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Light Water Reactor Sustainability Program

Industry Application Emergency Core Cooling System Cladding Acceptance Criteria Problem Statement

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DOE Office of Nuclear Energy

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ABSTRACT

The U. S. NRC is currently proposing rulemaking designated as “10 CFR 50.46c” to revise the loss-of-coolant-accident (LOCA)/emergency core cooling system (ECCS) acceptance criteria to include the effects of higher burnup on cladding performance as well as to address other technical issues. The NRC is also currently resolving the public comments with the final rule expected to be issued in April 2016. The impact of the final 50.46c rule on the industry may involve updating of fuel vendor LOCA evaluation models, NRC review and approval, and licensee submittal of new LOCA evaluations or re-analyses and associated technical specification revisions for NRC review and approval. The rule implementation process, both industry and NRC activities, is expected to take 4-6 years following the rule effective date. As motivated by the new rule, the need to use advanced cladding designs may be a result. 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 cost as a spectrum of fuel rod initial burnup states may need to be analyzed to demonstrate compliance. Consequently, there will be an increased focus on licensee decision making related to LOCA analysis to minimize cost and impact, and to manage margin.

The proposed rule would apply to a light water reactor and to all cladding types. The key points of the new rule are as follows:

- Cladding performance cannot be evaluated in isolation. Cladding performance and ECCS performance need to be considered in a coupled way.
- 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.” SECY-12-0034).

In the remainder of this report, we address the technical issues and approaches we will use to investigate this Industry Application within the Risk-Informed Safety Margin Characterization (RISMC) Pathway. Specifically, in Section 2 we review the RISMC vision, and discuss the industry application concept and its implementation strategy. In Section 3 we define the Risk-Informed Margin Management (RIMM) Industry Application #1 (IA1), Integrated Cladding ECCS/LOCA Performance Analysis, aiming to develop and demonstrate an advanced methodology and tool for industry to apply to large-break LOCA issue evaluation, decision making, and margin management. Sections 4, 5, and 6 introduce the subsequent phases of the IA1 project, and Section 7 proposes a schedule and budget for all phases of the project.

A review of phase 1 of this project (Problem Definition) was conducted on March 12-13, 2015 by the team members of this project.

CONTENTS

ABSTRACT iii

TABLES v

FIGURES vi

1. INTRODUCTION AND BACKGROUND 1

2. RISMC CONSIDERATIONS 3
   2.1 Attributes of RISMC 3
   2.2 Industry Applications 7

3. PROBLEM STATEMENT – PLANT MODEL SELECTION 8
   3.1 Figures Of Merit 9
   3.2 Core Physics Reference Design 11
   3.3 LBLOCA Transient Model 16
   3.4 Fuels Performance Models 18

4. INDUSTRY APPLICATION EARLY DEMONSTRATION 23
   4.1 Integrated Evaluation Model (IEM) Framework 23
   4.2 Realistic LBLOCA BEPU Analysis 25

5. STAKEHOLDER ENGAGEMENT 27
   5.1 Stakeholder Selection Strategy 27
   5.2 Value Proposition 29
   5.3 Selection Recommendation 32

6. INDUSTRY APPLICATION FULL DEMONSTRATION CONSIDERATIONS 33
   6.1 Traditional PRA and BEPU Integrated Approach 33
   6.2 Dynamic PRA and the Advanced RISMC Toolkit 35
   6.3 Research Needs for this Industry Application 39

7. WORK PLAN AND SCHEDULE 45

8. SUMMARY 46

9. REFERENCES 47

TABLES

Table 1. Comparison of “Risk-Informed” Analyses. 7

FIGURES

Figure 1. BEPU Paradigm in Practice. 2
Figure 2. Value Tree for “Risk-Informed” 6
Figure 3. Figure 2 from [14]. 10
Figure 4. Figure 1 from DG-1263. [32] 10
Figure 5. Example of AO Control (Watts Bar Nuclear Plant Unit 1 Cycle 13 Core Operating Limits Report (COLR). [36] 12
Figure 6. Core Design Strategy Scheme. 13
Figure 7. HELIOS-2 PWR Fuel Assembly Model. 13
Figure 8. BEAVRS Benchmark. [40] 14
Figure 9. RELAP5-3D Nodalization. 15
Figure 10. PWR Reload Pattern. 15
Figure 11. RELAP5 Model of the Primary System of a Typical Four Loop PWR. 17
Figure 12. RELAP5 Model of the Steam line of a Typical Four Loop PWR. 17
Figure 13. Illustration of Assembly Grouping and Homogenization in RELAP5 Calculations. 18
Figure 14. Illustration of Simulations of Various Accident Scenarios. 18
Figure 15. A Comparison of the Fink-Lucuta and MATPRO Empirical Models for the Thermal Conductivity of Full Density UO_2, as a Function of Temperature and Burnup. [46] 20
Figure 16. Segmentation at the Fuel-Cladding Gap for RELAP5-3D. [39] 21
Figure 17. Flow Chart of the RIMM Integrated Evaluation Model. 24
Figure 18. Monte Carlo Forward Propagation of Uncertainties. 25
Figure 19. Figure 7 and 8 from Frepoli. [12] 26
Figure 20. Characterization of an Operational Core. 31
Figure 21. Optimization of a Characterized Core. 31
Figure 22. Combined deterministic and probabilistic procedure for safety assessment of BDBA. [52] 34
Figure 23. RAVEN Simulation Platform Scheme. [58] 36
Figure 24. DET Scheme. 37
Figure 25. Hybrid Dynamic Event Tree (HDET) Scheme. [60] 38
Figure 26. HDET Methodology Applied to LOCA Scenarios. 38
Figure 27. The Acceptance Criteria. 40
Figure 28. Correspondence Between the Acceptance Criteria in Figure-of-Merit (FOM) Space and the Limit Surface in the Space of Controlled and Uncertain Variables. 40
Figure 29. Assembly Ellipsoids Intersecting the Limit Surface. 42
Figure 30. Iterating to Determine the Limit Surface. [61] 43
Figure 31. A Possible Correspondence Between the Limit Surface for Assembly #1 and the Figures Of Merit. 44
Figure 32. Synthesis of a Plant-Level Result, Given a Set of PCT and ECR Values. 44
Figure 33. RIMM IA#1 Project Phases. 45

Industry Application Emergency Core Cooling System Cladding Acceptance Criteria Problem Statement

1. INTRODUCTION AND BACKGROUND

The 10 CFR 50.46 rule (Code of Federal Regulations, Title 10, Part 50.46, Article (b) [1] [2] [3] [4]) defines the acceptance criteria for emergency core cooling systems for light-water nuclear power reactors. According to the rule:

[…] ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number 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 are calculated. […]

Currently the NRC allows two alternative methods to demonstrate compliance with the rule. A prescriptive, deterministic and conservative[1] method, Appendix K to part 50 provides required and acceptable features for the ECCS Performance Evaluation Models (EMs).

Otherwise […] the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. […]

The latter is what enables the so called Best-Estimate-Plus-Uncertainties (BEPU) methods. Regulatory Guides 1.157 [5] and 1.203 [6] provide further guidance on how to set up such methods in order to meet USNRC expectations.

Both of these methods are still constructed outside a risk-informed paradigm in the sense that safety analysis is formulated to support findings regarding reasonable assurance of adequate protection based on a specific surrogate for "safety." In particular, specific "design-basis" accidents serve as proxies for a certain portion of the spectrum of challenges to plant safety functions; satisfactory performance in these types of accidents (as demonstrated based on conservative analysis) is considered to be evidence of a certain kind of "safety."

Note that BEPU method today still contains a high degree of conservativism, mostly to cover a lack of knowledge in some phenomena and to ease licensing and implementation.

An illustration of the industry practice is shown in Figure 1. The column on the right is the "ideal final solution" in a situation of "perfect knowledge". In that situation, the EM 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 key 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 BEPU value is below the regulatory limit which is designed by regulators to be below the physical limit.

In practice, knowledge is not perfect and conservativism is added to enable the execution and licensing of the analysis. These are a combination of "penalizations", "accepted biases" in the tools and "estimate" of the uncertainty, often also affected by the limited size of the sample of the simulation "population."

---

[1] Note that potential sources of "non-conservativism" in the Appendix K of 10 CFR 50.46 rule were identified in 2001 by the USNRC in support of Risk Informed Regulation (http://pbadupws.nrc.gov/docs/ML0217/ML021720716.pdf)

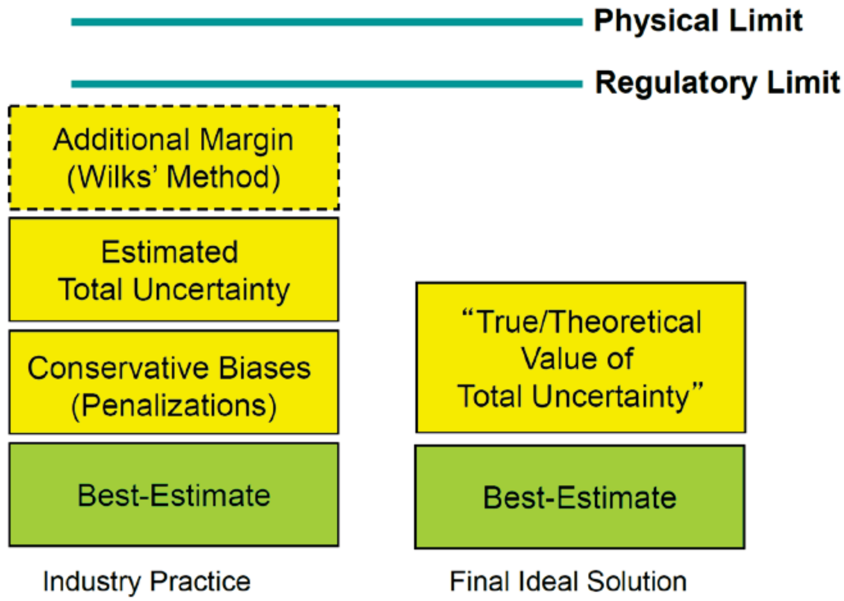

Figure 1. BEPU Paradigm in Practice.

Despite these shortcomings, in the last decade the use of best-estimate plus uncertainty Loss of Coolant Accident (LOCA) methodologies has become the de-facto standard in the industry. A large number of the Pressurized Water Reactor (PWR) fleet in the United States (U.S.) is currently analyzed using best-estimate methods. For PWRs, the two main methodologies applied in the U.S. are the AREVA realistic Large-Break LOCA (LBLOCA) [7] [8] and Westinghouse Electric Company (WEC) LBLOCA EM. [9] [10] [11]

The historical evolution of these methods can be followed with Westinghouse as an example, as they responded developing BE technology soon after the CSAU was proposed in the late 80's. [12] [13] WEC BELOCA methodology evolved from the original 1996 Safety Evaluation Report (SER), [9] which was based on a response surface technique to combine the uncertainties into the 2004 model, the Automated Statistical Treatment of Uncertainty Method (ASTRUM). [14] The only major distinction between the Code Qualification Document (CQD) EM [9] and the ASTRUM EM is the use of a direct Monte Carlo sampling of uncertainty. Through directly sampling the code (not a surrogate like a response surface) and reliance on non-parametric order statistics as a means to produce a 95/95 probability statement compliance with the 10 CFR 50.46 criteria is demonstrated. AREVA has a similar history.

In parallel to power uprates a trend of improving fuel utilization over the years has led to an increase of discharge burnup of the fuel, which almost doubled in the last two decades. The current regulatory limit is 62 MWd/kg, [15] however the limit is likely to be increased moving forward.

The NRC and the industry overall gathered a significant amount of data and information relative to fuel degradation effects under extended irradiation. As a result a wealth of information on the behavior of fuel at high burnup was collected over the same time period. Oxidation, embrittlement and deformation of fuel and fuel cladding under LOCA conditions have been extensively investigated.

A detailed review of these findings is beyond the scope of this report, however significant literature and regulatory material is available. This research expanded the knowledge base to support a better understanding of LOCA-related phenomena.

LOCA separate effect methods and tests have been carried out to gather knowledge on cladding creep, diffusion constants for α to $\alpha+\beta$ and $\alpha+\beta$ to β transformation temperatures, ductility, the effect of hydrogen and oxygen on residual cladding ductility, and integral quench tests for strength. LOCA phenomena such a ballooning, burst, oxidation, fuel relocation and possible fracture at quench have been further investigated. Methods for LOCA oxidation testing and details of high-temperature oxidation behavior of cladding material are available. This also includes two-sided oxidation and ring compression for ductility, in-reactor and out of reactor cladding deformation experimental data, experimental studies of coolability of a deformed core under LOCA conditions and additional data for validation of transient fuel performance computer codes.

As results of these findings, the NRC is considering a rulemaking that would revise requirements in 10 CFR 50.46 (the ECCS rule). [16] [17] [18] [19] The work sponsored by the NRC suggested that the current regulatory acceptance criteria are actually non conservative for higher burnup fuel (i.e., that embrittlement mechanisms not contemplated in the original criteria exist and the 17% limit on oxidation is not adequate to preserve the level of ductility that the NRC originally deemed to be warranted for adequate protection). Assuming that the ECCS rulemaking goes forward in essentially its current form, licensees will eventually need to perform and submit revised accident analysis that addresses the new acceptance criteria. Among the claims that need to be supported is the claim to have found the limiting case. Given that the limiting case has appropriate margin to acceptance criteria, it can then be argued that ECCS performance is adequate. Classically, identification of the limiting case has been done manually, and the complexity of the problem has led to numerous instances in which the analysis of record was based on a non-limiting case.

Overall the rule is expected to be more restrictive than the current rule, therefore the industry is responding by improving their methods to prevent undue restriction to the operation of the plants.

2. RISMC CONSIDERATIONS

This chapter summarizes the considerations on how the Risk-Informed Margin Management Industry Applications are part of the DOE Light Water Reactor Sustainability (LWRS) Program Risk Informed Safety Margin Characterization (RISMC) Technical Pathway. [20] A more detailed breakdown of project functions is available in Reference [21].

2.1 Attributes of RISMC

The premise of RISMC is that in making safety decisions (whether we need to modify our design or operating practices, whether our system risk is as low as reasonable practicable (ALARP), whether we have reasonable assurance of adequate protection), it is useful to analyze system performance in terms of “margin,” and moreover to do this in a “risk-informed” way. It is presently deemed potentially useful to have a “risk scale” that can be used to describe the degree of “risk informed” that a particular process or program takes when evaluating technical safety issues. The present section provides a proposed scale, with particular focus on safety margin characterization. The guidance is that a scale is used up to 11, where 11 is the maximum attainable degree of risk-informed an analysis could demonstrate – we will discuss the constituent parts of this scoring metric later in this section (multiple attributes combine to provide a potential total score of 11).

The current usage of “risk-informed” as it applies to NRC decision-making appears to have originated with NRC Chairman Jackson in the early or mid-1990’s. During the 1980’s and early 90’s, many, many papers were being written on the subject of “risk-based” regulation (emphasis added). The context of those papers was deciding whether regulatory burden could legitimately be reduced (or, in principle, whether it needed to be tightened), based on risk model results. Often, risk model results suggest that burden can be reduced; but then, as now, there was opposition to reducing burden significantly, based on PRA as the primary justification. For the traditionalists, “risk-based” was a non-starter.

Since the publication of RG 1.174, [22] it has been customary in commercial nuclear circles to cite it as a definitive source on what “risk-informed” means. Granted that RG 1.174 was a step forward in its time, trying to base all risk-informed developments on it has never been satisfactory from a technical point of view. RG 1.174 is about certain categories of license amendment requests, not regulatory theory. In particular, the inputs to integrated risk-informed decision-making include items such as whether current regulations are satisfied, and whether performance monitoring will be carried out to help ensure that the proposed plant modification (or changed operating practice) does not lead to a safety decrement.[2] And the RG "discussion" of defense in depth eventually caused ACRS to note that the RG's vagueness leading to "arbitrary appeals to defense in depth" as key tools for regulatory recidivists.

Classically, the phrase "risk-informed" describes a decision process, not a calculation or a margin characterization process. This process has the following characteristics:

- making better decisions through more realistic modeling, with "margin characterization" as a unifying theme, and
- addressing a broad range of issues of diverse kinds, and not just compliance with NRC regulations.

In the RISMC Pathway, we are concerned with scenarios that could impact safety or force downtime having nothing to do with public safety. Therefore, we are using the term "risk" in a broader sense than "risk to public health and safety."

Work discussed in RISMC reports in previous years included an MIT thesis by Lorenzo Pagani: [23]

- Safety margin is the difference between a characteristic value of the capacity and a characteristic value of the load.
- While [this measure] provides a first approximation of functional reliability, ranking different systems on safety margins alone can lead to erroneous results. The knowledge of the distance from failure in terms of safety margins is not sufficient to evaluate the risk of a system; the breadth of the uncertain distribution is the other important part of the assessment.

With the above in mind, the following definition was proposed for "Risk-Informed Safety Margin Characterization:"

- RISMC work should explicitly support the decision-makers (e.g., on major component replacement), and do this in part by providing a technically valid framework for assessing the expected value of doing further work to gather information to reduce uncertainty (testing, analysis, data collection)
- Characterization of RISM should imply:
  - The Pagani definition of margin, with agreed-upon designation of the "characteristic values" of load and capacity
  - Enough information about state-of-knowledge uncertainty and aleatory uncertainty to imply (together with the "margin") a failure probability
  - Margin and uncertainty to be evaluated from the perspective of the integrated safety case (not just DBA paths) (see "Scope Issues")

We now appreciate explicitly that if we define "capacity" in terms of a scenario-dependent quantity like peak clad temperature, then capacity is scenario-dependent, and that "failure" needs to be judged at the system level, as well as (perhaps) at the SSC level.

Arguably, the important thing is whether a comprehensive scenario set is modelled with a view to quantitative analysis of decision alternatives, as opposed to pass-fail compliance with requirements derived from surrogates formulated by engineering judgment (like large-break LOCA). If we do not model a scenario in a way that supports saying what is important and what is not, we are not being risk-informed.

In particular: classical LOCA analysis establishes whether a given configuration is sufficient, but not whether it is necessary. Classical safety analysis affords only limited insight into which of two "sufficient"

[2] Moreover, the intensity of public obeisance to RG 1.174 is waning. Quote from a recent draft NEI white paper: "In general, the existing guidance that addresses the treatment of uncertainty does not provide clear expectations on what information is needed and how the information should be interpreted in the context of risk-informed decision-making. This is applicable at both the practitioner and decision-maker levels. RG 1.174 is written at a very high, almost philosophical level and lacks details needed concerning the practical application of this concept."

systems would actually be preferable. On the other hand, safety margin characterization is "risk-informed" if it is based on the following:

- An issue space is formulated, implicitly defining a class of scenarios to be analyzed probabilistically and the figures of merit[3] to be evaluated probabilistically, "margin" then to be analyzed in terms of those figures of merit in those scenarios,
  - Uncertainties are identified and assigned appropriate distributions,
  - The state of knowledge within that issue space is delineated in terms of state-of-knowledge probability distributions on uncertain variables, or perhaps probability bounds analysis,

- The scenario set is analyzed in sufficient detail (with sufficient coverage of the issue space) to
  - Characterize margin[4] in the relevant figures of merit, including the comparison of absolute margin with variability and uncertainty;
  - Understand the significance of aleatory influences and epistemic influences separately;
  - Understand the probability of "failure" (the probabilistic weight of scenarios having zero or negative margin) at least semi-quantitatively;
  - Understand the main drivers (particular conditions under which margin is high or low), pointing to
    - stochastic variations, control of which would increase margin,
    - information that needs to be obtained in order to reduce epistemic uncertainty.

This definition does not address whether the analysis is good or poor: only whether it is carried out in a way that culminates in a probabilistic characterization of "margin" in a given issue space.

The essence of risk-informed is to create a basis for resource allocation (by licensee and by regulator) that does the best job we know how to do, consistent with our state of knowledge and institutional constraints. The analysis must be geared to supporting conclusions about which scenarios are more important than others, and how much more important, and how beneficial it would be to add preventive or mitigative measures beyond what is already there.

Degrees of Risk-Informed

Following is a first-cut allocation of un-normalized weights to the lineaments of risk-informed. The following are definitions of second-tier nodes on a value tree, where the first-tier node is "Risk-Informed."

Scope of Modeling

Formulation of a sufficiently broad issue space to comprise the scenarios that bear on the current decision, and the definition of figures of merit whose quantification brings everything relevant into the decision process. To get a high index in this area, the scope should include not only regulatory acceptance criteria, but also risk metrics, including impact of the subject change on operations. We will not address monetary costs, but we will try to address safety benefits and decrements as well as operational benefits and decrements (probability and severity of accidents, outages, derating, …).

Fidelity of Scenario Set Construction to Reality (including treatment of aleatory variables)

The whole point of being "risk-informed" is to be able to say that some scenarios are more important than others, or to say that a given set of scenarios has high or low absolute importance. Therefore the modeling of the scenarios within the defined scope is of central importance. Event trees and fault trees are better than reliance on a simple surrogate. But in order to get the highest score in this area, an analysis must simulate a set of time histories. It is generally understood that greater fidelity is achievable with event tree analysis than with analysis of a prescribed design-basis accident; but it is somewhat less widely appreciated that typical event tree analysis

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[3] Typically, these will be performance metrics in terms of which system success and system failure can be defined.

[4] Margin will be a pdf of where the performance metrics stand with respect to acceptance criteria (or, in the case of load metrics, how often they exceed the capacity of the SSCs subjected to those loads).

discretizes the scenario modeling in ways that cannot but be distortive, obliging the analysts to try to make sure that those distortions are "conservative." An example of this is acknowledged in a current draft of an NEI white paper: [24]

PRA models are approximate and are constructed by creating a discrete set of scenarios that encompass a range of scenarios. This is typically done in a bounding manner (e.g., choosing an initiating event that is bounding in terms of required plant and operator response to represent a group) to limit complexity in the PRA and to make the process more manageable. Thus there is a bias that is not readily measurable. While this may not be a significant issue for internal events, because the methods are relatively mature, characterizing these uncertainties can be resource intensive for fire and seismic PRAs due to lack of mature consensus methods and the larger associated uncertainties. [24]

Quantifying figures of merit by simulating a large number of time histories is still a research topic, because it is computationally intensive; so the grade in this area must logically depend on how good a job is done, which is a function of how many simulations are performed and how cleverly they are chosen.

Treatment of Uncertainties

One reason we do multiple simulations is to capture the effects of stochastic variations. Another is to be able to capture the effects of scenario phenomenology on component behavior.

To get the highest marks in this area, the treatment of epistemic uncertainty must be not only comprehensive, but open-minded. The decision should not be sensitive to choice of probability distributions on epistemic variables unless those distributions are largely determined by available evidence that has been brought to bear within a Bayesian framework using likelihood models that are appropriately open-minded (e.g., they do not pool trials that are not really exchangeable).

Figure 2 below provides a sort of value tree for "risk-informed." A true "value tree" would have weights summing to one; as noted earlier, these sum to 11.

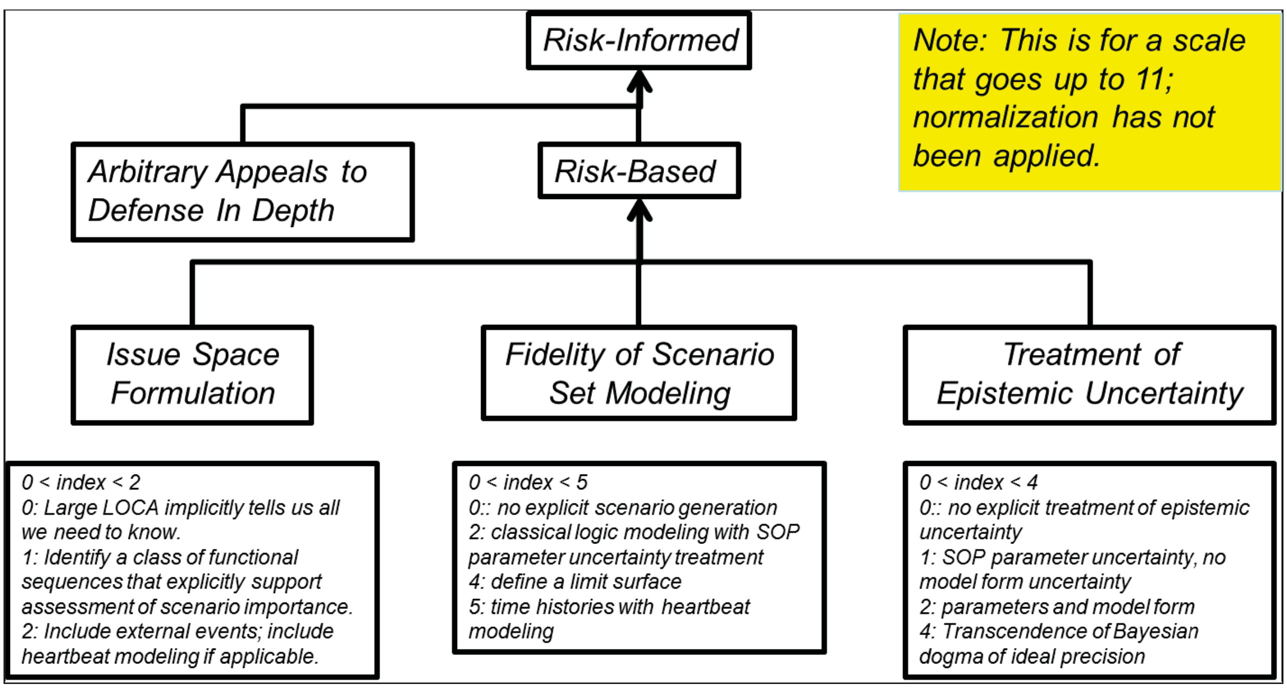


Figure 2. Value Tree for "Risk-Informed".

Examples

The table below notionally ranks certain fairly well-known examples of margin characterization, beginning with classical licensing. The CSNI examples were done over the last dozen years or so, trying to characterize margin in a risk-informed way, but starting from classical PSA models.

Table 1. Comparison of "Risk-Informed" Analyses.

Analysis	R-I Value	Comments
Classical Large LOCA	1	The argument was that if you are OK for Large LOCA you are probably OK for almost anything that would ever actually happen, so show you're OK for large LOCA, with "margin." Uncertainties are treated implicitly with analysis conservatism, in such a way as to support an "OK" conclusion, with high confidence.
BEPU LOCA	2	More realistic treatment of uncertainty than classical Large LOCA analysis, still on a narrowly defined scenario set, still embedded in the classical logic
CSNI analyses other than CSN [25] [26] [27]	4<answer<6	These looked at epistemic and stochastic uncertainties within a narrowly defined (assigned) scenario set.
R. Sherry *et al.* [28]	7	Analyzed Feed and Bleed within a space of stochastic variability and epistemic uncertainty and obtained interesting results.
V. Rychkov [29]	7.5	Analyzed Feed and Bleed for a French plant; somewhat like Sherry et al, but explicitly used the limit surface concept
Damage Domain Approach (Consejo de Seguridad Nuclear) [30]	8	This analysis worked to identify the limit surface (or, from their point of view, define the domain in parameter space within which "damage" occurs). Their results are limited by their simulation tools, but key results are not limited to details of probability density functions. They did a very careful job of thinking about the scope and details of generating their scenario set.
NRC PTS Study [31]	8	Extensive treatment of uncertainties and scenarios.

2.2 Industry Applications

Advanced safety analysis focuses on modernization of nuclear power safety analysis tools (including methods and data); implementing state-of-the-art modeling techniques; taking advantage of modern computing hardware; and combining probabilistic and mechanistic analyses to enable a risk-informed safety analysis process. The modernized tools will maintain the current high level of safety in our nuclear power plant fleet, while providing an improved understanding of safety margins and the critical parameters that affect them. Thus, the set of tools will provide information to inform decisions on plant modifications, refurbishments, and surveillance programs, while improving economics. The set of tools will also benefit the design of new reactors, enhancing safety per unit cost of a nuclear plant.

Risk-informed approaches provide a technical basis for understanding and managing hazards (i.e., safety risks). In addition, risk-informed approaches can be used to estimate costs (i.e., economic risks) to support safety decisions. While the focus of advanced safety analysis is on "facility" safety, it should be noted that these facilities are managed by diverse organizations (i.e., the nuclear industry, the Department of Energy (DOE), and associated oversight organizations). The benefits to be derived from the RISMC products will be applicable to all three groups.

The primary purpose of industry applications in advanced safety analysis is to demonstrate advanced risk-informed decision making capabilities in relevant industry applications. The end goal of these activities is the full adoption of the RISMC tools by industry applied to their decision making process.

The four elements of the above proposition are further explored below:

(a) Demonstrate

- Provide confidence and a technical maturity in the RISMC methodology (essential for broad industry adoption)
- Strong stakeholder interaction required
- Address a wide range of current relevant issues (see also item (d))
- Three phase approach
    (1) Problem definition (3-6 months)
    (2) Early Demonstration (eDemo) (limited scope) (6-12 months)
    (3) Complete Application and Validation (Long Term- Methods, Tools, Data) (1-5 years)

(b) Advanced

- Analyze multi-physics, multi-scale, complex systems
- Use of a modern computational framework
- A variety of Methods, Tools, and Data can be utilized (e.g. use of legacy tools and state-of-the-art tools)
- Be as realistic as practicable (with the use of appropriate supporting data)
- Consider uncertainties appropriately and reduce unnecessary conservatism when warranted

(c) Risk-Informed decision making capabilities

- Use of an integrated decision process
- Integrated consideration of both risks and deterministic elements of safety

(d) Relevant industry applications

- There are four Industry Applications (IA) carefully selected to cover a wide range of current industry issues (in order of importance):
    IA1 – Performance-Based ECCS Cladding Acceptance Criteria
    IA2 – Enhanced External Hazard Analyses (multi-hazard)
    IA3 – Reactor Containment Analysis
    IA4 – Long Term Coping Studies/FLEX

The focus of this report is on IA1, with emphasis on the first phase of the demonstration approach: Problem Definition. In the next chapters we will define the IA1 problem with an industry perspective, and debate its merits under a margin management point of view of decision making for the plant owner/operator.

3. PROBLEM STATEMENT – PLANT MODEL SELECTION

The RISMC toolkit is now sufficiently matured to offer a potential solution to the LOCA problem and provide to the plant operator a vehicle to manage the margins and inform decisions when compliance with 10 CFR 50.46 is challenged by changes in the operational envelope.

This is the driver behind the RISMC Industry Application 1 (IA1) and justification of this project called Risk Informed Margin Management (RIMM), [21] where margin is here relative to the 10 CFR 50.46 rule which is expected to be amended in 2016. The industry will need to comply with the new rule within the following four to six years (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 primary purpose of this report is to first "define the LOCA problem", then to lay out a roadmap to demonstrate application of the RISMC methodology. [20]

Considering most of the Large Break LOCA Analysis of Record (AORs) in the US are already licensed following the BEPU approach (RG 1.157 [5] and RG 1.203 [6]), the focus in this report is dedicated to those types of analyses. Moreover the new rule rollout plan assumes that plants currently operating with the Appendix K method as a licensing basis are the one retaining most of the margin in analytical spaces and will be able to comply the new rule by transitioning to BEPU methods. Moreover considering Appendix K plants re-analysis will require most of the effort, those are the last to be updated in the plan. A current industry estimate is that about half of the LOCA analyses will only need post-processing of existing results, and the other half will need new analyses.

Note that intermediate and small break LOCA scenarios are also significantly impacted by the new rule. However, to this day, there is no SBLOCA Analysis of Record (AOR) that follows the best-estimate approach and Intermediate Break LOCA scenario is not considered in the analyses on the basis that is considered not limiting.

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 cursory review of 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 Design
- Fuel Rod Design
- Containment
- Fluid systems

A review of the assumptions made of the above disciplines is presented in the sections below.

3.1 Figures Of Merit

The figures of merit (FOMs) are the relevant search parameters tied to the acceptance criteria of the rule. The proposed rule [16] [17] [18] [19] would replace the prescriptive analytical requirements of the previous rule (PCT<2200°F, MLO<17%, etc.), [1] [2] [3] [4] with performance-based requirements.

Quoting the rule:

"To demonstrate compliance with the requirements, ECCS performance would be evaluated using fuel-specific performance objectives and associated analytical limits that take into consideration all known degradation mechanisms and unique features of the particular fuel system, along with an NRC-approved ECCS evaluation model."

The proposed rule would apply to all fuel designs and cladding materials. The proposed rule would define two principle ECCS performance requirements:

- *Core temperature during and following the LOCA does not exceed the analytical limits for the fuel design used for ensuring acceptable performance.*
- *The ECCS provides sufficient coolant so that decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.*

[…]

The current requirement to maintain the calculated total cladding oxidation below 17 percent would be replaced with a requirement to establish analytical limits on peak cladding temperature (PCT) and integral time at temperature (ITT) that correspond to the measured ductile-to-brittle transition for the zirconium-alloy cladding material.

The Draft Regulatory Guide DG-1263 [32] defines an acceptable analytical limit on peak cladding temperature and integral time at temperature for the zirconium-alloy cladding materials tested in the U.S. Nuclear Regulatory Commission's (NRC's) loss-of-coolant accident (LOCA) research program. This analytical limit is based on the data obtained in the NRC's LOCA research program.

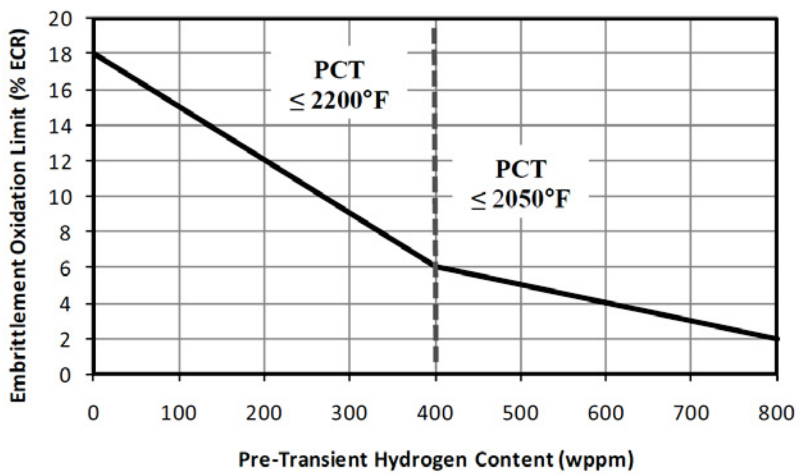


Figure 2. An acceptable analytical limit on peak cladding temperature and integral time at temperature (as calculated in local oxidation calculations using the CP correlation (Ref. 11))

Figure 3. Figure 2 from [14].

Referring to DG-1263 for example the analytical limit presented in Figure 3 will substitute the 17% limit. The hydrogen content (ppm) depends on the burnup value and material characteristic of the cladding, i.e. performance to embrittlement under irradiation for a specific cladding alloy.

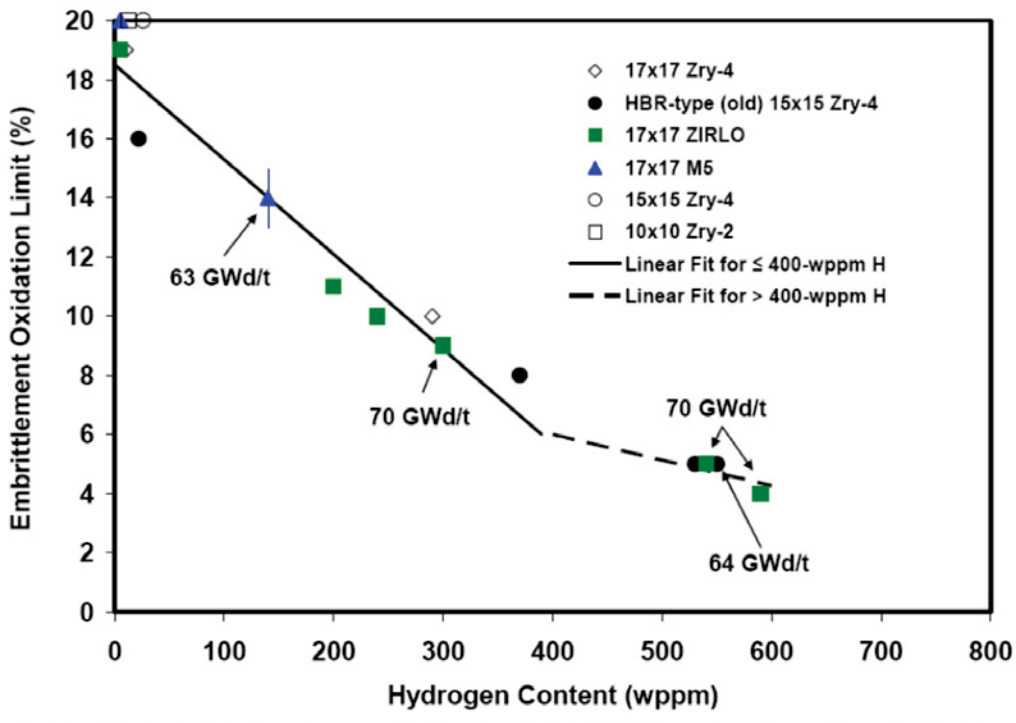


Figure 1. Ductile-to-brittle transition oxidation level (CP-ECR) as a function of pretest hydrogen content in cladding metal for as-fabricated, prehydrided, and high-burnup cladding materials. Samples were oxidized at ≤1,200 °C ±10 °C and quenched at 800 °C. For high-burnup cladding with about 550-wppm hydrogen, embrittlement occurred during the heating ramp at 1,160–1,180 °C peak oxidation temperatures (Ref. 8).

Figure 4. Figure 1 from DG-1263. [32]

The ductile-to-brittle threshold, defined in Figure 4, is an acceptable analytical limit on integral time at temperature (as calculated in local oxidation calculations using the Cathcart-Pawel (CP) correlation. [33]

The other Draft Regulatory Guides, DG-1261 [34] and DG-1262 [35] describe experimental techniques that are acceptable to the U.S. Nuclear Regulatory Commission (NRC) for determining these limits in general.

Note that the temperature limit (at least when the pre-transient hydrogen content is less than 400 wppm) is still the same, i.e. 2200°F. However the margin to embrittlement significantly decreases as the fuel is irradiated in the core and the cladding hydrogen concentration increases.

As a result of local oxidation, a measure of time-at-temperature is anticipated to be the controlling FOM under the new rule. In general term the two criteria ECR and PCT should be treated jointly. Further discussions of FOMs can be found in Section 6.3.

3.2 Core Physics Reference Design

Core design provides the state of the core at a given time in the cycle. A core design tool is used to compute core isotopic depletion. The calculation is a function of the specific out-of-core and in-core fuel management strategies. Out-of-core involves decisions such as cycle length, stretch out operations, feed fuel number, fissile enrichment, burnable poison loading and partially burnt fuel to reinsert, for each cycle in the planning horizon. In-core fuel management is based on decisions associated with determining the loading pattern, control rod program, lattice design, and assembly design presented.

Core design data is therefore needed to define the distribution of power and burnup in the reactor core at the time when the LOCA is postulated to occur.

Core design data is also utilized to determine the spectrum of axial power shapes and radial power distributions that need to be assumed in the LOCA analysis. The possible power shapes are a function of the reactor control logic adopted in the plant. For instance if the plant is subjected to load following maneuvers, the power oscillations which results from the following xenon transient are controlled by monitoring the Axial Offset (AO). These limits are defined in the Core Operating Limits Reports, like the example in Figure 5.

Moreover a LOCA analysis should also cover for the possibility that the plant was going through a power maneuver before the event. The resulting control rod movement, xenon transient, etc. results in significantly higher peaking factors than their baseload value and up to the Tech Spec limit. These are sometimes referred to as ‘augmented’ peaking factors.

The resulting power shape and peaking factors that may occur during this transient or similar Anticipated Operational Occurrences (AOOs) need to be assumed such that the space of possible axial power shapes can be considered in the LOCA analysis.

Core design data therefore feeds into setting up the core power distribution in the core as modeled by the LOCA simulator. Core design data also feeds into the fuel rod design analysis. Based on the relevant FOMs (PCT and ECR), it is possible to employ traditional (operator-split) methodologies, able to identify a reasonable number of limiting assemblies and critical moment of the cycle.

The new proposed approach modifies the analysis strategy, in order to take into account the effects of the burn-up rate in the identification of the limiting assemblies and cycle. Indeed, the maximum temperature and oxidation of the cladding must be casted as function of the burn-up.

The combination of burn-up and temperature in the definition of the limiting conditions leads to an additional requirement for the analysis of LOCA scenarios: detailed burn-up information. Hence, it may be desired to employ a new coupling strategy in order to fully assess the multiphysics characteristics of this LOCA problem.

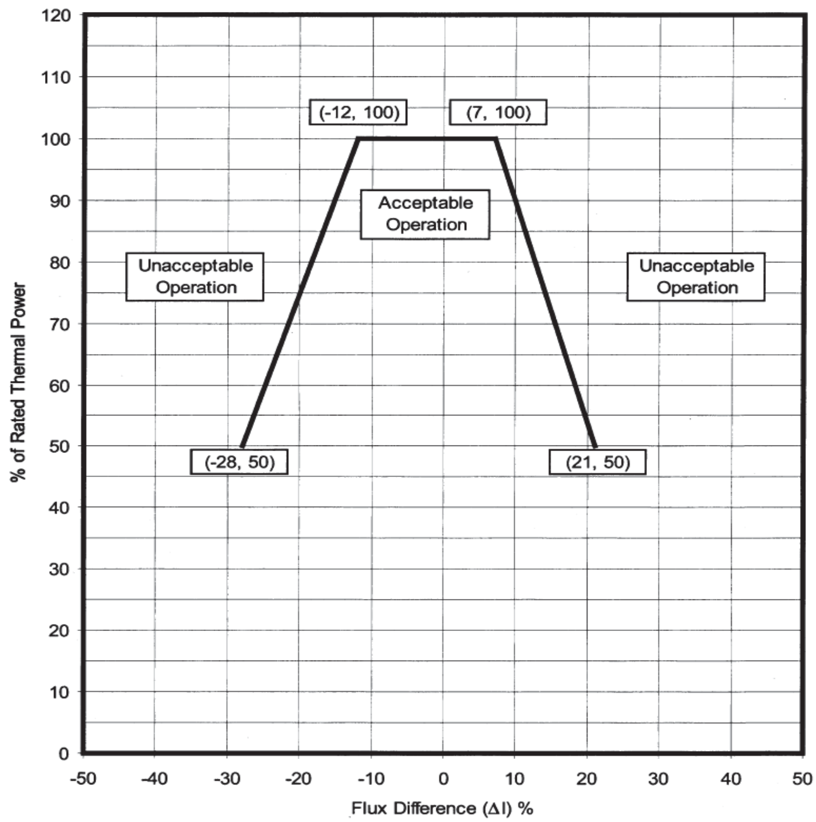


Figure 3
Axial Flux Difference Acceptable Operation Limits as a function of Rated Thermal Power (RAOC)

Figure 5. Example of AO Control (Watts Bar Nuclear Plant Unit 1 Cycle 13 Core Operating Limits Report (COLR). [36]

In order to obtain detailed burn-up information, a core design simulation strategy needs to be employed. Figure 6 shows the calculation flow for an exhaustive neutronic reactor design strategy. The core design is composed of several consequential steps, employed through as many different calculation methodologies (simulation codes).

The first phase is related to the generation of the homogenized neutron cross-sections, which are going to be used in the successive reactor calculation. The computation of these parameters is commonly performed through lattice codes. Simplistically, their principal goal is related to the computation of a detailed neutron flux in the core (2-Dimensional approximation). The calculation can be performed either considering individually all the unique assemblies constituting the reactor core (taking in account all the different compositions, geometry discontinuities, neighbor assemblies, etc.) or modeling the whole reactor core (exploiting eventual symmetries, in order to reduce it at least to ¼ core). The flux solution is then used to perform an energy-space collapsing to compute the homogenized neutron cross sections that will be used in the sub-sequential reactor calculation. Each lattice calculation is performed considering all the different combination of the state variables that characterize the reactor (burn-up, moderator temperature, boron concentration, etc.). The final homogenized cross section will be therefore available in tables tabulated versus the above-mentioned variables.

The second phase is related to the reactor calculation that approximates the reactor as constituted by homogenized regions (the reactor is seen as Cartesian or hexagonal regions where the physical property of the medium are considered homogenous). In the industry, neutronics packages already have an imbedded thermal-hydraulic model.

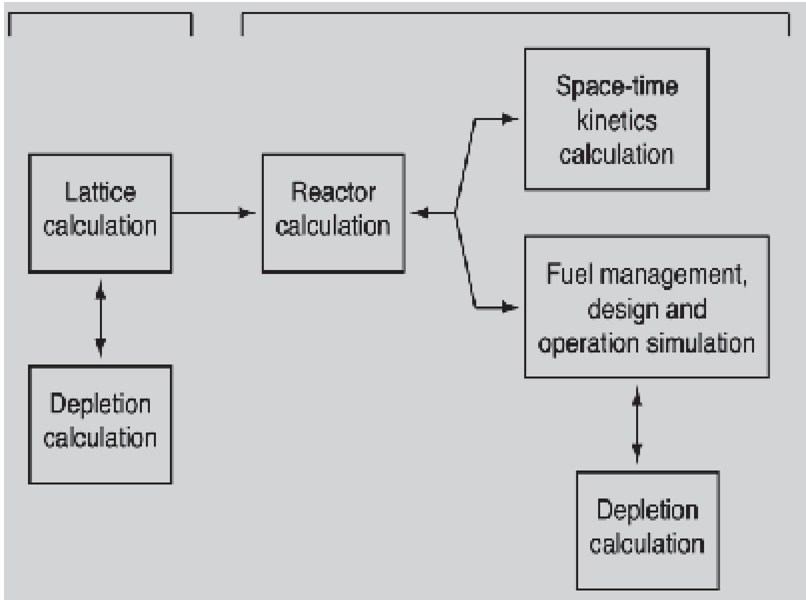


Figure 6. Core Design Strategy Scheme.

In the following paragraphs, the simulation packages that have been chosen are briefly described.

Lattice Code: HELIOS-2

The software HELIOS-2 (Studsvik ScandPower), [37] a well-known lattice code in the scientific community, employs two different methodologies for the solution of the transport equation, the method of the characteristics and the collision probability solver. Resonance self-shielding is calculated via the subgroup method, with a transport-based Dancoff calculation. The predictor-corrector method is used for depletion, and the depletion path allows arbitrary state changes, generalized decay capabilities, and branch-off calculations at any point in the solution path (in order to compute the flux solution for different combination of state variables).

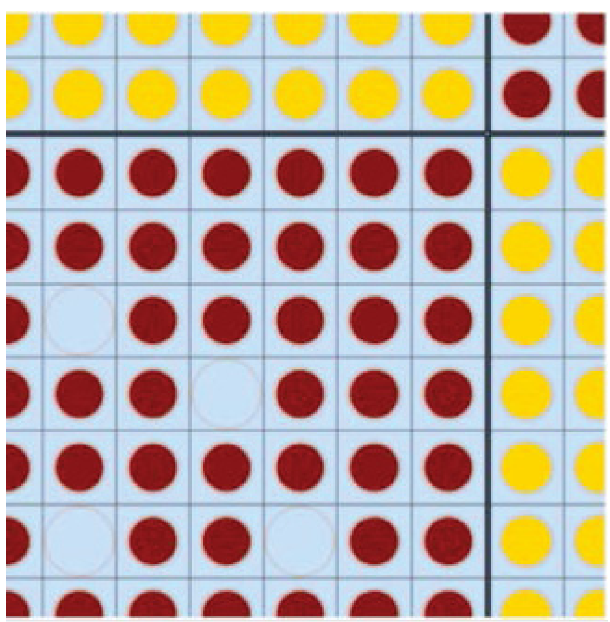


Figure 7. HELIOS-2 PWR Fuel Assembly Model.

HELIOS-2 has been validated against measured critical experiments, continuous-energy Monte Carlo calculations, and international isotopic benchmarks, delivering good accuracy for traditional, non-traditional, and experimental fuel designs. As an example, Figure 7 shows a fuel assembly modeled in HELIOS-2.

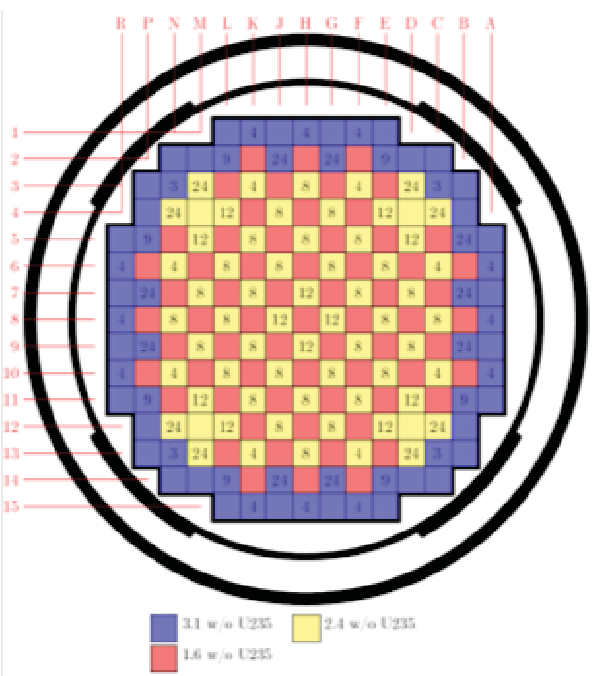
Reactor Physics Code: PHISICS

The PHISICS (Parallel and Highly Innovative Simulation for the INL Code System) code toolkit [38] is being developed at the Idaho National Laboratory. This package is intended to provide a modern analysis tool for reactor physics investigation. It is designed with the mindset to maximize accuracy for a given availability of computational resources and to give state of the art tools to the nuclear engineer. This is obtained by implementing several different algorithms and meshing approaches among which the user will be able to choose, in order to optimize his computational resources and accuracy needs. The software is completely modular in order to simplify the independent development of modules by different teams and future maintenance. 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 takes advantage of multiple computer cores (10 to 100 cores). In addition, the package is coupled with the system safety analysis code RELAP5-3D. [39]

Core Design Reference Plant

The reference plant, which has been chosen for the purpose of this project, is a typical Pressurized Water Reactor. The modeling of this reactor is based on the "Benchmark for Evaluation and Validation of Reactor Simulations (BEAVRS)." [40] BEAVRS is a detailed PWR benchmark containing real plant data for assessing the accuracy of reactor physics simulation tools for the first 2 operational cycles.

Figure 8 shows the radial layout of the reactor, including the reflector and the pressure vessel and the key plant parameters, respectively.



Reactor Core layout

Key plant parameters

Core Lattice	
No. Fuel Assemblies	193
Loading Pattern	w/o U-235
Region 1	1.61 %
Region 2	2.40 %
Region 3	3.10 %
Control	
Control Rod Material	Ag-80%, In-15%, Cd-15%
No. Control Rod Banks	57
No. Burnable Absorber Rods in Core	946
Burnable Absorber Material	Borosilicate Glass, 12.5 w/o B_2O_3
Performance	
Power	3411 MWth
Operating Pressure	15.51 MPa
Isothermal Coolant Temperature	564.82 K

Figure 8. BEAVRS Benchmark. [40]

Multi-Cycle Strategy

In order to assess the compliance of the existing power plants to the new rule, the LOCA accident scenario would be initiated from equilibrium cycle conditions. Hence, the reactor evolution needs to be followed for several operational cycles, until reaching a "reference" equilibrium cycle. Simplistically, from a loading point of view, the equilibrium cycle can be viewed as the cycle from which the fuel-reloading pattern is almost constant (i.e. same composition and spatial loading of the fuel batches). In this study, we assume the equilibrium cycle is reached after the 4th reloading (the 5th cycle is considered to be equivalent to the equilibrium cycle).

The PHISICS code is currently coupled with the Thermal Hydraulic (TH) code RELAP5-3D. For the first 4 cycles of the operation, the TH model of the reactor is going to be performed considering the reactor core only (without primary and secondary system). This choice has been taken since the first 4 cycles are used to compute the exposure history of the assemblies but are not active part of the LOCA simulation. 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 in the 5th cycle, the first four cycles are simulated using a core channel per fuel assembly will be modeled (193 in total). The radial reflector is modeled as a bypass channel (6% of the mass flow). Figure 9 shows the RELAP5-3D nodalization for the first 4 cycles.

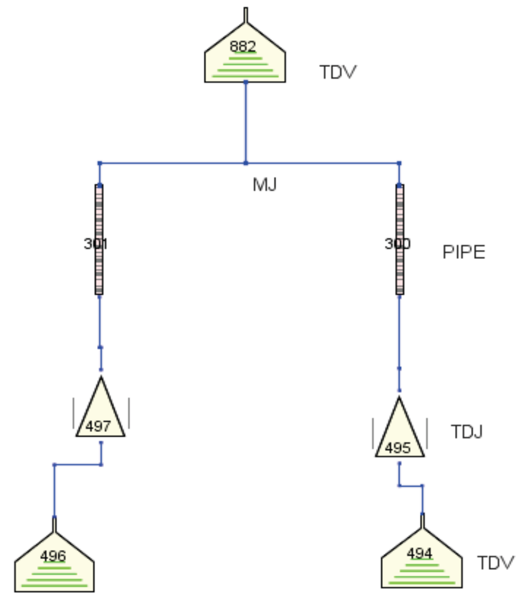


Figure 9. RELAP5-3D Nodalization.

The BEAVRS benchmark provides data for the first 2 cycles only (1 reloading pattern); for the sub-sequential 3rd, 4th, and 5th cycles a typical reloading pattern for large PWR is going to be used (see Figure 10).

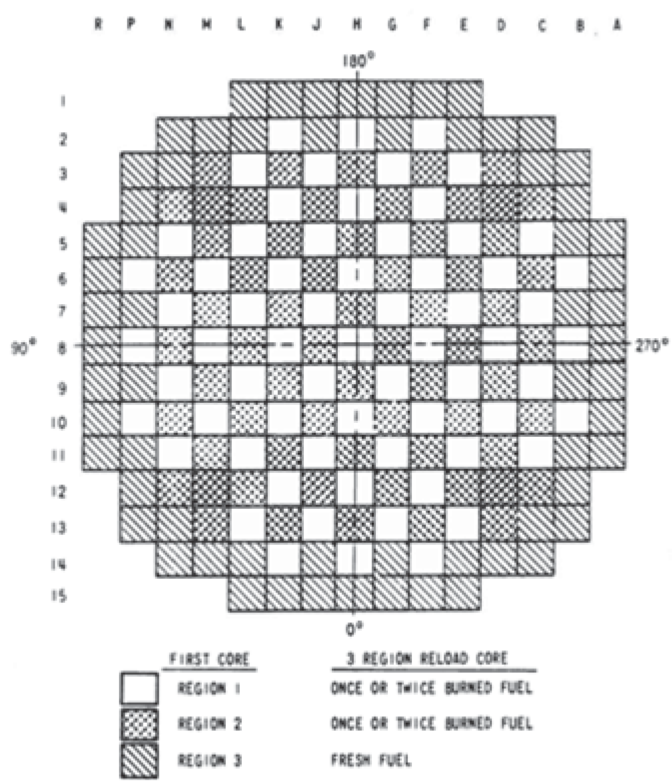


Figure 10. PWR Reload Pattern.

LBLOCA Transient Initialization

As mentioned in the previous sections, the 5th operational cycle will be the one at which the LOCA scenario is going to be initiated. The 4th cycle burn-up data will be used as initial conditions for the LOCA scenario analysis. In order to reduce the computational resources needed, the TH modeling is going to be simplified, passing from 193 to 6 core channels (see also Figure 13):

- The core will be divided into 3 burn-up regions, each of them divided in hot and average (6 core channels in total);
- For each of these regions, an at-risk assembly will be modeled separately.

A critical step in setting up a LOCA Evaluation Model is the LOCA transient initialization. A LOCA is initialized assuming the plan is operating at full power in state within the envelope defined by the tech spec limits.

The LOCA analysis requires the simulation of the LOCA transient from that state. In practice a LOCA simulation with RELAP5 requires restarting the calculation from a previously obtained steady state simulation. The steady-state simulation is intended to re-create the steady-state condition for the plant.

Steady-state acceptance criteria are established to ensure that computed fluid and core conditions are an accurate representation (within prescribed tolerances) of the target plant conditions characteristic of the design. The plant operator typically provides the target conditions to the analyst.

The steady-state acceptance criteria can be listed as a checklist and the verification and input adjustment can be automated.

Among other parameters, an important aspect of the transient initialization is to consider all permitted axial power shapes and peaking factors, assuming the plant was load following and subject to power maneuvers before LOCA is postulated to occurs. This information is also generated as part the ‘core design’ discipline discussed in this report.

3.3 LBLOCA Transient Model

For the initial demonstration, a RELAP5 plant model is selected for a four loop pressurized water reactor designed by Westinghouse. The RELAP5 model is a detailed representation of a typical four loop PWR power plant, describing all the major flow paths for both primary and secondary systems, including the main steam and feed systems. Also modeled are primary and secondary power operated relief valves (PORV) and safety valves. The emergency core cooling system (ECCS) was included in modeling the primary side, and the auxiliary feedwater system was included in the secondary side modeling. Figure 11 shows the diagram of the RELAP5 model of the primary system of the typical four loop PWR. Note that only two primary coolant loops are shown in Figure 11. However, each of the four primary coolant loops is represented in the RELAP5 model. The loops are designated as A, B, C, and D. Each modeled loop contained a hot leg, U-tube steam generator, pump suction leg, pump and cold leg. Attached to each cold leg is a low pressure injection connection and an accumulator with its associated piping. A high pressure injection connection is also attached to each cold leg. Heat structures are added to each volume in the primary loops to represent the metal mass of the piping and steam generator tubes. The reactor vessel model includes the downcomer, lower plenum, core, core bypass, upper plenum, and upper head. The following leakage paths are represented in the vessel model: downcomer to upper head, vessel inlet annulus to upper plenum, and upper plenum to the upper head by way of guide tube. Heat structures represent both external and internal metal mass of the vessel as well as the fuel rods. Decay heat was assumed to be at the ANS standard rate. [41]

The secondary system of the plant is also modeled. The steam generator secondary side model represents the major flow paths in the secondary and includes the downcomer, boiler region, separator and dryer region, and the steam dome. Due to the modeling constraints, the steam generator secondary separators and dryers are modeled with a single hydro-dynamic volume. Separation in the model thus takes place at a single elevation rather than at two locations (separator and dryer), as in the actual steam generator. The major flow paths of the steam line out to the turbine governor valves are modeled and are shown in Figure 12. Each line from the steam generator secondary out to the common steam header is modelled individually, and includes a main steam line isolation valve (MSIV), safety and PORV valves.

A simplified feedwater system is modeled, providing main and auxiliary feedwater to each steam generator. The control system models include: steam dump control system, steam generator level control, pressurizer pressure control system, and pressurizer level control systems, etc.

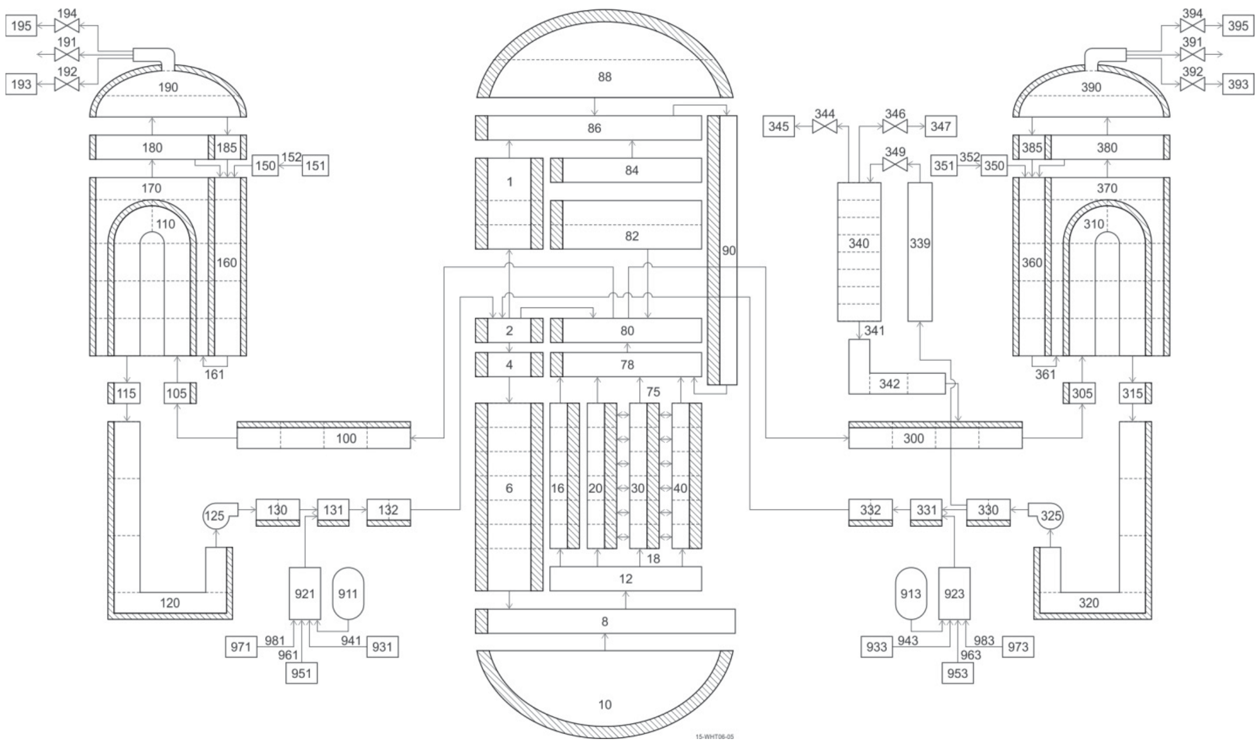


Figure 11. RELAP5 Model of the Primary System of a Typical Four Loop PWR.

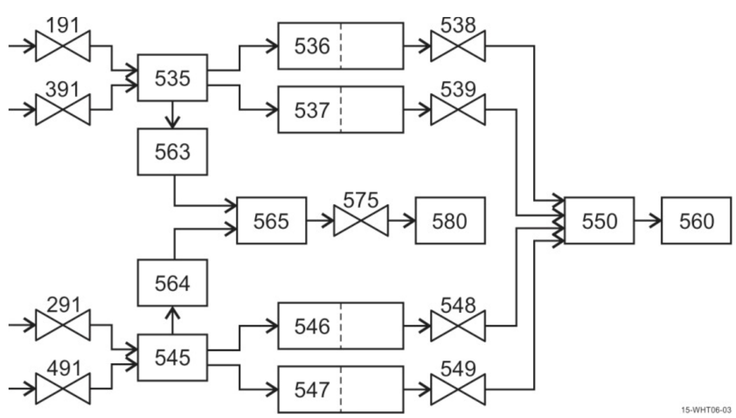


Figure 12. RELAP5 Model of the Steam line of a Typical Four Loop PWR.

The reactor core modeling will use different homogenization approaches for thermal fluid dynamics calculations and the heat conduction calculations in the solids. This approach will allow the RELAP5 cases to have a fast running time to facilitate the large number of RELAP5 runs required by the RISMC methodology. For the thermal fluids calculations, a multiple channel approach will be used, as illustrated in Figure 13. Specifically, the assemblies in a reactor core will be grouped together into various regions based on their burnup history. Two flow channels – one average channel and one hot channel – are built to represent each region. The channels are connected in the lateral direction to allow cross flows to be simulated. For the heat conduction calculations in the solids, one heat structure for each fuel assembly is built and connected to the average flow channel such that the average clad and fuel temperature for each fuel assembly can be calculated. Separate heat structures for the hot channel have also been built.

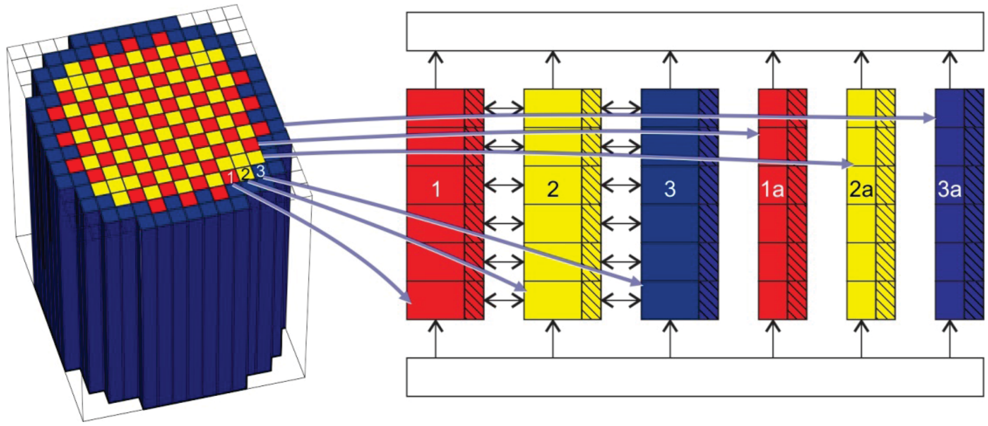


Figure 13. Illustration of Assembly Grouping and Homogenization in RELAP5 Calculations.

The RELAP5 model developed here can be used to perform simulations of various accident scenarios including large break loss of coolant accident (LBLOCA) which is the primary interest of this work. For LBLOCA scenario, the break is located at a cold leg of the PWR, as shown in Figure 14.

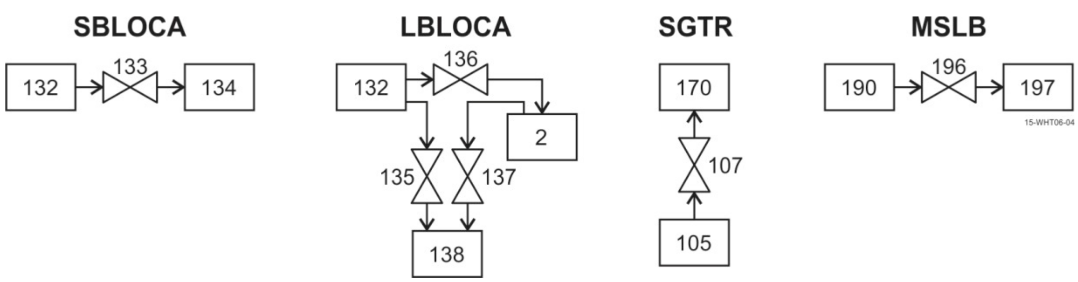


Figure 14. Illustration of Simulations of Various Accident Scenarios.

Following the initiation of LBLOCA, a fast depressurization of the primary system ensues. The consequent low pressure would activate the SCRAM signal to shut down the reactor. Water injection from the accumulators and later from the low pressure injection systems is activated to prevent core uncovering. The entire process lasts about 10 minutes. To be conservative, in our RELAP5 plant model simulation, the shutdown of the reactor following the initiation of LBLOCA is achieved through the negative reactivity feedback, rather than through the SCRAM of the reactor.

3.4 Fuels Performance Models

Fuel rod design data is necessary to define the state of the fuel pins modeled in the core. These calculations are performed by the fuel performance codes. The fuel performance codes receive inputs from core design for setting power shapes and burnup distributions in the fuel rod. Then the fuel rod designers need to ensure the integrity of the fuel is maintained during the operation of the plant in normal and offset conditions.

Ultimately the fuel performance code computes fuel pellet power and temperature distribution and rod internal pressure. These and other information are passed to the system code to initialize the status of the fuel pin before the LOCA transient.

The LOCA analysis is performed downstream of all the other analyses to confirm that safety limits criteria are not exceeded and in compliance with 10 CFR 50.46 acceptance criteria. The LOCA Analysis of Record (AOR) is designed to be applicable in several future fuel cycles and hold true even if core design and other plant parameters slightly change from cycle to cycle. The confirmation is part of the Core Reload Analysis Process.

Westinghouse generic bounding Reload Safety Evaluation Methodology called RSAC was introduced in 1985 and documented in the proprietary WCAP-9272. [42] The RSAC process is at the basis of many other methods and still used today.

The reload safety evaluation method from AREVA for B&W design was reviewed and approved in 2010 with the BAW-10179, Revision 8. [43]

The objective of the reload safety evaluation is to confirm the validity of the existing safety analysis, or Analysis of Record. The process is to confirm that the operational and design limits defined by the AOR still hold after the fuel reload. Small deviations are reconciled with simple evaluations that demonstrate margins are not reduced. Large deviations may trigger a full re-analysis.

Under the new rule this process (for the LOCA analysis at least) is expected to be revised. Also deviations will be more difficult to be reconciled under the new rule since it's not a simple PCT limit.

In the LOCA analyses, fuel rod performance models are indispensable in order to obtain accurate predictions on the three safety important figure of merits: PCT (Peak Clad Temperature), LMO (Localized Maximal Oxidation rate) and CWO (Core Wide Oxidation rate). Fuel performance models are traditionally divided into two categories: steady state model, and transient model. LOCA analyses need both models. The fuel performance capabilities can come from embedded fuel models inside a reactor system safety analysis code such as RELAP5, or through coupling with matured fuel performance codes such as FRAPCON, FRAPTRAN, even advanced codes currently being developed like BISON.

The steady state fuel performance models calculate the temperature, pressure, and deformation of a fuel rod as functions of time-dependent fuel rod power, burnup and coolant boundary conditions. The phenomena modeled by the code FRAPCON-3 [44] include:

- heat conduction through the fuel and cladding to the coolant;
- cladding elastic and plastic deformation;
- fuel-cladding mechanical interaction;
- fission gas release from the fuel and rod internal pressure; and
- cladding oxidation.

FRAPCON-3 is a Fortran 90 computer code that calculates the steady-state response of light-water reactor fuel rods during long-term burnup. The code is designed to perform steady-state fuel rod calculations and to generate initial conditions for transient fuel rod analysis by the FRAPTRAN computer code [45]. FRAPCON-3.5 is the latest version of FRAPCON-3, released in May 2014.

The transient fuel performance models calculate the transient performance of fuel rods during reactor transients and hypothetical accidents such as loss-of-coolant accidents, anticipated transients without scram, and reactivity-initiated accidents. The phenomena modeled by the code FRAPTRAN include

- heat conduction,
- heat transfer from cladding to coolant,
- elastic-plastic fuel and cladding deformation,

- cladding oxidation,
- fission gas release, and
- fuel rod gas pressure.

The initial conditions for the transient fuel performance models typically come from the steady state fuel performance models. For example, Burnup-dependent parameters for the FRAPTRAN code may be initialized from the FRAPCON-3 steady-state single rod fuel performance code. The Fuel Rod Analysis Program Transient (FRAPTRAN) is a Fortran language computer code that calculates the transient performance of light-water reactor fuel rods during reactor transients. FRAPTRAN-1.5 is the latest version of FRAPTRAN.

BISON [46] [47] is a finite element-based nuclear fuel performance code applicable to a variety of fuel forms including light water reactor fuel rods, TRISO particle fuel, and metallic rod and plate fuel. It solves the fully-coupled equations of thermomechanics and species diffusion, for 1D spherically symmetric, 2D axisymmetric or 3D geometries. Fuel models are included to describe temperature and burnup dependent thermal properties, fission product swelling, densification, thermal and irradiation creep, fracture, and fission gas production and release. Plasticity, irradiation growth, and thermal and irradiation creep models are implemented for clad materials. Models are also available to simulate gap heat transfer, mechanical contact, and the evolution of the gap/plenum pressure with plenum volume, gas temperature, and fission gas addition. BISON can be used for both steady state and transient fuel performance calculations.

RELAP5-3D [39] contains simplified fuel performance models. For steady state simulation, RELAP5-3D has default material properties for uranium dioxide, gap, and clad. However, the built-in material properties have many limitations. [48] For example, the uranium dioxide properties are for unirradiated fuel with an oxygen-to-metal ratio of 2.0 and 95% theoretical density. The effects of burnable poisons or other fissionable elements are not included. Irradiation strongly affects the uranium dioxide properties, as shown in the Figure 15, taken from the BISON theory manual. [46] Fink-Lucuta lines in Figure 15 are experimental data.

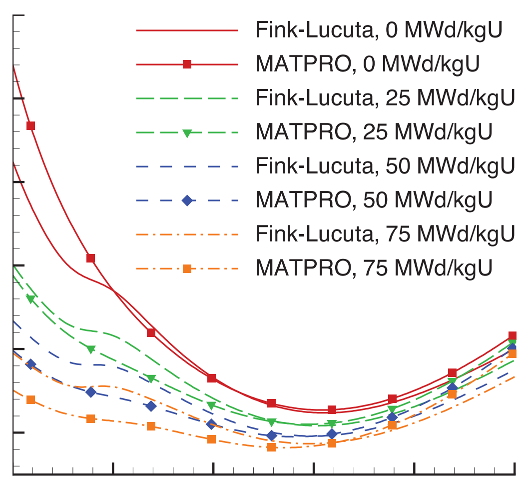


Figure 15. A Comparison of the Fink-Lucuta and MATPRO Empirical Models for the Thermal Conductivity of Full Density UO_2, as a Function of Temperature and Burnup. [46]

The only available built-in clad material is zircaloy. The zircaloy properties are for unoxidized metal only. The gap thermal properties in RELAP5-3D approximate those of a representative pressurized water reactor fuel rod at the end of an equilibrium cycle, and were calculated for a pressure of 4.1 MPa and a temperature of 300-3,000 K.

Knowing these limitations, RELAP5-3D allows users to input table data which can account for specific situations such as high burn-up or different types of materials. These temperature-dependent tables can be constructed according to empirical nuclear material models. Several of those models are described in the FRAPCON manual [44] and in the BISON theory manual. [46] The gap conductance treatment is more complex. If the embedded transient fuel performance models are not to be used, the user can iteratively adjust the gap thermal properties during steady state calculations to obtain a known or desired radial temperature distribution in the fuel rod; these values are then assumed to be constant. For the LOCA analysis, it is very important to obtain correct the steady state stored energy in the fuel rod before the LOCA transient begins. If users want to use the embedded transient fuel performance models, the table-type gap thermal properties cannot be used; instead, users can adjust the mole fractions of helium and other gases to obtain the desired radial temperature distribution in the fuel rod.

The transient fuel performance models in RELAP5-3D include dynamic gap conductance model, cladding deformation model, and metal-water reaction model. Both the gap conductance model and the cladding deformation model are based on simplified models from FRAP-T6. [49] The gap conductance model defines an effective gap conductivity based on a simplified deformation model generated from FRAP-T6. Major physical phenomena and geometry information include:

- The gas mixture thermal conductivity as a function of molar fractions of gas components and temperature;
- The offset of the longitudinal axis of the fuel pellets from the longitudinal axis of the cladding, as shown in Figure 16;
- Radial displacement of the fuel pellet surface due to thermal expansion and the fission gas induced swelling and densification;
- The radial displacement of the inner surface of the cladding due to thermal expansion, cladding creepdown, and elastic deformation caused by pressure difference across the clad inner and outer surfaces,
- The temperature jump distance terms which account for the temperature discontinuity caused by incomplete thermal accommodation of gas molecules to the surface temperature.

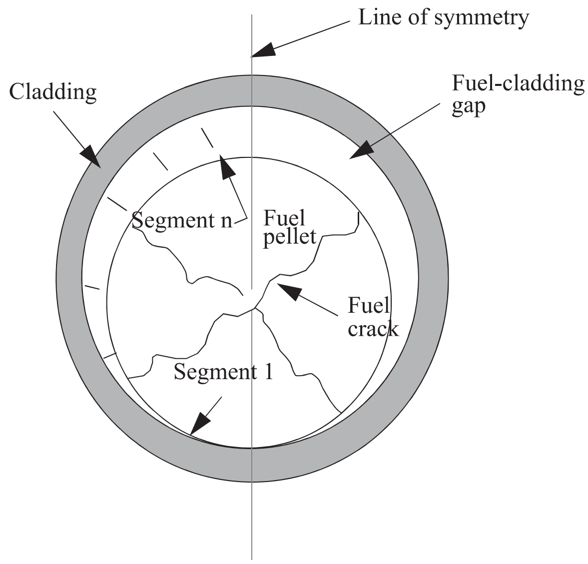


Figure 16. Segmentation at the Fuel-Cladding Gap for RELAP5-3D. [39]

The terms also account for the inability of gas molecules leaving the fuel and cladding surfaces to completely exchange their energy with neighboring gas molecules, which produces a nonlinear temperature gradient near the fuel and cladding surfaces.

The gap conductance model requires the following user inputs; many of those parameters should come from a steady state fuel performance code's results:

- Initial gap internal pressure, burn-up dependent value, from fuel performance code;
- Gap conductance reference volume, to obtain fuel pin outside pressure;
- Fuel surface roughness;
- Cladding inner surface roughness;
- Radial displacement due to fission gas-induced fuel swelling and densification, from fuel performance code;
- Radial displacement due to cladding creepdown, from fuel performance code.

An empirical cladding deformation model from FRAP-T6 has been incorporated into RELAP5-3D. The model may be invoked only in conjunction with the dynamic gap conductance model. The purpose of the model is to allow plastic deformation of the cladding to be accounted for in the calculation of fuel rod's cladding temperature during LOCA simulations; and to inform a user of the possible occurrence of rod rupture and flow blockage and hence the necessity of conducting more detailed simulations of the fuel rods' behavior. Thermal radiation heat transfer is included in the gap conductance calculation. The geometry information for the connected hydraulic volume such as the flow cross section area will be modified to account for the effects of fuel cladding swelling and rapture.

The metal-water reaction model in RELAP5-3D is normally used to model the reaction on the outside of the cladding. The metal-water reaction model can be coupled with the fuel rod deformation model for cylindrical heat structure geometries so that if a rod ruptures, the inside of the cladding can also react, although only at the elevation of the rupture (not over the entire rod length). The metal-water reaction heat source term for the cladding surface mesh point (outside and inside) is added into the first mesh interval on that surface. Although the metal-water reaction model was developed to model zirconium cladding oxidation in a steam environment, it has been generalized to model coolant-structure chemical interactions for which the parabolic rate equation applies. The metal-water reaction model calculates the thickness of the cladding annulus converted to oxide; however, it does not alter the thermal-physical properties of the cladding as the oxide layers develop. Similarly, although the model calculates the amount of hydrogen freed from each surface undergoing metal-water reaction, this hydrogen does not get included into the hydraulic equations, nor does the steam being consumed by the metal-water reaction get withdrawn from the hydraulic equations.

From the above review, we conclude that RELAP5-3D has limited fuel performance analysis capability for the demonstration LOCA analysis and flexible interfaces to bring additional steady state results from a standing-along fuel performance code. We envision four stages of coupling methods between system code and fuel performance code:

- Early demonstration run: use RELAP5-3D's internal fuel performance models with decoupled inputs from separate runs of FRAPCON-3.
- Realistic demonstration run: loosely couple RELAP5-3D with FRAPCON-3 and FRAPTRAN-1.5.
- Advanced demonstration run: loosely couple RELAP5-3D with BISON.
- Advanced demonstration run: strongly couple RELAP-7 with BISON.

RELAP5, FRAPCON, and FRAPTRAN are matured codes with many years' development and applications. They are ready for early demonstration of the RISMC methodology in ECC LOCA analysis with no or minimal code modification. The major works are in the preparation of input models and coupling these codes together. BISON is an advanced 3-D code which still is under development and validation stages. The development stage RELAP-7 is even several years behind BISON. When both advanced codes become ready, the unique strongly coupling capability will minimize the analysis uncertainties and potentially recover more margins with the application of RISMC methodology.

4. INDUSTRY APPLICATION EARLY DEMONSTRATION

Phase-II is the critical stage in the program because it's key in demonstrating the RIMM IEM concept and will provide sufficient information to develop a value proposition for industry stakeholder which may benefit from these developments. Once feasibility and value are demonstrated, the following phases will essential refine and further develop the concept and raise its technology readiness level to a point that can be directly absorbed in the market.

4.1 Integrated Evaluation Model (IEM) Framework

The analytical work required to demonstrate compliance with the 10 CFR 50.46 rule is typically performed by the vendors on behalf of the licensees (plant operators). Over the years, methodologies evolved from a deterministic Appendix K approach to Best-Estimate Plus-Uncertainty (BEPU) methods which spun off from the CSAU work in the 1980's. [9]

However, the identification of the "limiting case" has been accepted in the context that input and model uncertainties are sampled within realistic, albeit conservative ranges and combined via Monte Carlo (MC) techniques. Compliance is demonstrated by stating a 95/95 joint-probability statement from the sample of simulations with respect to the three main acceptance figure of merits (FOM): PCT, Maximum Local Oxidation and Core-Wide Oxidation. In the industry, this is typically referred to as the Wilks's approach, since the theoretical work stems from the 1940's work by Wilks on how to determine tolerance limits in the manufacturing industry. In practice this approach is a crude MC method with the minimum sample size required to stabilize the estimator of the 95th quantile.

There are many intrinsic limitations in this approach which are expected to escalate under the proposed new rule change because the FOM is much more complex multi-dimensional variable than a simple PCT or maximum local oxidation limit.

The recognized limitations in established industry methods are:

- There is little knowledge on why the so called limiting case is limiting. The applicant is simply interested in the final probabilistic statement that the ECCS design is acceptable under the rule.
- The result, specifically the 95/95 estimate, is strongly impacted by the limits of the sample size, seed issues and unquantified risk of exceedance of the regulatory limit. This is typically justified considering several layer of conservative biases embedded in the methodology
- There is no information or a simple method to perform global sensitivities. The sample size (statistics) is too small to provide a reliable answer
- There is limited information on which parameter is important
- Impact assessments on plant design changes and design optimization studies are extremely cumbersome, lengthy and unresponsive. Unless design changes are small
- There is limited information on the actual probability of exceeding a limit in a given design

Vendors and plant owners need to operate in a heavily regulated environment. The economics prevent deviations from well established procedures within the licensing basis of the Evaluation Models. The multi-physics problem is solved via operator splitting procedures. Interfaces across the disciplines and processes do not easily adapt to a new fully integrated method. The propagation of uncertainties across the various functional groups is addressed defining bounding assumptions at the interfaces which limit the possibility that the impact on an issue in a specific discipline crosses-over to the other physics.

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 those 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 RIMM project is expected to create value by anticipating these trends and focusing on developing a methodology that effectively addresses the limitations presented above. The primary goal is to explore an integrated approach for knowledge and uncertainty management, as illustrated in Figure 17. The resulting analytical tool is called RIMM Integrated Evaluation Model (RIMM IEM).

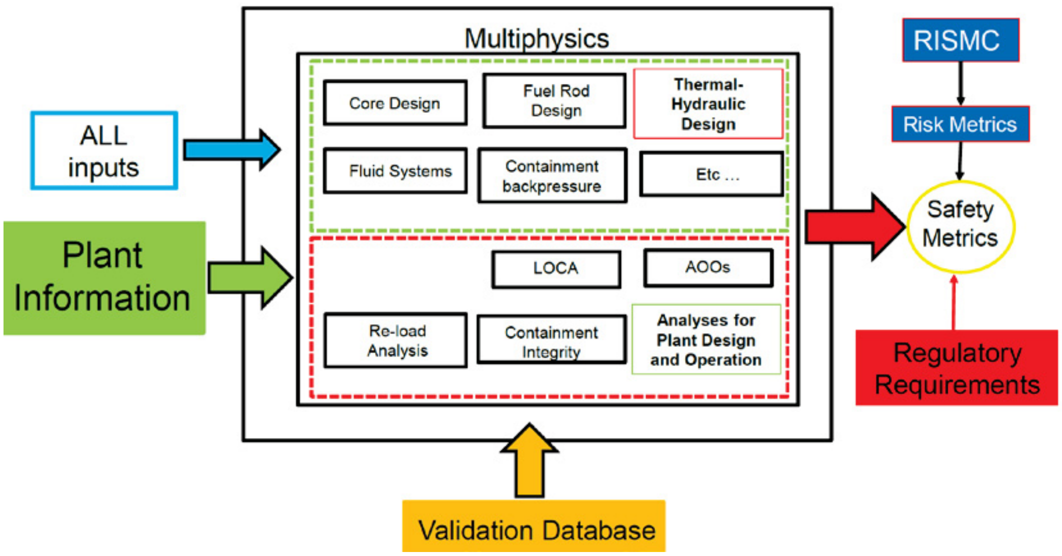


Figure 17. Flow Chart of the RIMM Integrated Evaluation Model.

The development of the Framework Demonstration represents a first proof of concept of the RIMM IEM. Most of the physical disciplines associated with a realistic Large Break LOCA (LBLOCA) Analysis will be approximated by Reduced Order Models (ROMs)[5]. The input and output data stream and uncertainty propagation techniques are prototyped at this stage. This will include the development of the emulators.

The high level functional requirements of the RIMM IEM can be summarized as follows:

1. Create a "multiphysics" knowledge database architecture which collects and organizes actual data sets and simulations results in a format suitable for the framework
2. Develop a representative PWR plant model which properly integrates functions covering thermal-hydraulics, core physics, fuel performance, and all the disciplines needed to complete a demonstrative LOCA analysis
3. Develop an Integrated Evaluation Model (the RIMM IEM) able to probe from the knowledge database and inform plant operators on their plant margins relative to the ECCS design acceptance criteria
4. Enable design parameters optimization studies within the envelope of acceptable results with respect to 10 CFR 50.46
5. Provide a responsive but robust analytical framework to support plant operators in addressing 10 CFR 50.46 reporting requirements or changes regulated under 10 CFR 50.59.
6. Build the methodology in an architecture that can fit within a risk-informed framework which is the ultimate goal of the RISMC

[5] The term 'emulators' refers to the probabilistic surrogate models that approximate a desired input/output relationship. Reduce Order Model (ROM) denotes simplified physical models (sometimes referred to as "glass box" models). These are used for two different purposes. For instance in the Framework Demo, we'll be building emulators of the ROMs.

The final product is amenable for consideration by both plant operators and vendors. At a minimum it will provide a common framework for managing knowledge across the industry stakeholders. A more detailed discussion on stakeholder engagement can be found in Section 5.

4.2 Realistic LBLOCA BEPU Analysis

Once the initial development of the Framework Demo is completed, the data from the ROM will be replaced by data obtained from actual simulators. Here is where the RELAP5 PWR model integrated with the core physics and fuel rod performance functions will be integrated into the Framework Demo.

The evolution of the LBLOCA analysis and how it is currently performed in the industry is described next. In a nutshell, once data is gathered and the NPP model developed, the analysis is fully automated with the simulation of several instances of LOCA scenarios which represent realizations randomly sampled following a crude Monte Carlo procedure. The procedure considers the most important sources of uncertainties which have been properly characterized with their own probability density function. About 40 uncertainty parameters are typically considered in the analysis. [12]

Figure 18 depicts the process. The vector x is the vector of randomly sample input settings, while the vector y is the vector of the outcomes, typically the target FOMs, like Peak Clad Temperature (PCT), (Maximum Local Oxidation (MLO) and Core-Wide Oxidation (CWO)).

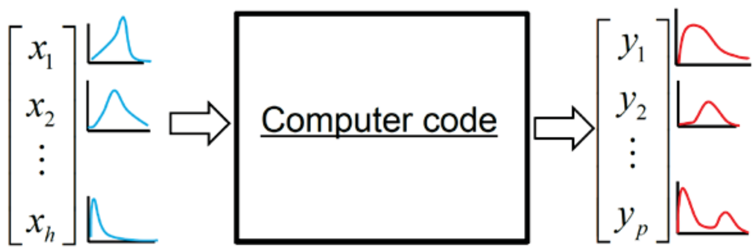


Figure 18. Monte Carlo Forward Propagation of Uncertainties.

The outcome of the analysis results in a sample of results which can be collected as shown in Figure 19. The red marks identify double-ended guillotine cases while the green triangles are split breaks which stretch down to an equivalent break size of 1.0 ft2 assumed to be minimum size of a large break.

The compliance is demonstrated by ensuring that the maximum PCT value, the maximum MLO value and maximum CWO value in the 124-set are below the acceptance criteria, which in the current rule are respectively 2200 oF, 17% and 1%. The procedure guarantees that a joint-probability of 95% on the three criteria is satisfied with 95% confidence.

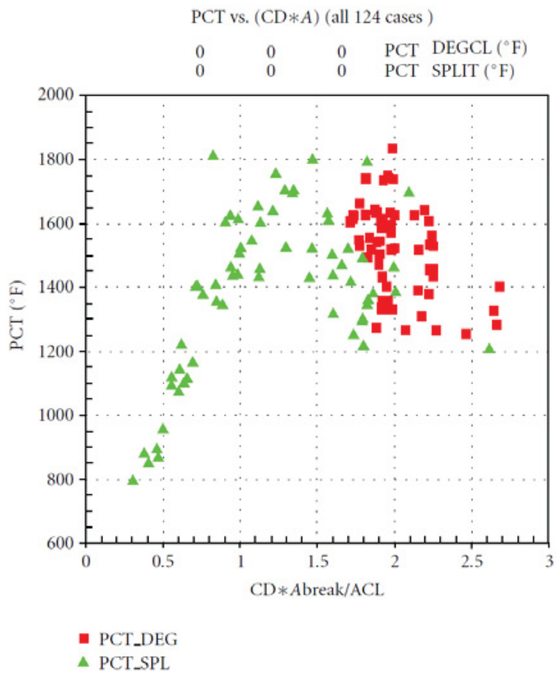


Figure 7: Peak clad temperature (PCT) from the ASTRUM 124 run set.

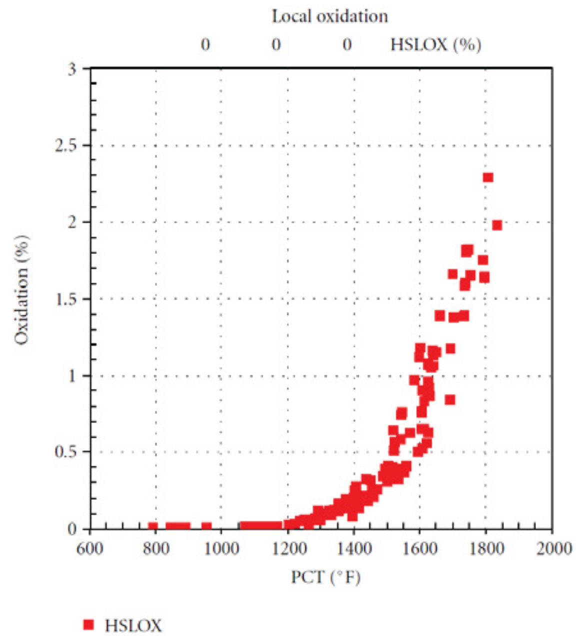


Figure 8: Oxidation and PCT from the ASTRUM 124 run set.

Figure 19. Figure 7 and 8 from Frepoli. [12]

The BEPU approach consists of several logic steps in the evaluation process. A few key steps include:

- Selection of accident scenario, safety criteria and the figure of merits (FOM) involved in the acceptance criteria;
- Identification and ranking of relevant physical phenomena and selection of appropriate thermal hydraulic parameters as well as development of their probability density functions based on safety criteria;
- Developing computer model for accident scenario;
- Random sampling of the selected thermal hydraulic parameters and plant configurations according to the probability density functions. The sample size (N) depends on the particular statistical approach used. N times RELAP5-3D runs will be performed with the random sampled input parameters;
- Post processing the output files of the N times RELAP5-3D runs to develop probability distribution functions of the FOM.

For the LBLOCA BEPU demonstration, a list of input parameters with uncertainty considered to be sampled will be assembled collaboratively between FPoliSolutions and the INL LOCA analysis team. The uncertainty analysis methodology to be used will be the nonparametric statistics method, which was originally presented by Wilks. In this approach, all the uncertainty parameters are sampled simultaneously in each RELAP5-3D run. Using the well-known Wilks formula, the minimum number of sample size can be determined for a desired population proportion with a certain tolerance interval. For example, if we are interested in determining bounding values (95th percentile with 95% confidence level) of the peak clad temperature and maximum local oxidation rate, the number of RELAP5-3D runs is found to be 59.

5. STAKEHOLDER ENGAGEMENT

The participation and collaboration of the nuclear industry is fundamentally important for the execution of the Industry Applications vision: to address relevant issues, realistically, and provide the tools for the industry to evaluate and make long term decisions in an effective manner.

Hence we envision industry participating and providing critical information to the demonstration defined in the previous chapters.

5.1 Stakeholder Selection Strategy

Below, we discuss the considerations used in the selection of a fuel vendor and/or an owner-operator stakeholder for the collaboration demonstration of the RIMM IA1 methodology and tool.

Existing RELAP5 Model

Early in the INL RIMM IA1 project the team decided to use the RELAP5 system thermal-hydraulic code for demonstrating the methodology. RELAP5 LOCA simulations will be used to build the database for the RIMM tool for the IA1 application. Section 5.3 has more information on this consideration.

Westinghouse 3-Loop or 4-Loop Plant

The project team also decided to target a Westinghouse 3-loop or 4-loop pressurized water reactor (PWR) for the IA1 demonstration. This consideration was based on the large population of Westinghouse 4-loop PWRs, and also due to the familiarity of the RIMM IA1 project team with modeling LOCA for this design. This consideration was then extended to include the Westinghouse 3-loop PWRs due to similarity with the 4-loop PWRs in the context of LOCA analysis.

High Impact of Proposed 50.46c Regulation on Some Designs

The proposed 50.46c LOCA/ECCS rule will have a greater impact on those reactors that are limited by low PCT and/or ECR margin. The NRC has assigned all of the plants to one of three tracks for compliance with the proposed 50.46c regulation based on their review of a PWR Owners Group (PWROG) report that characterizes the PCT and ECR margin. The PWROG report was last updated in the fall of 2013. [50] The most recent NRC evaluation was dated 12/30/2014. [51] The Track 2 and 3 plants are identified below with the Westinghouse 3-loop and 4-loop plants highlighted in bold print. The Track 3 plants are expected by NRC to have the most work effort to comply, followed by Track 2, and then Track 1 (not shown). Note that the Westinghouse plants in Track 3 are all placed in that track due to SBLOCA, which is not the scope of RIMM IA1.

Track 2	**Track 3**
Beaver Valley 2	**Beaver Valley 1 (SBLOCA low margin)**
Braidwood 1&2 & Byron 1&2	ANO-2 App K
Catawba 1 & 2	Palo Verde 1&2&3 App K
D. C. Cook 1	St. Lucie 2 App K
Indian Point 2&3	**Surry 1&2 (SBLOCA low margin)**
McGuire 1 & 2	**Summer 1 (SBLOCA low margin)**
Watts Bar 1	Waterford 3 App K
North Anna 2	
Point Beach 1	
Ginna	
Seabrook 1	
Surry 1&2	
Turkey Point 3&4	
Wolf Creek	

Low PCT Margin

The current LOCA PCT margins (2200°F is the regulatory limit for PCT) for the Westinghouse 3-loop and 4-loop plants were obtained from the annual licensee reports required by 50.46(a)(3). The ECR margins are not currently reported, but high PCT generally indicates high ECR.

Four-Loop	
Braidwood/Byron [ML14097A402]	Unit 1 – 2023°F / Unit 2 – 2045°F
Callaway [ML14084A547]	2055°F
Catawba [ML14205A287]	2070°F
Comanche Peak [ML14136A029]	Unit 1 – 1629°F/ Unit 2 – 1850°F
Cook [ML14245A016]	Unit 1 – 2090°F/ Unit 2 – 1954°F
Diable Canyon [ML14219A412]	Unit 1 – 2124°F/ Unit 2 – 2125°F
Indian Point [ML14085A416]	Unit 2 – 2119°F/ Unit 3 – 2046°F
McGuire [ML14205A287]	2086°F
Millstone-3 [ML14188C056]	1933°F
Seabrook [ML14205A443]	1919°F
Sequoyah [ML14330A326]	1940°F
South Texas [ML14112A459; ML14205A014; ML12310A383]	Unit 1 – 2120°F/ Unit 2 – 2117°F
Vogtle [ML14344A056]	Not submitted to NRC
Watts Bar [ML14119A332]	1865°F
Wolf Creek [ML14094A427]	2175°F
Three-Loop	
Beaver Valley [ML14309A392]	Unit 1 – 1840°F/ Unit 2 – 1832°F
Farley [ML14344A056]	Not submitted to NRC
Harris [ML14139A137]	2073°F
North Anna [ML14188C056]	Unit 1 – 1866°F/ Unit 2 – 1909°F
Robinson [ML14337A046]	2088°F
Summer [ML14149A321]	1960°F
Surry [ML14188C056]	2081°F
Turkey Point [ML14351A073]	2124°F

Stakeholder Knowledge and Willingness to Participate

A key consideration in the selection of a vendor or owner-operator stakeholder is their knowledge and experience related to LOCA analysis, and their willingness to participate in a R&D project of this nature. This is the most important consideration in that the INL RIMM project will require the stakeholder to contribute resources to perform project activities. It is recognized that potential industry stakeholders have limited resources with LOCA-related knowledge.

Survey of Existing RELAP5 Models

Early in the INL RIMM IA1 project the team decided to use the RELAP5 system thermal-hydraulic code for demonstrating the methodology. RELAP5 LOCA simulations will be used to build the database for the RIMM tool for the IA1 application. The project team also decided to target a Westinghouse 3-loop or 4-loop pressurized water reactor for the IA1 demonstration. Based on those decisions one of the main considerations in the selection of the industry stakeholder was the availability of an existing RELAP5 plant model. This chapter summarizes the availability of RELAP5 models for Westinghouse 3-loop and 4-loop plants.

AREVA has RELAP5 models for all of the PWRs that it has performed safety analysis using this software. The subset that are of Westinghouse 3-loop or 4-loop design are the following:

- McGuire (4-loop / Duke)
- Catawba (4-loop / Duke)
- Sequoyah (4-loop / TVA)
- North Anna (3-loop / Dominion)
- Harris (3-loop / Duke)
- Robinson (3-loop / Duke)

The following owner-operators have RELAP5 models of Westinghouse 4-loop or 3-loop plants:

- Duke [McGuire and Catawba (4-loop)]
- TVA [Watts Bar (4-loop)]
- STNOC [South Texas Project (4-loop)]
- Dominion [Millstone-3 (4-loop); North Anna (3-loop)]

INL has RELAP5 plant models for the following Westinghouse 3-loop and 4-loop plants:

- Robinson (3-loop / Duke)
- Seabrook (4-loop / NextEra FPL)
- Diablo Canyon (from Zion / 4-loop / PGE)
- Zion (4-loop / Exelon)
- North Anna (3-loop / Dominion)
- Surry (3-loop / Dominion)
- SNUPPS (4-loop / WCNOC & Ameren UE)

5.2 Value Proposition

The RIMM IA1 methodology and tool will provide a means of quantifying the impact on the key LOCA analysis figure-or-merits (peak cladding temperature (PCT), equivalent cladding reacted (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 a LOCA evaluation model. The information that the tool provides can then be used for decision making and margin management.

Value Proposition for the Industry (Notional)

Nuclear installation designers, vendors and licensees (plant 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. 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.

For instance, the propagation of uncertainties across the various functional groups is addressed 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.

Such 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 margin resides 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 RIMM project (part of the RISMC IA1) is expected to create value by anticipating these trends and focusing on developing a methodology that effectively addresses the limitations presented above. The primary goal is to explore an integrated approach for knowledge and uncertainty management, as illustrated in Figure 17.

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

- Provide a responsive toolbox for the plant operator which enables rapid decisions on considered changes within the LOCA issue space (as regulated under the new 10 CFR 50.46c). The goal is to greatly reduce the response cycle.
- Enable factoring-in current knowledge in the process to enhance safety and operation optimization, two objectives not necessarily in conflict
- Quantify currently unquantified uncertainties to the extent practical and trends to a truly realistic representation of the LOCA which provides insights on the design otherwise distorted by biases
- An approach that can lead to new knowledge and understanding of the LOCA scenario which could be 'locked' in the engineering assumption of licensing calculations. Enable a more effective 'exploration' of the issue space.
- Eliminate the issues associated with the so called Wilks' approach (variability in the estimator, i.e. risk of under-prediction of or over-prediction of FOM, lack of knowledge in what is truly limiting in the design, incapacity to perform sensitivity studies, impact assessment etc.)
- A 'plug-and-play' design of the multiphysics tool which enables plant owners and vendors to consider and further develop RIMM Framework for use within their established codes and methods.

Note that the RIMM IEM is not intended to replace licensed AORs but rather to replace or aid the 'engineering judgment' applied in managing those AORs. In other words, the RIMM IEM is a margin management tool. This objective is achieved by representing the plant realistically with all the uncertainties included and by considering and managing the entire body of knowledge.

For practical purposes, the value proposition can be broken up in two parts:

1. Characterization

As a first step, the owner/operator will use the RIMM IEM tool to "characterize" the core designed for operation. Figure 20 illustrates this process, where the IEM maps an envelope of maximum ECR as a function of cycle exposure. This allows the operator to have a realistic assessment of an operating core, and conceivably be more prepared for a quick response re-analysis in case a problem might occur.

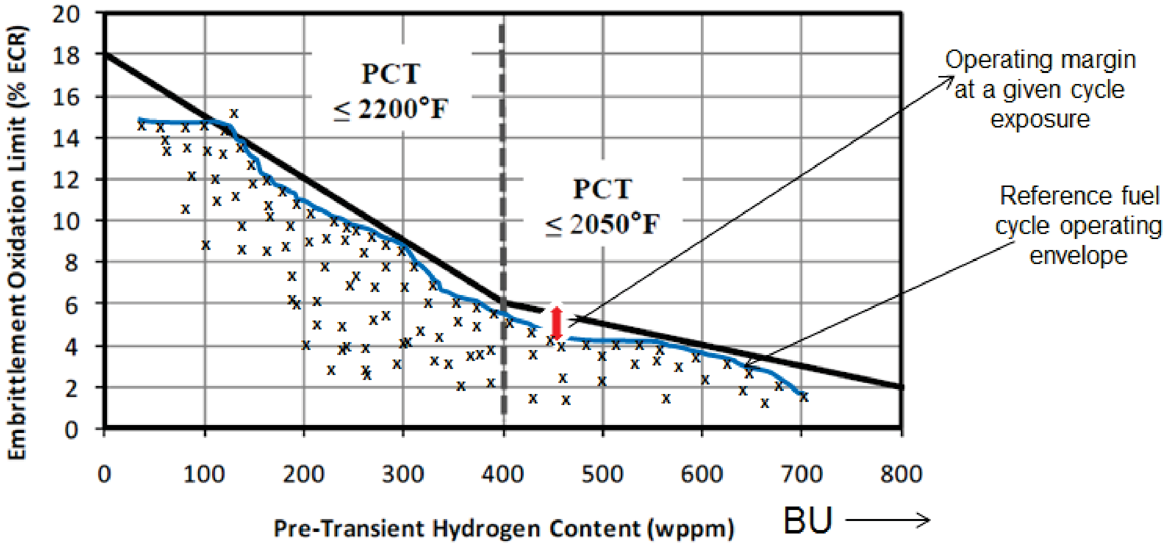


Figure 20. Characterization of an Operational Core.

2. Optimization

The characterization of a reference core with the IEM tool is intended to simplify the existing reload analysis process, although not intended to replace the existing licensing process. In principle, a reload engineer that has trained the IEM tools to analyze a given core design can re-analyze such reference design in much faster time than using a traditional reload design analysis process.

Eventually, we will be able to incorporate optimization schemes in the IEM toolset that can quickly reshape a desired parameter envelope (in this case ECR) as an optimization feature of a core design process, as illustrated in Figure 21. In practice, such 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 (see also Figure 17).

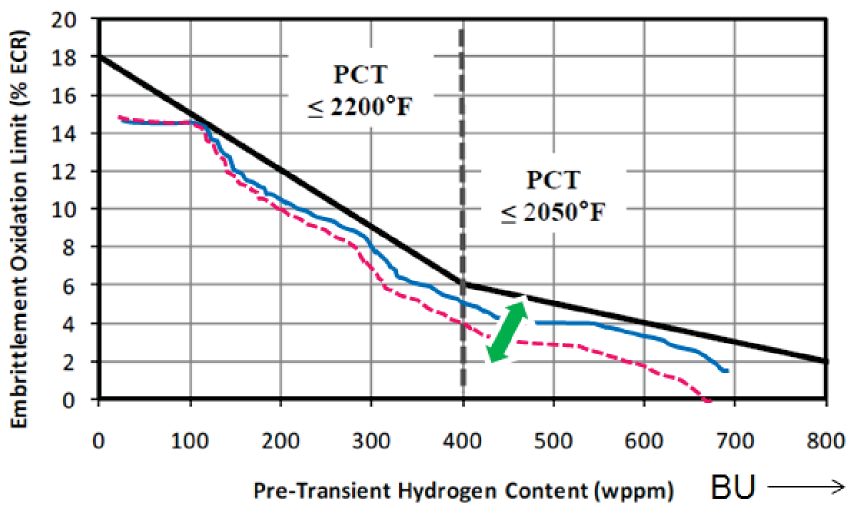


Figure 21. Optimization of a Characterized Core.

Note that the concept described above is simplified, and it serves the purpose of only illustrating the concept proposed. In practice, the IEM will need to evaluate a multi-dimensional problem not easily visualized (a more detailed discussion of this topic is seen in Section 6.3). Computational constraints to analyzing highly complex systems with many variables to be considered have kept us in the past from executing these types of schemes.

Today, with the development of a RISMC Toolkit built in a state-of-the-art computational environment, we will be able to implement complex multi-physics schemes solving fully coupled systems problems in real time, not practically executed in the past.

Fuel Vendor Value Proposition

For a fuel vendor the RIMM IA1 methodology and tool is envisioned as a LOCA analysis scoping tool with the following potential uses:

- Provide quantitative estimates of design or operational margin loss or gain associated with various combinations of changes in LOCA analysis inputs
- Provide quantitative estimates of impact on the LOCA analysis figures-of-merit due to errors in LOCA analysis inputs
- Develop marketing strategies related to LOCA analysis
- Provide LOCA inputs related studies in response to customer inquiries and requests
- Respond to LOCA-related regulatory inquiries and requests for additional information

It is also possible that the RIMM IA1 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 due to LOCA analysis input errors that are required by 10 CFR 50.46.

Owner-Operator Value Proposition

For an owner-operator the RIMM IA1 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 errors in LOCA analysis inputs (including reporting of LOCA analysis ΔPCT and ΔECR due to LOCA analysis input errors that are required by 10 CFR 50.46)

A less likely but possible type of potential application is to use the RIMM IA1 methodology and tool as an independent owner-operator LOCA analysis capability. This capability could be used to perform vendor-independent LOCA 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.

5.3 Selection Recommendation

Vendor Selection Recommendation

The fuel vendors own the NRC-approved LOCA evaluation models and perform all of the LOCA analyses for the plant owner-operators. Therefore, they have the plant LOCA analysis database, the fuel design database, and the staff experience to provide excellent collaboration on the RIMM IA1 project. Collaboration with a fuel vendor may involve overcoming the following obstacles:

- Vendors are likely to expect funding of any work effort to support the project
- Vendors are typically short on LOCA staff resources
- Vendors will want to influence the project direction
- Vendors will want to review and restrict publication of project deliverables

- Vendors will be wary of anything that involves their customers
- Vendors will be wary of anything that may cause regulatory interaction

Evaluation of Westinghouse

Westinghouse is considered to be the industry leader on best-estimate LOCA analysis. The Westinghouse "Full Spectrum LOCA" (FSLOCATM) LOCA evaluation model is currently under NRC review, and is the most advanced LOCA analysis methodology. The WCOBRA/TRAC-TF2 computer code is the most advanced LOCA system thermal-hydraulic analysis code. Their "Advanced Statistical Treatment of Uncertainty Methodology" (ASTRUM) is widely used for the LOCA analyses of record. Westinghouse is also the designer of the 4-loop and 3-loop PWRs that have been selected for the RIMM IA1 project, and so they have all of the design information. Westinghouse uses mainly ZIRLO™ cladding at present, with two reactors currently using the advanced Optimized-ZIRLO™ cladding design. Westinghouse has limited RELAP5 code experience.

Evaluation of AREVA

AREVA's "Realistic Large-Break LOCA" methodology is RELAP5-based, and positions AREVA as an experienced RELAP5 user. AREVA also maintains several conservative Appendix K LOCA evaluation models for various PWR designs. AREVA's LOCA methods have been applied to both Westinghouse 3-loop and 4-loop PWRs. AREVA's advanced M5 cladding alloy is considered to be the least impacted by the proposed 50.46c rule due to its low corrosion rate during normal operation. There may an issue with access to Westinghouse PWR design information through AREVA as the use of the information may be restricted.

Owner-Operator Selection Recommendation

The owner-operator selection process initially down-selected from the broader population by focusing on staff knowledge related to LOCA analysis and the scope of other related analyses performed in-house. This initial screen resulted in APS, Duke, Dominion, STP, and TVA as being potential collaborators for RIMM IA1 collaboration.

6. INDUSTRY APPLICATION FULL DEMONSTRATION CONSIDERATIONS

Phase III of the Industry Applications considers the full spectrum of demonstrations with all advanced features of the RISMC toolkit (concurrently in development while early RISMC demonstrations take place). These will include applications of RAVEN and RELAP7, also including other models in the MOOSE framework, as needed.

Below, are discussions of a few ideas that will be explored in FY2016 and beyond.

6.1 Traditional PRA and BEPU Integrated Approach

Nuclear reactor safety study traditionally has been advanced in two separate domains with only loose integration: deterministic safety analysis method and probabilistic risk analysis method. For example, deterministic safety analysis such as LOCA calculations done by a safety system code needs the input from PRA analysis for constructing the scenario sequences. In these LOCA calculations, the safety component availability and timing information in the scenario comes from PRA analysis. The PRA requires occasional runs of system codes (often conservative codes like MAAP for fast execution) to obtain information of failure or success. However, no uncertainties in the system analysis model are accounted for. Moreover, in the current PRA, the functional failure probability that the loads will exceed the capacity is assigned as zero for the success sequences and as unity otherwise, based on thermal-hydraulic calculations or engineering judgments for comparison of the characteristic value of load and capacity. [52] However, even in the success sequence, there is always possibility that functional failure will occur because both the load and capacity have uncertainties which can be described by probability density functions. On the contrary, in the failure path, there is possibility that the functional failure probability is less than unity.

One of major attributes of the RSMIC methodology is the strong coupling between PRA and deterministic system analysis tools with full consideration of as many as possible uncertainty sources. [53] Other studies on integrated risk and safety margin approach have also emerged in recent years. For example, Pagani et al. [54] analyzed the passive cooling in a gas-cooled fast reactor under a loss-of-coolant accident condition to quantify the role of functional failures, and compared failure probabilities of passive system with those of alternative active design. The BEPU was proposed for calculating the probability density function of the load (for example, PCT, MLO, and CWO) against the capacity. By BEPU application, the functional failure probability can be defined as the probability that the load exceeds the safety limit, and is also called exceedance probability. [55] Korean researchers proposed the combined deterministic and probabilistic procedure (CDPP) for safety assessment of the beyond design basis accidents (BDBAs), [52] where the conditional exceedance probability (CEP) acts as go-between deterministic and probabilistic methods. The combined approach is accompanied by BEPU method, and results in more reliable values of core damage frequency (CDF) and conditional core damage probability (CCDP). In the PRA process, a conservative thermal hydraulic analysis is usually used, while combined approach could replace conservative analysis with best estimate analysis with uncertainty quantification. This method had been successfully applied to the LB-LOCA assessment for Korean APR-1400 design [52] and to reevaluate the SBO risk in the OPR-1000 nuclear power plant. [56] With this method, the CDF for a key event timing has been quantified. It was often believed that static PRA analysis cannot address timing issues.

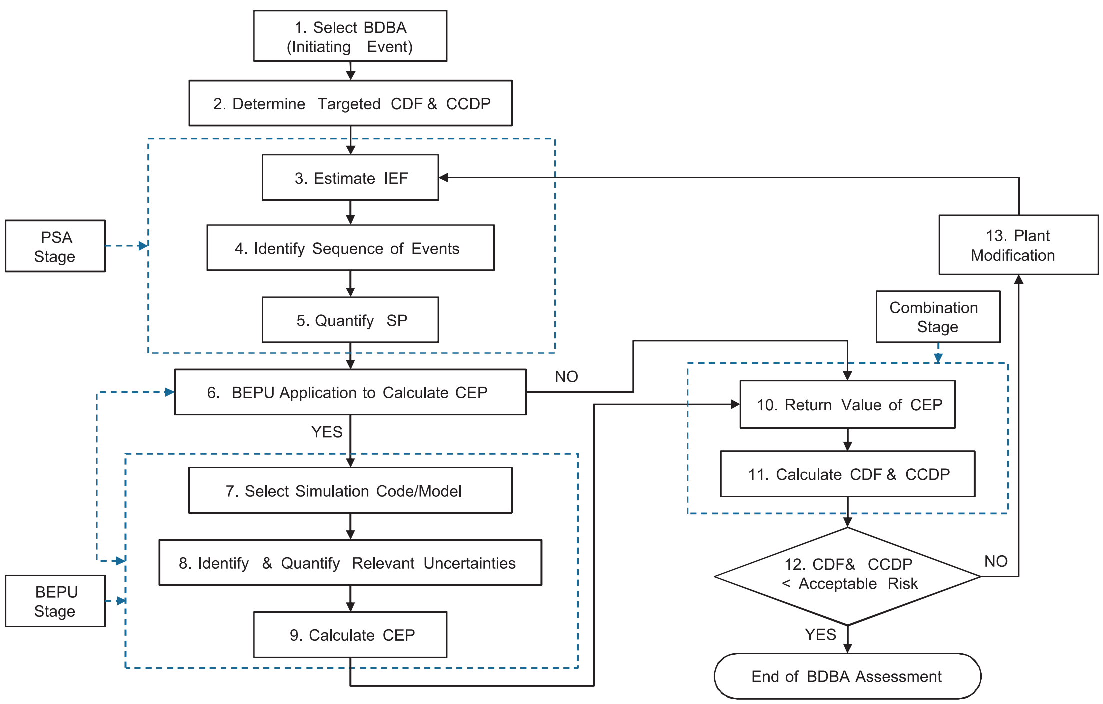


Figure 22. Combined deterministic and probabilistic procedure for safety assessment of BDBA. [52]

We can borrow a lot of ideas from the Korean CDPP method to combine PRA and BEPU method together for the RISMC ECC demonstration and compare this method with dynamic PRA method for efficiency and effectiveness. Figure 22, borrowed from the original paper, [52] describes the steps of CDPP method. The PRA stage identifies and selects event sequences for BEPU stage. The BEPU stage generates the conditional exceedance probability (CEP) by many Monte Carlo runs (i.e., 1000). This failure information is not used and wasted in the traditional BEPU analysis. The final stage combines the results and updates CDF. Feedback loops can be used to modify the plant configuration (i.e., power uprate) and operation procedures to recover margin or reduce risk.

6.2 Dynamic PRA and the Advanced RISMC Toolkit

In order to assess the compliance of the operating nuclear power plants to the new rule, a rigorous Probabilistic Risk Assessment (PRA) needs to be employed. As already mentioned, the new safety margins criteria are related to the burn-up level reached by the assemblies when the LOCA scenario is initiated. This means that the limits cannot be seen as static thresholds but must be considered in a dynamic environment, since they evolve during the operation of the reactor.

Another aspect that needs to be considered in such analysis is the presence of several uncertainties associated with the key parameters of the plant that, depending on their value, can lead to completely different accident scenarios.

From a practical point of view, the goal of the PRA analysis of LOCA events can be summarized as follows:

- Computation of the probability of exceeding the burn-up dependent limits;
- Sensitivity analysis on the uncertain parameters that can influence the LOCA scenario and sub-sequential ranking (i.e. from the most to the least impactful parameter);
- Identification of the uncertain parameters' margins through the research of the reliability (or limit) surface (i.e. hyper surface that identifies the transition between success and failure of the plant).

The PRA analysis tool of choice is RAVEN. [57] RAVEN is a generic software framework to perform parametric and probabilistic analysis based on the response of complex system codes. RAVEN is capable to communicate with any system code. This communication is achieved by the implementation of Application Programming Interfaces (APIs). These interfaces are used to allow RAVEN to interact with any code as long as all the parameters that need to be perturbed are accessible by inputs files or via python interfaces. RAVEN is currently coupled with several simulation tools, among which RELAP5-3D (chosen as system code for the LOCA analysis).

The probabilistic and parametric framework represents the core of the RAVEN analysis capabilities. The main idea behind the design of the system is the creation of a multi-purpose framework characterized by high flexibility with respect to the possible performable analysis. The framework must be capable of constructing the analysis/calculation flow at run-time, interpreting the user-defined instructions and assembling the different analysis tasks following a user specified scheme.

In order to achieve such flexibility, combined with reasonably fast development, a programming language naturally suitable for this kind of approach was needed: Python.

Hence, RAVEN is coded in Python and is characterized by an object-oriented design. The core of the analysis performable through RAVEN is represented by a set of basic components (objects) the user can combine, in order to create a custom analysis flow. A list of these components and a summary of their most important functionalities are reported as follows:

- Distribution: In order to explore the input/output space, RAVEN requires the capability to perturb the input space (initial conditions of a system code). The initial conditions, that represent the uncertain space, are generally characterized by probability distribution functions (PDFs), which need to be considered when a perturbation is applied. In this respect, a large library of PDFs is available.
- Sampler: A proper approach to sample the input space is fundamental for the optimization of the computational time. In RAVEN, a "sampler" employs a unique perturbation strategy that is applied to the input space of a system. The input space is defined through the connection of uncertain variables and their relative probability distributions.
- Model: A model is the representation of a physical system (e.g. Nuclear Power Plant); it is therefore capable of predicting the evolution of a system given a coordinate set in the input space.
- Reduced Order Model (ROM): The evaluation of the system response, as a function of the coordinates in the input space, is very computationally expensive, especially when brute-force approaches (e.g. Monte Carlo

methods) are chosen as the sampling strategy. ROMs are used to lower this cost, reducing the number of needed points and prioritizing the area of the input space that needs to be explored. They can be considered as an artificial representation of the link between the input and output spaces for a particular system.

- Postprocessor: In order to analyze the data generated from the exploration of the uncertain domain, post-processing capabilities are needed. Under this category, RAVEN collects all the statistical tools, data mining algorithms and uncertainty quantification capabilities.

The list above is not comprehensive of all the RAVEN framework components (visualization and storage infrastructure).

Figure 23 shows a general overview of the elements that comprise the RAVEN statistical framework, including the ones not explained above. [58]

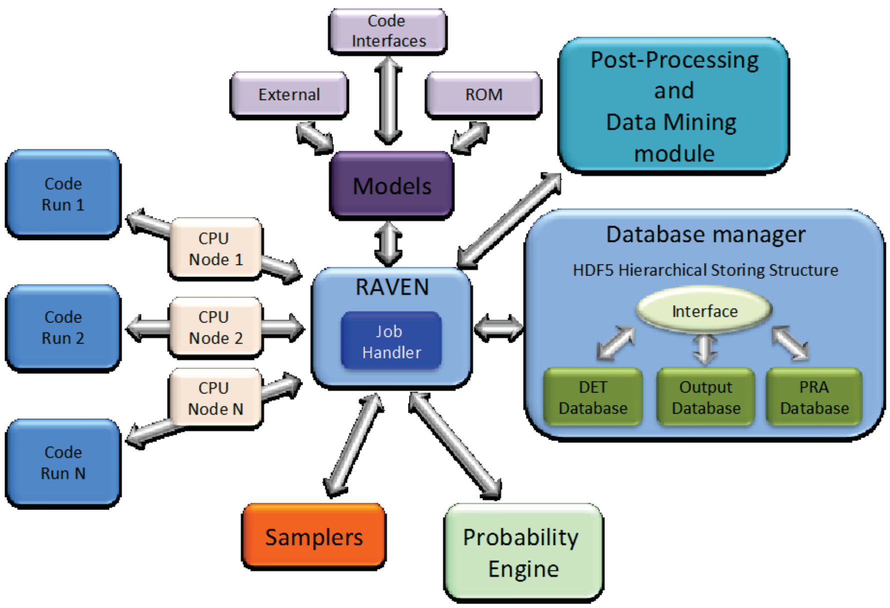


Figure 23. RAVEN Simulation Platform Scheme. [58]

In order to assess the probability of exceeding the burn-up dependent limit, a sampling of the parameters affected by uncertainties is needed. This kind of analysis is characterized by high level of complexity (influence the computation time of the simulation codes), high dimensionality (uncertain parameters to take in consideration) and a high discontinuity (presence of safety systems that can suddenly start operating). The approach that is currently used to perform such analysis is based on the well-known Monte Carlo technique; although its obvious validity, the Monte Carlo approach can be extremely expensive, overall in cases where complex physics are involved (run time of relative simulation tools is quite high). In order to overcome the computation burden of the Monte Carlo method, a Dynamic Event Tree (DET) methodology will be used. The DET technique brings several advantages, [59] among which being order of magnitude faster than Monte Carlo, while accounting for the time evolution of the system in the accident sequence. In DET, event sequences are run simultaneously starting from a single initiating event. The branches start at user specified times and/or when an action is required by the operator and/or the system. This approach creates a deterministic sequence of events based on the time of their occurrence. This leads to a more realistic and mechanistically consistent analysis of the system taken in consideration. The DET methodologies are designed to take the timing of events explicitly into account, which can become very important especially when uncertainties in complex phenomena are considered. The main idea of this methodology is to let a code (e.g., PHISICS-RELAP5) determine the pathway of an accident scenario within a probabilistic "environment". Figure 24 schematically shows the DET logic. As already mentioned, the accident sequence starts with an initiating event. Based on a user defined branching logic, driven by Probabilistic Distribution Functions (PDFs), an event occurs at a certain time instant. The simulation spoons n different branches. In each of them, the branching event determines a different consequence (carrying on associated probabilities). Each sequence continues until another event occurs and a new set of branching is spooned. The simulation ends when an exit condition or a maximum mission time is reached

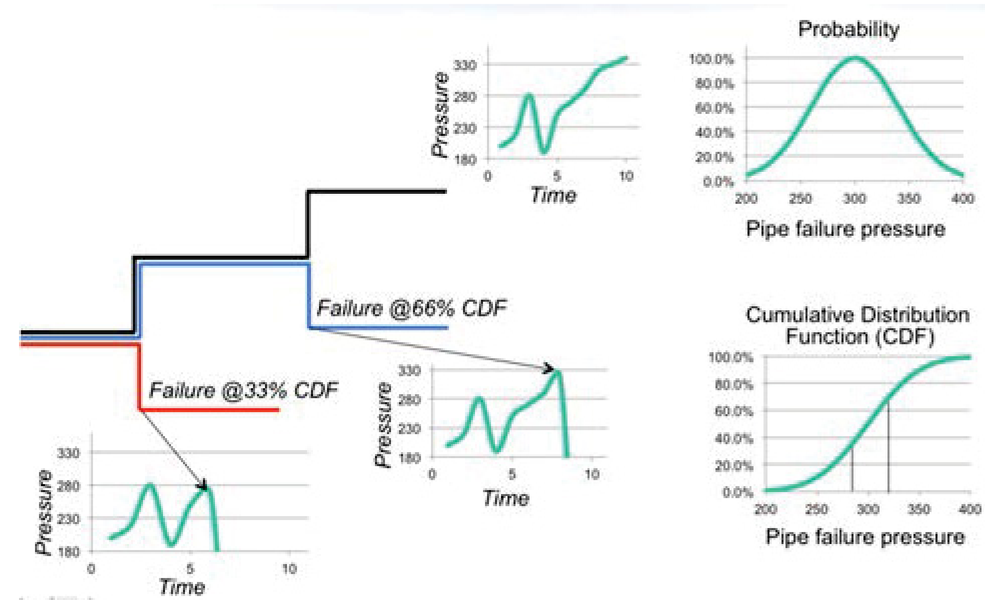


Figure 24. DET Scheme.

In the particular case of the LOCA scenarios, the basic DET methodology needs to be replaced with the Hybrid Dynamic Event Tree (HDET). [60] The HDET method represents an evolution of the DET methodology for the simultaneous exploration of the epistemic (lack of knowledge on the part of the analyst conducting the modeling and simulation) and stochastic uncertainty space. In similar traditional methods the uncertainties, depending on their classification, are generally treated employing a Monte-Carlo sampling approach (epistemic) and DET methodology (aleatory). The HDET methodology implemented in RAVEN is capable to use DET and MC contemporaneously and in addition provides additional sampling strategies to the user. The epistemic or epistemic-like uncertainties can be sampled through the following strategies:

- Monte-Carlo;
- Grid Sampling;
- Stratified (e.g., Latin Hyper Cube).

Figure 25 schematically conceptually shows how the HDET methodology works. The user defines the parameters that need to be sampled by one or more different approaches. The HDET module samples those parameters creating a N-Dimensional Grid characterized by all the possible combinations of the input space coordinates coming from the different sampling strategies. Each coordinate in the input space represents a separated and parallel standard DET exploration of the uncertain domain.

The HDET methodology allows the user to completely explore the uncertain domain employing one methodology. The addition of Grid Sampling strategy among the approaches usable, allow the user to perform a discrete parametric study, under aleatory and epistemic uncertainties.

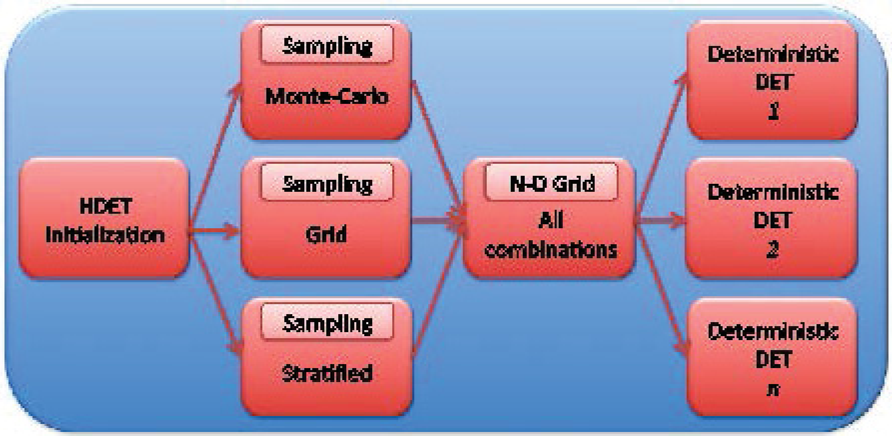


Figure 25. Hybrid Dynamic Event Tree (HDET) Scheme. [60]

Since the intrinsic dynamic characteristic of the DET methodology, in general, and the HDET, in particular, the LOCA scenario analysis represents a perfectly suited case analyzable with this methodology. In Figure 26, the scheme of the HDET applied to LOCA scenario is reported.

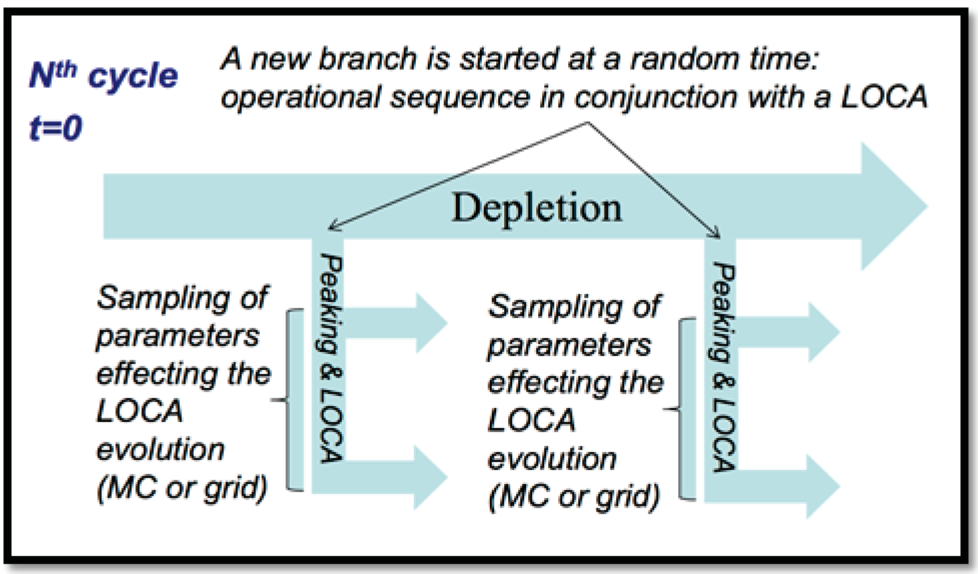


Figure 26. HDET Methodology Applied to LOCA Scenarios.

More details for the LOCA scenario considering the DET branching could be performed on the time at which the LOCA starts during one fuel cycle of the reactor.

The exploration of the system response using the HDET will ultimately lead to the knowledge of several possible outcomes of the LOCA accident scenario (in terms of PCT and corresponding burn-up and oxidation) with their corresponding probability. A post processing function, build within RAVEN, will allow the analyst to combine this information to assess what is the final probability to exceed the new limits.

After this preliminary analysis is completed it will be possible to perform a sub-sequential investigation where the computation of a sensitivity coefficient will allow to establish what are the most relevant uncertainties effecting the success/failure probability.

Finally using the RAVEN feature to utilize an accelerated search of reliability surface, it will be possible to use the HDET methodology to determine region of the input space that either leads to a positive/negative final outcome of the LOCA accident.

6.3 Research Needs for this Industry Application

It is useful to identify two classes of research needs: Methodology and Modeling. The present section is focused on Methodology; for present purposes, the following Modeling "givens" are presupposed.

The FOMs: PCT and ECR

Based on work done so far, there is general agreement that it is useful to apply the limit surface concept to the problem. This implies formulation and application of an emulator. A plant-level emulator of results for the limiting assembly would be reasonably straightforward to develop (given that we are only attempting to track PCT and ECR, and not (for example) time dependence), but for scrutability reasons, there is an argument for carrying out the analysis in sufficient detail to support discussion of the outcomes for particular assemblies, and only later assembling these results into a plant-level result.

A possible process is outlined below for purposes of illustration. Every step mentioned has precedent, but as far as we know, this particular combination of undertakings and outputs has no collective precedent, nor is it really known a priori what detailed form of the result will be most useful to plant management. The "methodology" research needs for the demo are therefore essentially figuring out the best things to do in particular, by trying out ideas on a less-than-full-scope but meaningful problem, and see what works, and solve the problems that emerge in the course of doing this.

Steps in a Candidate Process

Process Objective:
Characterize Margin in Large LOCA for a representative design, based on:

- representative initial conditions and other conditioning assumptions (time in cycle, age of assemblies, …),
- representative ranges of uncertain parameters (discharge coefficients, loss coefficients, heat transfer coefficients, …), and a representative selection of things that vary from one time history to another (system actuation times, system failure times, human actions, …).

Figure 27 below shows the acceptance criteria articulated in the proposed rule for peak clad temperature and clad oxidation. "Margin" will be measured relative to those acceptance criteria

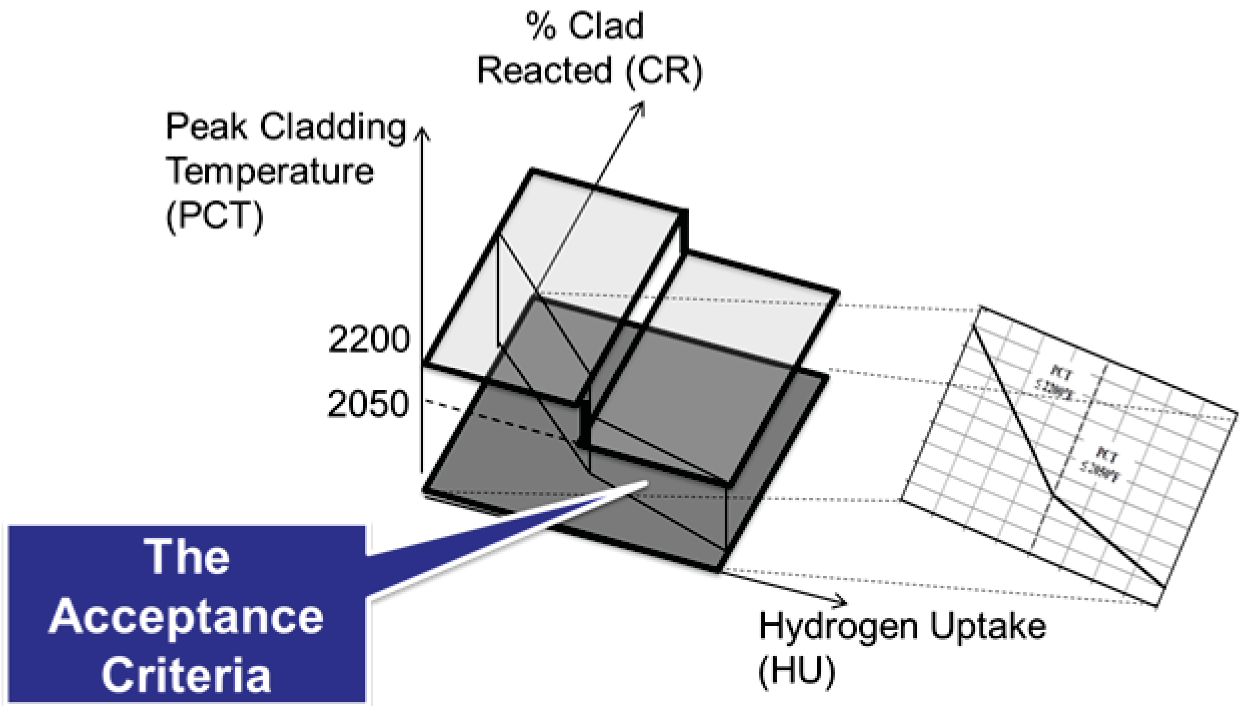


Figure 27. The Acceptance Criteria.

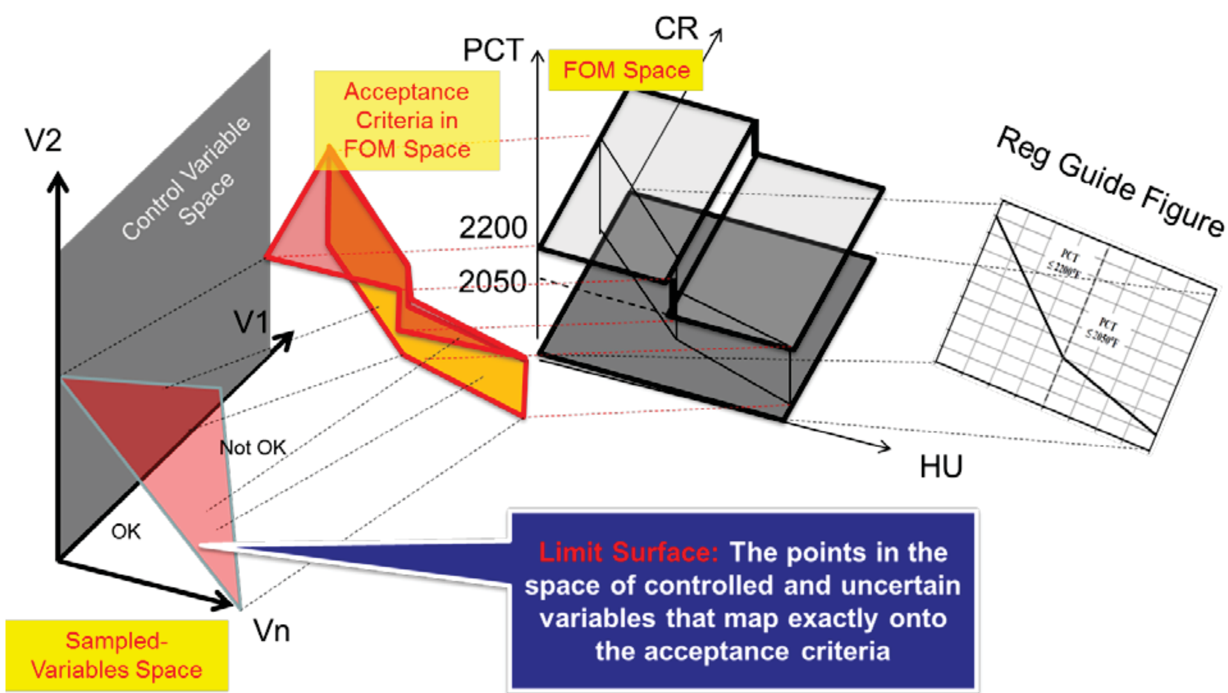


Figure 28. Correspondence Between the Acceptance Criteria in Figure-of-Merit (FOM) Space and the Limit Surface in the Space of Controlled and Uncertain Variables.

The steps outlined below lead to a risk-informed characterization of that margin, reflecting the variability and uncertainties that pertain to Large-Break LOCA:

1. **Formulate the Issue Space.**

The term "issue space" refers to a considered problem specification, without which analysis results are very difficult to interpret.

> ## The Issue Space
>
> Any given analysis of performance is predicated on many things, some tacit, some explicit. In order to be interpretable, the analysis needs to be carried out on the basis of clearly stated assumptions, including some indication of which degrees of freedom are allowed to vary in the analysis, and which are fixed. Let us call this specification the "issue space" for that particular evaluation. It includes:
>
> - Model parameter value ranges
> - Variability or uncertainty in initial conditions
> - Scope of equipment failures analyzed:
>   - Some in scope, some not
>   - Some assumed always to occur, some assumed not to occur
>   - Sometimes failure time (or item-specific cumulative damage threshold) is among the sampled parameter values
> - Scope of modeled variations in what operators may do
> - Parameters and degrees of freedom that are fixed in the analysis, and an indication of the implications of this for interpretation of the analysis results
>
> The issue space associated with a given evaluation depends on the decision being supported, along with numerous other factors.
>
> As formulated, this concept is meant to include classical safety analysis as a special case: for example, analysis of system performance conditional on a limiting single failure subsequent to a specific initiating event such as loss of offsite power. The issue space covers all scenarios characterized by a single failure, operating conditions chosen to be within technical specifications but otherwise limiting, limited or no credit for human action, etc. But in classical safety analysis, the point is to show that limiting performance is still acceptable, while for present purposes, the point is to show the spectrum of behavior within the issue space, and the probabilities of the various outcomes.

For purposes of the demo, the issue space needs to be reduced relative to an actual licensing analysis, but still large enough to illustrate the process. This will require a project decision regarding level of detail of the model to be applied. Specifically, the issue space formulation requires us to specify

- enough structure in the model of the core to allow us eventually to determine which site (rod, assembly, …) is limiting with respect to PCT and/or which site is limiting with respect to ECR, and
- the variables that influence these outcomes.

For purposes of early demonstration, the level of detail in the core model and the scope of the variables considered may be reduced while the details of the methods are being refined. Figure 26 notionally illustrates a correspondence between the acceptance criteria and a space comprising both controlled and uncertain variables

2. **Partition the Issue Space into a subspace of Control Variables and a subspace of Other Variables.**

The Control-Variable subspace represents the space within which operators can choose to situate the plant (power level, core configuration, etc.). The other variables are either uncertain (e.g., certain loss coefficients) or uncontrollable (e.g., failure times). Eventually, we will want to present results in the Control Variable subspace. This is suggested notionally in Figure 28.

3. **Assign appropriate ranges to all the variables in the Issue Space.**

4. **Select an illustrative core design and a set of representative initial conditions for the LOCA analysis.**

For the first exercise, choose a few representative assemblies and a representative setting of the control variables.

Note: This is not the most general thing to do, but for purposes of the demo, it is a defensible downselection.

Figure 29 notionally illustrates what we have in mind for this phase. Each ellipsoid in the figure represents a particular assembly, situated in the (core power, burnup) plane, and elongated along the notional "uncertain variable(s)" axis.

The red triangle is the notional limit surface in the space we are talking about. Formally, we could find the whole thing, but it may turn out to be a good idea to simply find the limit cookies (the portion of the surface intersected by the ellipsoids).

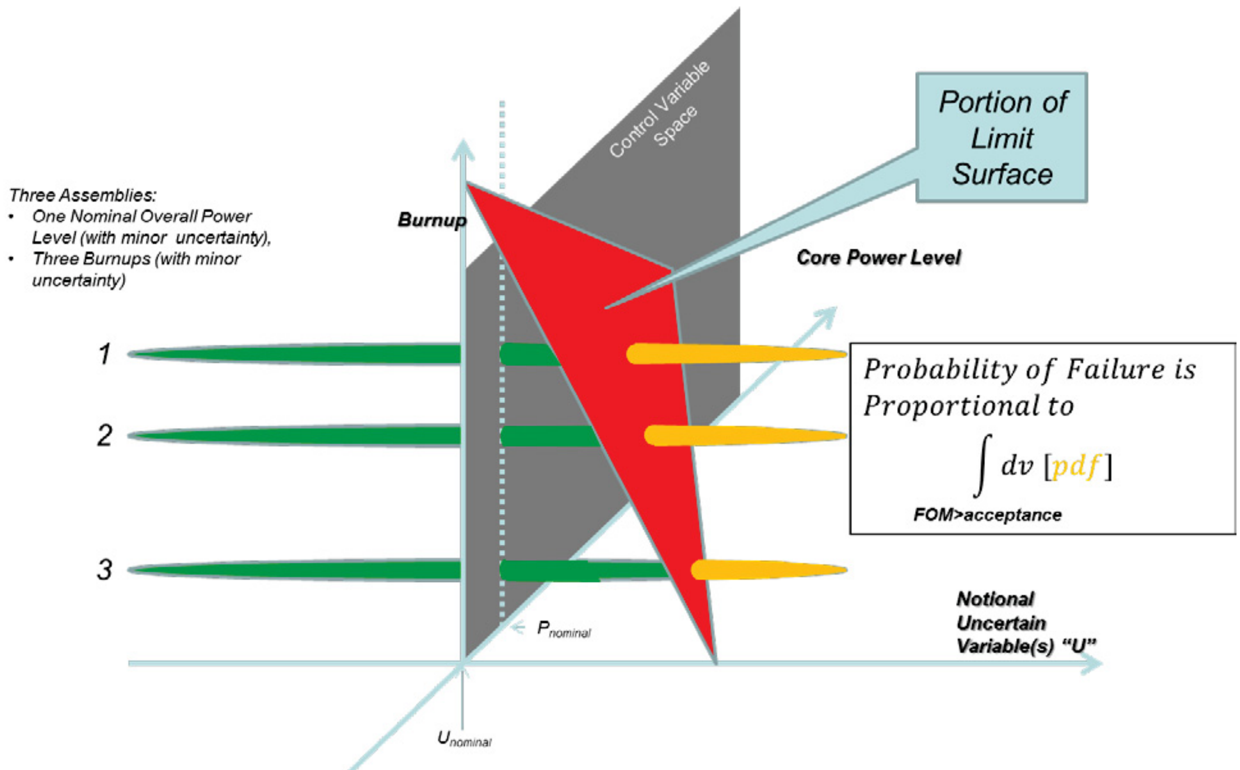


Figure 29. Assembly Ellipsoids Intersecting the Limit Surface.

5. Find the limit surface in the Issue Space for each assembly.

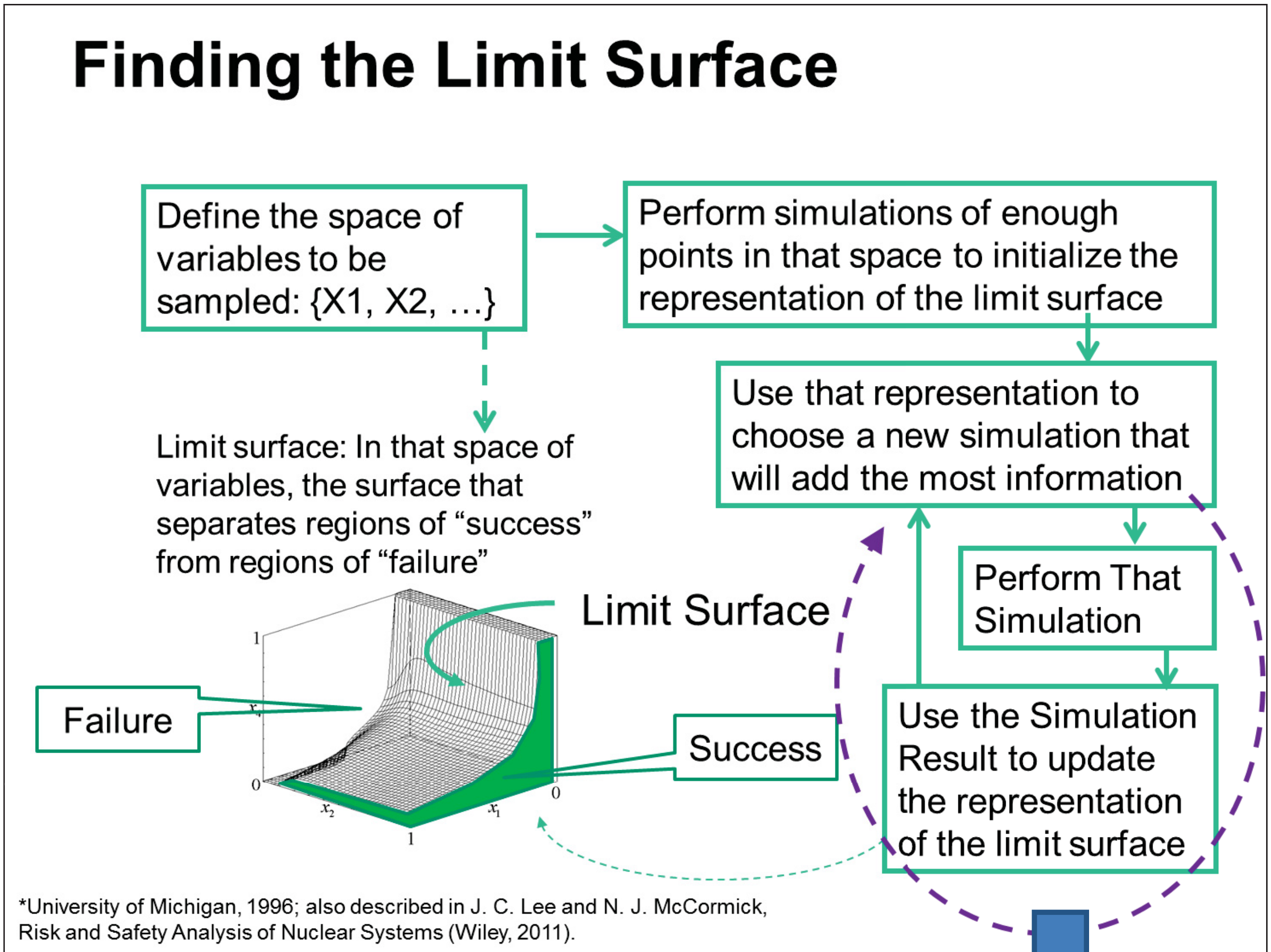
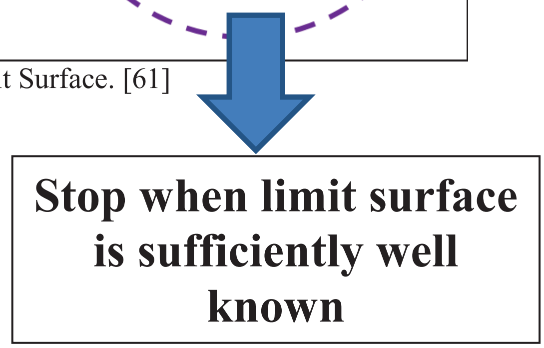


Figure 30. Iterating to Determine the Limit Surface. [61]



A tested way of finding the limit surface is illustrated above in Figure 30. [61] The team will need to experiment with emulator types to determine which is best for this application.

Figure 31 notionally shows the correspondence between a particular assembly’s limit surface in the space of controlled and sampled variables, and the FOMs.

6. Formulate a plant-level description of “margin” conditional on a given setting of the controlled variables.

A final step is formulation of a result with immediate operational significance. Knowing each assembly’s limit surface is, in a way, too much information for some purposes. It is desirable to show a plant-level result, relating, for a given setting of the controlled variables, the limiting assembly’s standing with respect to the FOMs.

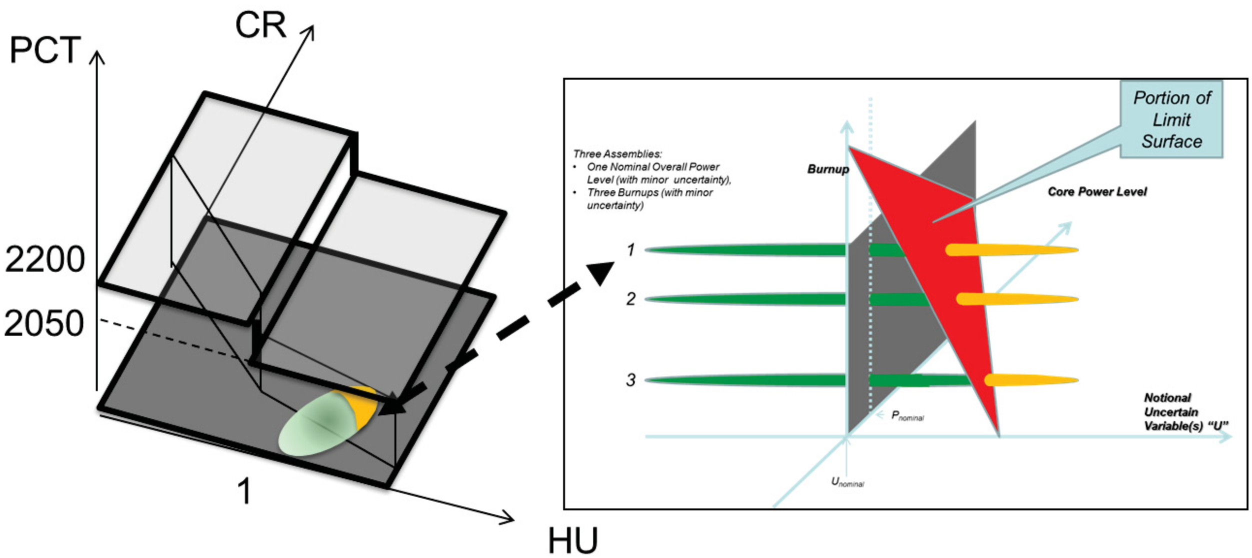


Figure 31. A Possible Correspondence Between the Limit Surface for Assembly #1 and the Figures Of Merit.

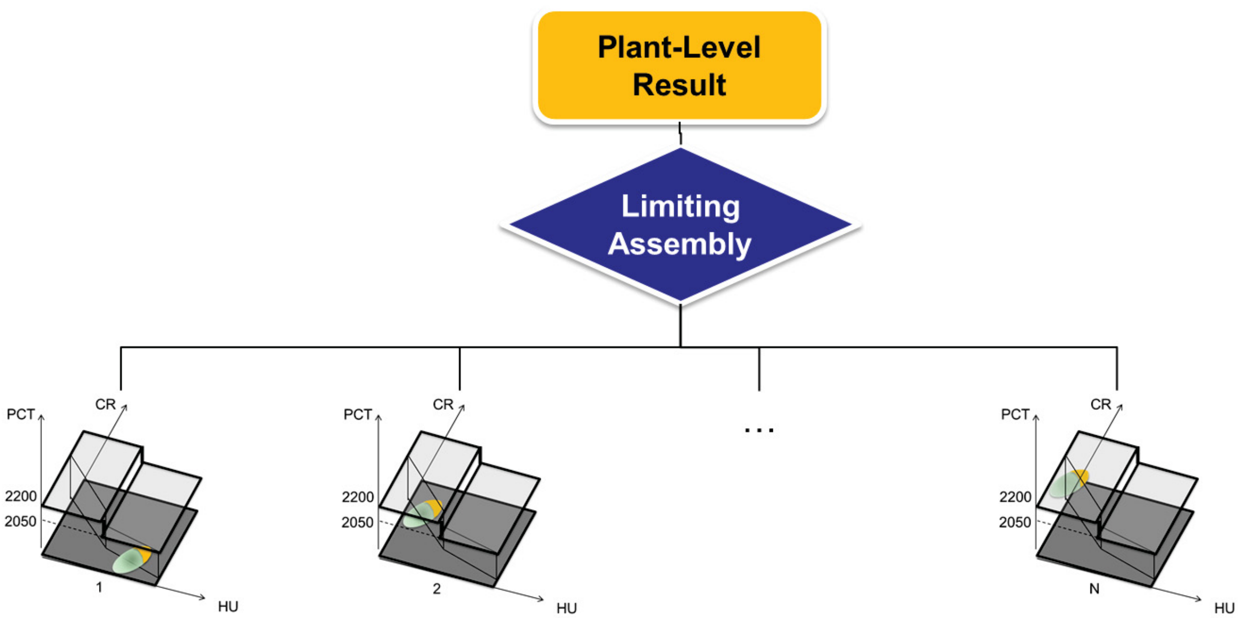


Figure 32. Synthesis of a Plant-Level Result, Given a Set of PCT and ECR Values.

Figure 31 implicitly suggests that #1 is limiting, because the limit surface is closest to its nominal position, and its ellipsoid extends farthest beyond the limit surface. This is an artifact of the figure; ellipsoids of different sizes, corresponding to assembly-specific uncertainties, would suggest a different conclusion. Moreover, since the limit surface is defined by the more limiting of PCT and ECR, we do not, strictly speaking, know for sure whether the three notional assemblies illustrated are even limited by the same acceptance criterion.

For example, as outlined, the process contemplates defining a collection of entities (assemblies?), providing a {PCT, ECR} result for every entity modeled, and deriving the plant-level result as the limiting entity [entities?] for any given configuration. This appears to have the advantage of scrutability, but if every entity needed its own emulator, it would also sound like a lot of work. The major reason to proceed this way is that it is probably more transparent than working only with the plant-level result. But we may find that it is perfectly satisfactory to work directly with a plant-level model.

Summary

The demo applies ideas and techniques that have precedents (limit surface, emulators, …) and have previously been applied to problems of meaningful complexity and scale. [62] We believe that the present application (characterizing margin to newly proposed acceptance criteria) is feasible. However, while arguably within the state of the art, or at least not too far beyond it, the application is not within the current state of practice. The research needs will be met by

- carefully scoping an exercise that is feasible but also illustrative of the real problem,
- doing the exercise,
- establishing whether the form and content of the results are useful,
- and, if they are not, iterating until they are.

7. WORK PLAN AND SCHEDULE

The RISMC program and the plan for the industry application was presented in the INL report [21]. The demonstration objectives are:

1. Provide confidence and a technical maturity in the RISMC methodology (essential for broad industry adoption)
2. Strong stakeholder interaction required
3. Address a wide range of current relevant issues (see also item (d))
4. Three phase approach:
   - Problem definition (3-6 months) – (on going)
   - Early Demonstration (eDemo) (limited scope) (6-12 months)
   - Complete Application and Validation (Long Term- Methods, Tools, Data) (1-5 years)

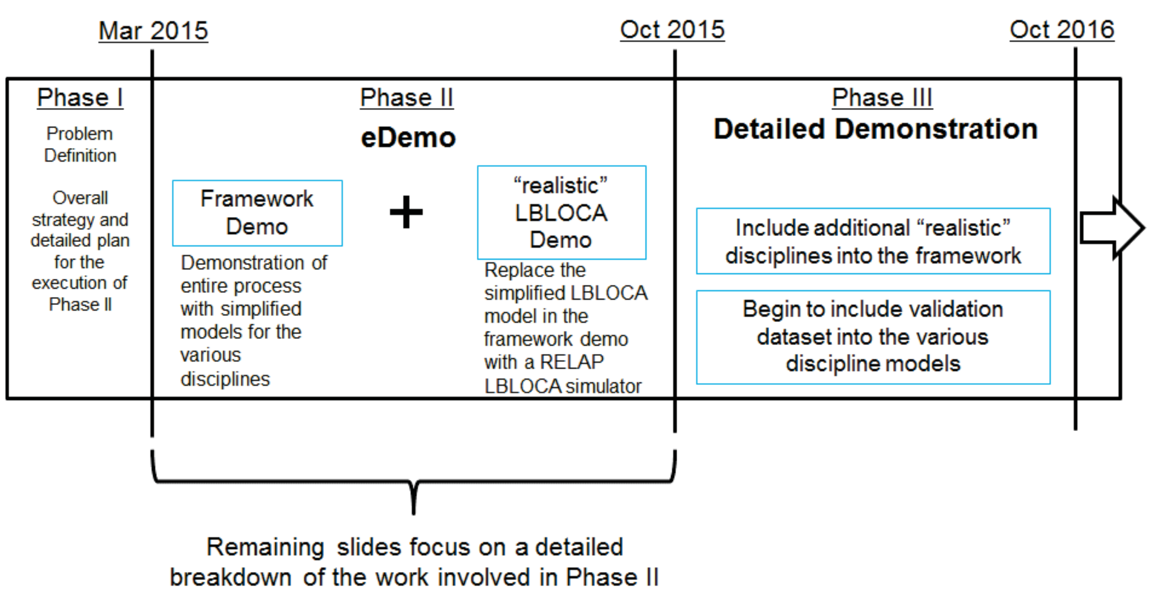


Figure 33. RIMM IA#1 Project Phases.

The program and program objective for the Industry Application 1 (IA1) are shown in Figure 33. The eDemo is the first Milestone in 2015. The approach for Phase II is based on two-tiers.

In Tier 1 the Framework Demo is developed. The Framework Demo is a simplified version (prototype) of the integrated evaluation model. Each disciplines is modeled with very simple reduced order models (ROMs). The goal is to identify all the inputs and disciplines involved and compute the approximated value of the outputs to construct a first tier “knowledge database”. The “knowledge database” is then analyzed with GP emulators for the purpose of illustrating the RIMM machinery in its entirety.

In Tier-2, the LOCA simulator is inserted in the Framework to obtain a more realistic and credible LOCA EM. Completion of Tier-2 represents the completion of the eDemo (Phase-II).

As the program enters subsequent phases (Phase-III and beyond) each discipline (simply represented by ROMs in Phase-II) is properly replaced by realistic simulators, therefore improving the fidelity and quality of the “knowledge database”.

8. SUMMARY

This report summarized the activities conducted during Phase-I of the RIMM Project, the RISMC Industry Application 1 (IA#1). The drivers behind the RISMC IA#1 were outlined in Section 1. In short, the RIMM Integrated Evaluation Model (IEM) is expected to be a margin management tool for the industry to cope with the challenges associated with the more restrictive LOCA rule, the 10 CFR 50.46s which is expected to be amended in 2016. The industry will need to comply with the new rule within the following five to ten years and the RIMM IEM is anticipated to be an effective tool for plant operators and vendors to facilitate these activities.

The primary purpose of this report was to first “define the LOCA problem,” then to lay out a roadmap to demonstrate application of the RISMC methodology which is realized by developing the RIMM IEM.

Sections 3 and 4 describe the current situation under the current rule. These sections described how the industry approaches the LOCA analysis, particularly in the context of BEPU methodologies. The reader can identify what are the challenges and trends toward the rollout of the new rule. Hence, these sections define the problem

Section 5 suggests a value proposition or vision for this technology. This proposition is to be presented to selected industry stakeholders, as discussed, to demonstrate this technology for realistic scenarios within their systems.

Note that the RIMM IEM is not intended to replace licensed 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 to provide an analytical machinery that can represent a power plant realistically with all the key uncertainties included and that considers all physical disciplines involved in an integrated fashion, i.e. an Integrated Evaluation Model. The tool will enable a plant operator to manage the entire body of knowledge – called the “Knowledge Database” - to inform decisions that minimize or manage the risks of exceeding the LOCA criteria while allowing a safe and economical operation of the plant by eliminating or reducing the need for expensive re-analyses.

Nuclear installation designers, vendors and licensees (plant 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. 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. The method tends to be ‘not responsive’ or 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 margin resides 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 their 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 RIMM project (part of the RISMC IA1) is expected to create value by anticipating these trends and focusing on developing a methodology that effectively addresses the limitations presented above. The vision for the RIMM IEM is summarized in the following propositions:

- Provide a responsive toolbox for the plant operator which enables rapid risk-informed decisions on considered changes within the LOCA issue space (as regulated under the new 10 CFR 50.46c).
- Enable factoring-in current knowledge in the process to enhance safety and operation optimization, two objectives not necessarily in conflict
- Quantify currently unquantified uncertainties to the extent practical and trends to a truly realistic representation of the LOCA which provides insights on the design otherwise distorted by undue biases.
- Use a risk-informed approach that can lead to new knowledge and understanding of the LOCA scenario which could be 'locked' in the engineering assumption of licensing calculations. Enable a more effective 'exploration' of the issue space.
- Eliminate the issues associated with the so called Wilks' approach (variability in the estimator, i.e. risk of under-prediction of or over-prediction of FOM, lack of knowledge in what's truly limiting in the design, incapacity to perform sensitivity studies, impact assessment etc.)
- Provide a 'plug-and-play' design of the multiphysics tool which enables plant owners and vendors to consider and further develop a RIMM Framework for use within their established codes and methods.

An overall plan is developed in this report. The focus in 2015 is an early demonstration (eDemo) of the RIMM IEM. The plan and eDemo problem definition is discussed throughout this report.

9. REFERENCES

1. G. Hache and H. Chung, "The History of the LOCA embrittlement criteria," OECD, 2001.
2. "Report of Advisory Task Force on Emergency Cooling," TID-24226.
3. "Interim Acceptance Criteria for Emergency-Core-Cooling Systems for Light-Water Power Reactors," U.S. Federal Register 36 (125). 1971, p. 12247-12250.
4. W. Cottrell, "ECCS rulemaking hearings," Nuclear Safety, 15, pp. 30-55, 1974.
5. U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.157 - Best-Estimate Calculations of Emergency Core Cooling System Performance," U.S. NRC, Washington D.C, 1989.
6. U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.203 Transient and Accident Analysis Methods," U.S. NRC, Washington, DC, December 2005.
7. B. Dunn and e. al., "Realistic Large Break LOCA Methodology for Pressurized Water Reactors, Revision 2," AREVA NP Inc., 2010.
8. P. R. Martin and L. D. O'Dell, "Development Considerations of AREVA NP Inc.'s Realistic LBLOCA Analysis Methodology," *Science and Technology of Nuclear Installations,* vol. 2008, p. 13, 2008.
9. M. Y. Young, S. M. Bajorek, M. E. Nissley and L. E. Hochreiter, "Application of Code Scaling Applicability and Uncertainty Methodology to the Large Break Loss of Coolant," *Nuclear Engineering and Design,* vol. 186, no. 1-2, 1998.

10. M. E. Nissley, "Method for Satisfying 10 CFR 50.46 Reanalysis Requirements for Best-Estimate LOCA Evaluation Models," Westinghouse Electric Compnay, LLC, Pittsburgh, PA, 2004.

11. M. E. Nissley, C. Frepoli, K. Ohkawa and K. Muftuoglu, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Westinghouse Electric Company, Pittsburgh, PA, 2003.

12. C. Frepoli, "Review Article - An Overview of Westinghouse Realistic Large Break LOCA Evaluation Model," *Science and Technology of Nuclear Installations,* vol. 2008, p. 15, 2008.

13. C. Frepoli and K. Ohkawa, "Westinghouse Experience in Licensing and Applying Best-Estimate LOCA Methodologies within the Industry: Past, Present and Future," in *Workshop on Best Estimate Methods and Uncertainty Evaluations*, Barcelona, Spain, 2011.

14. U.S. Nuclear Regulatory Commission, "Safety Evaluation by the Office of the Nuclear Reactor Regulation Topical Report WCAP-16009-P. Revision 0, "Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)"," U.S. NRC, Washington, DC, November 5, 2004.

15. U.S. Nuclear Regulatory Commission, ADAMS Accession Number ML051670395, U.S. NRC, Washington, DC, June 10, 2005.

16. "http://pbadupws.nrc.gov/docs/ML1211/ML121100200.pdf".

17. "http://www.gpo.gov/fdsys/pkg/FR-2014-03-24/pdf/2014-05562.pdf".

18. "http://pbadupws.nrc.gov/docs/ML1435/ML14358A493.pdf".

19. R. Borchard and M. Johnson, "10 CFR 50.46c Rulemaking: Request to Defer Draft Guidance and Extension Request for Final Rule and Final Guidance," U.S. Nuclear Regulatory Commission, Washington, DC, March 4, 2013.

20. C. Smith, C. Rabiti, R. Martineau, "Risk-Informed Safety Margins Characterization (RISMC) Pathway Technical Program Plan," INL-EXT-11-22977, Revision 2, September 2014

21. R. H. Szilard, C. l. Smith, R. Youngblood, "RISMC Advanced Safety Analysis Project Plan – FY 2015 – FY 2019," INL-EXT-14-33186, September 2014.

22. U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.174, U.S. NRC, Washington D.C, 1989.

23. L. Pagani, "On the Quantification of Safety Margins" (MIT, 2004).

24. Draft NEI White Paper, NEI-NRC Risk-Informed Steering Committee (RISC), January 2015.

25. Task group on Safety Margins Action Plan (SMAP). "Safety Margins Action Plan – Final Report" NEA/CSNI/R(2007)9.

26. F. Fouet, P. Probst, J.M. Lanore, "Impact on Safety Margins of a 10% Power Up-Rate for Zion NPP SBO Transient," Proceedings of PSAM-10.

27. Assessing Safety Margins: the impact of a power uprate on risk from Small and Medium LOCA scenarios, Vinh N. Dang, Tae-Wan Kim, Martin A. Zimmermann, and Annalisa Manera, Proceedings of PSAM-10.

28. "Technical Framework for Management of Safety Margins--Loss of Main Feedwater Pilot Application, EPRI 1023032," S. Hess (Project Manager), J. Gabor, R. Sherry, and D. True, Principal Investigators (EPRI, 2011).

29. Failure domain characterization: Loss of Main and Auxiliary Feed Water Example. V. Rychkov, EDF R&D, IDPSA workshop, Stockholm, 19-21 November 2012

30. Application of the Damage Domain approach to the calculation of exceedance frequencies, Javier HORTALa *, José M. IZQUIERDOa, César QUERALb, and Luisa IBAÑEZ, Proceedings of PSAM-10.
31. Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61), NUREG/1806 (NRC, 2007).
32. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1263, ADAMS Accession Number ML12284A323, U.S. NRC, Washington, DC.
33. J. V. Cathcart, R. E. Pawel, R. A. McKee, R. E. Druscel, G. J. Yurek, J. J. Cambell and S. H. Jury, "Zirconium Metal-Water Oxidation Kinetics IV. Reaction Rate Studies", ORNL/NUREG-17, Aug. 1977.
34. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1261, U.S. NRC, Washington, DC.
35. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-1262, ADAMS Accession Number ML12284A324, U.S. NRC, Washington, DC.
36. U.S. Nuclear Regulatory Commission , ADAMS Accession Number ML093350101, U.S. NRC, Washington, DC.
37. 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 (2009) PHISICS.
38. C. Rabiti, Y. Wang, G. Palmiotti, A. Alfonsi, A. Epiney, H. Hiruta, T. Grimmet, *PHISICS: Parallel and Highly Innovative Simulation for INL Code System,* INL/EXT-11-21384, Novembre 2011.
39. The RELAP5-3D© Code Development Team, "RELAP5-3D© Code Manual Volume I: Code Structure, System Models, and Solution Methods," Idaho National Laboratory Report, INEEL-EXT-98-00834, Rev. 4, June, 2012.
40. N. Horelik, B. Herman, B. Forget, K. Smith, "**B**enchmark for **E**valuation **A**nd **V**alidation for **R**eactor **S**imulations (BEAVRS) v1.01," Proc. Int. Conf. Mathematics and Computational Methods Applied to Nuc. Sci. & Eng., Sun Valley, Idaho, 2013.
41. American National Standard for Decay Heat Power in Light Water Reactors, ANSI/ANS-5.1-1979, August 1979.
42. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
43. AREVA NP, BAW-10179NP-A, "Safety Criteria and Methodology for Acceptance Cycle Reload Analyses," ADAMS Accession Number ML102210443, Lynchburg, VA, Revision 8, May 2010.
44. U.S. NRC, "FRAPCON-3.5: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup," NUREG/CR-7022, Vol. 1 Rev.1 PNNL-19418, Vol. 1 Rev.1, May 2014.
45. U.S. NRC, "FRAPTRAN-1.5: A Computer Code for the Transient Analysis of Oxide Fuel Rods," NUREG/CR-7023, Vol. 1 Rev.1 PNNL-19400, Vol. 1 Rev.1, May 2014.
46. J. D. Hales, et al., "BISON Theory Manual, The Equations behind Nuclear Fuel Analysis," Idaho National Laboratory, Jan., 2015.
47. J. D. Hales, et al., "BISON Users Manual," Idaho National Laboratory, Jan., 2015.
48. INL, *RELAP5-3D Code Manual Volume V: User's Guidelines*, INEEL-EXT-98-00834, Rev. 4, June, 2012.

49. L. J. Siefken et al., FRAP-T6: A Computer Code for the Transient Analysis of Oxide Fuel Rods, NUREG/CR-2148, EGG-2104, Idaho National Engineering Laboratory, May 1981.
50. U.S. Nuclear Regulatory Commission , ADAMS Accession Number ML13329A086; ML13329A090, U.S. NRC, Washington, DC, Fall 2013.
51. U.S. Nuclear Regulatory Commission , ADAMS Accession Number ML14358A493, U.S. NRC, Washington, DC, December 2014.
52. Kang, D.G., et al., “A Combined Deterministic and Probabilistic Procedure for Safety Assessment of Beyond Design Basis Accidents in Nuclear Power Plant: Application to ECCS Performance Assessment for Design Basis LOCA Redefinition,” Nuclear Engineering and Design, Vol. 260, pg. 165–174, 2013.
53. C. Smith, et al., “Risk Informed Safety Margin Characterization (RISMC) Advanced Test Reactor Demonstration Case Study,” Idaho National Laboratory, INL/EXT-12-27015, August 2012.
54. Pagani, L.P., Apostolakis, G.E., Hejzlar, P., 2005. The impact of uncertainties on the performance of passive systems. Nucl. Technol. 149 (2), 129–140.
55. NEA, 2007. Safety Margins Action Plan – Final Report, NEA/CSNI/R(2007)9, Task Group on Safety Margins Action Plan (SMAP), Committee on the Safety of Nuclear Installations. OECD Nuclear Energy Agency.
56. D.G. Kang and S.H. Chang, “The safety assessment of OPR-1000 nuclear power plant for station blackout accident applying the combined deterministic and probabilistic procedure,” Nuclear Engineering and Design, Vol. 275, pp.142-153 (2014).
57. A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, and R. Kinoshita, “Raven as a tool for dynamic probabilistic risk assessment: Software overview,” in Proceeding of M&C2013 International Topical Meeting on Mathematics and Computation, CD-ROM, American Nuclear Society, LaGrange Park, IL (2013).
58. D. Mandelli, C. Smith, Z. Ma, T. Riley, J. Nielsen, A. Alfonsi, C. Rabiti, J. Cogliati, ” Risk-Informed Safety Margin Characterization Methods Development Work,” INL/EXT-14-33191, INL, September 2014.
59. A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, R. Kinoshita, A. Naviglio, “Dynamic Event Tree Analysis Through RAVEN”, International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2013), September 22-26, Columbia, SC, USA, (2013).
60. A. Alfonsi, C. Rabiti, D. Mandelli, J. Cogliati, and R. Kinoshita, “RAVEN: Development of the adaptive dynamic event tree approach,” Tech. Rep. INL/MIS-14-33246, Idaho National Laboratory (INL), (2014).
61. J. C. Lee and N. J. McCornick, Risk and Safety Analysis of Nuclear Systems, Wiley, 2011.
62. R. Youngblood and C. Smith, “Technical Approach and Results from the Fuels Pathway on an Alternative Selection Case Study,” INL/EXT-13-30195, INL, September 2013.