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# Light Water Reactor Sustainability Program

# A Strategic Approach to Employ Risk-Informed Methods to Enable Margin Recovery of Nuclear Power Plants Operating Margins

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# A Strategic Approach to Employ Risk-Informed Methods to Enable Margin Recovery of Nuclear Power Plants Operating Margins

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# SUMMARY


The existing fleet of the nuclear power plants (NPPs) in the U.S. has been designed and constructed based on the defense-in-depth safety principle. Design basis safety analyses have been performed using deterministic approaches, which normally employ conservative models and assumptions to provide tolerances to account for uncertainties. The conservatisms associated with the current design basis safety analysis process provide sufficient margin such that the probability of damage to the plant should be negligible even under the worst considered plant conditions. However, culminations of conservatisms in the current process may reflect unrealistic operating situations that limit the operating flexibility of the current fleet and can result in adverse impacts on plant economics while not producing significant additional benefits to plant safety. Additionally, while the deterministic approaches use prescribed enveloping design basis accident scenarios, NPP operating experience has shown that more complicated scenarios, such as those that resulted from beyond design basis accident sequences during the Three Mile Island and Fukushima accidents, need to be considered. Operating experience has also indicated that use of probabilistic risk assessment approaches can be valuable to support decision making related to prioritizing safety improvements and determining the extent to which the benefits obtained are sufficient to justify the costs and expenditure of resources.

The Risk-Informed Systems Analysis Pathway of the U. S. Department of Energy Light Water Reactor Sustainability Program initiated research tasks to develop risk-informed multi-scale and multi-physics high fidelity analytical capabilities. These capabilities are intended to support the industry to assess and recover margins associated with the conservatisms in the current design basis process such that the existing NPPs can operate more efficiently and with more operational flexibility while continuing to meet all existing regulatory requirements and maintaining high levels of safety. The objective of the initiative is the development of an integrated evaluation approach that combines the plant PRA methods and models with Multi-Physics Best Estimate Plus Uncertainty analyses in a seamless fashion. The integrated evaluation framework that is developed will enable plant system configuration variations to be studied with speed and precision, including detailed risk and benefit assessments associated with the adoption of advanced nuclear technologies by the operating light water reactor plants in their pursuit of both safety and operational performance enhancements. This approach is used to identify the actual margins that are available for the accident scenarios so that decision makers (plant owner and regulator) can identify areas of excess margin. This will provide the potential for NPPs to reallocate that margin to higher priority applications and provide commensurate operational cost reductions.

The focus of this report is to present an integrated research and development roadmap to identify and perform high-value evaluations of advanced nuclear technology concepts with the ultimate goal of identifying the technical (e.g., benefits to risk, safety, and operational margins) and economic (e.g., business and cost) elements associated with industry adoption. The integrated evaluation approach will support the development and deployment of the advanced nuclear technologies that are capable of achieving substantial safety and economic improvements as well as timely widespread adoption by the U.S. nuclear industry.

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# ACRONYMS

| | |
|---|---|
| 1D/2D/3D | One, Two, or Three Dimensional (respectively) |
| AC | alternating current (electrical power) |
| AOO | Anticipated Operational Occurrence |
| ATF | Accident Tolerant Fuel |
| BDBA | Beyond Design Basis Accident |
| BEPU | Best Estimate Plus Uncertainty |
| BWR | Boiling Water Reactor |
| CCF | Common Cause Failure |
| CC/SW | Component Cooling/Service Water |
| CDF | Core Damage Frequency |
| CMFD | Coarse Mesh Finite Difference |
| CPR | Critical Power Ratio |
| CSAU | Code Scaling, Applicability and Uncertainty |
| CTF | COBRA-TF (Coolant Boiling in Rod Arrays – Two Fluid subchannel thermal-hydraulics analysis code) |
| CVCS | Chemical and Volume Control System |
| DBA | Design Basis Accident |
| DNB | Departure from Nucleate Boiling |
| DNBR | Departure from Nucleate Boiling Ratio |
| DOE | U. S. Department of Energy |
| ECCS | Emergency Core Cooling System |
| EMDAP | Evaluation Model Development and Application |
| EPRI | Electric Power Research Institute |
| ESFAS | Engineered Safety Feature Actuation System |
| FSAR | Final Safety Analysis Report |
| FW | Feedwater |
| FY | Fiscal Year |
| INL | Idaho National Laboratory |
| LERF | Large Early Release Frequency |
| LB-LOCA | Large Break Loss of Coolant Accident |
| LOCA | Loss of Coolant Accident |
| LOOP | Loss of Offsite Power |
| LOTUS | LOCA analysis toolkit for the US |
| LWR | Light Water Reactor |

| | |
|---|---|
| LWRS | Light Water Reactor Sustainability |
| MAAP | Modular Accident Analysis Program |
| MACCS | MELCOR Accident Consequence Code System |
| MCPR | Minimum Critical Power Ratio |
| MDNBR | Minimum Departure from Nucleate Boiling Ratio |
| MELCOR | Methods for Estimation of Leakages and Consequences of Releases |
| MOC | Method of Characteristics |
| MOOSE | Multi-Physics Object-Oriented Simulation Environment |
| MP-BEPU | Multi-Physics Best Estimate Plus Uncertainty |
| NPP | Nuclear Power Plant |
| NRC | U.S. Nuclear Regulatory Commission |
| PCT | Peak Clad Temperature |
| PIRT | Phenomenon Identification and Ranking Table |
| PRA | Probabilistic Risk Assessment |
| PWR | Pressurized Water Reactor |
| R&D | Research and Development |
| RD&D | Research, Development and Deployment |
| RCCA | Rod Cluster Control Assembly |
| RCS | Reactor Coolant System |
| RELAP5 | Reactor Excursion and Leak Analysis Program 5 |
| RELAP-7 | Reactor Excursion and Leak Analysis Program 7 |
| RHRS | Residual Heat Removal System |
| RIA | Reactivity Insertion Accident |
| RI-MP-BEPU | Risk-Informed Multi-Physics Best Estimate Plus Uncertainty |
| RISA | Risk Informed Systems Analysis |
| RPS | Reactor Protection System |
| SAMA | Severe Accident Mitigation Alternatives |
| SAPHIRE | Systems Analysis Programs for Hands-on Integrated Reliability Evaluations |
| SB-LOCA | Small Break Loss of Coolant Accident |
| SGTR | Steam Generator Tube Rupture |
| SRP | Standard Review Plan |
| SSC | systems, structures and components |
| TRACE | TRAC/RELAP Advanced Computational Engine |
| TRL | Technical Readiness Level |
| U.S. | United States |

# A Strategic Approach to Employ Risk-Informed Methods to Enable Margin Recovery of Nuclear Power Plants Operating Margins

## 1. INTRODUCTION

The nuclear industry in the United States has achieved superb performance in safety and reliability throughout the operating fleet of nuclear power plants (NPPs). The nuclear power industry also has a long history of working toward achieving improved plant economics through the introduction of many innovative technologies, such as longer operating cycles (many existing boiling water reactors (BWRs) currently are operating on a 24-month cycle), higher fuel enrichment (the fuel enrichment is close to the licensing limit of 5%), higher burnup of discharged fuel, and power uprates. Such innovations have led to remarkable improvements in the economic performance of existing NPPs. For instance, the total additional power generated from U.S. NPP power uprates is equivalent to the power that would result from building approximately eight new 1,000-MWe NPPs [1].

However, the current market conditions associated with the electric power market (abundant supplies of low-cost natural gas because of hydraulic fracturing extraction techniques, governmental subsidies for wind and solar generation technologies, etc.) have put significant strains on existing NPP economic competitiveness. At the present time, the operating costs of many NPPs are not able to match the production costs of alternative sources (such as natural gas fired plants and production tax credits associated with solar and wind power). Therefore, additional innovations will need to be introduced to the existing fleet of NPPs to remedy this situation.

To achieve these objectives, a great interest exists from the operating fleet to adopt technologies that can support cost reduction and operational performance enhancements. For example, increasing fuel enrichment and discharge burnup levels could be used to extend PWR operating cycles from the current 18 months to 24 months. Such extensions would provide for greater plant capacity factors resulting in improved plant economics. However, the impact of longer operating cycles, higher fuel enrichment, and plant power uprates is that these enhancements will result in larger activity inventory in the core and the plant response during operational transient and postulated accident conditions could get closer to the prescribed safety limits. Consequently, some of the innovations as well as plant aging have the potential to erode the available safety margins of these plants. Such potential impacts will need to be evaluated and managed if the operating fleet is to successfully obtain the benefits from these enhancements.

It is noted that the term "safety margins" is used to ensure that the structures, systems, and components (SSCs) in an NPP can perform their intended functions under both normal and abnormal operating conditions. The application of safety margins compensates for uncertainties in the phenomena and model data, and variability in the initial and boundary conditions associated with the analysis of events that can impact plant safety. Safety margins can also compensate (at least to some extant) for phenomena that may not have been foreseen during the design process. Simply put, safety margins provide allowances for insufficient knowledge or uncertainties associated with the design and operation of the plants.

Safety margins provide a buffer between the expected plant response during anticipated events and the point at which conditions will likely threaten plant safety (i.e., core damage or release of fission products to the environment). Since it takes time for the operating parameters in transients to overcome these buffers, the existence of safety margin allows plant safety systems and operating personnel to react to these events and mitigate their consequences.

The safety margin, in absolute terms, is defined as the distance between a regulatory acceptance criterion (or safety limit) and a physical limit, which is a critical value of a safety metric (e.g., pressure or temperature) at which the system or barrier loses its function. However, the actual value at which failure occurs is often difficult to determine in practice due to the complex physical phenomena that inhibits direct experimental measurement. As a result, the nuclear industry relies on highly sophisticated reactor systems safety analysis codes (such as RELAP5 and TRACE) to evaluate plant response to postulated transient and accident conditions. These codes are validated against a broad range of experimental data that span the applicable operating range of plant performance during transient and accident conditions. The use of these codes in reactor safety analyses requires that an "adequate safety margin" exists by demonstrating that the calculated safety metric value remains under the regulatory acceptance criterion.

The basic concept associated with margins is shown schematically in Figure 1 [2]. In its most simplistic form, the margin is represented as a distance between the load that an SSC experiences ($L$) and the capacity ($C$) that the SSC is capable of withstanding. Because there are uncertainties associated with both the load and the capacity, these parameters are more properly represented as distributions and the evaluation of margin becomes represented as the probability that the load experienced would exceed the SSC capacity [i.e., $P(L > C)$]. However, the determination of the actual distributions associated with SSC loads and capacities would be extremely time consuming and expensive to obtain. Thus, the approach chosen during the early days of reactor development and regulation was to specify a safety limit as a conservative point that would be used to ensure the probability that the load experienced by an SSC during some analyzed event would not exceed the system capacity would be acceptably low. Probably the most well-known application of this paradigm is the specification that fuel peak clad temperature (PCT) should not exceed 2200°F during plant transient or accident conditions.

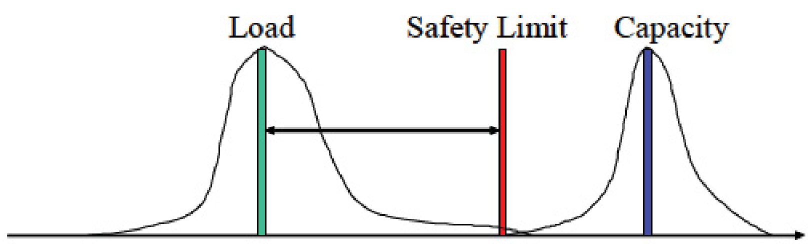


Figure 1. Safety margin concept [2].

The margin to physical limit (the point of failure) from the nominal operating point consists of two parts as shown in Figure 2 [3]. In this framework, the distance between the operating point and the prescribed safety limit (regulatory limit) is under the control of the licensee, and this is called the "licensing margin". However, the distance between the safety limit and the physical limit is controlled by the regulatory authority. It should be noted that the majority of the "margin recovery" activities described in this report target the former of these regimes as these margins are under the control of the licensee (e.g., the NPP owner/operator).

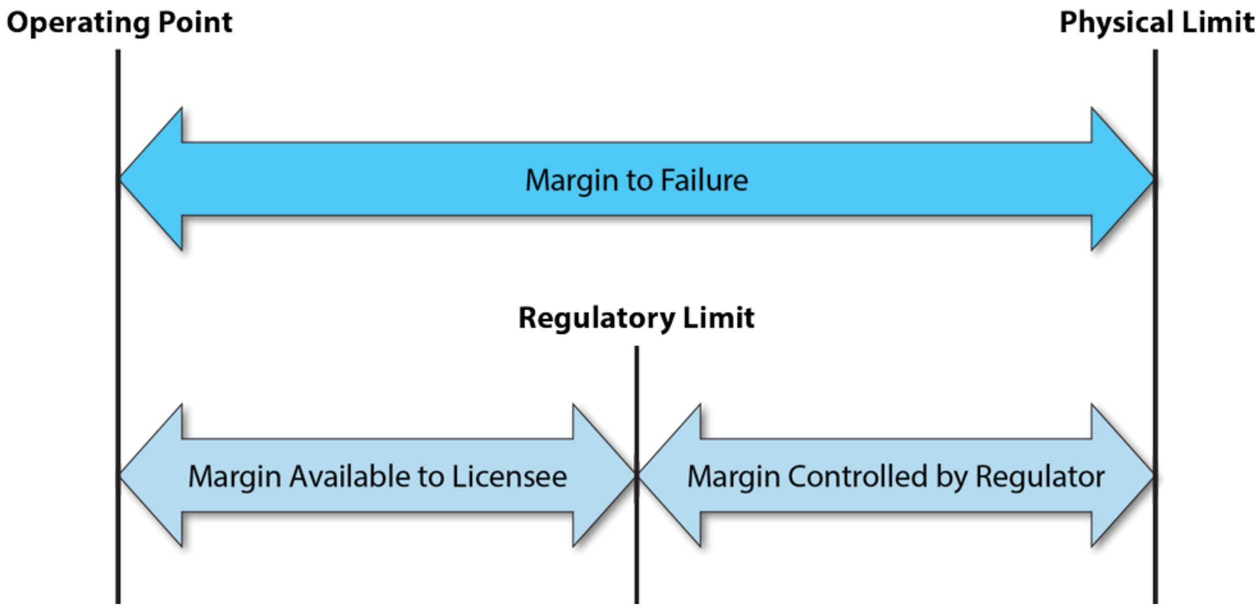


Figure 2. Safety margin applied to NPPs [3].

Because the regulatory authority sets the regulatory limit at a value that is also less than the point at which the unacceptable consequence would occur, there exists additional margin to the point at which failure will occur. For ease of analysis, the licensing margin is hereafter referred to as safety margin on the basis of analyses in this document. The commonly accepted methods to determine safety margins are the conservative approach (taking the highest possible uncertainty into account) such as use of the models prescribed in Appendix K to 10 CFR Part 50 – ECCS Evaluation Models [4] and Best Estimate Plus Uncertainty (BEPU) approaches that explicitly evaluate uncertainties and incorporate them into the decision process.

Over the past several decades the nuclear industry has been able to recover safety margins through multiple approaches such as plant equipment upgrades and modernization and the application of more sophisticated analytical capabilities. For instance, to evaluate the Emergency Core Cooling System (ECCS) performance, conservative models and assumptions were specified in Appendix K to 10 CFR Part 50. However, use of these conservative ECCS Evaluation Models provided conservative (e.g., minimal or pessimistic) evaluations of the safety margins. With improved understanding of plant transient and accident phenomena, efforts have been made to mitigate the conservative biases and assumptions in the evaluation model methodology, allowing a licensee to move toward BEPU methodologies. The 1988 amendment to the 10 CFR 50.46 rule allowed the use of realistic physical models to analyze loss-of-coolant accident (LOCA) and ECCS performance, with the provision that due allowance be given to any remaining uncertainties in the code, data, or modeling [5]. Consequently, BEPU modeling and simulation methodologies have been developed to demonstrate margin recovery, and are now extensively employed throughout the nuclear industry. The regulatory expectations for use of BEPU methodologies for NPP transient and accident analyses are specified in Regulatory Guide 1.203 [6]. This regulatory guidance provides a comprehensive description of the Evaluation Model Development and Application (EMDAP) that provides an integrated approach to the conduct NPP safety analyses.

The key aspect of the BEPU methodology is to quantify and propagate uncertainties in the calculations across all constituent phenomena that are modeled (reactor physics, thermal hydraulics, material properties, etc.). However, the computational constraints that arise due to the complex systems and interdependencies of variables historically have prevented the nuclear power industry from executing such multi-physics schemes. Because of these limitations, the existing BEPU methodology primarily focuses on the uncertainties in thermal hydraulics. This methodology is depicted in Figure 3 as an example of a LOCA analysis.

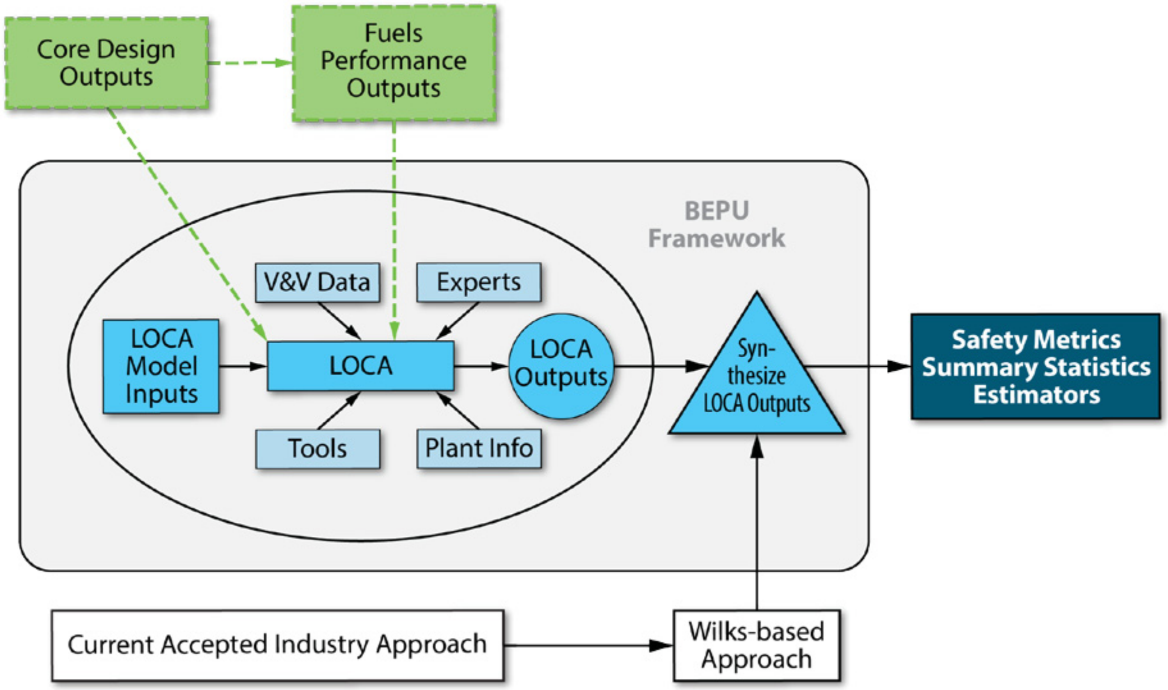


Figure 3. Schematic illustration of the current BEPU process for LOCA analysis.

It is noted that the current BEPU methodology still typically contains a high degree of conservatism, primarily as a means to mask knowledge gaps related to certain phenomena, and to simplify licensing and implementation. Further, because the complete propagation of uncertainties across the various disciplines can be prohibitively expensive (in terms of computational capability and time), bounding assumptions are often used where multiple phenomena need to be modeled and assessed to address uncertainty considerations. This approach, in turn, limits the ability to consistently propagate uncertainties in multi-physics simulations. Therefore, existing BEPU methods, as currently applied, often provide limited information on the actual margins available in the plants. As a consequence, a portion of the margins that exist in the plants continue to reside in engineering judgment and conservative assumptions, and from which it has proven to be extremely challenging to obtain economic benefits.

Moving forward, as more automation is adopted into plant processes, it is anticipated that the nuclear industry will develop better standardized databases and improved interfaces that function across the various engineering disciplines. Such standardization and increased automation will be capable of enabling new paradigms to evaluate and manage uncertainties across various disciplines and support a more integrated multi-physics approach that can be applied to the safety analysis problem. This will become more important as the industry adopts new and advanced nuclear technologies such as flexible operating strategies, further increases fuel enrichment and discharge burnup to extend operating cycle length, implements digital instrumentation and control upgrades, and deploys advanced nuclear fuel technologies, including accident-tolerant fuel (ATF). The evaluation and adoption of any of these enhancements will require detailed analyses of the fuel and SSC behavior within the context of entire plant system dynamics. Fortunately, because of the advancements in computing power over the past several decades, multi-physics simulations are now practical within the context of uncertainty quantification and sensitivity analysis (i.e., multi-physics best estimate plus uncertainty (MP-BEPU) methodologies).

Currently, the BEPU approach is predominantly applied to analyses of the predefined design basis accidents that are limited to single failures in active safety systems. Moving forward, a comprehensive listing of postulated initiating events (IEs) for all plant states should be prepared to ensure that the analysis is complete. An initiating event is an event that leads to an anticipated operational occurrence (AOO) or a postulated accident condition. A realistic analysis would include operator errors and equipment failures (both within and external to the facility), human-induced or naturally caused events, and internal or external hazards that, directly or indirectly, challenge one or more of the systems required to maintain the safety of the plant. Hence, probabilistic risk assessment (PRA) models are needed to provide grouping of postulated initiating events and their associated transients, and systematic enumeration of accident sequences with all logical combinations of failures and successes of safety and non-safety systems. Within the United States, and to a lesser extent internationally, PRA is being used increasingly as an important element in regulatory decision making. PRA methods not only determine the risk metrics, such as core damage frequency (CDF) or large early release frequency (LERF), but also determine what are the most probable accident sequences and the SSCs that contribute the most to the overall plant risk. Hence, combing PRA and MP-BEPU analysis would provide a comprehensive assessment of plant risks and permit a comprehensive quantification of margins. In addition, by providing a probabilistic evaluation, the results can be used to prioritize analyses to concentrate on those that have the highest likelihood of resulting in undesired consequences; thus ensuring efficient use of resources. In this report this approach is referred to as a risk-informed approach to recover safety margins.

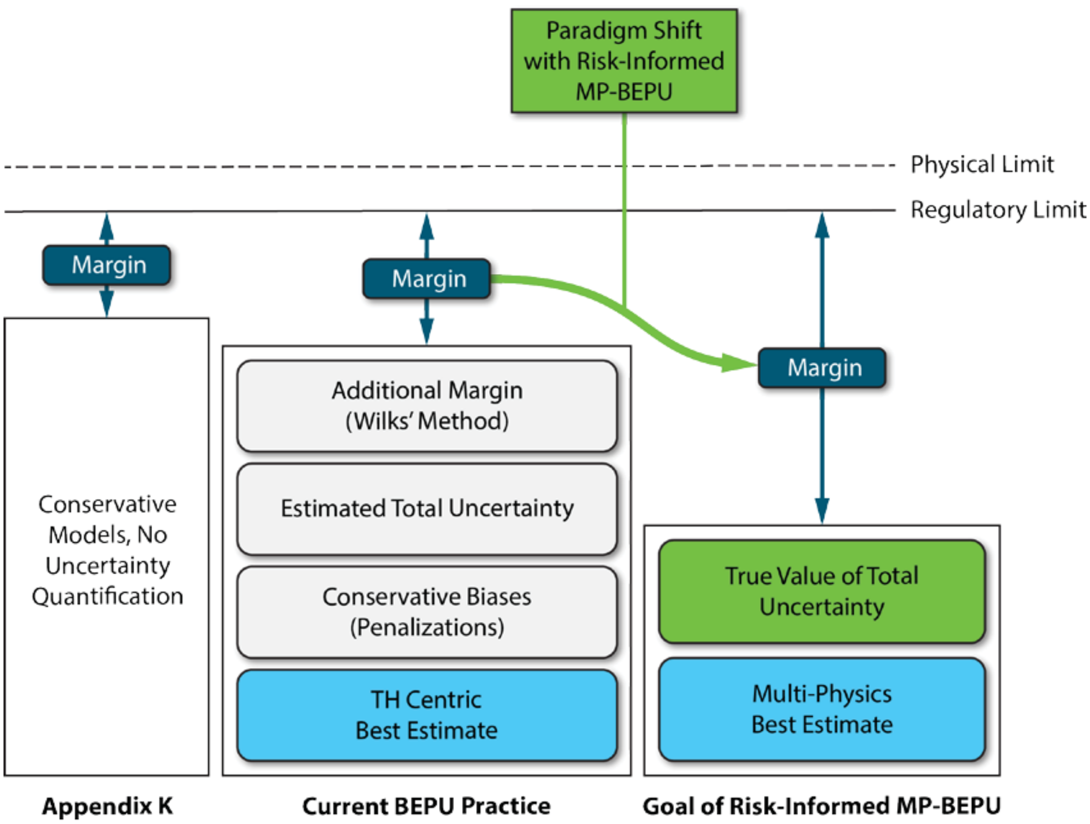


Figure 4. Schematic illustration of the objective of risk-informed multi-physics best estimate plus uncertainty (RI-MP-BEPU) framework.

This research and development (R&D) roadmap proposes to develop a Risk-Informed Multi-Physics Best Estimate Plus Uncertainty (RI-MP-BEPU) framework to conduct comprehensive investigations of design basis requirements and their implementation through plant processes and systems (SSCs, maintenance, surveillance, testing, qualification and quality requirements of encompassed systems, Technical Specifications, limiting conditions of operations, etc.) to identify and recover margins associated with uncertainties and conservatisms of legacy licensing, design, and analysis. The RI-MP-BEPU framework is an extension of the LOCA analysis toolkit for the U.S. (LOTUS) framework [7] being developed for LOCA applications in response to the proposed new rulemaking in 10 CFR 50.46c [8]. This approach is shown schematically in Figure 4.

The proposed RI-MP-BEPU framework will take advantage of modern high-fidelity probabilistic computing to support best estimate modeling that will support consistent uncertainty propagation and quantification. This approach will permit sensitivity analyses within a multi-scale and multi-physics environment to fully realize the benefits of multi-physics simulations. Utilizing state-of-the-art computational architectures, RI-MP-BEPU will integrate various simulation tools across the full spectrum of plant analysis activities, including Core Design, Fuels Performance, Component Aging and Degradation, Systems Analysis, Containment Response, Radionuclides Transport and Release, and Risk Assessment. This will allow complex multi-physics and risk-informed approaches to be implemented so that fully coupled NPP systems problems can be solved in a reasonable time. The approach is intended to be applied to identify the actual margins that are available for the spectrum of accident sequences assessed with the intent of permitting decision makers (both utility and regulatory) to identify where areas of excess margin exist for which the cost of the excess margin does not warrant the benefits. This will provide the potential for NPPs to reallocate that margin to higher priority applications and provide commensurate operational cost reductions.

Figure 4 above compares the RI-MP-BEPU approach with the existing Appendix K and BEPU approaches. The column on the right in Figure 5 represents the "ideal final solution" in a situation of "perfect knowledge." In this situation, RI-MP-BEPU is able to predict the "true" best-estimate or "nominal" state of the device (given plant, scenario, etc.), and can then account for all uncertainties in what is called the "true/theoretical value of total uncertainty." Compliance with acceptance criteria is demonstrated by showing that the MP-BEPU value is below the regulatory limit, which is designed by regulators to be below the physical limit. As discussed in the previous paragraph, for situations where the actual margin is larger than necessary, the RI-MP-BEPU framework will provide critical information that would permit reduction of margins that are overly conservative, thus providing the ability of operating NPPs to recoup monetary savings from these reductions.

Three elements are involved in establishing the RI-MP-BEPU framework. The first element is the integration and coupling of computer codes. The second element includes the uncertainty quantification and sensitivity analysis in the multi-physics simulations. The third element is the PRA integration. Section 4 of this report provides a detailed description of the RI-MP-BEPU framework.

The outcome of this research effort will be a Risk Informed Systems Analysis (RISA) R&D plan that is integrated with industry efforts to recover operating margins, reduce operating costs, and improve operational flexibility and efficiency, with the ultimate goal of risk-informing NPP activities while maintaining plant safety.

## 2. DESCRPTION OF NUCLEAR POWER PLANT CONDITIONS

An NPP could be either in operational states or accident states, as illustrated in Figure 5. Since 1970, the American Nuclear Society (ANS) classification of plant conditions has been widely used to divide plant conditions into four categories in accordance with the anticipated frequency of occurrence and potential radiological consequences to the public [9,10]. The four categories are:

Condition I: Normal Operation and Operational Transients

Condition II: Faults of Moderate Frequency (i.e. events that are expected to occur several times during the plant's lifetime)

Condition III: Infrequent Faults (i.e., events that may occur during the lifetime of the plant)

Condition IV: Limiting Faults (i.e. postulated accidents that are not anticipated to occur during the lifetime of the plant).

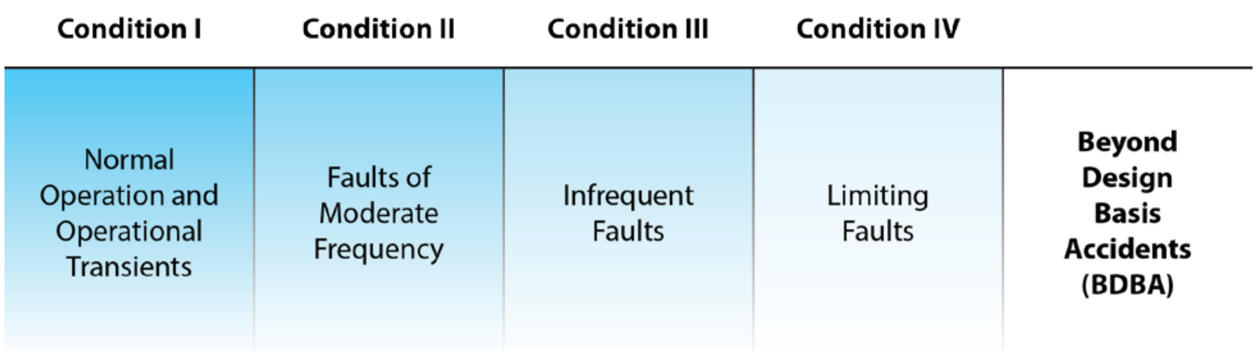


Figure 5. Illustration of NPP states.

The basic principle applied while relating design requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to the public, and those extreme situations which pose the greatest risk to the public should be those that are least likely to occur.

Condition I is defined as plant operation within specified operational limits and conditions. Since the NPP is designed to operate continuously and to be able to respond to anticipated transient events, Condition I events do not result in any adverse impacts on the plant or the public.

The Condition II and Condition III faults are typically referred to as AOOs. An AOO is an operational process that deviates from normal operation and is expected to occur at least once during the operating lifetime of an NPP. In view of appropriate design provisions, AOOs do not cause any significant damage to items that are important to safety and do not lead to accident conditions; however, they may result in a reactor scram. Plants should be able to handle the full range of these AOOs with no radioactivity release and return to normal operation. AOOs typically include events such as loss of normal electrical power, turbine trip, failure of control equipment, and loss of power to the main coolant pumps.

Condition IV faults are postulated design basis accidents (DBAs) that are not expected to occur during the operational lifetime of a NPP. An NPP is designed to withstand DBAs according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within acceptable regulatory limits. These postulated DBAs determine the criteria for the design and evaluation of various safety-related systems and equipment. For DBAs, the possibility of limited damage to the fuel is accepted, but offsite consequence release limits should not be exceeded.

All of the AOOs and DBAs are defined in Section 15.0 of the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP – NUREG 0800) [11]. Each operating NPP demonstrates how it meets these requirements in the plant safety analysis report.

In addition to Conditions I through IV described above, an NPP could also undergo an event that is considered to be beyond the design basis accident conditions. A Beyond Design Basis Accident (BDBA) involves accident conditions that are more severe than a design basis accident and that has the potential to result in core degradation. (Note that these events alternatively are known as severe accidents.) In a severe accident, the fuel rod integrity, primary system, and containment integrities may be breached, and the radioactive source term could be released to the environment. BDBA is used as a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely to occur.

## 2.1 Condition I – Normal Operation and Operational Transients

Condition I events are expected to occur frequently or regularly over the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter (e.g., system pressure) and the value of that parameter that would require either automatic or manual protective action (e.g., pressure scram setpoint value). Analysis of each fault condition is generally based upon a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

A typical list of Condition I events is shown in the following:

1. Steady state and shutdown operations
   a. Power operation with power between 5% and 100% of rated thermal power
   b. Startup with $k_{eff} \geq 0.99$ and power $\leq$ 5% of rated thermal power
   c. Hot standby (subcritical, residual heat removal system (RHRS) isolated)
   d. Hot shutdown (subcritical, RHRS in operation)
   e. Cold shutdown (subcritical, RHRS in operation)
   f. Refueling (plant shutdown with core alterations in progress)
2. Operation with permissible deviations

   Various deviations which may occur during continued operation, as permitted within accepted boundaries, must be considered in conjunction with other operational modes. These include:

   a. Operation with components or systems that are out of service as permitted by the plant Technical Specifications
   b. Leakage from fuel with clad defects
   c. Radioactivity in the reactor coolant
      i. Fission products
      ii. Corrosion products
      iii. Tritium
   a. Operation with steam generator leaks up to the maximum allowed by the Technical Specifications
   b. Testing as allowed by the Technical Specifications.

   It is noted that most of the items listed in this subsection are deviation only in the strictest sense of the term (i.e., they constitute normal operational maneuvering and actions conducted within the framework of the plant's operating license and Technical Specifications).
3. Operational transients
   a. Plant heatup and cooldown
   b. Step load changes

c. Ramp load changes
d. Load rejection up to and including design full load rejection transient.

## 2.2 Condition II – Faults of Moderate Frequency

The faults that fall into this condition, at worst (i.e., under the assumption of the single failure criterion), result in a reactor trip with the plant able to subsequently return to operation. These faults do not propagate to cause more serious faults (i.e., Condition III or IV events). In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system or secondary system overpressurization.

The following is a list of faults included in this category:

1. Feedwater (FW) system malfunctions that result in a decrease in FW temperature
2. FW system malfunctions that result in an increase in FW flow
3. Excessive increase in secondary steam flow
4. Inadvertent opening of a steam generator relief or safety valve
5. Loss of external electrical load
6. Turbine trip
7. Inadvertent closure of main steam isolation valves
8. Loss of condenser vacuum and other events resulting in turbine trip
9. Loss of nonemergency alternating current (AC) power to the station auxiliaries
10. Loss of normal feedwater flow
11. Partial loss of forced reactor coolant flow
12. Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low power startup condition
13. Uncontrolled RCCA bank withdrawal at power
14. RCCA misalignment (dropped assembly, dropped assembly bank, or statically misaligned assembly)
15. Startup of an inactive reactor coolant pump at an incorrect temperature
16. Chemical and Volume Control System (CVCS) malfunction that results in a decrease in the boron concentration in the reactor coolant
17. Inadvertent operation of the Emergency Core Cooling System (ECCS) during power operation
18. CVCS malfunction that increases reactor coolant inventory
19. Inadvertent opening of a pressurizer safety or relief valve
20. Break in instrument line or other lines from reactor coolant boundary that penetrate containment.

## 2.3 Condition III – Infrequent Faults

Condition III occurrences are faults that may occur very infrequently during the lifetime of a plant. They will result in the failure of only a small fraction of the fuel rods, although sufficient fuel damage may preclude resumption of operation for a considerable time after the event. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers.

The following is a list of faults included in this category:

1. Steam system piping failure (minor)
2. Complete loss of forced reactor coolant flow
3. Rod Control Cluster Assembly (RCCA) misalignment (single rod cluster control assembly withdrawal at full power)
4. Inadvertent loading and operation of a fuel assembly in an improper position
5. Small break loss of coolant accidents (SB-LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary
6. Postulated radioactive ground release due to liquid tank failures
7. Spent fuel cask drop accidents.

## 2.4 Condition IV – Limiting Faults

Condition IV occurrences are faults that are not expected to take place during the lifetime of the plant, but are postulated because their consequences could include the release of a significant amount of radioactive material to the environment. They are the most drastic postulated events that are designed against, and represent limiting design cases. In plant licensing, Condition IV faults are analyzed to ensure that they do not result in a fission product release to the environment resulting in an undue risk to public health and safety in excess of the guideline values specified in 10 CFR 100. A single Condition IV fault is not expected to cause a consequential loss of required functions of those systems needed to cope with the fault, including those of the ECCS and the containment.

The following is a list of faults classified in this category:

1. Major secondary system pipe rupture up to and including double-ended rupture of the largest system pipe (PWR and BWR)
2. Feedwater system pipe break
3. Reactor coolant pump shaft seizure (locked rotor) (PWR)
4. Reactor coolant pump shaft break (PWR)
5. Seizure of one recirculation pump (BWR)
6. Spectrum of RCCA ejection accidents (PWR)
7. Control rod drop accident (BWR)
8. Steam generator tube failure
9. Large break loss of coolant accidents (LB-LOCA) resulting from the spectrum of postulated piping breaks, including double-ended rupture, within the reactor coolant pressure boundary (PWR and BWR)
10. Design basis fuel handling accidents.

## 2.5 Beyond Design Basis Accidents

Beyond design basis accidents (BDBA) are very low frequency events, brought about by multiple failures, which may result in changes to the reactor core configuration and significant radioactive material releases from the damaged core. In worst case severe accident scenarios, the reactor core becomes molten and the reactor containment is breached. These beyond-design-basis accidents are not usually analyzed in safety analysis reports; however, they are included in PRA studies.

BDBAs are accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely to occur during the operational lifetime of the NPP. In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand. As the regulatory process strives to be as thorough as possible, BDBA sequences are analyzed to understand the full scope and capability of a design. Examples of BDBAs include Station Blackout, Anticipated Transients Without Scram, etc.

# 3. RISK-INFORMED APPROACH TO RECOVER MARGINS

The safety assurance of an NPP is to protect the onsite operating staff, the public, and the environment from the hazards of the radioactive materials generated as a result of the fission process. Safety systems are designed and installed to detect the initiating conditions of undesired abnormal situations and mitigate their consequences. The design, operation, and analysis of an NPP have to comply with strictly prescribed safety regulations and standards issued by regulators such that the nuclear safety objectives are achieved and an adequate level of protection of public health and the environment is provided.

Safety analyses are analytical evaluations of physical phenomena occurring at NPPs. These evaluations are made for the purpose of demonstrating that safety requirements, such as ensuring the integrity of barriers against the release of radioactive material and various other acceptance criteria, are met for all postulated initiating events that could occur over a broad range of operational states, such as those described in Section 2. Within the context of a plant probabilistic risk assessment, these analyses would include assessments for different levels of availability of the safety systems. Safety analysis methods are developed and utilized to evaluate the plant behavior and safety margins.

Defense-in-depth is one of the fundamental safety principles in the design and construction of the existing NPPs in the U.S. This principle has a strong influence in safety culture and licensing requirements, and also in plant operations. As a key element of the defense-in-depth principle, design basis safety analyses are performed using deterministic approaches that use mathematical models to calculate the time history of essential plant variables after the initiating event occurs to study the plant response. The current regulatory framework is based largely on deterministic approaches that employ safety margins, operating experience, and accident analyses. The purpose of deterministic calculations of plant response (physics, chemistry, thermal-hydraulics, etc.) to postulated events is to verify the safety design, which demonstrates that the licensing requirements are fulfilled, and to make realistic safety assessments for actual or anticipated events. Essential parameters include fuel and cladding temperature, fuel rod surface heat flux, reactor pressure, as well as temperature and pressure in the reactor containment, etc. The design bases of each nuclear unit are documented in its Final Safety Analysis Report (FSAR), which is updated periodically as the Updated Safety Analysis Report. NPP operation, including maintenance and surveillance of safety-related equipment, is controlled and restricted by Technical Specification requirements. Deterministic analyses involve standard good engineering practices, calculations, and judgments. The design bases include the assumption of worst-case conditions for accident analyses. Examples of these worst-case conditions include the assumptions of an initial reactor power of greater than 100%, restrictive power distributions within the core, conservative engineering factors, the minimum required accident mitigation equipment available, and pipe breaks of all possible sizes. A specific set of rules and acceptance criteria are applied. Typically, the analyses focus on neutronic, thermal hydraulic, radiological, thermomechanical, and structural aspects, which are often analyzed with different computational tools. The computations are usually carried out for predetermined operating modes and operational states, and the events include AOOs, DBAs, BDBAs, and severe accidents with core degradation. The results of computations are spatial and time dependent of various physical variables (thermal power of the reactor; pressure, temperature, etc.) or, in the case of an assessment of radiological consequences, radiation doses to workers or the public. The deterministic methods generally assume a bounding set of fault conditions. The adoption of conservative assumptions relating to plant and system performance is an accepted approach to address uncertainty when performing these deterministic analyses. Two types of deterministic approaches are widely used in the nuclear industry. These are indicated as Options I and II as shown in Table 1, which is adapted and modified from IAEA Specific Safety Guide No. SSG-2 [12].

Table 1. Comparison of options for performing safety analysis.

| Option | Computer Codes | Availability of Plant Systems | Initial and Boundary Conditions |
|---|---|---|---|
| I. Conservative | Conservative | Conservative Assumptions | Conservative input data |
| II. Best Estimate Plus Uncertainty | Best Estimate - TH centric | Conservative Assumptions | Mixture of realistic and conservative inputs, boundary conditions, and data. Includes assessment of uncertainties |
| III. Risk-Informed Multi-Physics Best Estimate Plus Uncertainty (RI-MP-BEPU) | Best Estimate – Multi-Physics simulations | Derived from PRA | Realistic inputs, boundary conditions, and data with assessment of uncertainties |

Option I is a Conservative Deterministic approach. In this option, the computational models are conservative and are intended to produce demonstrably conservative results. The selected initial and boundary conditions, including the time available for operator actions, are assumed to have conservative values. The most severe single failure of the safety systems that are designed to mitigate the consequences of the accident is assumed. There is a large number of NPPs in the U.S. that still use this option to license their plants to demonstrate compliance to the acceptance criteria specified in 10 CFR Appendix K.

Option II uses the best estimate models in the code instead of conservative models, together with more realistic initial and boundary conditions. The approach typically employs an EMDAP as described in Regulatory Guide 1.203 [6]. Uncertainties are identified and propagated through the calculations so that the uncertainty in the calculated results can be estimated. A high probability that acceptance criteria would not be exceeded should be demonstrated. The uncertainties associated with the use of a best estimate computer code and realistic assumptions for the initial and boundary conditions should be combined statistically. Any dependence between uncertainties, if present, should be taken into account. In addition, it should be verified that the ranges of parameters that are applied are realistic. Sensitivity studies should be performed, especially to detect the possibility of any "cliff edge effects." Different variations of Option II are used in practice depending upon the experimental data. This is because whenever extensive data are available, the tendency is to use realistic input data, and whenever data are scarce, the tendency is to use conservative input data.

With deterministic approaches, the premise is that with an adequate selection of the analysis cases, the use of bounding codes and assumptions and the selection of suitable acceptance criteria provide confidence that plant operation will result in negligible damage even under the worst postulated plant conditions. However, the exclusive use of the deterministic analysis of DBAs to assess the plant safety could be insufficient due to the following two reasons:

1. analyses are limited to several "worst case" postulated accidents (which have very low likelihood of occurrence),
2. a limited number of faults are considered.

The combination of these two factors could lead to the prediction of incorrect progression of accidents or exclude some important physical phenomena and produce non-representative, or misleading insights. As a result, decisions made based on these types of analysis might not always be the most prudent in terms of plant risk minimization. Even though the current BEPU approaches address certain conservatisms, substantial conservatisms are still built into the current BEPU process. Consequently, more sophisticated methodologies should be developed to increase the "Realism" in plant safety analyses. It should be noted that the objective of developing methods that support increased realism for NPP safety analyses is not limited to obtaining a more detailed understanding of where conservatisms have been previously applied. The increased realism in analysis also should be used to identify specific instances where margins are sufficiently robust such that the additional margins do not provide sufficient benefits to justify their costs. It is relaxation of these margins that will provide the capability to enhance plant economic performance (and concomitant societal benefit) without substantively reducing plant safety.

To address the first concern, as alluded in the Introduction section, MP-BEPU methodologies will be developed in the conduct of Margin Recovery and Operating Cost reduction research, development, and deployment (RD&D) activities. The intent is to develop an approach that is both rigorous and cost effective such that it can replace the conservative Appendix K approach as well as make the current BEPU approach more cost effective. The MP-BEPU approach within the Light Water Reactor Sustainability (LWRS) RISA Pathway includes BEPU simulations in core design, fuels performance, components aging and degradation, systems analysis, containment response, and radionuclides release, transport, and dose consequences. This proposed approach has the potential to address far more conservatisms than the current BEPU approach.

One use of the PRA techniques is to address the concern of analyzing a limited number of faults (single failure criterion) considered in the deterministic (design basis) analyses. PRA is an established technique to numerically quantify risk measures in a NPP. The PRA approach uses historic data (e.g., service experience) and analytical techniques to estimate the likelihood of risks versus their consequences. A key benefit of PRA is that it evaluates a (relatively) complete spectrum of possible incident scenarios that may occur at an NPP (i.e., it is not limited to the smaller subset of AOOs/DBAs that are required for regulatory/licensing submittals nor is it limited to postulating single failures). Additionally, because the PRA uses probabilistic techniques, by its very nature it is capable of evaluating and addressing uncertainties. Within application to commercial NPP operation, risk is defined as the product of the likelihood and consequences of events (including rare events such as severe accidents) with results generally reported as an annual frequency of undesired consequences (typically core damage and large early release of fission products to the environment). Rather than focusing on incredulous events (e.g., the double-ended guillotine pipe rupture of a main coolant line), PRA includes the full range of potential events that have the potential to impact NPP safety. Basically, an NPP PRA answers three questions:

1. What can go wrong?
2. How likely is it?
3. What are the consequences?

The PRA augments and complements traditional deterministic engineering analyses by providing quantitative measures related to public safety (i.e., in the form of frequency of core damage and release of radioactive materials to the environment) and thus provides a means of addressing the relative significance of issues in relation to plant safety.

Throughout the history of commercial nuclear power, the nuclear industry, regulatory agencies, and the nuclear energy research community have continued to research and implement new and better methods to operate, maintain, test, and analyze NPPs and equipment to reduce risk and to ensure safety. A well-integrated PRA and MP-BEPU analysis framework, or the so-called RI-MP-BEPU described in this R&D plan, will provide the capability to increase safety analysis "Realism" such that excess safety margins can be recovered.

The belief that significant margins are available for recovery with minimal impact on safety is a consequence of analyses of NPP performance and studies of plant risk that have been performed. In these assessments, the safety goals established by the US Nuclear Regulatory Commission are taken as the fundamental policy objectives of protecting the health and safety of the general public [13]. The metrics that are evaluated in an NPP PRA (Core Damage Frequency and Large Early Release Frequency) provide estimates for the point at which an NPP accident would need to progress before it would have any possibility to challenge the public health objectives specified in the NRC policy statement. Although an NPP PRA intends to provide a realistic assessment of the frequency of occurrence of these events, from the perspective of the policy objectives described in 51 FR 30028 [13], they can be considered to represent a conservative method of demonstrating that the policy objectives are met. A recent Electric Power Research Institute (EPRI) sponsored study [14] showed that, based on PRA results from U.S. operated NPPs, there is roughly a factor of 100,000 margin to the quantitate health objective (QHO) for an individual fatality to a member of the public due to an NPP accident. In addition, this study estimated a factor of ~70 of margin for the risk of an individual fatality due to latent cancer using worst case assumptions and a factor of ~300 margin when the risk is weighed by the estimated frequency of event occurrence. In addition to these results, additional EPRI-sponsored analyses [14,15] has shown a roughly 10-fold decrease in plant CDF since the early 1990s. Given these data, there is ample reason to expect that there are margins that are available for which the cost of their maintenance does not provide sufficient public benefit and the reallocation of these margins to improve NPP economics would have greater public benefits in the form of improved plant operational performance and economics.

It is proposed that a well-integrated RI-MP-BEPU safety analysis framework can provide additional information to demonstrate compliance to the requirement, "sufficient safety margin is maintained," specified in RG 1.174 [16] in the assessment of these margins and the considerations of their reallocation. The RI-MP-BEPU framework would remedy the current incoherence between the probabilistic and deterministic safety analysis approaches due to a lack of consistency between the accident sequences considered and a lack of a uniform assessment of acceptable risk of undesired consequences.

The RI-MP-BEPU framework, as schematically illustrated in Figure 6, will be used to assess the benefits of advanced nuclear technologies in terms of safety, operational performance, and economics at existing NPPs. Successful application of such methods by NPPs will accelerate the implementation of an economically optimal combination of advanced nuclear technologies to improve safety and performance while also reducing maintenance and operational costs.

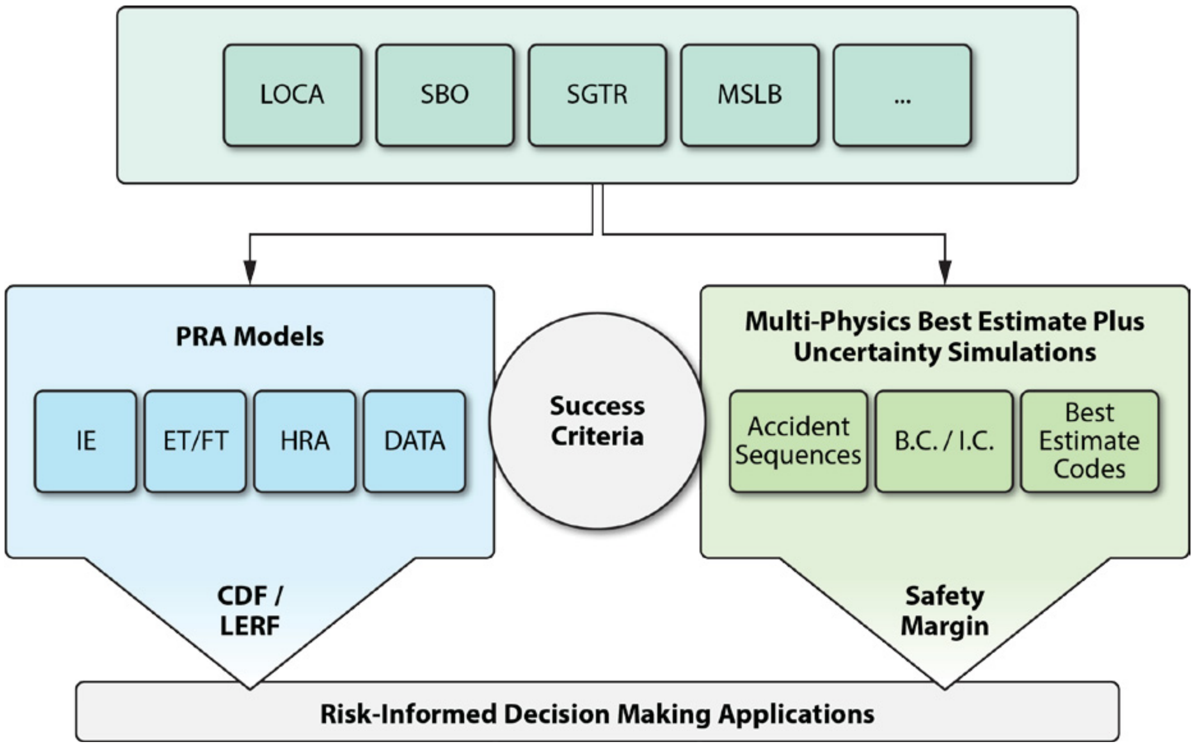


Figure 6. Illustration of the well-integrated risk-informed multi-physics best estimate plus uncertainty approach to recover safety margins.

In summary, Table 1 above provides a summarized comparison of the three different options discussed in this document to perform safety analysis. Option I (conservative approach similar to Appendix K analysis methods) and variations of Option II (current BEPU approach) are currently widely used in the nuclear power industry. Option III (RI-MP-BEPU) is what this R&D aims to achieve to modernize the safety analysis of NPPs to accelerate the deployment of advanced nuclear technologies into the existing operating NPPs. Option III is intended to provide increased realism for NPP safety analyses to permit obtaining a more detailed understanding of where conservatisms have been applied in previous methods. The increased realism can then be used to identify specific instances where excess margins exist and do not provide sufficient benefits to justify their costs.

As an example of a potential application of the RI-MP-BEPU approach, the interaction between the trip limit and the respective safety limit for a particular parameter provides a measure of operational flexibility. The trip limit, which is of considerable significance to safety margin, is determined based on plant response to AOOs and DBAs. To meet the acceptance criterion for a specific analyzed event, the trip limit is set such that the AOO or DBA does not pose any safety concerns to the plant systems and operators. To ensure this, the trip limit is always set conservatively (with due consideration given to uncertainties and other influencing parameters such as instrument drift rates) so that the safety limits are not jeopardized even during the worst case postulated event progression. However, this approach comes at the cost of reducing available operating margin and flexibility, a result of which is an increased likelihood of unnecessary trips due to the reduced operating margin. The RI-MP-BEPU approach could be used to provide more realistic information about the physical behavior of the plant, assist in identifying the most relevant safety parameters and allow more realistic comparison with acceptance criteria.

Table 2. Illustration of margin space to be explored.

| **“System”** | **Performance Metric** | **Example of Margin Contributors** |
|---|---|---|

| | | |
|---|---|---|
| Nuclear Power Plant | Safety margin | *L* = scenarios are modeled that represent component failures/successes leading to an increased core coolant temperature<br>*C* = ability of the fuel/clad to withstand elevated core coolant temperature |
| Facility Fire Model | Safety & economic margin | *L* = critical specific fire scenarios are modeled that provide impact to core damage frequency<br>*C* = ability of the plant systems to withstand fire events |
| Structures such as the Core Internals | Economic margin | *L* = scenarios are modeled that account for potential costs of off-normal conditions and replacement due to core internal degradation issues<br>*C* = ability of the core internals to withstand radiation embrittlement and corrosion |
| Component such as an Emergency Diesel Generator | Seismic margin | *L* = scenarios are modeled that estimate the energy transferred from an earthquake using non-linear soil-structure interaction analysis<br>*C* = ability of a diesel generator to withstand the energy transferred from the earthquake |

In practical engineering applications, the actual loads ($L$) and capacities ($C$) associated with the plant systems are uncertain; as a consequence, most engineering margin evaluations should account for the uncertainties and therefore contain a probabilistic element. The RI-MP-BEPU approach provides probabilistic safety margins that are defined by the probability that the load exceeds the capacity, under simulated scenario conditions. For example, in a loss-of-coolant accident analysis in which the calculated peak clad temperature during the event (loading condition $L$) is a distribution, and $C$ is the peak clad temperature capacity, the probabilistic margin would be represented by the expression $Pr(L > C)$. Table 2 gives an example of the types of probabilistic margins that the RI-MP-BEPU framework aims at investigating.

For accident scenarios in which the margin is smaller, RI-MP-BEPU is necessary to quantify conservatisms. For AOOs, the use of RI-MP-BEPU may avoid the selection of unnecessary restrictive limits and set points, and may provide a more precise evaluation of actual margins relating to the limits and set points. This, in turn, would provide additional operational flexibility and reduce unnecessary reactor scrams or actuations of the protection systems. An immediate potential application of this approach would be to support flexible NPP operation in which NPPs vary plant output in response to changes in load on the electrical grid (so called “load following” operation). This represents a potentially high value application of the RI-MP-BEPU approach as a number of NPPs are being increasingly required to operate within this framework (as compared to their historical operation as base load generating units).

# 4. DESCRIPTION OF RISK-INFORMED MULTI-PHYSICS BEPU ANALYSIS FRAMEWORK

As discussed in Section 3 in the conduct of this RD&D area, the probabilistic margin approach will be used to support recovery of safety margins and to quantify impacts on economics and reliability. Toward this end, an RI-MP-BEPU analysis framework will be built to treat uncertainties directly and to avoid conservatisms associated with the current analysis approaches. Further, this approach will be used in risk-informed margins management to present results to decision makers as it relates to margin evaluation, management, and recovery strategies.

Three components are involved in the development of the RI-MP-BEPU framework. The first component is the integration and coupling of computer codes across various physics. A "plug-and-play" multi-physics computer codes integration environment will be developed such that all the computer codes involved in the analyses will be brought under one roof. The second component is to develop a multi-physics BEPU methodology for the FSAR Chapter 15 accident and transient analyses with the emphasis on consistent uncertainty propagation and quantification in multi-physics simulations. The third component is to apply the integrated PRA/MP-BEPU (the so-called RI-MP-BEPU) approach to systematically explore the potential to recover excess margins for the AOOs and DBAs for the existing operating fleet of NPPs. In the initial stages of the RD&D effort, the Department of Energy (DOE) will work with partner utilities/NPPs to identify high-value applications to demonstrate the RI-MP-BEPU approach.

## 4.1 Code Integration and Coupling for Risk-Informed Multi-Physics BEPU Simulations

### 4.1.1 Code Integration

Performing safety analysis for an NPP involves multiple disciplines including core design, fuel/clad performance, components aging and degradation, systems analysis, containment analysis and radioactive material release and consequence analysis. Traditionally, the analyses are performed sequentially. This means that the analysis performed to address one portion of the physics does not necessarily provide sufficient consideration for the downstream analyses that need to be performed. As a result, the boundary conditions used for one set of analysis frequently assume conservative values from the upstream analyses. As a minimum, this sequential processing of information is inefficient and results in excess expenditures of time and resources. This often is exacerbated by the need to revise previous analyses due to results obtained during later analyses. This need to operate in a cyclic manner can add significantly to the expense and time required to conduct these analyses.

In addition, different models and assumptions went into the computer codes developed for each of the physics being analyzed. As a result, the conventional approach and methods are strongly "code-oriented." The analyst has to be familiar with the details of the codes utilized, in particular with respect to their input and output structures. This represents a significant barrier for widespread use outside of the small pool of experts that develop and apply the codes. It becomes apparent how difficult it is to make changes and accelerate progress under such a paradigm, especially in a heavily regulated environment where even a single line change in a code can carry a heavy cost of bookkeeping and regulatory review. This "divide-and-conquer" approach currently is adopted in the industry where every physics is resolved independently and coupling is addressed by complex interface procedures. The current process is labor intensive and inefficient. There are significant assumptions and engineering judgments used in setting up those procedures that makes the propagation of uncertainties across the disciplines complex and potentially prone to errors. More importantly, continued use of these current methods has a significant bias to retain excess analytical margins, which cannot be exploited at a later time to enhance operational and economic performance.

To address the conservatisms built into the current practice, it is essential to propagate uncertainties across the stream of physical disciplines and manage the data stream. The use of an integrated approach in managing the data stream is probably the most important aspect of what is proposed here. This also is well suited with the current trends in industry to enhance automation and develop integrated databases across their organizations.

Our vision is to move toward to a "plug-and-play" or task-oriented approach, as illustrated in Figure 7, where the codes are integrated together under one roof and each code is simply treated as a module "under the hood" that provides the input-output relationship for a specific analytical discipline. The focus shifts on managing the data stream at a system level, as depicted in Figure 7. The "plug-and-play" multi-physics environment is essentially a workflow engine with capability to drive physics simulators, model complex systems and provide risk assessment capabilities. The various components, as depicted in Figure 7, are described in the following subsections.

As shown in Figure 7, the plug-and-play multi-physics environment retrieves all values of interest from output files and stores them in a more compact manner in HDF5 format [17]. HDF5 is a data model, library, and file format for storing and managing data. It supports an unlimited variety of data types, and is designed for flexible and efficient input/output and for high-volume and complex data. HDF5 is portable and is extensible, allowing applications to evolve in their use of HDF5. The HDF5 technology suite includes tools and applications for managing, manipulating, viewing, and analyzing data in the HDF5 format. The data are also easily accessible for use in other codes. Provided that the needed data were calculated and stored, any arbitrary codes can be added into the multi-physics integration environment in an ad-hoc manner and access previously generated data. This flexibility in storage allows for a plug-and-play environment.

The philosophy of integrating various computer codes under one roof provides the opportunity to propagate uncertainties consistently in multi-physics simulations. Subsection 4.2 has more details on the uncertainty quantification methodologies used in RI-MP-BEPU.

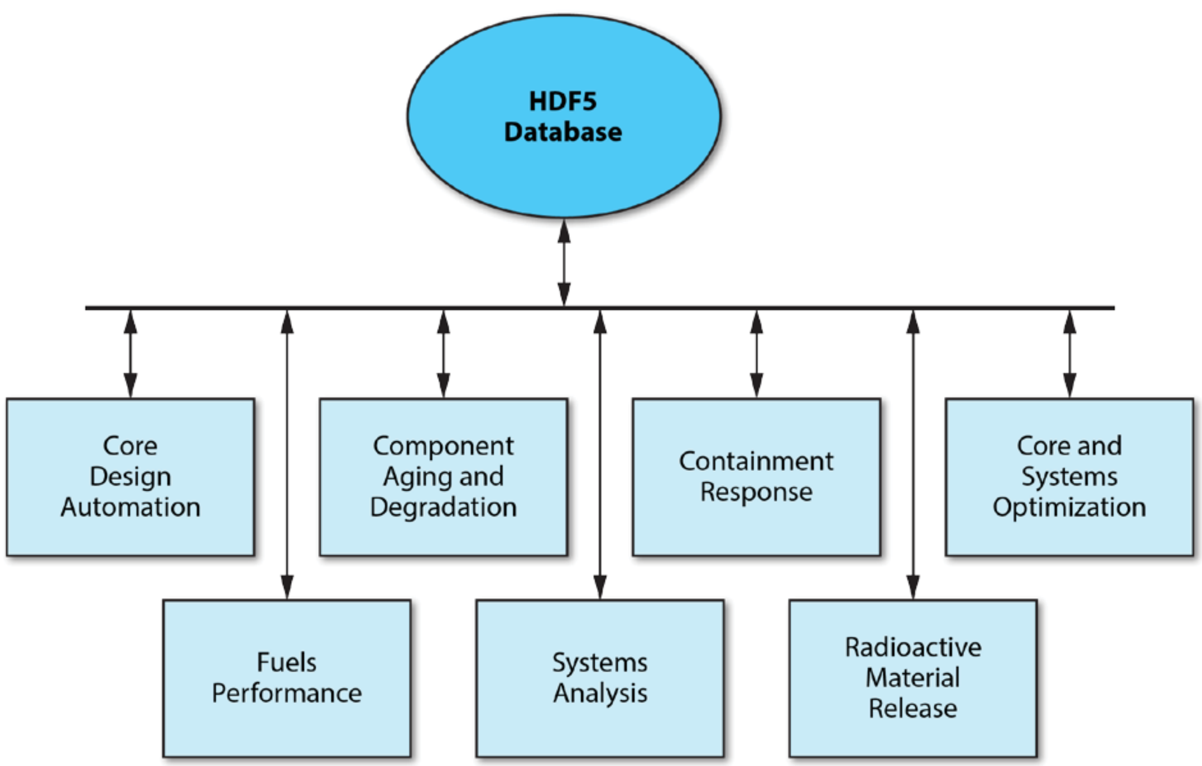


Figure 7. Notional illustration of "plug-and-play" multi-physics integration.

### 4.1.2 Code Coupling

The RELAP5-3D code is a widely used computer code for reactor systems safety analysis. However the reactor core modeling in RELAP5-3D plant models is normally done with simplified models. For certain transient calculations, such as non-LOCA transients where the departure from nucleate boiling ratio (DNBR) has to be evaluated, more detailed subchannel analysis (e.g., using a code such as COBRA-TF) has to be performed and coupled such that RELAP5-3D/COBRA-TF calculations will be required. Similarly, RELAP5-3D has simple fuel rod performance models (mainly heat conduction and rod distortion) under transient conditions. To obtain a more detailed and realistic assessment of the fuel rods behavior under transient conditions, coupled RELAP5-3D/BISON and RELAP5-3D/FRAPTRAN calculations are required. In this RD&D project, code coupling activities will be carried out to couple RELAP5-3D and COBRA-TF, and RELAP5-3D and BISON, as well as RELAP5-3D and FRAPTRAN for transient analyses of fuel performance. The coupling between RELAP5-3D and COBRA-TF will be performed to support DNBR calculations for non-LOCA transients. Currently, this is being planned for 2019. For this coupling, two approaches are being investigated. The first approach is one-way coupling from RELAP5-3D to COBRA-TF while the second is two-way coupling between the codes. The coupling between RELAP5-3D and BISON is a longer term activity that is needed to provide more complete understanding of the behavior of the reactor fuels under transient conditions.

It should be noted that although this RD&D effort will concentrate on use of RELAP5-3D as the systems code which will be coupled to other “specialty” codes, this HDF5 “plug and play” format will permit the process to be extended to include other codes as desired. For example, the US NRC applies the TRACE systems code for the purposes of confirmatory regulatory analysis. The structure of the RI-MP-BEPU approach will support the coupling of TRACE and other codes within the framework; thus enabling a broad user base across the industry.

## 4.2 Uncertainty Quantification and Sensitivity Analysis for Multi-Physics Best Estimate Simulations

Since uncertainties exist in the current approach to estimate and manage safety margins, significant research efforts are being made in seeking techniques to obtain more complete characterizations of analytical results. Traditionally the safety margin estimation is mostly based on conservative evaluation model calculations. Thus, the analytical safety margin has a high level of conservatism that can present a skewed understanding of the actual operating situation and limit the potential for enhancement of plant performance. More realistic analyses should be used to evaluate the evolution and consequences of plant transients and accidents. The use of Best Estimate analysis together with an evaluation of the Uncertainties, or the so-called Best Estimate Plus Uncertainty approach, is increasing for the following reasons:

1. The use of conservative assumptions may sometimes exclude or mask certain important physical phenomena. Hence there is a potential to overlook some key sequences of events that are important in assessing the safety of the plant.
2. The use of conservative approaches tends to produce pessimistic results, resulted from the prediction of an incorrect progression or unrealistic timescales of events, and often does not show the true margins to the acceptance criteria that apply in reality. This situation is exacerbated by the fact that the current process for performing safety analysis is sequential in practice with each step providing conservatisms related to its specific physics. This often results in a compounding effect such that the final analytical results are extremely conservative leading to the specification of overly conservative operational limitations and requirements. On the surface, this may seem acceptable from a safety perspective; however, since plant resources are finite, overly conservative limits can skew the

distribution of resources away from where they could provide more value (in terms of both safety and economic performance).

3. A Best Estimate Plus Uncertainty approach provides more realistic information about the physical behavior of the plant, assists in identifying the most relevant safety parameters, and allows more realistic comparison with acceptance criteria. Thus, by providing more realistic outcomes the approach has the potential to permit reductions in unnecessary operational restrictions and requirements with the ability to enhance operational and economic performance without substantively reducing plant safety.

As an outcome of implementation of the updated regulations in 1988 to allow best estimate methods to be used in ECCS/LOCA analysis, the code scaling, applicability, and uncertainty (CSAU) methodology was developed and documented in NUREG-5249 [18]. Accompanying NUREG-5249, the USNRC released Regulatory Guide 1.157 [19], best-estimate calculations of emergency core cooling system performance, which provides specific details describing acceptable best-estimate LOCA methodologies. The CSAU methodology represents a framework for deriving a quantifiable degree of assurance from a best estimate analysis tool. The CSAU framework outlines a procedure that leads from the identification and characterization of the dominant phenomena influencing the key acceptance parameter, fuel peak clad temperature (PCT), to quantify a best-estimate of the consequences of an LB-LOCA and its associated uncertainty. Additional guidance on use of an EMDAP for application of analytical methods and computational codes to support NPP accident and transient analysis is provided in Regulatory Guide 1.203 [6].

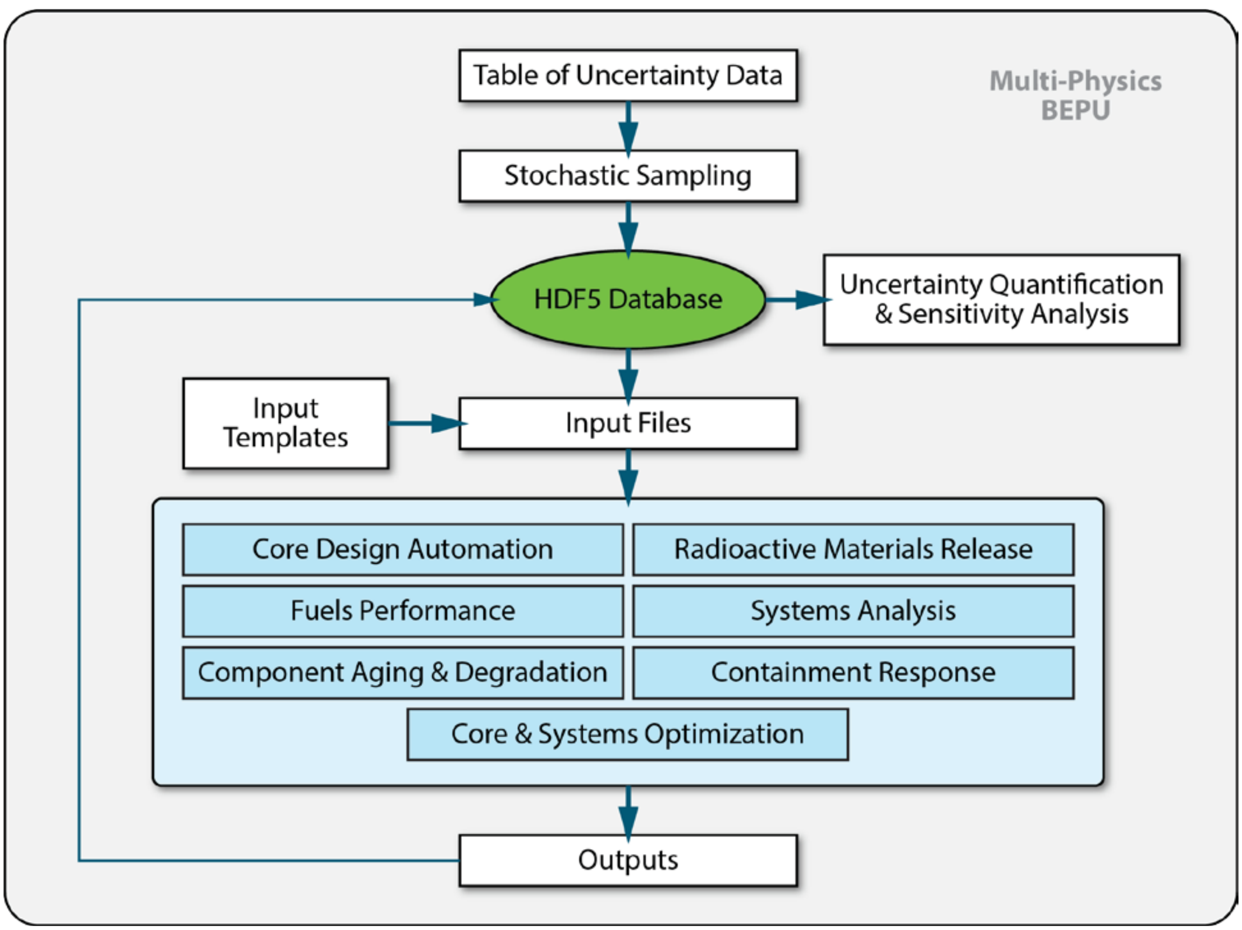


Figure 8. Notational illustration of multi-physics best estimate plus uncertainty (MP-BEPU) analysis approach.

In the conduct of this RD&D, the CSAU methodology will be extended to support multi-physics simulations to develop the so-called MP-BEPU methodology. This is shown schematically in Figure 8 in which the MP-BEPU approach serves as a wrapper that is applied to each of the constituent single physics-based codes shown schematically in Figure 7. Regardless the specific codes used to model the physics involved, the methodology discussed here is really a different strategy in managing the uncertainties. In the MP-BEPU methodology, uncertainties are propagated directly from all the uncertain design and model parameters. The interactions between the various model parameters are directly solved within the MP-BEPU framework. This not only facilitates the automation of the process but it is also mathematically more robust because the advanced procedures considered to propagate uncertainties and/or perform global sensitivity and risk studies require that the inputs sampled be independent. This requirement is hard to achieve following the traditional "divide-and-conquer" approach. Note that in the current process applied for safety analyses (i.e., the sequential process described previously) the condition of independence is very difficult to achieve (and typically is either just assumed or ignored).

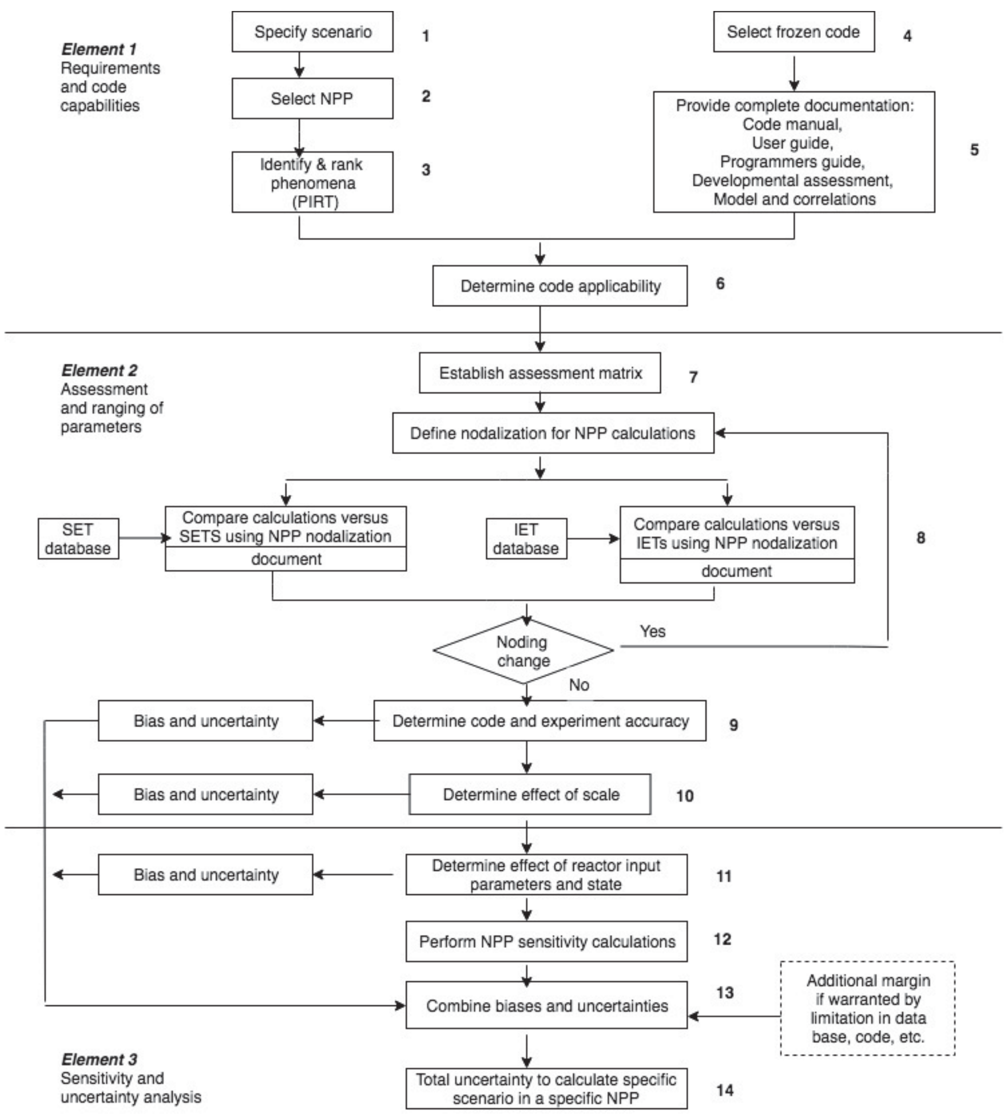


Figure 9. The CSAU methodology framework [18].

The development of the MP-BEPU methodology will closely follow the procedures outlined in the CSAU framework. As is graphically shown in Figure 9, this framework consists of three elements and 14 steps that build on a qualitative understanding of the ECCS/LOCA problem to define the necessary tasks to derive a quantitative solution. The CSAU elements and steps for this process are described in more detail below.

### 4.2.1 Element 1 – Requirements and Code Capabilities

The first CSAU element, which consists of six steps, establishes the foundation of understanding to guide methodology development. It emphasizes defining the problem and capturing the knowledge that will be used to provide the fundamental technical basis for decisions making downstream in the methodology development process. Steps 1, 2, 4, and 5 shown in Figure 10 identify the problem through specification of the event scenario, plant type, computer codes and versions, and computer codes documentation, respectively. Step 3, identify and rank phenomena (using the Phenomena Identification and Ranking Table – PIRT – process), is a particularly important step given the substantial effort required to develop a CSAU-based methodology. It is recognized that different processes and phenomena would have different influences on the plant behaviors. In the PIRT process, engineering judgment or expert opinion is formulized to aid both the methodology development and regulatory review. This step provides the basis to reduce the analysis effort to a manageable set of phenomena ranked with respect to their importance on the safety metrics. In the traditional BEPU approach, the PIRT process is an ad-hoc process with the PIRT being developed based on expert's opinions and judgments.

Here we note that Step 4 (Select frozen code) of the CSAU process represents a process that potentially conflicts with modern software development processes. In the modern approach to software development, the process is executed in a continuous development (daily builds, sprints, SCRUM process, etc.) with code verification performed on a nearly continuous basis via regression testing typically performed daily to achieve some selected fraction of code coverage and less frequently (large scope regressions) for more (nearly) complete coverage. Therefore, code verification is an integral and ongoing part of the development and delivery process. Because the codes selected for use in RISA applications are standard codes with extensive validation bases, once the integrated regression testing procedures are implemented, further development and application of the codes can proceed using standard modern software development processes. Such an approach will be essential to apply these codes within timeframes that are needed by the industry.

Step 6 in Element 1 serves to establish computer codes' applicability to the analysis problem. This is done by defining a cross reference of phenomena and plant components to the computer code's models, correlations, and nodalization capability.

### 4.2.2 Element 2 – Assessment and Ranging of Parameters

This element of the CSAU methodology establishes the methodology's and computer codes' pedigree to perform a best-estimate plus uncertainty analysis. This is done by verification and validation of models, uncertainty and sensitivity analyses. Step 7 defines the codes' assessment matrix. Fuels performance codes like FRAPCON/FRAPTRAN and BISON and systems analysis codes like RELAP5-3D and TRACE include a large number of closure-relationships to address the broad spectrum of possible phenomena. It is important to assess every code model and correlation to support the subset of important phenomena anticipated during a transient. As indicated previously, because the codes selected for use in RISA applications have had extensive verification and validation performed on them, the models and correlations that they currently contain can be considered adequate for use in RISA applications. Therefore, the modifications to these codes will concentrate on those aspects that pertain specifically to the Use Case applications of interest. As a result, PIRTs and subsequent sensitivity analyses will be carried out to identify any critical parameters or additional data (including any potential experimental programs) needed for code assessment to support the industry application. It should be noted that due to the high costs of experimental testing, maximal use of available experimental data that currently exists will need to be used and future experiments will need to concentrate only on high-value areas where gaps exist. A comprehensive resource that lists and classifies existing experimental data that has been developed is provided in a 2014 EPRI report [20].

Step 8 defines the NPP systems nodalization, which presents an inherent code uncertainty. However, nodalization-induced code uncertainty is deemed to be of lesser importance relative to the practical requirements of model accuracy and calculation efficiency. The objective is to define the minimum noding needed to capture the important phenomena.

Step 9 is to determine code and experiment accuracy. Code assessment using the test matrix from Step 7 and the nodalization of the NPP model from Step 8 are used to accomplish this step. Code accuracy is quantified for bias and deviations through confirmatory code uncertainty and benchmarks. This step also serves as a validation for Step 6, code applicability, and sets up the tasks of Element 3, sensitivity and uncertainty analysis. The demonstration of code accuracy or adequacy has been a required component in computer codes development and applications. For a CSAU-based evaluation methodology, the emphasis is focused on evaluating the important individual phenomena to the overall code uncertainty.

Step 10 determines the test scalability. In the long history of code models correlations development, computer code models and correlations have often been tuned to particular data sets. This approach to computer development can create result biases and uncertainties associated with the scaling of the problem of interest. Scaling uncertainties can be evaluated using data from a suite of test programs generated at various scales.

### 4.2.3 Element 3 – Sensitivity and Uncertainty Analysis

Given the inherent uncertainty and complexity of the multi-physics processes that occur during transients, a best-estimate statement of compliance to acceptance criteria must be provided statistically. This CSAU element focuses on setting-up, executing, and evaluating a transient analysis. As a statistically based methodology, the problem setup involves identifying and evaluating the impact of the biases and uncertainties for transient contributors identified from CSAU Elements 1 and 2. Execution involves the convolution of these uncertainty contributors and the final result is evaluated from the number of calculations necessary to provide a statistically meaningful set.

Step 11 in this element addresses the uncertainties associated with the measurable states that define a plant's operating condition, such as pressures, temperatures, water levels, etc.

The objective of CSAU Steps 12 and 13 is to combine the biases and uncertainties of the important individual contributors as identified in Step 9 and Step 11 through the execution of a large number of plant simulations. The convolution of the many uncertainty contributors to a specific transient is an inherently statistical approach. The two most commonly used approaches are either parametric or nonparametric. For instance, the response surface method is a parametric method. The number of calculations required for that approach is dependent on the number of uncertainty parameters considered. The nonparametric approach decouples the association between the number of uncertainty parameters and the number of required calculations. The desired quantification of the selected figure of merit (e.g., minimum departure from nucleate boiling ratio (MDNBR)) uncertainty is the identification of a specific result that represents coverage of the results domain at or above 95% with a 95% confidence. The 95/95 coverage/confidence has been recognized by the USNRC as having sufficient conservatism for use in transient analyses.

The final step, Step 14, in the CSAU process is to identify the total uncertainty. The total uncertainty can be quantified relative to a "best-estimate" figure-of-merit.

## 4.3 Probabilistic Risk Assessment (PRA) Methods Integration

Probabilistic Risk Assessment (PRA), also called Probabilistic Safety Assessment, is defined in the ASME/ANS PRA Standard [21] as "a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public." It is an engineering analysis approach that models the entire plant and its constituent systems in an integrated fashion to discover subtle interrelationships and evaluate the risk and vulnerabilities associated with plant operations. PRA answers three fundamental questions, which are called the "risk triplet": What can go wrong? How likely is it? And, what are the consequences?

PRA uses a logical deductive approach to systematically identify potential NPP accident scenarios and estimate the likelihoods and consequences of these scenarios. PRA can quantify risks associated with plant performance measures and provide insights into the strengths and vulnerabilities of the design and operation of an NPP. An NPP PRA model typically starts with the identification of the hazards and a spectrum of initiating events. Event trees are then used to model the response of the plant's systems and operators to each initiating event, with different plant system and human action combinations. PRA models provide sequences that, depending on successes or failures of relevant systems, lead to either a safe or a core damage end state. Fault trees are used to identify all combinations of equipment failures and human errors that could lead to system failures, and quantify the overall failure probability for each plant system represented in the event trees. Data analysis is performed to estimate the frequencies of initiating events and probabilities of the basic events representing equipment failures and unavailabilities, while human reliability analysis estimates the probabilities of human errors. The event trees and fault trees are linked and quantified to calculate the likelihood of all the sequences that lead to the same outcome (e.g., core damage (output as CDF) or large early release (output as LERF)). PRA makes use of realistic assessments of the performance of the equipment and plant personnel by considering a wide range of faults, taking an integrated look at the plant as a whole such that system inter-dependencies can be accounted for, and using realistic acceptance criteria for the performance of the plant and systems. Through developing event trees and fault tress, PRA evaluates a large list of possible accidents in a systematic way, selects those that are evaluated to occur above a certain frequency and evaluates the effectiveness of protection/mitigation for them.

Another important attribute of PRA is that it involves analyses of both single and multiple failures. Multiple failures often lead to situations beyond the plant design basis and, in some cases, are more likely to occur than single failures. By addressing multiple failures, a PRA can cover a broad spectrum of potential accidents at a plant and provide a realistic and comprehensive assessment of them.

PRAs are generally divided into three different levels. Level one efforts identify potential plant damage states that lead to core damage and their associated probabilities. Level two efforts model damage progression and containment strength for establishing fission-product release categories and their likelihood of occurrence. Level three efforts evaluate the potential offsite consequences of radiological releases and the probabilities associated with the occurrence of those consequences.

Traditional PRA methodologies, such as the event tree/fault tree methods, have the limitation that they do not explicitly trace time elements in the plant system and human response models, but rather are static and binary logic-based. Alternatively, Dynamic PRA methodologies have been investigated to improve traditional static PRAs. For example, simulation-based PRA methodologies treat time elements and the dynamical interactions among the plant systems and human actions explicitly in the PRA model by generating numerous simulations to represent accident scenarios. Significant research efforts have been spent on developing different Dynamic PRA methodologies [22,23] while the methodologies have been applied in applications such as Dynamic PRA modeling of a digital feedwater control system [24] and simulation-based dynamic approach for external flooding analysis [25]. However, to date, the application of Dynamic PRA technologies has not been widely adopted across the commercial nuclear industry. This is due to several reasons with the foremost being the complexity associated with the models and the requisite computational capabilities required for their evaluation. Additionally, in application to regulatory issues, no consensus set of standards exist for their application (as exists for traditional PRA methods in reference [21]).

PRA is an important element of the NRC's licensing and regulatory processes. The NRC's PRA Policy Statement issued in 1995 [26] encourages the staff and industry to increase the use of PRA in all nuclear regulatory matters "to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach" and "support the traditional defense-in-depth philosophy." The PRA models and results, applied as part of the so called "risk-informed" approaches, have since been widely used by the NRC and the nuclear industry in their decision making processes. Using PRA in the decision making process has aided licensees in aging management, the definition of enhanced strategies for in-service inspection, and in determining which design modifications are desirable from both risk-reduction and cost-benefit standpoints for the improvement of plant safety. For example, the NRC uses PRA in nuclear reactor regulatory activities such as the development of regulations and guidance, licensing decisions and certification of reactor designs, oversight of licensee operations and facilities, and the evaluation of operational experience. These are indicative of a trend towards a modern risk informed approach to safety regulation in which PRA is used to provide inputs to decisions concerning safety.

Integrating PRA with MP-BEPU approaches would expand the analysis scope from the Design Basis Space to the much broader and comprehensive risk space with which any possible transient in the plant would be included. Figure 10 provides the schematic illustration of the risk-informed multi-physics BEPU analysis framework indicating its relationship to the MP-BEPU approach described previously.

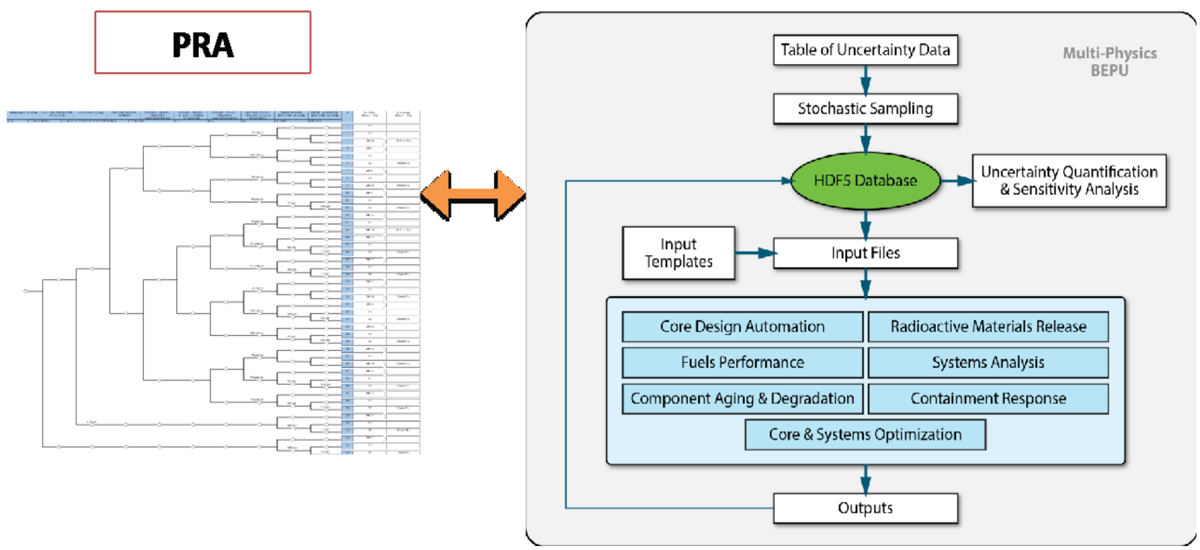


Figure 10. Notional illustration of the implementation of risk-informed multi-physics best estimate plus uncertainty analysis approach.

As a risk-informed multi-physics analytical framework, RI-MP-BEPU is not intended to replace a licensing Analysis of Record, but rather to augment the "engineering judgment," which is typically applied in their management and maintenance. The goal is an analytical and computational approach that can represent a power plant realistically with all the uncertainties included and that considers all relevant phenomenology that are involved in an integrated fashion. The objective is that such an approach can be used to perform comprehensive evaluations of plant safety to assess actual margins to permit effective decision-making. The ultimate outcome of this research will permit comprehensive evaluations across the full spectrum of anticipated plant events from which margins can be critically assessed and from which decisions can be made for those situations where margins are considered to be larger than necessary (and from which cost/benefit tradeoffs can be evaluated).

The plug-and-play approach employed in the RI-MP-BEPU framework will enable plant owners and reactor vendors to consider and further customize the multi-physics analysis framework for use within their established codes and methods. Therefore, it could potentially become the engine for license-grade methodologies at some point in the future. In other words, it is possible that RI-MP-BEPU 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 changes that are required by 10 CFR 50.46c.

# 5. RESEARCH AND DEVELOPMENT ACTIVITIES

One of the objectives of the RISA Pathway is to apply methods and tools to assess the safety, risk, and economic impacts of advanced nuclear technologies so that they can be more easily adopted by the existing fleet of NPPs. A fundamental objective of this research is to support reductions in the capital and operating costs of these NPPs to enhance their economic competitiveness while maintaining high levels of safe and efficient operation. One requirement for the licensing and deployment of any advanced nuclear technology that can impact plant safety margins will be an assessment of its performance under postulated transient and accident conditions. The postulated events that will require such analyses are defined in Section 15.0 of the U.S. NRC Standard Review Plan (SRP; NUREG 0800) [11]. The RD&D plan described in this report will focus resources on analyses of the anticipated performance of proposed advanced nuclear technologies across the full spectrum of plant transients and accidents, including AOOs, design basis accidents (DBAs), and BDBAs.

AOOs span the full range of conditions for which light water reactors (LWRs) must be evaluated for advanced new nuclear technologies to be licensed and deployed in the existing fleet. AOOs and DBAs constitute the spectrum of design basis events that are required for analysis in the licensing of an NPP in the United States. Because the set of DBAs required for licensing represent some of the most extreme conditions that an NPP could reasonably be expected to experience during the course of its operating life, these events can be used in the initial evaluations as a valuable representation of the potential benefits that can be provided by advanced nuclear technologies.

## 5.1 Analysis Acceptance Criteria

The acceptance criteria are very different for LOCA and non-LOCA events. Their respective acceptance criteria are described below:

### 5.1.1 LOCA

The following analysis acceptance criteria, which are excerpted from 10 CFR 50.46, apply for LOCA:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F
2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation
3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react
4. Calculated changes in core geometry shall be such that the core remains amenable to cooling
5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

### 5.1.2 NON-LOCA

The following are the specific acceptance criteria for AOOs and have been excerpted from Section 15.0 (Subsection I.2.A) of the NRC Standard Review Plan [11]:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs and that the critical power ratio (CPR) remains above the minimum critical power ratio (MCPR) safety limit for BWRs.
3. An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the reactor coolant system or reactor containment barriers.

**Condition II Events**

1. Same as Criterion (1) above for AOOs
2. Same as Criterion (2) above for AOOs
3. By itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV category without other incidents occurring independently or result in a consequential loss of function of the reactor coolant system or reactor containment barriers.

**Condition III Events**

1. No more than a small fraction of the fuel elements in the reactor are damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time.
2. For PWRs, the release of radioactive material may exceed guidelines of 10 CFR Part 20, but shall not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. For BWRs, the offsite release of radioactive material is limited to a small fraction of the guidelines of 10 CFR Part 100, which may be the result of the failure of a small fraction of the fuel elements in the reactor.
3. A Condition III incident shall not, by itself, generate a Condition IV fault or result in a consequential loss of function of the Reactor Coolant System (RCS) or reactor containment barriers.

**Condition IV Events**

A postulated accident could result in sufficient damage to preclude resumption of plant operation. For the postulated accidents (which are Condition IV events) or design basis accidents, the following acceptance criteria, which are excerpted from Section 15.0 (Subsection I.2.B) of the NRC Standard Review Plan [11] apply:

1. Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.
2. Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs. If the minimum DNBR or MCPR does not meet these limits, then the fuel is assumed to have failed.
3. The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.
4. A postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

## 5.2 Accident Categorization

Because LOCA and non-LOCA events have different analysis acceptance criteria, it is appropriate to group these postulated events into the following categories. The categories indicated below are from Section 15.0 of the NRC Standard Review Plan [11].

### 5.2.1 NON-LOCA

As described in Section 15.0 of the NRC Standard Review Plan [11]. AOOs and postulated accidents can be grouped into the following seven types:

1. Increase in Heat Removal by the Secondary System

   A malfunction which causes an increase in heat removal by the secondary system results in a decrease in the temperature of the coolant in the primary system. In the presence of a negative moderator temperature coefficient, this can result in an increase in the core power level and a reduction in the MDNBR. In addition, if the malfunction is due to an increase in feedwater flow, this can cause overfilling of the steam generator. These transients are primarily "system-driven" in that the system transient results are not dictated by specifics of the fuel assembly geometry, but rather by the response of the reactor coolant system to the transient conditions. The details of the fuel assembly and fuel rod design are not modeled in the system transient and are not critical parameters. The analyses of these events are performed to confirm that the primary coolant temperature reduction and associated insertion of positive reactivity do not result in an excessively large power increase that challenges the DNB limit for the plant. A number of events have been postulated that could result in an increase in heat removal from the Reactor Coolant System by the secondary system. Analyses are presented for several such events which have been identified as limiting cases.

   a. Feedwater system malfunction causing a reduction in feedwater temperature
   b. Feedwater system malfunction causing an increase in feedwater flow
   c. Excessive increase in secondary steam flow
   d. Inadvertent opening of a steam generator relief or safety valve causing a depressurization of the Main Steam System
   e. Spectrum of steam system piping failures inside and outside containment.

   The above are considered to be Condition II events, with the exception of a major steam system pipe break, which is considered to be Condition IV event.

2. Decrease in Heat Removal by the Secondary System

   A malfunction that causes a decrease in heat removal by the secondary system results in an increase in the temperature of the coolant in the primary system. The heat up and expansion of the coolant can lead to a reduction in the DNBR, a primary or secondary system pressure increase, or pressurizer overfill. As with the cool-down events described in 1 above, these events are primarily system-driven. The details of the fuel assembly and fuel rod are not modeled in the system transient and are not critical parameters. For example, the loss of normal feedwater/feedwater pipe break events are driven by the heat transfer between the primary and secondary systems and, in particular, the performance of the auxiliary feedwater system. The analyses of these events are performed to confirm that limits on reactor coolant system pressure, pressurizer water volume, and secondary side pressure are met. A number of non-LOCA heatup transients and accidents have been postulated that could result in a reduction of the capacity of the secondary system to remove heat generated by the core and transported in the Reactor Coolant System.

   a. Steam pressure regulator malfunction or failure that results in decreasing steam flow
   b. Loss of external electrical load
   c. Turbine trip
   d. Inadvertent closure of main steam isolation valves
   e. Loss of condenser vacuum and other events causing a turbine trip
   f. Loss of nonemergency AC power to the plant auxiliaries

g. The loss of nonemergency AC power event can also result in a flow coastdown due to a loss of power to the reactor coolant pumps.
h. Loss of normal feedwater flow
i. Feedwater system pipe break

The above faults are considered to be Condition II events, with the exception of a feedwater system pipe break, which is considered to be a Condition IV event.

3. Decrease in Reactor Coolant System Flow Rate

A malfunction that causes a decrease in the reactor coolant flow rate results in an increase in the temperature of the primary coolant in the core, and a decrease in the ability of the coolant to remove heat from the fuel. This can cause a reduction in the MDNBR and, in the case of a Locked Rotor event, a rapid increase in reactor coolant system pressure. As with the cooldown and heatup events described in 1 and 2, the reactor coolant system response to a loss of flow is "system driven" in that the system transients are dictated by the system transient conditions. An evaluation will be performed to confirm that the DNBR results for the partial/complete loss of forced coolant flow and Locked Rotor events remain valid. The peak cladding temperature is calculated for the Locked Rotor event. A number of faults are postulated that could result in a decrease in reactor coolant system flow. These include:

a. Partial loss of forced reactor coolant flow
b. Complete loss of forced reactor coolant flow
c. Reactor coolant pump shaft seizure (locked rotor)
d. Reactor coolant pump shaft break.

Item (a) is considered to be a Condition II event, Item (b) a Condition III event and Items (c) and (d) are Condition IV events.

4. Reactivity and Power Distribution Anomalies

Several non-LOCA transients are characterized by changes, either locally or globally, in core reactivity or power shape. The resulting increase in core power, or the core power peaking factor, could cause a reduction in the MDNBR. A number of faults have been postulated that could result in reactivity and power distribution anomalies. Reactivity changes could be caused by RCCA motion or ejection, boron concentration changes, or addition of cold water to the Reactor Coolant System. Power distribution changes could be caused by RCCA motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. The RCCA Ejection event is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. In the case of the RCCA Ejection event, the concern is the post-DNB pellet temperature and enthalpy increase. (Note that the RCCA ejection event is classified as a DBA event and not an AOO.) A number of faults are postulated that could result in an unintended increase in reactivity or reactor power. These include:

a. Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition
b. Uncontrolled RCCA bank withdrawal at power
c. RCCA misalignment
d. Start of inactive reactor coolant pump at an incorrect temperature
e. Malfunction or failure of a flow controller in a boiling water reactor
f. CVCS malfunction that results in a decrease in boron concentration in the reactor coolant
g. Inadvertent loading and operation of a fuel assembly in an improper position

h. Spectrum of RCCA ejection accidents
i. Spectrum of rod drop accident in a BWR.

Items (a), (b), (d), (e), and (f) above are considered as Condition II events, Item (g) is a Condition III event, and Item (h) is an ANS Condition IV event. Item (c) entails both Condition II and III events.

5. Increase in Reactor Coolant Inventory

These non-LOCA events are characterized by an increase in reactor coolant system water inventory. Several events have been postulated which could cause an increase in reactor coolant inventory or a change in boron concentration in the reactor coolant. The events typically analyzed are:

a. Inadvertent operation of Emergency Core Cooling System (ECCS) during power operation
b. Chemical and Volume Control System (CVCS) malfunction that increases reactor coolant inventory.

These events are considered Condition II transients.

6. Decrease in Reactor Coolant Inventory

These non-LOCA events are characterized by a decrease in reactor coolant system water inventory. Events that result in a decrease in reactor coolant inventory are described in the following:

a. Inadvertent opening of a pressurizer safety or relief valve
b. Failure of small lines carrying primary coolant outside containment
c. Steam generator tube rupture
d. Boiling Water Reactor piping failure outside containment.

Items (a) and (b) above are considered to be Condition II events and Item (c) is considered to be a Condition IV event. Item (d) can be a Condition II or III event depending on the size of the break.

7. Radioactive Release from a Subsystem or Component

A number of events have been postulated which could result in a radioactive release from a subsystem or component. These events are:

a. Postulated radioactive release due to liquid-containing tank failure
b. Design basis fuel handling accidents
c. Spent fuel cask drop accident.

Events (a) and (b) are considered as Condition III events and Event (c) is considered a Condition IV event.

### 5.2.2 LOCA and Beyond Design Basis Events

An important accident sequence to analyze is various LOCAs that can occur as a result of a spectrum of postulated piping breaks within the reactor coolant pressure boundary. Additionally, plant risk assessments have indicated that the risks of core damage and large early release also can occur due to events that are outside of the classical design basis accident sequences that are assessed as part of the licensing process.

As discussed previously, the Large Break LOCA (LB-LOCA) event represents one of the required accident analysis sequences required for NPP design and licensing bases. As a result, any effort at margin recovery will, at a minimum, require analysis of this accident. For application of RISA, LB-LOCA analyses will be performed using the LOTUS framework.

Compared to LB-LOCA events, Plant PRAs have indicated that the Small Break LOCA event also can challenge the ability to maintain adequate core cooling. These events typically have a larger impact on CDF and LERF than the LB-LOCA event due to (1) a much higher initiating event frequency, and (2) fewer alternative mechanisms for providing core cooling than for the LB-LOCA case. Small Break LOCA transients are characterized by a gradual top-down draining of the reactor coolant system, with low flow rates in the core relative to those occurring at steady-state or for Large Break LOCA transients. The hydraulic losses in the core due to frictional drag, form loss, and acceleration are small, and reasonable variations in flow resistance are expected to have a negligible effect on Small Break LOCA analysis results. For application of RISA, Small Break LOCA analyses also will be performed using the LOTUS framework.

In addition to the design basis events described in the SRP, BDBA events also are of importance for the assessment of potential benefits of advanced technologies. The most significant of these events has already been the subject of extensive analyses with the primary focus being on station blackout events (both short-term station blackout and long-term station blackout for both BWR and PWR NPPs).

## 5.3 Prioritization of Accident Analysis in this RD&D

The analysis approaches for all the transients in the Chapter 15 of an NPP FSAR will be modernized with RD&D activities described in this document. However, some transients are more limiting than others and they will be investigated first in our RD&D activities. The following is a list of accidents that will be investigated in the first 2 years followed by the analyses of the remaining transients required in Chapter 15 of the FSAR.

1. Large Break LOCA (LB-LOCA)
2. Medium Break LOCA (MB-LOCA)
3. Small Break LOCA (SB-LOCA)
4. Loss of off-site Power (LOOP)
5. Loss of main feedwater - PWR
6. Loss of main feedwater - BWR
7. Main steam line break (MSLB)
8. Loss of component cooling/service water (CC/SW)
9. Steam generator tube rupture (SGTR)
10. Turbine trip without bypass (PWR)
11. Turbine trip without bypass (BWR)
12. Transient Overpower Accidents – PWR rod ejection
13. Transient Overpower Accidents – BWR rod drop
14. Site induced accidents such as fire events
15. PWR locked rotor
16. BWR recirculation pump shaft seizure
17. Inadvertent RCS blowdown (PWR)
18. Inadvertent RCS blowdown (BWR)
19. BWR recirculation pumps trip
20. Fuel handling accident for extended burnup and increased enrichment.

# 6. INDUSTRY ENGAGEMENT

The RISA Pathway of the LWRS program will develop pilot demonstration projects in collaboration with university partners (such as Texas A&M University) and industry partners (such as EPRI, NPP utility owners/operators, reactor vendors, and engineering service providers) to demonstrate margin recovery through the implementation of advanced modeling and simulation tools and safety analysis methodologies.

Even though the R&D will be conducted jointly, each organization will have a specific focus within the broader scope of advancing the technology readiness level (TRL) [27], as illustrated in Figure 11. The margin recovery portion of the RISA Pathway of the LWRS program will focus on the risk-informed multi-physics BEPU technology development and demonstration (TRLs 5 and 6). RISA has the role of performing, high-impact research to demonstrate margin recovery through advanced modeling and simulation. Industry partners will focus on technology deployment and business development (TRLs 7, 8, and 9). It will be the responsibility of industry organizations (EPRI, operating utilities, reactor vendors, etc.) to evaluate the margins gained and to develop the business case for industry to adopt advanced technologies.

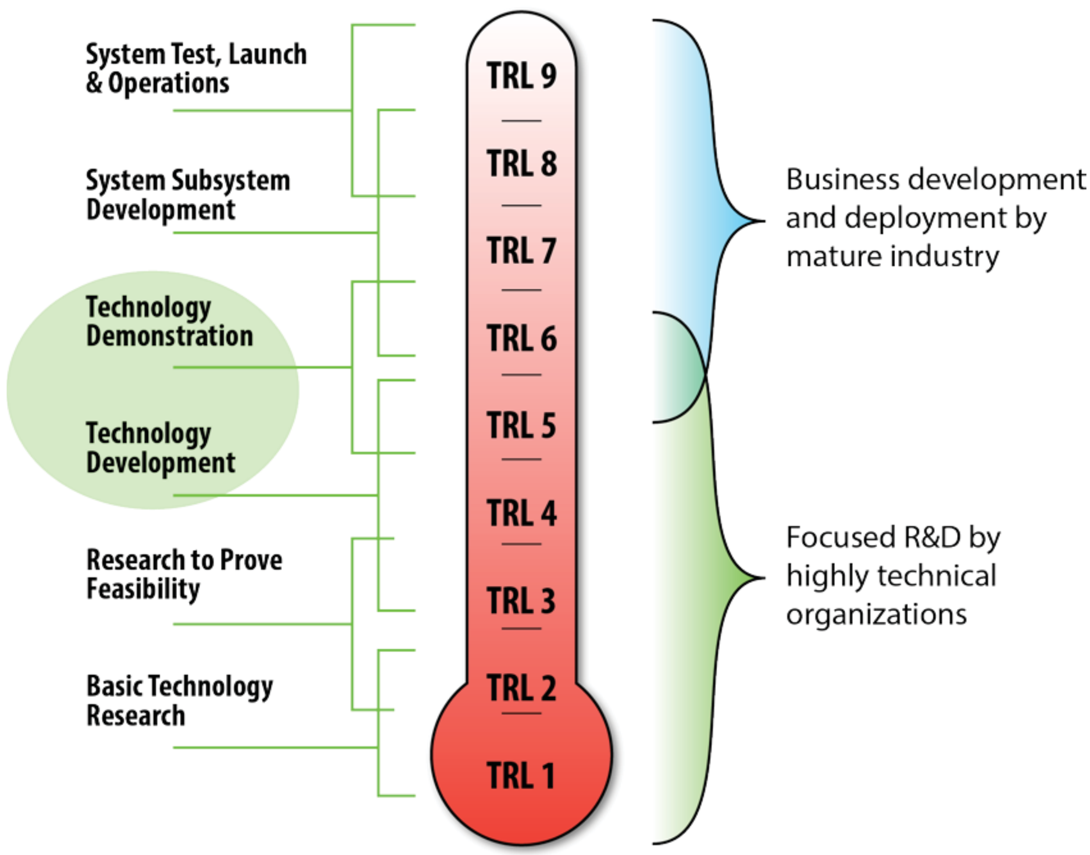


Figure 11. Illustration of technology readiness level [27].

A collaborative approach will be required to develop a well-structured and clearly defined path for development, demonstration, licensing, and deployment of margin recovery technologies with a shared emphasis on DOE and industry objectives. This structure will permit direct industry involvement during the development and initial demonstrations of the methods and tools, which should ensure their compliance with industry needs. The approach also will provide a mechanism for the developers to receive timely industry feedback through direct participation in first-of-a-kind applications of the methods and tools.

As presented in the recent report on RISA use cases (INL/EXT-18-51012 [28]), the RISA Pathway within the LWRS program has been developed to conduct collaborative research to apply risk-informed technologies to assist operating NPPs to reduce costs and support adaptation to the changing market conditions related to electric power generation. The RISA strategy emphasizes the identification and conduct of scalable pilot applications with partnering NPPs to address issues of current relevance to both the host utility and the industry as a whole.

The RISA Pathway is being performed within the framework of specific use cases that are intended to provide a pathway for rapid technology development, deployment, and dissemination throughout the operating U.S. nuclear power industry to address issues of economic, operational, or safety significance. Four specific use case categories have been incorporated into the RISA Pathway:

1. Enhanced resilient plant concept (including adoption of accident tolerant fuel technologies)
2. Cost reduction and risk categorization
3. Margin recovery and operating cost reduction
4. Market economics (wholesale pricing, energy policy, etc.).

Each of these Use Cases addresses one or more issues of critical importance to ensuring the safe and economic operation of the U.S. fleet of commercial nuclear plants. The following provides a breakdown of each collaborator's specific activities for applications within these use cases:

- Idaho National Laboratory (INL) will develop, validate, and demonstrate the methods and tools that will be used in the conduct of RISA applications to utility use case applications. Specifically, INL will integrate the high-fidelity modeling and simulation tools currently under development in DOE's modeling and simulation programs (i.e., Consortium for Advanced Simulation for Light Water Reactors (CASL) and Nuclear Energy Advanced Modeling and Simulation (NEAMS)) with other existing tools used by industry into a user-friendly multi-physics framework with built-in uncertainty quantification and sensitivity analysis capability. The multi-physics best estimate plus uncertainty framework will be integrated with the PRA tools to yield the integrated risk evaluation model. INL will work with industry partners to apply these tools to provide integrated analyses of the selected FSAR Chapter 15 events for initial applications of the RISA approach to utility identified use case applications.
- Industry partners will support the conduct of analyses that assess the benefits of advanced technologies to enhance NPP safety, operational, and economic margins. Initial use case applications that have been identified by potential host utilities have been described in report INL/EXT-18-51012 [28]. Those applications that are specifically related to the margin recovery use case are described in more detail in Section 7 of this report. As an independent R&D organization, EPRI may provide guidance and input on the technical requirements for advanced technologies to be commercially viable in operating NPPs. Within this perspective, EPRI may assist as an interface between NPP owners/operators and other key stakeholders, including U.S. DOE, fuel vendors, and engineering services providers. As part of this collaboration, EPRI may be a valuable contributor to the technical evaluation and cost/benefit assessment associated with the industry pilot demonstrations. Through its access to its members, EPRI can also sponsor relevant analyses at operational NPPs to assess and demonstrate the benefits of advanced technology deployment.
- University partners will provide a supporting role to various use case applications throughout the RISA Pathway. Within the margins recovery use case, a university partner will develop a high-fidelity, generic four-loop PWR RELAP5-3D plant model based on the host utility's operating plant. In this model all of the proprietary information will be replaced with representative, yet non-

proprietary, information to facilitate collaborations between the involved partners. For this particular use case application, the university partner and the host utility have a long history of collaboration with the LWRS program in the Industry Applications project for ECCS/LOCA to develop new methodologies to help the industry transition to the proposed new rulemaking in 10 CFR 50.46c. This collaboration will be carried out further, exploring margins in the entire FSAR Chapter 15 analysis.

- For each of the respective use case applications, the host NPP utility will provide in-kind contributions to the project in the form of plant data, geometry specifications, plant operating conditions, and simulation scenarios.
- Industry partners (utilities, reactor vendors, and engineering service providers) will contribute to the development of the RISA methods and tools with a focus on identifying characteristics and capabilities required for industry acceptance and adoption.

# 7. IUDUSTRY PILOT DEMONSTRATION PROJECTS

The RI-MP-BEPU framework will be applied to the industry issues through industry pilot demonstration projects. The pilot demonstration projects are developed with the industry partners with the aim at achieving operating cost reductions for the fleet of operating NPPs and providing a mechanism for deployment in advanced reactor designs. The focus of the industry pilot demonstration projects is to conduct comprehensive assessments of a few selected high-value advanced nuclear technologies to demonstrate margin recovery using advanced modeling and simulation tools with the intent of accelerating their development and use by the industry. The specific applications of the pilot demonstration projects are (1) fuel enrichment and burnup extension to enable 24-month fuel cycle for PWRs, and (2) digital instrumentation and controls (I&C) upgrades risk assessment. The adoption of these advanced technologies has the potential to either improve the fuel cycle economics or to reduce operating costs while simultaneously maintaining or enhancing plant safety.

## 7.1 Fuel Enrichment and Burnup Extension to Enable 24-Month PWR Cycles

Extending the fuel discharge burnup level can present significant economic benefits to the current fleet of operating LWRs. It allows for longer cycles and improved resource utilization. The major economic gain of longer cycles is due to the increased capacity factor resulting from decreased refueling times as a fraction of total operating time, as well as fewer assemblies to be discharged for a given amount of produced energy. Significant progress has been achieved in the past to increase fuel discharge burnup. This has given the utilities considerable reductions of fuel cycle costs. Further burnup increases would incentivize the utilities to achieve additional fuel cycle cost reductions. Previous studies conducted by EPRI found that fuel costs decrease with increasing discharge burnups, with the use of fuel enrichment less than the current limit of 5 w/o [29]. Additional studies also showed that enhancements of fuel enrichment greater than 5 w/o (up to 6 w/o) can result in additional decreases in fuel costs and further increases in discharge burnups for both BWRs and PWRs [30].

Increasing the fuel assembly discharge burnup is the most efficient means to achieve the fuel cycle cost reduction. Advanced fuel assembly designs offer sufficient margins to be used with higher enrichments in smaller fuel reload batch sizes and with increasingly heterogeneous core loading schemes. However, there are many technological challenges posed by the increased burnup such as the corrosion and hydrogen pickup of the clad, fuel thermal conductivity degradation, dimensional changes of the fuel assembly structure, the rim effect of the pellet, and the potential for increase of fission gas release. Increasing discharge burnup generally tends to increase steady state core peaking factors. This can also result in increased peaking factors under accident conditions such as LOCA, dropped or misaligned control rods, ejected control rods, or steam line break. Transient analysis should be performed to demonstrate acceptable consequences for these accident conditions. Additionally, increased discharge burnup results in increased loads and more demanding requirements for the fuel assemblies, which could raise the risk of fuel failures. These issues would be exacerbated considering the current industry trend of adopting flexible operation strategies to maximize the revenue of the NPPs.

In the United States the main licensing challenges for high burnup fuel are design basis accident condition analyses, especially for LOCA and RIA. (Note that for international applications, analyses related to beyond design basis accident (BDBA) conditions or Design Extension Conditions would provide an additional licensing challenge.) The design basis Large Break LOCA sequence can be divided into three phases: blowdown, refill and reflood. During the blowdown phase, ballooning and burst of the cladding occur since the rod internal pressure becomes much higher than the system pressure of the reactor pressure vessel and the strength of the fuel cladding decreases as the temperature increases. During the refill phase, the cladding is severely oxidized by steam and it becomes embrittled. During the reflood phase, the embrittled cladding may rupture by thermal shock caused by rapid cooling. Under RIA conditions, the fuel clad local conditions such as oxide layer thickness, oxide spalling and local hydrides have strong influence on fuel rod safety and the fuel rod local conditions are a strong function of the burnup level. Another obstacle is that there is no internationally accepted RIA failure limit at high burnup. Fuel rod burst, fuel pellet fragmentation, relocation, and dispersal during these transient conditions are the main obstacles for fuel discharge burnup to be extended above the current limit of 62 GWD/MTU. A major open question is how far experimental data for accident conditions can be extrapolated above the burnups covered by the currently available data. The current enrichment limit of 5 w/o U-235 represents a world-wide established limit for fabrication, transport and storage of nuclear fuel for LWRs. It is anticipated that considerable effort would be necessary to obtain regulatory approvals to extend either the current enrichment or burnup limits associated with nuclear fuel.

This pilot demonstration project will assist in establishing licensing criteria for fuel rods at burnup extensions beyond 62 GWD/MTU. The process involves the review of the licensing limits defined in Section 4.2 of the NRC Standard Review Plan (NUREG-0800) [11]. The objective of the evaluation process is to assess the need for modifications or additions to the existing licensing criteria, or to demonstrate the applicability of the present limits to rod average burnup levels beyond 62 GWD/MTU. (Note that for this limit the burnup value refers to the rod average burnup value.)

The licensing requirements of the fuel system specified in Section 4.2 of SRP are:

1. Fuel damage is not expected during Condition I and Condition II events. It is not possible, however, to preclude a very small number of rod failures. These are within the capability of the plant clean-up system and are consistent with plant design bases.
2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged although sufficient fuel damage might occur to preclude immediate resumption of operation. The fraction of fuel rods damaged must be limited to meet the dose guidelines of 10 CFR 100.
3. The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

Evaluation of burnup extension involves multiple disciplines to address the challenging licensing issues under transient conditions. The RI-MP-BEPU framework provides the approach needed to accelerate the licensing and deployment of this technology. Core design will be performed for a four-loop Westinghouse design PWR with increased enrichment up to 6 w/o to achieve a twenty-four month cycle and around 75 GWD/MTU discharge burnup. The VERA-CS code will be used to provide pin-resolved power distributions for the core design followed by detailed fuel performance calculations for individual fuel rods in the core using the FRAPCON and BISON fuel performance analysis codes. The risk-informed transient safety calculations will subsequently be performed to address the challenging issues of burst potential evaluation under LOCA conditions and fuel fragmentation, axial relocation, pulverization and dispersal issues during LOCA and RIA events. The analyses will apply both deterministic (VERA-CS, FRAPCON/FRAPTRAN, BISON, RELAP5-3D, Methods for Estimation of Leakages and Consequences of Releases (MELCOR)) and probabilistic (Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), RAVEN) methods using the LOTUS controller. Tightly coupled calculations between RELAP5-3D and BISON, and RELAP5-3D and FRAPTRAN during LOCA and RIA conditions are required to provide detailed assessment of the fuel rod burst potential, fuel fragmentation, relocation and dispersal. The tightly coupled calculation capability for RELAP5-3D/BISON and RELAP5-3D/FRAPTRAN will be developed as part of this demonstration pilot project. Rigorous Uncertainty Quantification and Sensitivity Analysis techniques will be integrated into the RI-MP-BEPU framework to alleviate concerns (to the extent practicable) on extrapolating experimental data.

The near term goals of this activity will be to: 1) conduct evaluations of non-burst potential under transient (e.g., LOCA) conditions for increased discharge burnup fuel (from the current limit of 62 GWD/MTU up to 75 GWD/MTU rod averaged burnup); and 2) collaborate with the ongoing Nuclear

Science User Facility project to provide comprehensive uncertainty quantification and uncertainty analysis to inform the experiments that will be conducted at the TREAT reactor for high burnup fuel. Increased fuel discharge burnup requires the use of higher enrichment fuel. Since the cost of enrichment is non-linear, at some point the increased enrichment costs may offset the benefits of improved fuel cycle efficiency and reduced number of fuel assemblies to be discharged. This project will investigate the combination of the optimal level of enrichment, discharge burnup and fuel cycle length to achieve optimal improvements in fuel cycle economics under a realistic utility environment. The long term goal of this activity will be to work with industry partners (including EPRI and industry lead utilities conducting evaluation and testing of fuel enrichment and burnup extensions) to address the challenging licensing issues of fuel fragmentation, relocation and dispersal and evaluate the economic improvements from the adoption of enhanced fuel technology.

### 7.1.1 RD&D Activities

- Core design analysis with VERA-CS for a four-loop PWR to achieve extended burnup, 3/31/2019.
- Expand the model fidelity, capability and quality assurance of the 4-loop generic PWR model based on host utility's input, 9/30/2019.
- Demonstrate coupled RELAP5-3D/COBRA-TF calculation capability to compute DNBR for non-LOCA transients, 9/30/2019.
- Perform risk-informed MP-BEPU analysis for SBLOCA, LBLOCA, RIA for the extended burnup core design to evaluate extended burnup fuel burst potential, 9/30/2019.
- Demonstrate coupled RELAP5-3D/FRAPTRAN and RELAP5-3D/BISON calculation capabilities to better simulate extended burnup fuel behavior under transient conditions, 9/30/2020.
- In collaboration with the Nuclear Science User Facility project, perform rigorous uncertainty quantification and sensitivity analysis to design identified experiments of high burnup fuel in the TREAT reactor, 9/30/2020.
- Perform the evaluation of extended burnup fuel for risk significant transients for a four-loop PWR, 9/30/2020.
- Perform the evaluation of extended burnup fuel for risk significant transients for a BWR, 9/30/2019.
- Evaluate the impact of flexible operating strategies on high burnup fuel, 9/30/2021.

## 7.2 Digital Instrumentation and Controls Risk Assessment

Digital instrumentation and controls (I&C) technology provides the potential solution to address the reliability and obsolescence issues of the legacy analog I&C systems of the operating LWR fleet. Many analog I&C architectures consisted of discrete and separate analog components in each echelon of defense from signal detection, through processing, and system logic for actuation and control. Digital technology offers the inherent capabilities of integration, interconnectivity, and standardization that can combine formerly discrete systems (e.g., RPS and ESFAS) into a single digital I&C system. Digital systems can significantly reduce nuclear power plant operations and maintenance costs, and improve plant performance and availability, while maintaining or even improving plant safety. However, if not correctly designed, these same inherent capabilities for integration, interconnectivity and standardization can create single sources of failure that can adversely affect multiple control functions; therefore, these are potential sources of common cause failure (CCF) [31]. A CCF is the malfunction of two or more plant components or functions due to a single failure source. That single failure source may be a random failure of a single hardware resource that is shared among multiple control functions, or a defect in a standard design that is shared among multiple control functions. The issue of digital CCF has been difficult to address and has been the reason some nuclear plant operators have deferred upgrades of these critical plant systems, opting rather to maintain them with costly engineering and maintenance efforts.

Advances that support the licensing case to enable the transition from analog I&C technologies to digital technologies in the U.S. nuclear industry are greatly needed. A commercial nuclear utility is collaborating with the RISA Pathway of the LWRS Program to develop and demonstrate a process for evaluating the reliability and risk associated with a safety-related digital I&C system to support its design, qualification, and use. This pilot demonstration project will define a technical basis and approach that can be used by utility design, operational, and licensing staff to address the unique issues associated with replacing an analog, nuclear-qualified, safety-related I&C system with a modern digital system.

The goal of this research is to assure the long-term safety and reliability of vital engineered systems, reduce uncertainty in licensing costs and time, and support integration of digital systems in the plant and more efficient upgrades of technology for the entire lifecycle of NPPs. This pilot demonstration project will define a risk-informed analysis process—steps, methods, and tools—to assess the safety risk and operational cost risk for a digital replacement to the reactor protection system, which consists of reactor trip system and ESFAS systems, at a three-loop Westinghouse reactor plant from the conceptual design of the system through what the operating utility deems necessary and adequate for regulatory approval and to provide information to support that utility's decision-making process with respect to procuring the system. The pilot demonstration application will propose solutions where process gaps or unacceptable uncertainties are identified. A well-structured and risk-informed process will be developed and applied to evaluate a specific design considered by the host utility for replacement of RPS with the desired integration with other safety- and non-safety systems. The process must be well-structured to ensure that it is thorough and assures that the necessary safety and reliability requirements have been considered and met for both safety and cost due-diligence. It must be risk-informed; that is, the likelihood and consequences of any failures or spurious operations must be acceptable compared with both likelihood and consequences compared to the current analog system that has operated acceptably.

Recognizing the fact that thoroughly analyzing the RPS is a large and challenging work scope, the process begins with evaluating only a channel logic level conceptual design with a detailed design and functional specification of a proposed I&C system to validate the chosen evaluation process. This initial evaluation will be conducted as a table-top exercise to provide critical assessments of the feasibility and value of the process. It will propose solutions where process gaps or unacceptable uncertainties are identified. Selective "deep dive" analyses will be considered and performed where appropriate. The evaluation will then go through various stages of risk and reliability evaluations sequentially, including review and analysis of existing data, susceptibility analysis, reliability assessment, and coping analysis. The long-term goal of this pilot application will be to specify each of these analyses in sufficient detail to support the evaluation of an integrated RPS and/or ESFAS replacement using the risk-informed, graded approach to safety significance.

Generally, digital systems cannot be proven to be error-free; therefore, these systems are considered susceptible to CCF because identical copies of the software-based logic and architecture are present in redundant divisions of safety-related systems. Also, some errors that are labeled as "software errors" actually result from errors in the higher level requirements specifications used to direct the system development but that fail in some way to represent the actual process. Such errors place further emphasis on the need for diversity to avoid or mitigate CCFs. The current NRC position to address uncertainty of software design related to common cause failures of digital safety-related RPS and ESFAS systems calls for implementation of a diverse analog system. This requirement has proven to be a considerable operational and cost burden for the industry and has served as one of the major impediments to adoption of digital I&C technology for safety related systems in the US. This pilot demonstration project aims at addressing this important concern during the development of the well-structured, risk-informed, graded assessment approach as well as during the table-top implementation of the process.

CCFs have the potential to create unanalyzed malfunctions that may not be bounded by previous plant analyses; thereby, creating unanalyzed plant conditions that may challenge plant safety. Plant safety is assured for events that have been considered in the plant's transient and accident analyses, which is typically Chapter 15 of most Updated Safety Analysis Reports (UFSARs). While the plant may be safe for other events, there is no certainty of safety without additional analysis. Potential CCFs in new digital safety or non-safety systems can create unanalyzed plant conditions because the plant transient and accident analyses for currently operating U.S. plants were based on the failures considered applicable to the analog I&C technology when those analyses were performed. CCFs have been considered in the plant's probabilistic risk assessment models. The PRA models will be expanded to include the new digital CCFs not encompassed by the current PRA. The RI-MP-BEPU framework is being developed to provide the capability to assess CDF and LERF as well as to demonstrate that the plants are protected based on specific regulatory acceptance criteria against the effects of AOOs and postulated accidents with a concurrent CCF in the digital protection system.

### 7.2.1 RD&D Activities

- Develop a strategy to perform risk assessment for digital instrumentation and controls upgrades, July 31, 2019.
- Develop a static reliability model based on a conceptual design of digital RPS, September 30, 2019.
- Collaborating with the lead industry partner and Plant Modernization Pathway of the LWRS Program, develop static reliability model for the digital RPS to be procured by the host utility, September 30, 2020.
- Develop RELAP5-3D plant system models with detailed control systems modeling of the digital RPS for the host utility's plant, September 30, 2020.
- Collaborating with the plant modernization Pathway and the lead industry utility partner, conduct comprehensive assessment to identify all the potential CCFs caused by digital I&C systems, June 30, 2021.
- Perform RI-MP-BEPU analyses for accident scenarios with concurrent CCFs, September 30, 2021.
- Demonstration of dynamic reliability studies for digital I&C upgrades risk assessment, September 21, 2021.

# 8. DESCRIPTION OF COMPUTER CODES

Both existing and advanced analysis tools will be utilized in the application of RISA to the Use Case applications identified previously. Due to the high costs associated with the qualification and regulatory acceptance of analytical tools, it is anticipated that the licensing of advanced nuclear technologies will rely predominantly on the current suite of tools used to assess AOO/DBA/BDBA events. However, because of the large uncertainties that currently exist for advanced nuclear technologies, the existing tools will need to be informed and enhanced to support the licensing and deployment of these technologies. The codes that have been identified for use in the execution of this research plan are detailed below.

## 8.1 Core Design and Analysis

### 8.1.1 VERA-CS

VERA-CS [32] includes coupled neutronics, thermal-hydraulics, and fuel temperature components with an isotopic depletion capability. The neutronics capability employed is based on MPACT [33], a three-dimensional (3-D) whole core transport code. The thermal-hydraulics and fuel temperature models are provided by the COBRA-TF (CTF) subchannel code [34]. The isotopic depletion is performed using the ORIGEN [35] code system.

#### *8.1.1.1 MPACT*

As stated in the MPACT Theory Manual [33], MPACT is a 3-D whole core transport code that is capable of generating subpin-level power distributions. This is accomplished by solving an integral form of the Boltzmann transport equation for the heterogeneous reactor problem in which the detailed geometrical configuration of fuel components, such as the pellet and cladding, is explicitly retained. The cross-section data needed for the neutron transport calculation are obtained directly from a multigroup cross section library, which has traditionally been used by lattice physics codes to generate few-group homogenized cross sections for nodal core simulators. Hence, MPACT involves neither *a priori* homogenization nor group condensation to achieve the full core spatial solution.

The integral transport solution is obtained using the method of characteristics (MOC), and employs discrete ray tracing within each fuel pin. MPACT provides a 3-D MOC solution; however, for practical reactor applications, the direct application of MOC to 3-D core configuration requires considerable amounts of memory and computing time associated with the large number of rays. Therefore, an alternative approximate 3-D solution method is implemented in MPACT for practical full core calculations, based on a “2-D/1-D” method in which MOC solutions are performed for each radial plane and the axial solution is performed using a lower-order one-dimensional (1-D) diffusion or SP3 approximation. The core is divided into several planes, each on the order of 5 to 10 cm thick, and the planar solution is obtained for each plane using 2-D MOC. The axial solution is obtained for each pin, and the planar and axial problems are coupled through transverse leakage. The use of a lower order 1-D solution, which is most often the nodal expansion method with the diffusion or P3 approximation, is justified by the fact that most heterogeneity in the core occurs in the radial direction rather than the axial direction. Alternatively, a full 3-D MOC solution can be performed if necessary, if the computational resources are available.

The Coarse Mesh Finite Difference (CMFD) acceleration method, which was originally introduced to improve the efficiency of the nodal diffusion method, is used in MPACT for the acceleration of the whole core transport calculation. The basic mesh in the CMFD formulation is a pin cell, which is much coarser than the flat source regions defined for MOC calculations. (Typically there are approximately 50 flat source regions in each fuel pin.) The concept of dynamic homogenization of group constants for the pin cell is the basis for the effectiveness of the CMFD formulation to accelerate whole core transport calculations. The intra-cell flux distribution determined from the MOC calculation is used to generate the homogenized cell constants, while the MOC cell surface-averaged currents are used to determine the radial nodal coupling coefficients. The equivalence formalism makes it possible to generate the same transport solution with CMFD as the one obtained with the MOC calculation. In addition to the acceleration aspect of the CMFD formulation, it provides the framework for the 3-D calculation in which the global 3-D neutron balance is performed through the use of the MOC generated cell constants, radial coupling coefficients, and the nodal expansion method-generated axial coupling coefficients.

In the simulation of depletion, MPACT can call the ORIGEN code, which is included in the SCALE [36] package. However, MPACT has its own internal depletion model, which is based closely on ORIGEN, with a reduced isotope library and number of isotopes. The internal depletion model will be used for in the Use Case applications where MPACT is applied.

#### *8.1.1.2 COBRA-TF*

Coolant Boiling in Rod Arrays – Two Fluid (COBRA-TF) [34] is a transient subchannel code based on the two-fluid formulation, in which the conservation equations of mass, energy, and momentum are solved for three fields, namely the vapor phase, continuous liquid, and entrained liquid droplets. The conservation equations for the three fields and for heat transfer from and within fuel rods are solved using a semi-implicit finite-difference numerical scheme, with closure equations and physical models to account for interfacial mass transfer, interfacial drag forces, interfacial and wall heat transfer, inter-channel mixing, entrainment, and thermodynamic properties. The code is applicable to flow and heat transfer regimes beyond critical heat flux (CHF), and is capable of calculating reverse flow, counter flow and crossflow with either three-dimensional (3D) Cartesian or subchannel coordinates for thermal-hydraulic or heat transfer solutions. It allows for full 3D LWR core modeling and has been used extensively for LWR LOCA and non-LOCA analyses including the departure from nucleate boiling (DNB) analysis.

The COBRA-TF (CTF) code was originally developed by the Pacific Northwest Laboratory and has been updated over the last few decades by several organizations. CTF is being further improved as part of the VERA multi-physics software package as part of the CASL DOE Modeling and Simulation Hub. These enhancements include:

- Improvements to user-friendliness of the code through creation of a preprocessor utility
- Code maintenance, including source version tracking, bug fixes, and transition to modern Fortran
- Incorporation of an automated build and testing system using CMake/CTest/Tribits [37]
- Addition of new code outputs for better data accessibility and simulation visualization
- Extensive source code optimizations and full parallelization of the code, enabling fast simulation of full core subchannel models
- Improvements to closure models, including Thom boiling heat transfer model, Yao-Hochreiter-Leech grid-heat-transfer enhancement model, and Tong factor for the W-3 critical heat flux correlation
- Addition of consistent set of steam tables from the IAPWS-97 standard [38]
- Application of an extensive automated code regression test suite to prevent code regression during development activities
- Code validation study with experimental data.

In a steady-state or transient CTF simulation subchannel data, such as flow rate, temperature, enthalpy, pressure, and fuel rod temperatures are projected onto a user-specified or pre-processor generated mesh and written to files in a format suitable for visualization. The freely available Paraview [39] software is used for visualizing the 3-D data that results from large, full core models and calculations.

## 8.2 Fuel Performance

The following codes are currently used throughout the US commercial nuclear industry for fuel performance analysis.

### 8.2.1 FRAPCON/FRAPTRAN

FRAPCON/FRATRAN is a suite of codes developed by Pacific Northwest National Laboratory for the US NRC for the purposes of performing fuel performance analyses under steady state (FRAPCON) and transient (FRAPTRAN) conditions. FRAPCON [40] is used to analyze the steady-state response of light-water reactor fuel rods. The code calculates the temperature, pressure, and deformation of a fuel rod as functions of time-dependent fuel rod power and coolant boundary conditions. The phenomena modeled by FRAPCON include: (1) heat conduction through the fuel and cladding to the coolant; (2) cladding elastic and plastic deformation; (3) fuel-cladding mechanical interaction; (4) fission gas release from the fuel and rod internal pressure; and (5) cladding oxidation. The code contains necessary material properties, water properties, and heat-transfer correlations.

The Fuel Rod Analysis Program Transient (FRAPTRAN [41]) is a Fortran computer code that calculates the transient performance of light-water reactor fuel rods during reactor transients and hypothetical accidents such as loss-of-coolant accidents, anticipated transients without scram, and reactivity-initiated accidents. FRAPTRAN calculates the temperature and deformation history of a fuel rod as a function of time-dependent fuel rod power and coolant boundary conditions. Although FRAPTRAN can be used in "standalone" mode, it is often used in conjunction with, or with input from, other codes. The phenomena modeled by FRAPTRAN include (1) heat conduction, (2) heat transfer from cladding to coolant, (3) elastic-plastic fuel and cladding deformation, (4) cladding oxidation, (5) fission gas release, and (6) fuel rod gas pressure.

### 8.2.2 FALCON

The Fuel Analysis and Licensing Code—New (FALCON) is a state-of-the-art LWR fuel performance analysis and modeling code [42] that was developed by EPRI and has been validated to high fuel burnup conditions. It is based on a finite-element numerical structure and is capable of analyzing both steady-state and transient fuel behaviors. FALCON employs a robust numerical scheme with fully coupled thermal and mechanical iterations to perform steady-state and transient analyses. The code incorporates pellet and cladding material and behavior models required for steady-state and transient fuel performance analysis. FALCON has been benchmarked and validation over a range of representative cases of test reactor experiments and commercial reactor fuel rod data. As an EPRI developed product, FALCON is used by a number of operating utilities to analyze fuel performance at their operating NPPs.

### 8.2.3 BISON

BISON [43] is a finite element-based nuclear fuel performance code applicable to a variety of fuel forms including LWR fuel rods, tristructural isotopic particle fuel, and metallic rod and plate fuel. This advanced fuel performance code is being developed at INL and offers distinctive advantages over FRAPCON/FRAPTRAN such as 3-D simulation capability, etc. BISON solves the fully coupled equations of thermomechanics and species diffusion, for either 1-D spherical, 2-D axisymmetric or 3-D 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 also are 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 has been coupled to the mesoscale fuel performance code, MARMOT, demonstrating its fully coupled multiscale fuel performance capability. BISON is based on the Multi-Physics Object-Oriented Simulation Environment (MOOSE) framework; therefore, BISON can efficiently solve problems using standard workstations or very large high-performance computers. BISON is currently being validated against a wide variety of integral LWR fuel rod experiments.

## 8.3 Components Aging & Degradation

### 8.3.1 Grizzly

Grizzly [44] is being developed to simulate the progression of aging mechanisms in LWR SSCs, and to assess their ability to safely perform their intended engineering functions after being subjected to aging. Grizzly is planned to ultimately have capabilities for modeling a variety of structures, but current development is focused on reactor pressure vessels (RPVs) and concrete structures because of the essential functions and extreme difficulty of mitigating degradation or replacement of those components. For RPVs, Grizzly has a modern, flexible architecture for multidimensional engineering fracture mechanics analysis, which allows it to compute the probability of fracture in the presence of a population of pre-existing flaws that can serve as fracture initiation sites under a given transient event. It also has a set of models being developed to predict microstructure evolution under irradiation, which will be used to provide improved predictive models of embrittlement that can be applied for long-term operation scenarios. For concrete structures, Grizzly has coupled physics models to predict expansive mechanisms, including alkali-silica reaction and radiation-induced volumetric expansion, and their effects on the mechanical response of the structure, including fracture and damage.

## 8.4 Systems Analysis Codes

The following codes (and specific versions thereof adapted for use by industry and NRC) represent the current suite of tools to conduct analyses of AOO / DBA / BDBA events at commercial nuclear power plants (NPPs) operating in the United States. Reference [45] provides additional summary descriptions of these codes, including their capabilities, computational structure, available documentation, range of applicability, limitations, and relevant precautions.

The following codes (or specific modifications of them developed by the fuel vendors) have widespread use throughout the industry for the assessment of AOO and DBA events.

### 8.4.1 RELAP5-3D

The RELAP5-3D [46] code has been developed for best-estimate transient simulation of LWR coolant systems during postulated accidents. Specific applications of the code have included simulations of transients in LWR systems such as loss of coolant, anticipated transients without scram (ATWS), and operational transients, such as loss of feedwater flow, loss of offsite power, station blackout, and turbine trip. RELAP5-3D, the latest in the series of RELAP5 codes, is a highly generic systems code that, in addition to calculating the behavior of the reactor coolant system during a transient, can be used to simulate a wide variety of hydraulic and thermal transients in both nuclear and nonnuclear systems involving mixtures of vapor, liquid, noncondensable gases, and nonvolatile solutes.

RELAP5-3D is suitable for the analysis of all transients and postulated accidents in LWR systems, including both large- and small-break LOCAs, as well as the full range of operational and postulated transient applications. Additional capabilities include space reactor simulations, gas-cooled reactor applications, fast breeder reactor modeling, and cardiovascular blood flow simulations.

The RELAP5-3D code is based on a nonhomogeneous and nonequilibrium model for the two-phase system that is solved by a fast, partially implicit numerical scheme to permit economical calculation of system transients. The objective of the RELAP5-3D development effort from the outset was to produce a code that included important first-order effects necessary for accurate prediction of system transients but that was sufficiently simple and cost effective so that the conduct of parametric or sensitivity studies would be possible.

The code includes many generic component models from which general systems models can be developed and the progress of various postulated events can be simulated. The component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor kinetics, electric heaters, jet pumps, turbines, compressors, separators, annuli, pressurizers, feedwater heaters, ECC mixers, accumulators, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and noncondensable gas transport.

The system mathematical models are coupled into an efficient code structure. The code includes extensive input checking capability to help the user discover input errors and modeling and input inconsistencies. Also included are free-format input, restart, renodalization, and variable output edit features. These user conveniences were developed in recognition that the major cost associated with the use of a system transient code generally is in the engineering labor and time involved in accumulating system data and developing system models, while the computational cost associated with generation of the final result is usually small.

### 8.4.2 TRACE

The TRAC/RELAP Advanced Computational Engine (TRACE) [47] is a modernized best-estimate thermal-hydraulics code designed to consolidate the capabilities of the NRC's three legacy safety analysis codes: TRAC-B (BWR), TRAC-P (PWR), and RELAP. It is able to analyze a full spectrum of transients and accidents including large and small break LOCAs in both BWRs and PWRs. The capability also exists to model thermal hydraulic phenomena in both one and three dimensions. TRACE is currently the NRC's primary thermal-hydraulics analysis tool. A comprehensive validation matrix, including separate and integral effect tests, has been developed for the overall code assessment and validation.

As part of the international CAMP-Program sponsored by the USNRC, TRACE has been coupled with the Purdue Advanced Reactor Core Simulator (PARCS) to analyze the interactions of the plant dynamic thermal-hydraulic performance and the neutron kinetics for the reactor core.

### 8.4.3 MAAP

The Modular Accident Analysis Program (MAAP) [48] is an integral systems analysis code developed by the Electric Power Research Institute (EPRI) to simulate the response of light water reactors (LWRs) during severe accidents. As an EPRI developed code it is only available to EPRI members; however because all US operated NPPs (as well as a large number of international NPPs) are EPRI members, they are able to utilize the MAAP code for the conduct of severe accident analyses. Given a set of initiating events and operator actions, MAAP predicts the plant's response as the accident progresses. The code is used for the following:

- Predicting the timing of key events (e.g., core uncovery, core damage, core relocation to the lower plenum, and vessel failure)
- Evaluating the influence of mitigation systems and operator actions
- Predicting the magnitude and timing of fission product releases
- Evaluating uncertainties and sensitivities associated with severe accident phenomena.

MAAP results are used to determine success criteria and accident timing for probabilistic risk assessments (PRAs) to obtain estimates of CDF and LERF. MAAP is an integral systems analysis code that treats the full spectrum of important phenomena that could occur during an LWR accident.

### 8.4.4 RELAP-7

The RELAP-7 [49] (Reactor Excursion and Leak Analysis Program) code is the next generation nuclear reactor system safety analysis code being developed at Idaho National Laboratory (INL). The code is based on INL's modern scientific software development framework, MOOSE (Multi-Physics Object Oriented Simulation Environment), and uses modern numerical methods, allowing for implicit time integration, second-order schemes in both time and space, and strongly coupled multi-physics. The overall design goal of RELAP-7 is to take advantage of the previous thirty years of advancements in computer architecture, software design, numerical integration methods, and physical models with the ultimate goal of developing a reactor systems analysis capability that retains and improves upon RELAP5-3D's capabilities and extends the analysis capability for all reactor system simulation scenarios.

The RELAP-7 code will become the next generation tool in the RELAP reactor safety/systems analysis application series. The key to the success of RELAP-7 is the simultaneous advancement of physical models, numerical methods, and software design while maintaining a solid user perspective.

## 8.5 Containment Response

### 8.5.1 MELCOR

The Methods for Estimation of Leakages and Consequences of Releases (MELCOR) [50] is a computational code developed by the Sandia National Laboratory (SNL) for the US NRC, US DOE, and the International Cooperative Severe Accident Research Program (CSARP). The MELCOR code is primarily used by the NRC, US national laboratories, and university researchers for the conduct of severe accident analyses. Similar to MAAP, the code simulates the response of LWRs during severe accidents and is also used to determine success criteria and accident timing for NPP PRAs to obtain estimates of CDF and LERF. Given a set of initiating events and operator actions, MELCOR predicts the plant's response as the accident progresses. The code is used for the following:

- Prediction of the timing of key events (e.g., core uncovery, core damage, core relocation to the lower plenum, vessel failure)
- Evaluation of the influence of mitigation systems and operator actions
- Prediction of the magnitude and timing of fission product releases
- Evaluation of uncertainties and sensitivities associated with severe accident phenomena.

Similar to MAAP, MELCOR results are used to determine success criteria and accident timing for NPP PRAs to obtain estimates of CDF and LERF.

### 8.5.2 GOTHIC

GOTHIC [51] is a comprehensive software package for efficient analysis of thermal hydraulic transients involving water, steam and noncondensing gases. It is a versatile, general purpose accurate thermal-hydraulics software package for modeling a wide range of systems and events. GOTHIC solves the conservation equations for mass, momentum and energy for multicomponent, multi-phase compressible flow in lumped parameter and/or multi-dimensional (1-, 2-, or full 3-D) geometries. The ability to combine these different nodalization options in a single model allows GOTHIC to provide computationally efficient solutions for multiscale applications.

The GOTHIC code has become an industry standard for nuclear containment and general purpose thermal-hydraulic analyses. It is used extensively in the nuclear utility industry for safety-related applications

## 8.6 Radioactive Material Release

### 8.6.1 MACCS

The MELCOR Accident Consequence Code System (MACCS) [52] is the NRC code used to perform probabilistic offsite consequence assessments for hypothetical atmospheric releases of radionuclides. The purpose of this code is to simulate the impact of severe accidents at NPPs on the surrounding environment. The code models atmospheric transport and dispersion, emergency response and long-term protective actions, exposure pathways, early and long-term health effects, land contamination, and economic costs. The MACCS code can be used for a variety of applications including: (1) probabilistic risk assessment of NPPs and other nuclear facilities, (2) sensitivity studies to gain a better understanding of the parameters important to PRA, and (3) cost-benefit analysis. MACCS was designed as a tool for level-three Probabilistic Safety Assessment analysis and is used by U.S. NPP license renewal applicants to support the plant specific evaluation of severe accident mitigation alternatives (SAMA) as part of an applicant's environmental report for license renewal. MACCS is also used in severe accident mitigation design alternatives and severe accident consequence analyses for environmental impact statements for new reactor applications. The NRC uses MACCS in its cost-benefit assessments supporting regulatory analyses that evaluate potential new regulatory requirements for NPPs.

## 8.7 Risk Assessment

The following codes represent the current suite of mature as well as advanced tools that are still being developed to perform PRAs of commercial NPPs operating in the United States.

### 8.7.1 SAPHIRE

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) [53] is a software application developed for performing a complete PRA using a personal computer running the Microsoft Windows operating system. It was developed by INL for the U.S. NRC.

SAPHIRE enables users to supply basic event data, create and solve fault and event trees, perform uncertainty analyses, and generate reports. In that way, analysts can perform PRAs for any complex system, facility, or process. For NPP PRAs, SAPHIRE can be used to model a plant's response toinitiating events, quantify core damage frequencies, and identify important contributors to core damage (Level 1 PRA). The program also can be used to evaluate containment failure and release models for severe accident conditions given that core damage has occurred (Level 2 PRA). In so doing, the analyst can build the PRA model assuming that the reactor is initially at full power, low power, or shutdown. In addition, SAPHIRE can be used to analyze both internal and external events and, in a limited manner, to quantify the frequency of release consequences (Level 3 PRA).

### 8.7.2 CAFTA

The Computer Aided Fault Tree Analysis System (CAFTA) [54] is a computer program developed by EPRI to create, edit, and quantify reliability models, utilizing fault trees and event trees. CAFTA is used to build PRA models to assess Level 1 (core damage) and Level 2 (large early release) events. Given a set of initiating events, basic events, and operator actions, CAFTA quantifies the top gate of the fault tree. CAFTA is used to perform the following analyses:

- Develop, manage and evaluate fault and event trees
- Generate and analyze cutsets
- Evaluate the influence of modeled events
- Perform risk ranking evaluations
- Conduct sensitivity analyses.

CAFTA interfaces with multiple programs within the EPRI Risk and Reliability Workstation Suite of risk assessment tools to permit rapid and comprehensive risk assessments. Since CAFTA was developed by EPRI, it has been used by operating utilities in their conduct of plant risk assessments. The code has been developed and is maintained under a quality assurance program, which is in compliance with U.S. 10 CFR 50, Appendix B, and ISO 9001 quality assurance requirements.

### 8.7.3 RAVEN

RAVEN [55] is a software framework that is designed to perform parametric and stochastic analyses based on the response of complex systems codes. It is capable of communicating directly with the system codes described above that currently are used to perform plant safety analyses. The provided Application Programming Interfaces allow RAVEN to interact with any code as long as all the parameters that need to be perturbed are accessible by input files or via python [56] interfaces. RAVEN is capable of investigating system response and exploring input spaces using various sampling schemes such as Monte Carlo, grid, or Latin hypercube. However, RAVEN's strength lies in its system feature discovery capabilities such as: constructing limit surfaces, separating regions of the input space leading to system failure, and using dynamic supervised learning techniques.

### 8.7.4 EMRALD

EMRALD [57] is a dynamic PRA tool being developed at INL based on three phase discrete event simulation. Traditional PRA modeling techniques are effective for many scenarios, but it is hard to capture time dependencies and any dynamic interactions using conventional techniques. EMRALD modeling methods are designed around traditional methods yet enable an analyst to probabilistically model sequential procedures and see the progression of events through time that caused the outcome. Compiling the simulation results can show probabilities or patterns of time correlated failures.

An open communication protocol using the very common messaging platform XMPP [58], allows for easy coupling with other engineering tools. This coupling allows for direct interaction between the PRA model and physics-based simulations, so that simulated events can drive the PRA model and sampled PRA parameters can affect the simulation environment. The capabilities included in EMRALD permit PRA models to more easily and realistically account for the dynamic conditions associated with the progression of plant transient and accident sequences including accounting for the occurrence of modeled operator actions taken to mitigate the event.

## 8.8 Integration Tools

### 8.8.1 LOTUS

LOTUS [7] is a multi-physics best estimate plus uncertainty (MP-BEPU) analysis framework being developed at INL. It established the automation interfaces among the various disciplines depicted in Figure 7 of Section 4.1 such that uncertainties can be propagated consistently in multi-physics simulations. These disciplines include: 1) Core Design Automation which focuses on automating the cross section generation, core design, and power maneuvering process, 2) Fuel Performance which focuses on automating the interface between core design and fuel performance calculations, and the interface between fuel performance and systems analysis, 3) Components Aging and Degradation which focuses on automating the interface between core design and systems analysis with component aging and degradation, 4) System Analysis which focuses on automating the process required to setup large numbers of system analysis code runs needed to facilitate RISA applications on LOCA and other accident scenarios, 5) Containment Response which focuses on automating the interface between systems analysis and containment response, 6) Radioactive Material Release which focuses automating the interface between systems analysis, containment response and radioactive material release, 7) Uncertainty Quantification and Risk Assessment which focuses on uncertainty quantification and sensitivity analysis in multi-physics simulations and on establishing the interfaces to enable combined deterministic and probabilistic analysis, and 8) Core Design and Plant Systems Optimization which focuses on developing a core design and plant modifications optimization tool that can perform in-core and out-of-core design optimization.

LOTUS integrates existing computer codes as well as advanced computer codes that are being developed under various DOE programs to provide feedback and guide development of advanced tools. Regardless of the specific codes used to model the physics, the methodology discussed here is a paradigm shift in managing the uncertainties and assessing risks.

Conventional methods are strongly “code-oriented.” The analyst has to be familiar with the details of the codes utilized, in particular with respect to their input and output structures. This represents a significant barrier for widespread use. It becomes apparent how difficult it is to make changes and accelerate progress under such a paradigm, especially in a heavily regulated environment where even a single line change in a code carries a heavy cost of bookkeeping and regulatory review.

LOTUS’s vision is to move toward to a “plug-and-play” approach where the codes are simply modules “under the hood” that provide the input-output relationships for a specific discipline. The focus shifts to managing the data stream at a system level. LOTUS is essentially a workflow engine with the capability to drive physics simulations, model complex systems, and provide risk assessments. A plug-and-play approach will enable plant owners and vendors to consider and further customize the LOTUS framework for utilizing their established codes and methods. Therefore, it could potentially become the engine for license-grade methodologies. In other words, it is possible that LOTUS technology could be advanced in the future to a level of fidelity and maturity such that it could be used for licensing or regulatory applications.

# 9. PROJECT SCHEDULE

This project plan is to be conducted in collaboration with work being performed as part of broader industry efforts to develop, mature, license, and deploy advanced nuclear technologies in the industry to increase plant operational flexibility and reduce operational costs. In particular, the project activities and schedule are developed to be performed collaboratively with industry R&D efforts being led by EPRI with specific attention to conducting the identified R&D activities in a manner that efficiently and cost-effectively utilizes resources. The schedule also accounts for the fact that different advanced nuclear technologies are being developed and studied that have different timeframes for licensing and deployment.

The project schedule is shown in Table 3. It should be noted that this schedule reflects current industry objectives and priorities for the licensing and deployment of the various advanced technologies. This schedule is anticipated to evolve as additional information is obtained and interactions between industry, DOE, and NRC occur.

Table 3. Timeline for margin recovery RD&D activities.

| | **FY 2019** | | | | **FY 2020** | | | | **FY 2021** | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Q1 | Q2 | Q3 | Q4 | Q1 | Q2 | Q3 | Q4 | Q1 | Q2 | Q3 | Q4 |
| RI-MP-BEPU Framework Development & Optimization | | | | | | | | | | | | |
| **Burnup Extension** | | | | | | | | | | | | |
| Core Design with Extended Burnup | | | | | | | | | | | | |
| Expand the Generic PWR RELAP5 Model to Analyze More Transient Scenarios | | | | | | | | | | | | |
| RELAP5-3D/COBRA-TF Coupling for DNBR Calculations | | | | | | | | | | | | |
| LB-LOCA | | | | | | | | | | | | |
| SB-LOCA/MB-LOCA | | | | | | | | | | | | |
| RELAP5-3D/BISON & RELAP5-3D/FRAPTRAN Coupling | | | | | | | | | | | | |
| Perform Risk-Informed Design of Experiments Analyses to Support the Experiments of Extended Burnup Fuel in TREAT | | | | | | | | | | | | |
| Loss of Off-Site Power (LOOP) | | | | | | | | | | | | |
| Loss of Feedwater (PWR) | | | | | | | | | | | | |
| Main Steam Line Break (MSLB) | | | | | | | | | | | | |
| Loss of Component Cooling/Service Water (CC/SW) | | | | | | | | | | | | |
| Steam Generator Tube Rupture (SGTR) | | | | | | | | | | | | |
| PWR Rod Ejection | | | | | | | | | | | | |

| | FY 2019 | | | | FY 2020 | | | | FY 2021 | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Q1 | Q2 | Q3 | Q4 | Q1 | Q2 | Q3 | Q4 | Q1 | Q2 | Q3 | Q4 |
| PWR Locked Rotor | | | | | | | | | | | | |
| Inadvertent RCS Blowdown (PWR) | | | | | | | | | | | | |
| Fuel Handling Accident for Extended BU and Enhanced Enrichment | | | | | | | | | | | | |
| BWR Recirculation Pump Shaft Seizure | | | | | | | | | | | | |
| BWR Rod Drop | | | | | | | | | | | | |
| Loss of Feedwater (BWR) | | | | | | | | | | | | |
| Turbine Trip Without Bypass (BWR) | | | | | | | | | | | | |
| Inadvertent RCS Blowdown (BWR) | | | | | | | | | | | | |
| BWR Recirculation Pumps Trip | | | | | | | | | | | | |
| **Digital I&C Risk Assessment** | | | | | | | | | | | | |
| Develop a strategy to perform risk assessment for digital I&C upgrades | | | | | | | | | | | | |
| Develop a static reliability based for a conceptual digital RPS design | | | | | | | | | | | | |
| Perform static reliability studies for a conceptual digital I&C design | | | | | | | | | | | | |
| Develop static reliability model for host utility's plant specific digital RPS | | | | | | | | | | | | |
| Develop RELAP5-3D plant system models with detailed control systems modeling of the digital RPS for the host utility's plant | | | | | | | | | | | | |
| Identification of CCFs caused by digital I&C | | | | | | | | | | | | |
| Analyses of accident scenarios with concurrent digital I&C CCFs | | | | | | | | | | | | |
| Demonstrate dynamic reliability study methodology for digital I&C | | | | | | | | | | | | |

# 10. ANTICIPATED OUTCOMES

The intrinsic value of successful RD&D in this area is expected to be significant. The integrated RI-MP-BEPU approach to be deployed in this project has the potential to accelerate the development and deployment of advanced nuclear technologies to simultaneously enhance the safety and reduce the operating costs of NPPs. The integrated RI-MP-BEPU evaluation approach will allow a comprehensive, integrated, and risk-informed evaluation of plant upgrades to increase safety margins and reduce plant risk (in terms of CDF and LERF) such that plant operational flexibility can be increased and operating costs can be reduced.

The concept of safety margins has served as a fundamental cornerstone of the design, operation, and maintenance of commercial NPPs throughout the history of the industry. NPP operation is predicated on ensuring an adequate level of margin exists for all critical parameters (e.g., fuel and cladding temperature, DNBR, containment pressure, etc.) over the spectrum of postulated plant conditions ranging from normal plant operations to transient and accident conditions. Due to limitations in the state of knowledge (i.e., uncertainties) related to phenomenology and plant response, operating limits have typically been set in a manner that is conservative. Because individual margins are set conservatively, culminations of these conservatisms may reflect unrealistic operating conditions that limit the operating flexibility of the plants and can result in adverse effects on economics, which would not be optimal when evaluated on a risk/benefit basis. In many cases, it is believed that there exist excess margins for which the benefits provided are not sufficient to justify the costs incurred to achieve and maintain them. As a result, it is believed that portions of these margins could be recovered and repurposed to provide significant enhancement in plant economics while providing negligible impact on plant safety.

The RISA Pathway of the LWRS Program has initiated tasks to develop risk-informed multi-scale and multi-physics high fidelity analytical capabilities to support the industry to identify, assess, and recover margins associated with those that are due to over-conservatisms in the current design basis process. The ultimate objective of this research will be to recover and reallocate these excess margins to permit NPPs to operate more economically while providing negligible impact on plant safety. To achieve this objective an integrated evaluation approach which combines the plant PRA methods with MP-BEPU analyses will be developed and demonstrated. This integrated RI-MP-BEPU evaluation framework is intended to enable plant system configuration variations to be studied with speed and precision, including conduct of detailed risk and benefit assessments associated with the adoption of advanced nuclear technologies by the fleet of operating LWR plants in their pursuit of both safety and operational performance enhancements.

This report presents an integrated R&D roadmap to identify and perform high-value evaluations of advanced nuclear technology concepts with the ultimate goal of identifying the technical (e.g., benefits to risk, safety, and operational margins) and economic (i.e., business and cost) elements associated with industry adoption. The integrated evaluation approach is intended to support the development and deployment of advanced nuclear technologies that are capable of achieving substantial safety and economic improvements as well as timely widespread adoption by the U.S. nuclear industry.

One significant benefit of the RI-MP-BEPU evaluation approach will be its capability to support more accurate and efficient cost/benefit evaluations of advanced technology being considered for adoption within the nuclear industry. One of the roadblocks to deployment of advanced technologies in operating NPPs is the high costs and lengthy durations associated with obtaining regulatory approvals. Much of these costs occur in responding to regulatory questions related to uncertainties. Because the RI-MP-BEPU approach explicitly accounts for and propagates uncertainties, the approach is anticipated to provide a more streamlined process to support more timely and efficient regulatory reviews. Streamlining this process by providing more comprehensive analyses that explicitly evaluate uncertainties as an integral part of the process would significantly accelerate and streamline decision making - both for the NPP and the regulatory authority. As a result, the RI-MP-BEPU approach that will be developed in the research is anticipated to provide a significant impact to facilitate deployment of advanced technologies at NPPs and this support the enhancement of the economic competitiveness of the U.S. LWR fleet.

# 11. REFERENCES


1. https://www.nei.org/resources/statistics/us-nuclear-plant-actual-and-expected-uprates
2. S. Hess, J. Gaertner, N. Dinh, and R. Szilard; "Risk-Informed Safety Margin Characterization"; *Proceedings of 17th International Conference on Nuclear Engineering*; 12 – 16 July 2008; Brussels, Belgium (Paper ICONE17-75064).
3. NEA/CSNI/R(2007)9; *Task Group on Safety Margins Action Plan (SMAP) Safety Margins Action Plan – Final Report;* https://www.oecd-nea.org/nsd/docs/2007/csni-r2007-9.pdf
4. https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-appk.html
5. https://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0046.html
6. NRC, *Transient and Accident Analysis Methods*; U.S. NRC Regulatory Guide 1.203; December 2005, https://www.nrc.gov/docs/ML0535/ML053500170.pdf
7. H. Zhang, R. Szilard, A. Epiney, C. Parisi, R. Vaghetto, A. Vanni, K. Neptune; "Industry Application ECCS/LOCA Integrated Cladding/Emergency Core Cooling System Performance: Demonstration of LOTUS-Baseline Coupled Analysis of the South Texas Plant Model"; INL Technical Report INL/EXT-17-4246, June 2017.
8. https://www.nrc.gov/docs/ML1109/ML110970044.pdf
9. ANS 51.1; "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants (Replaces ANSI N18.2); 1983 (withdrawn in 1998)
10. ANSI/ANS-52.1-1978; "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," (Replaces ANSI N18.2); 1983 (withdrawn in 1998)
11. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*; NUREG-0800; Section 15.0 "Introduction – Transient and Accident Analyses" Rev 3; March 2007; United States Nuclear Regulatory Commission; Washington, DC.
12. *Deterministic Safety Analysis for Nuclear Power Plants*, IAEA Specific Safety Guide No. SSG-2. https://www.pub.iaea.org/MTCD/publications/PDF/Pub1428_web.pdf
13. NRC, *Safety Goals for the Operation of Nuclear Power Plants: Policy Statement,* U.S. Nuclear Regulatory Commission; 51 FR 30028; 4 August 1986
14. EPRI, "Insights on Risk Margins at Nuclear Power Plants: A Technical Evaluation of Margins in Relation to Quantitative Health Objectives and Subsidiary Risk Goals in the United States," Electric Power Research Institute. Palo Alto, CA: 2018. 30020122967
15. EPRI, "Safety and Operational Benefits of Risk-Informed Initiatives," Electric Power Research Institute, Palo Alto, CA, 2008, 1016308
16. *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*; U.S. NRC Regulatory Guide 1.174 Revision 2; May 2011; https://www.nrc.gov/docs/ML1731/ML17317A256.pdf
17. https://www.hdfgroup.org/HDF5/
18. B. Boyack, et al., "Quantifying Reactor Safety Margins", NUREG/CR-5249, 1989.
19. Best-Estimate Calculations of Emergency Core Cooling System Performance, U.S. NRC Regulatory Guide 1.157, May 1989.
20. EPRI, *Data Sources for Capability Assessments of Next Generation Safety Analysis Codes*, Electric Power Research Institute, 2014. 3002003110.
21. ASME and ANS, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sb-2013, 2013.
22. N. Siu, "Risk Assessment for Dynamic Systems: an Overview," Reliability Engineering and System Safety, 41, pp. 43–73, 1994.
23. T. Aldemir, "A Survey of Dynamic Methodologies for Progabilistic Safety Assessment of Nuclear Power Plants," Annals of Nuclear Energy, 52, pp. 113-124, 2013.

24. T. Aldemir, et al., “Probabilistic Risk Assessment Modeling of Digital Instrumentation and Control Systems using Two Dynamic Methodologies,” Reliability Engineering and System Safety, 95, pp. 1011–1039, 2010.
25. Z. Ma, et al., “A Simulation-based Dynamic Approach for External Flooding Analysis in Nuclear Power Plants,” Proceedings of the 20th Pacific Basin Nuclear Conference, Springer, 2017.
26. 60 FR 42622, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” Federal Register, Volume 60, Number 158, p. 42622, Washington, DC, August 16, 1995.
27. https://www.nasa.gov/pdf/458490main_TRL_Definitions.pdf.
28. Ronaldo Szilard, et al., *RISA Industry Use Case Analysis*, INL/EXT-18-51012, August 2018.
29. “Optimum Cycle Length and Discharge Burnup for Nuclear Fuel – A Comprehensive Study for BWRs and PWRs: Phase 1: Results Achievable Within the 5% Enrichment Limit”, EPRI. Palo Alto, CA: 2001. 1003133
30. “Optimum Cycle Length and Discharge Burnup for Nuclear Fuel: Phase 2: Results Achievable With Enrichments Greater than 5 w/o”, EPRI. Palo Alto, CA and U. S. Department of Energy, Washington, DC: 2002. 1003217
31. Ken Thomas and Ken Scarola, *Light Water Sustainability Program Strategy for Implementation of Safety Related Digital I&C Systems*, INL/EXT 18 45683, June 2018.
32. CASL-U-2014-0014-002, “VERA Common Input User Manual, Version 2.0.0,” Revision 2, February 2015.
33. CASL-U-2015-0077-000, “MPACT User’s Manual Version 2.0.0,” February 2015.
34. CASL-U-2015-0055-000, “CTF – A Thermal-Hydraulic Subchannel Code for LWRs Transient Analysis,” February 2015
35. https://www.ornl.gov/division/rnsd/projects/origen
36. https://www.ornl.gov/scale
37. https://cmake.org/
38. http://www.iapws.org/relguide/IF97-Rev.html
39. https://www.paraview.org/
40. *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; October 2014; United States Nuclear Regulatory Commission; Washington, DC
41. *FRAPTRAN 1.4: A Computer Code for the Transient Analysis of Oxide Fuel Rod*s; NUREG/CR-7023, Vol. 1; March 2011; United States Nuclear Regulatory Commission; Washington, DC
42. Fuel Performance Analysis Capability in FALCON. EPRI, Palo Alto, CA: 2002. 1002866.
43. J. D. Hales, et al., “BISON Theory Manual, The Equations behind Nuclear Fuel Analysis,” Idaho National Laboratory, Jan., 2015.
44. Spencer, B., M. Backman, P. Chakraborty, D. Schwen, Y. Zhang, H. Huang, X.-M. Bai, and W. Jiang, *Grizzly Usage and Theory Manual Version 1.0 Beta*, INL/EXT-16-38310, Idaho National Laboratory, Idaho Falls, ID, Mar. 2016.
45. *Framework for Risk-Informed Safety Margin Characterization*. EPRI. Palo Alto, Ca; 2009. 1019206.
46. https://relap53d.inl.gov/SitePages/Home.aspx (accessed 12 September 2018)
47. TRACE V5.0 Theory Manual: Field Equations, Solution Methods, and Physical Models https://www.nrc.gov/docs/ML0710/ML071000097.pdf (accessed 12 September 2018)
48. Modular Accident Analysis Program: A Software Toll for Analyzing Nuclear Plant Accident Scenarios; EPRI 1025795; July 2012; Electric Power Research Institute; https://www.epri.com/#/pages/product/000000000001025795/ (accessed 12 September 2018)
49. RELAP-7 Theory Manual, INL/EXT-14-31366, Idaho National Laboratory, February 2014.
50. NUREG/CR-6119 MELCOR Computer Code Manuals (3 Volumes) https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6119/ (accessed 12 September 2018)

51. http://www.numerical.com/gothic.php.
52. H-N Jow, et al., "MELCOR Accident Consequence Code System (MACCS)," NUREG/CR-4691, 1989.
53. https://saphire.inl.gov/ (accessed 12 September 2018)
54. Risk and Reliability Workstation; EPRI 1020712; July 2012; Electric Power Research Institute; https://www.epri.com/#/pages/product/000000000001020712/ (accessed 16 April 2018)
55. D. Mandelli, et al., "BWR Station Blackout: A RISMC Analysis Using RAVEN and RELAP5-3D", *Nuclear Technology*, vol. 193, 161-174, January 2016.
56. https://www.python.org/
57. https://emrald.inl.gov/SitePages/Overview.aspx (accessed 12 September 2018)
58. https://xmpp.org/