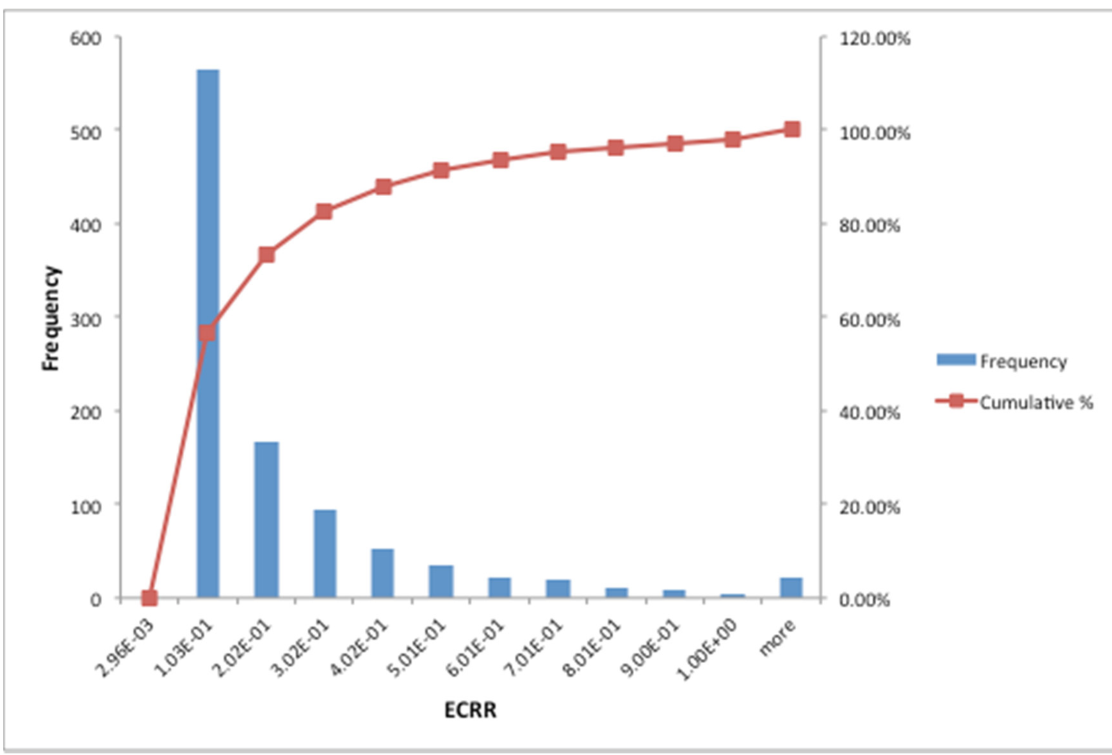


Figure 36. PDF and CDF for ECRR at EOC.

Table 10. Summary of the 95/95 Estimators for PCT and ECR for the STP Core Design.

	PCT (K)		ECR (%)	
	$\mu_{95\%}$	$2.11 * SE_M$	$\mu_{95\%}$	$2.11 * SE_M$
BOC	1134.91	13.47	1.03	0.08
100 Days	1174.53	6.75	1.49	0.09
200 Days	1214.18	5.49	1.68	0.03
300 Days	1271.04	6.05	1.86	0.06
EOC	1308.86	14.05	3.23	0.11

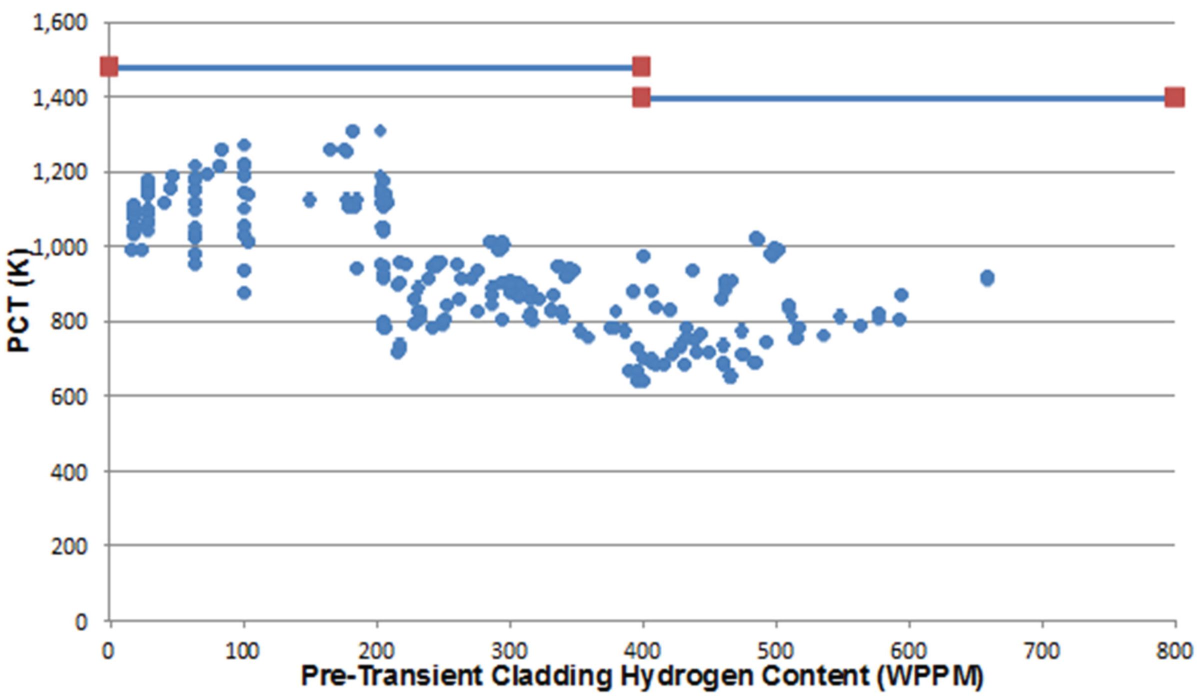


Figure 37. PCT versus Pre-Transient Cladding Hydrogen Content.

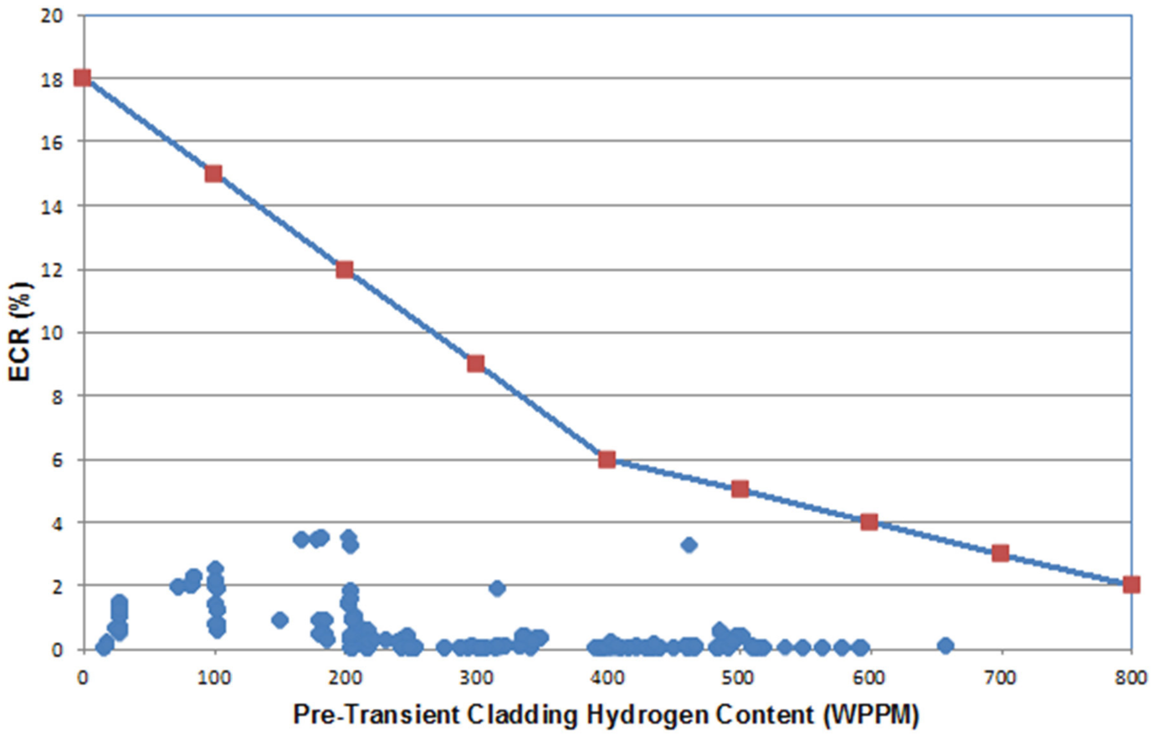


Figure 38. ECR versus Pre-Transient Cladding Hydrogen Content.

5.4.2 Sensitivity analysis

Sensitivity analysis seeks to determine the contribution of the uncertainty in a single model input to the uncertainty in model results. Sensitivity analysis provides a clearer picture of how system inputs correlate to system outputs. Parameters with negligible or no contribution to the system response can be removed in future studies while those parameters with significant contribution present a guide to where areas of future research should be focused on reducing the input uncertainty. The same data used for the uncertainty quantification study is examined for the sensitivity analysis. For demonstration purpose, only the results at 300 days are presented here.

The sensitivity indices for PCTR and ECRR at 300 days using Pearson Correlation Coefficients, Sobol Indices and Delta Moment Independent Measures are shown in Figures 39 and 40 respectively.

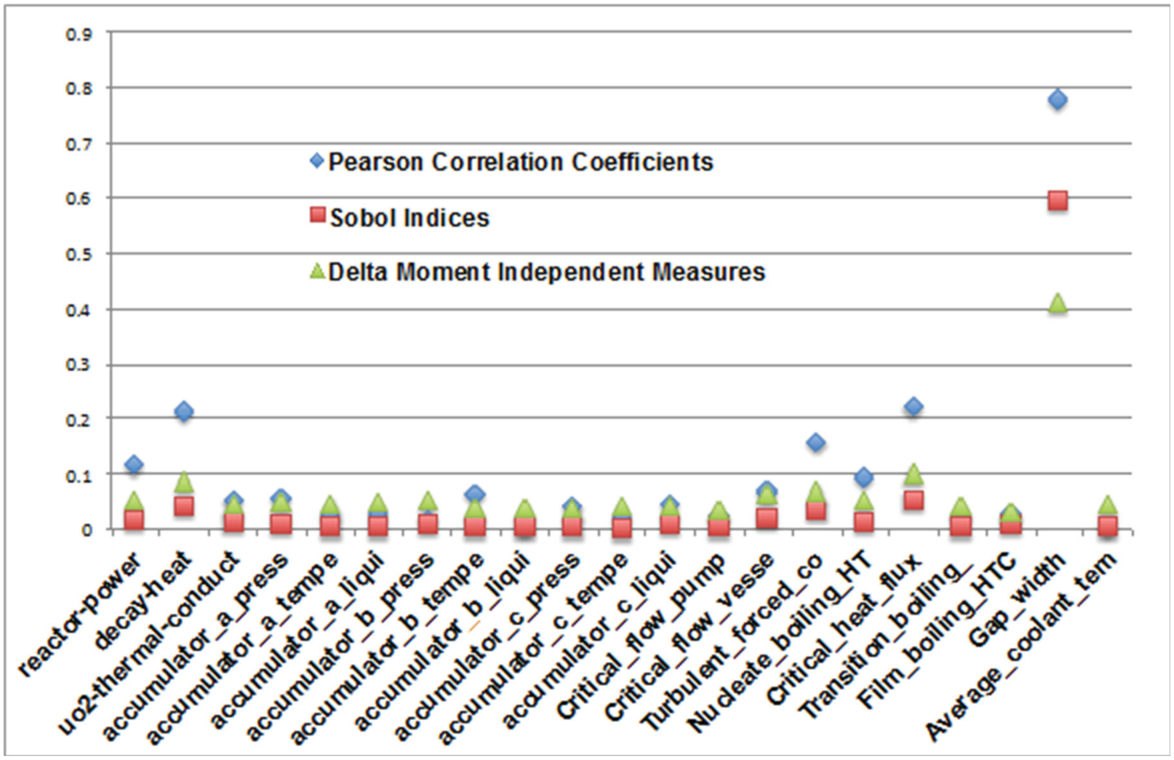


Figure 39. Comparison of Sensitivity Measures for PCTR at 300 Days.

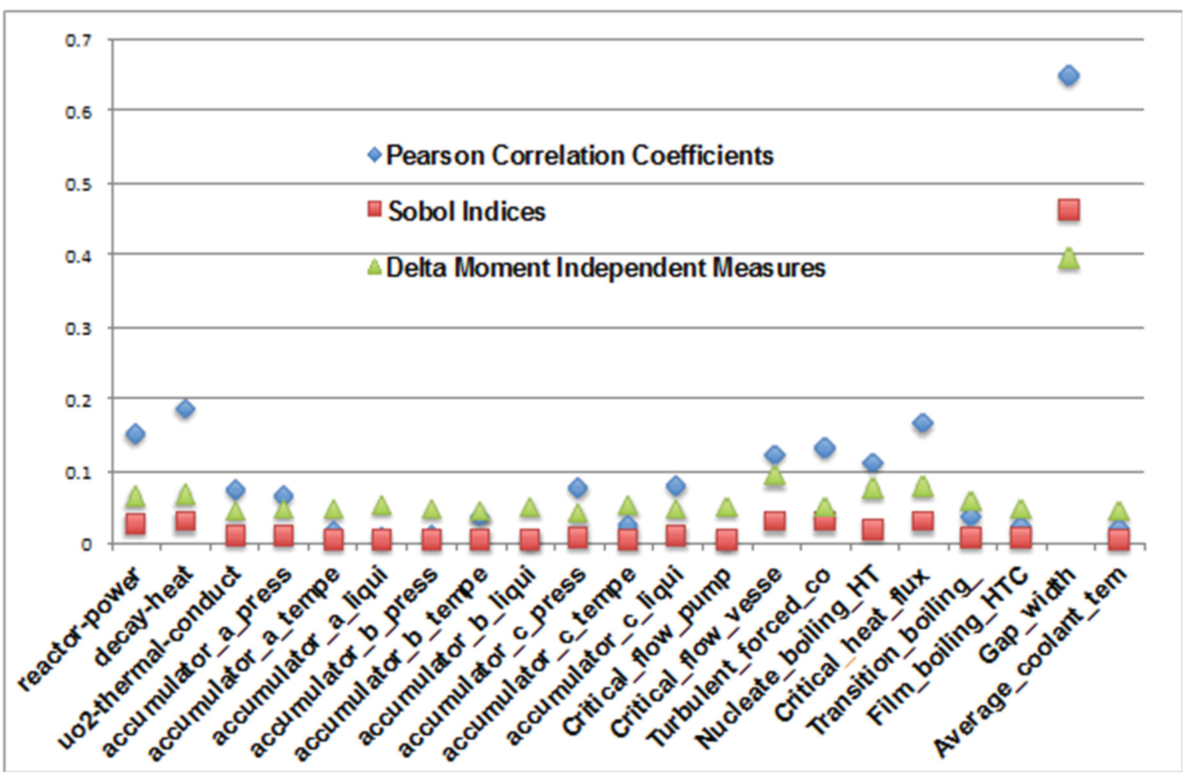


Figure 40. Comparison of Sensitivity Measures for ECRR at 300 Days.

6. CONCLUSIONS, FUTURE WORK AND THE PATH FORWARD

This section presents the conclusion of the analysis results. The future work and path forward are also discussed.

6.1 Results Conclusions

The results from the previous sections indicate that the PCT and ECR responses are well characterized by performance based modeling under large break LOCA conditions for the generic PWR model built based the STP plant. Both PCT and ECR comply with the proposed acceptance criteria with sufficient margins available.

6.2 Industry Application ECCS/LOCA Future Work

The **first area of future work** going forward in the LOTUS framework development would be on further development of the software of LOTUS for the advanced capability (LOTUS-A). A LOTUS Software Requirements Specifications (SRS) and Software Design Description (SDD) will be developed in order to guide the software development of LOTUS.

The **second area of future work** is on tightly coupled codes for transient calculations. We rely on the simplified fuel performance model in RELAP5-3D to capture the fuel behavior under LOCA conditions. Since the metal-water reaction model developed by Cathcart is implemented in RELAP5-3D, the ECR values calculated using RELAP5-3D are adequate to demonstrate the compliance to the 10 CFR 50.46c rules. However, in order to provide more mechanistic modeling of fuel behaviors under LOCA conditions, fuels performance codes have to be coupled with system analysis codes. Fuels performance codes such as FRAPTRAN and BISON will be coupled with system analysis codes such as RELAP5 and RELAP-7 to provide a comprehensive evaluation of clad inner surface oxidation, balloon and burst potential, fuel fragmentation, relocation and dispersal taking into consideration burnup under LOCA and other transient conditions. It is noted that clad inner surface oxidation, fuel fragmentation, radial relocation and dispersal models are to be developed in FRAPTRAN and BISON. This effort will help the development of FRAPTRAN (NRC) and BISON (CASL) on their development activities to address these very challenging issues under LOCA conditions.

The **third area of the future work** is to extend the LOTUS analysis capability to support the development and licensing of the accident tolerant fuel (ATF). Accident tolerant fuel aims at developing advanced cladding materials and fuel designs to achieve superior performance under accident conditions such as LB-LOCA. It has the potential to enable fuel to have high enrichment and to achieve much higher burnup beyond the current licensing limit of 62 GWd/tHM and to extend the “coping time” under accident conditions such that the fuel cycle economics and reactor safety can be greatly improved. The LOTUS framework can be used to perform risk-informed simulations to assess the design and experiments of ATF concepts.

The **fourth area of future work** going forward in the LOTUS framework development would be on developing the core design optimization capability. Figure 41 shows a schematic illustration of the optimization scheme to be developed within the LOTUS framework.

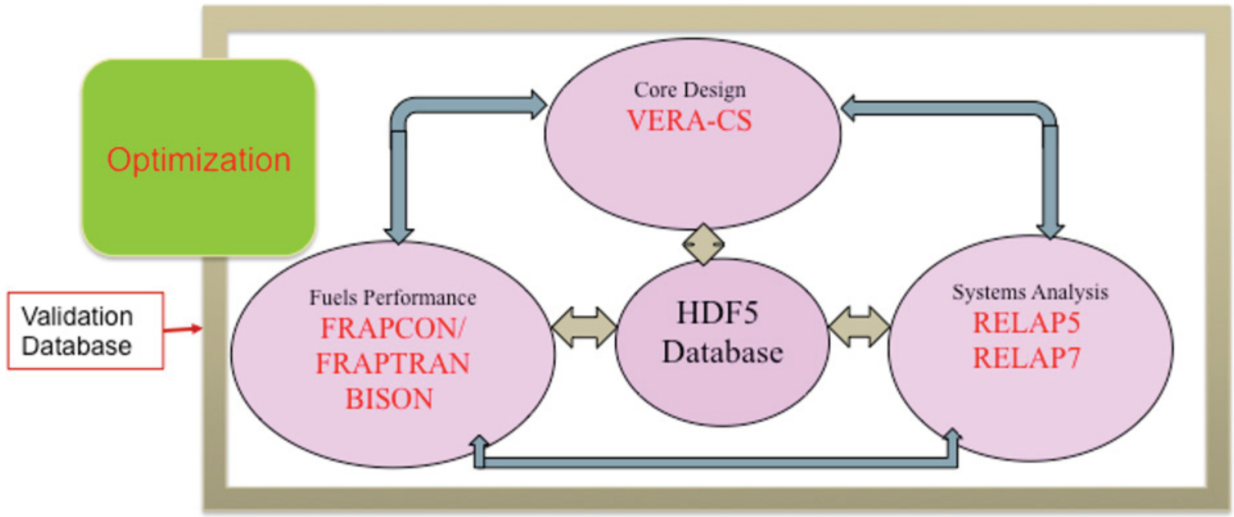


Figure 41. Schematic Illustration of Core Design Optimization Development for LOTUS.

6.3 Path Forward

The idea behind the Industry Application ECCS/LOCA is the development of an Integrated Evaluation Model that is motivation to revisit how risks and uncertainties are quantified across the physical disciplines for the safety analysis in the proposed NRC rule 10 CFR 50.46c. The use of an integrated approach in managing the data stream across the various engineering calculations in this Industry Application is one of the most challenging aspects of the INL research and development. The Integrated Evaluation Model developed by INL for Industry Application ECCS/LOCA (LOTUS) is proposing a solution to this problem.

The importance of the LOTUS framework also extends to current and future nuclear fuels applications. The progress shown on the Industry Application ECCS/LOCA will also provide primary benefits for other technical challenges such as the evaluation and characterization of accident tolerant fuels (ATF) being researched by DOE. As shown in Figure 42, ATF has the "game changing" attributes that have the potential to transform the nuclear industry. Figure 42 is reproduced from a presentation made by Scot Greenlee who is the General Chair & Senior Vice President of Engineering and Technical Services, Exelon Generation Nuclear. Having an integrated multi-physics toolkit that is fuel/clad-, fuel cycle-, and scenario-centric provides a ready platform for the analysis of novel fuel and cladding systems. The coupled multi-physics, multi-scale LOTUS analysis framework allows plant system configuration variations to be studied with speed and precision, including detailed assessment of introducing ATF into current LWR plants for design enhancements. One added benefit of LOTUS is that it allows analyzing inverse problem configurations which are not easily done with traditional sequential processes. This is an important attribute in terms of evaluating ATF composition, behavior, and characteristics under accident scenarios given certain expected safety enhancement such as extending "coping time".

Figure 42. Illustration of “Game Changers” in Delivering Nuclear Promises (Reproduced from Scot Greenlee’s Presentation at 2016 American Nuclear Society Utility Working Conference). [23]

The LOTUS framework can expedite the burnup extension studies. The technical basis for burnup extension (to higher limits) will require additional data to better model fuel behavior at the expected new burnup limit. A comprehensive evaluation of balloon and burst potential, taking into consideration burnup, is needed to gauge both the magnitude and disposition of the issue. The RISMC IA ECCS/LOCA framework has tools suitable for this evaluation. These RISMC tools could be used to perform a sensitivity/probabilistic evaluation, taking into consideration a targeted plant’s systems, to obtain fuel rod balloon and burst potential/pin count. Such higher fuel burnup applications and the potential for evaluation models that offer solutions to these issues is appealing to organization such as Southern Co., Exelon, and the Tennessee Valley Authority.

Lastly, an important facet of the engagement related to RISMC Tool development is continued interaction with the U.S. Nuclear Regulatory Commission. Engagement with the NRC includes technical briefings by the RISMC research team, updates on the RISMC Tools development and applications, and overview of the IA ECCS/LOCA approach being used by RISMC, highlighting the advanced tools and integration approach we are using to solve this complex issue. The value of interacting with the NRC on the RISMC research and development, especially for Industry Application ECCS/LOCA is echoed by the industry and academic communities.

7. REFERENCES

1. U.S. NRC, Draft Regulatory Guide DG-1263, ADAMS Accession Number ML111100391, U.S. NRC, Washington, DC, http://pbadupws.nrc.gov/docs/ML1111/ML111100391.pdf.
2. R. Szilard, et. al., "Industry Application Emergency Core Cooling System Cladding Acceptance Criteria Early Demonstration," Idaho National Laboratory, INL/EXT-15-36541, September 2015.
3. R. Szilard, et. al., "Loss of Coolant Accident /Emergency Core Coolant System Evaluation of Risk-Informed Margins Management Strategies for a Representative Pressurized Water Reactor", INL/EXT-16-39805, September 2016.
4. R. Szilard, et. al., "R&D Plan for RISMC Industry Application #1: ECCS/LOCA Cladding Acceptance Criteria," Idaho National Laboratory, INL/EXT-16-38231, April 2016.
5. R. H. Szilard, et. al., "Industry Application Emergency Core Cooling System Cladding Acceptance Criteria Problem Statement," INL-EXT-15-35073, April 2015.
6. B. E. Boyack, R. B. Duffey, P. Griffith, et al., "Quantifying Reactor Safety Margins," NUREG/CR-5249, 1989.
7. B. E. Boyack, I. Catton, R. B. Duffey, et al., "Quantifying Reactor Safety Margins – 1: An Overview of the Code Scaling, Applicability, and Uncertainty Evaluation Methodology," *Nuclear Engineering and Design*, Vol. 119, no. 1, pp. 1-15, 1990.
8. S. S. Wilks, "Determination of Sample Sizes for Setting Tolerance Limits," The Annals of Mathematical Statistics, Vol. 12, no. 1, pp. 91-96, 1941.
9. South Texas Project Electric Generating Station (STPEGS) Updated Final Safety Analysis Report (UFSAR), Rev. 18, available at the NRC website.
10. RELAP5-3D Code Manual Volume I: Code Structure, System Models and Solution Methods, INEEL-EXT-98-00834, Rev. 4, June, 2012.
11. A. Epiney, et. al., "PHISICS Multi-Group Transport Neutronic Capabilities for RELAP5," *Proc. Int. Congress Advances in Nuclear Power Plants (ICAPP 2012)*, Chicago, Illinois, June 24 –28, 2012, American Nuclear Society (2012).
12. C. Rabiti, et. al., "New Simulation Schemes and Capabilities for the PHISICS/RELAP5-3D Coupled Suite," *Nuclear Science and Engineering*, **Vol. 182**, 104-118, January 2016.
13. N. P. Luciano, et. al., "THE NESTLE 3D NODAL CORE SIMULATOR: MODERN REACTOR MODELS," ANS MC2015 Joint International Conference on Mathematics and Computation (M&C), Supercomputing in Nuclear Applications (SNA) and the Monte Carlo (MC) Method, Nashville, TN, April 19-23, 2015, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2015).
14. C.A. Wemple, H-N.M. Gheorghiu, R.J.J. Stamm'ler, E.A. Villarino, "The HELIOS-2 Lattice Physics Code," 18th AER Symposium on VVER Reactor Physics and Reactor Safety, 19-23 September 2011, Eger, Hungary.

15. A. Epiney, et. al., “RISMC Industry Application #1 (ECCS/LOCA) - Core characterization automation: Reference PWR designs for IA#1,” American Nuclear Society Summer Meeting, June 11-15, 2017.
16. http://frapcon.labworks.org/.
17. Hongbin Zhang, et al., “Comparisons of Wilks’ and Monte Carlo Methods in Response to the 10CFR50.46(c) Proposed Rulemaking,” *NUTHOS-11: The 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety,* Gyeongju, Korea, October 9-13, 2016.
18. E. Borgonovo, “A New Uncertainty Importance Measure,” Reliability Engineering and System Safety, Vol. 92, pp. 771-784 (2007).
19. E. Plischke, E. Borgonovo, and C.L. Smith, “Global Sensitivity Measures from Given Data,” European Journal of Operational Research, Vol. 226, pp. 536-550 (2013).
20. http://salib.readthedocs.io/en/latest/.
21. T. Bahadir, S. Lindahl, S. Palmtag, “SIMULATE-4 Multigroup Nodal Code with Microscopic Depletion Model”, Proceedings of M&C, Avignon, France, September 12-15, 2005.
22. B. Harding, C. Tremblay, and D. Cousineau, "Standard Errors: A Review and Evaluation of Standard Error Estimators using Monte Carlo Simulations," The Quantitative Methods of Psychology, Vol. 10, no. 2, (2014).
23. Scot Greenlee, “Delivering the Nuclear Promise”, a presentation at 2016 American Nuclear Society Utility Working Conference.