INL/EXT-18-45220

Light Water Reactor Sustainability Program

R&D Roadmap for Enhanced Resilient Plant Systems, Metrics, Scenarios, Risk Analysis, and Modeling and Simulation

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EXECUTIVE SUMMARY

The nuclear industry and the U.S. Department of Energy are jointly developing accident tolerant fuel (ATF) which has the potential to greatly improve the safety and economic performance of the current fleet of NPPs. The combination of ATF with the optimal use of FLEX equipment (deployed as an industry response to the Fukushima Daiichi Nuclear Power Plant (NPP) accident in 2012), enhancements on plant components and systems, incorporation of augmented cooling systems or new passive cooling systems, and improvements in fuel cycle efficiency, have the potential to render the NPPs more efficient and resilient to events (both internally and externally initiated) that have the potential to challenge the integrity of the fuel or plant containment. In this report, combinations of these plant improvements are called Enhanced Resilient Plant Systems. In order to accelerate the development and deployment of the components that comprise Enhanced Resilient Plant Systems, a well-integrated toolkit that is fuel/clad-, fuel cycle-, systems-, and scenario-centric is required to provide a ready platform for the analysis of the advanced ATF fuel and cladding materials that are being developed, as well as plant systems modifications to achieve enhanced resiliency. Since the deployment of these enhancements will comprise a risk/benefit tradeoff of their impacts on plant safety and economics, the envisioned toolkit will need to be capable to characterizing the multidimensional aspects associated with these evaluations and provide decision-makers with information that can optimize the technological, economic, and safety benefits that are to be obtained.

The Risk-Informed Systems Analysis (RISA) Pathway of the DOE Light Water Reactor Sustainability (LWRS) Program initiated several research tasks to develop analytical capabilities to support the industry in the transition to the ATF and Enhanced Resilient Plant Systems. The general idea behind the initiative is the development of an integrated evaluation approach which combines the plant probabilistic risk assessment (PRA) models with Multi-Physics Best Estimate Plus Uncertainty (MP-BEPU) analyses in a seamless fashion. The integrated evaluation framework aims at enabling plant system configuration variations to be studied with speed and precision, including detailed risk and benefit assessments of introducing ATF and Enhanced Resilient Plant Systems into current LWR plants to achieve both safety and operational performance enhancements.

The focus of this report is to present an integrated research and development (R&D) roadmap to identify and perform high value evaluations of proposed ATF and Enhanced Resilient Plant Systems concepts to identify both the technical (e.g. benefits to risk, safety, and operational margins) and the economic (i.e. business and cost) elements associated with industry adoption of the technology. The integrated evaluation approach will be developed to support the development, testing, qualification, licensing, and deployment of ATF and Enhanced Resilient Plant Systems technology that is capable of achieving substantial safety and economic improvements as well as timely widespread adoption by the US nuclear industry.

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ACRONYMS

1D/2D/3D	One, Two, or Three Dimensional (respectively)
AOO	Anticipated Operational Occurrence
API	Application Programming Interface
ATF	Accident Tolerant Fuel
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
BEPU	Best Estimate Plus Uncertainty
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analysis System
CDF	Core Damage Frequency
CHF	Critical Heat Flux
CMFD	Coarse Mesh Finite Difference
CSARP	International Cooperative Severe Accident Research Program
CTF	COBRA-TF (Coolant Boiling in Rod Arrays – Two Fluid subchannel thermal-hydraulics analysis code)
DBA	Design Basis Accident
DNB	Departure from Nucleate Boiling
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
FALCON	Fuel Analysis and Licensing Code—New
FLEX	Diverse and Flexible Coping Strategy
FY	Fiscal Year
HRA	Human Reliability Analysis
IE	Initiating Event
INL	Idaho National Laboratory
IREM	integrated risk evaluation model
JH	JENSEN HUGHES
LERF	Large Early Release Frequency
LB-LOCA	Large Break Loss of Coolant Accident

LOCA	Loss of Coolant Accident
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LOTUS	LOCA analysis toolkit for the US
LTSBO	Long Term Station Blackout
LWRS	Light Water Reactor Sustainability
MAAP	Modular Accident Analysis Program
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
MSPI	Mitigating Systems Performance Index
NEM	Nodal Expansion Method
NOED	Notice of Enforcement Discretion
NPP	Nuclear Power Plant
NRC	US Nuclear Regulatory Commission
O&M	Operation & Maintenance
ODE	Ordinary Differential Equation
PARCS	Purdue Advanced Reactor Core Simulator
PDE	Partial Differential Equation
PIRT	Phenomenon Identification and Ranking Table
PNNL	Pacific Northwest National Laboratory
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
R&D	Research and Development
R&R	Risk and Reliability
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RELAP5	Reactor Excursion and Leak Analysis Program 5
RELAP-7	Reactor Excursion and Leak Analysis Program 7
RIA	Reactivity Insertion Accident
RISA	Risk Informed Systems Analysis
RISMC	Risk Informed Safety Margin Characterization

ROP	Reactor Oversight Process
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SDP	Significance Determination Process
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratory
SRP	Safety Review Plan
SSC	systems, structures and components
STSBO	Short Term Station Blackout
TRACE	TRAC/RELAP Advanced Computational Engine

1. INTRODUCTION

Existing nuclear power plants (NPP) are facing significant challenges of having comparatively high operating costs while simultaneously being required to implement measures to enhance plant safety. There are a number of initiatives underway in the commercial nuclear power industry to enhance the safety and improve the economic competiveness of existing NPPs. These initiatives include efforts on developing Accident Tolerant Fuel (ATF) [1, 2], implementing a Diverse and Flexible Coping Strategy (FLEX) [3] to provide additional mitigation capability in the unlikely occurrence of a beyond design basis event, and an industry-wide initiative entitled, "Delivering the Nuclear Promise: Advancing Safety, Reliability, and Economic Performance" [4]. The collective changes resulting from these separate initiatives have the potential to produce greater contributions in the aggregate to plant efficiency and resiliency, especially if integrated through systematic efforts and analyses.

While the research and development on ATF are still ongoing, the main attributes of ATF include improved fuel and cladding properties, improved clad reaction with steam, slower hydrogen generation rates, better fission product retention, and improved fuel cladding interactions. These attributes are intended to lead to higher melting temperature of the fuel/cladding, longer time windows (coping time) for performance of operator mitigating actions with higher likelihoods of successful completion (i.e. improved human reliability), lower hydrogen (or other combustible gas) generation, and fewer fission products released during severe accident conditions. The combinations of ATF, optimal use of FLEX, enhancements to plant components and systems, incorporation of augmented or new passive cooling systems, as well as improvements on fuel cycle efficiency are called Enhanced Resilient Plant Systems.

The objective of this research effort is to identify and perform high value evaluations of proposed accident tolerant fuel (ATF) / Enhanced Resilient Plant Systems concepts to identify both the technical (e.g. benefits to risk, safety, and operational margins) and the economic (i.e. business and cost) elements associated with industry adoption of the technologies. This research will develop an integrated approach to support the development, testing, qualification, licensing, and deployment of ATF and Enhanced Resilient Plant Systems technologies that are capable of achieving substantial safety and economic improvements as well as timely widespread adoption by the US nuclear industry.

The key metrics that will be used to evaluate the resiliency enhancements for a NPP include:

1) Increased Coping Time compared to the current fuel/plant systems

2) Decreased Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) compared to the current fuel/plant systems

3) Increased safety margins such as more margins on fuel/clad temperature, or reduced hydrogen gas generation, compared to the current fuel/plant systems

4) Improved plant economics during normal operations

Plant resiliency enhancements can be demonstrated by meeting one or more metrics described above. For the first metric, the definition of coping time has evolved over time. In the U.S. NRC's Regulatory Guide 1.155 for Station Blackout, the coping time is defined as the limited time the batteries are capable of providing electrical power for the essential safety systems. In evaluating the risks and benefits of ATF and Enhanced Resilient Plant Systems, the definition of coping time may include the following:

Coping Time is the available time for NPP operators to mitigate an event that has the potential to result in significant core damage or a large early release of radioactive materials to the environment.

The mitigation actions may include: recovery of offsite power, recovery of on-site emergency diesel generators (EDG), initiation of high or low pressure recirculation, manual initiation of auxiliary feedwater (AFW) if automatic actuation fails, initiation of feed and bleed cooling, etc. In addition to providing more time to implement specified mitigation measures to address the event, increased coping times also would be expected to reduce human error probabilities and enable better utilization of the FLEX equipment combined with ATF and Enhanced Resilient Plant Systems configurations. As a result of these improvements the plant CDF and LERF are expected to be reduced. The CDF and LERF reductions could bring direct economic benefits to NPPs with the potential to reduce costs and regulatory burden associated with mitigating systems performance index (MSPI), significance determination process (SDP), notices of enforcement discretion (NOED), and plant maintenance and refueling outages.

The outcome of this research effort will be a Risk Informed Systems Analysis (RISA) R&D plan that is integrated with ongoing DOE and industry efforts to mature and deploy ATF / Enhanced Resilient Plant Systems concepts in a timeframe identified by the industry as necessary to meet critical decision-making milestones (e.g. second license renewal applications). The RISA methods and tools developed in this plan will be used to assess the benefits of advanced nuclear fuel concepts in terms of safety, operational performance, and economics at existing nuclear power plants (NPPs). Successful application of such methods by NPPs will support the implementation of an economically optimal combination of advanced fuels and plant system resiliency enhancements to improve safety and performance while also reducing maintenance and operational costs.

2. Technical Approach

In order to fully realize the benefits of ATF and Enhanced Resilient Plant Systems in terms of risk benefits and cost reduction and to ease the transition to accommodate the ATF and Enhanced Resilient Plant Systems designs, it is imperative to perform comprehensive risk-informed evaluations of the design changes at the plant, systems, and components level in an integrated manner. Accomplishment of this objective entails the development of an Integrated Risk Evaluation Model (IREM) as shown in Figure 1.

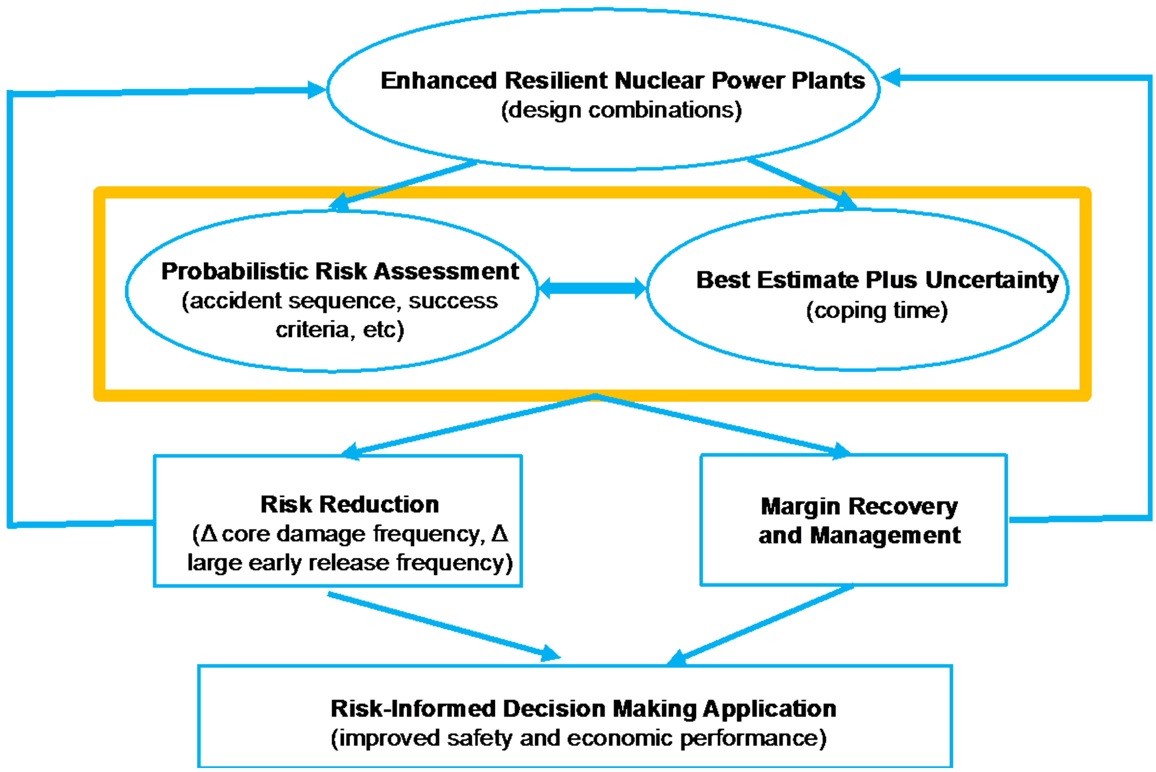

Figure 1. Schematic Illustration of the Integrated Risk Evaluation Model (IREM) for Enhanced Resilient Plant Systems.

Risk assessments of NPPs typically are performed by combining probabilistic risk assessment (PRA) methods with best estimate plus uncertainty (BEPU) methods to assess the plant physics (neutronics, thermal hydraulics, materials response, etc.) to postulated transient and accident events. PRA methods evaluate scenarios in order to determine accident sequences that include failure of Structures, Systems, and Components (SSCs) given a set of prescribed initiating events. PRA methods not only estimate risk metrics such as core damage frequency (CDF) and large early release frequency (LERF), but also determine what the most probable accident sequences are and the components that contribute the most to the overall plant risk. Modern BEPU methods employ multi-physics analysis tools in order to assure that plant safety systems can prevent the occurrence of core damage or a given set of accident conditions. One approach developed by the RISA (formerly known as Risk-Informed Safety Margin Characterization (RISMC)) Pathway, called LOTUS, is a Multi-Physics Best Estimate Plus Uncertainty (MP-BEPU) reactor safety analysis framework. This approach integrates single physics computer codes for Core Design, Fuels Performance, and Systems Analysis in a seamless fashion. Within the LOTUS framework uncertainties can be propagated consistently throughout the multi-physics simulations. The risk benefits can be quantified from the results obtained from the simulation with the reduction of the risk metrics such as CDF and LERF evaluated. From these results potential cost reductions can be identified and quantified. The results will be a characterization of SSC risk reductions as a function of key mitigation actions (such as for increasingly longer coping times during severe accident sequences), from which associated economic and regulatory benefits can be evaluated.

With the selection of candidate ATF and Enhanced Resilient Plant Systems designs (e.g., FLEX, new passive cooling systems, etc.), IREM will be used to perform detailed PRA/BEPU analyses. Specifically, the following analysis steps are to be carried out:

1. Identify a set of accident sequences for both pressurized and boiling water reactors that might be mitigated by accident tolerant fuel/enhanced resilient nuclear power plant systems. The candidate scenarios include short and long-term station blackout, loss-of-coolant, loss of feedwater, anticipated transients without scram (ATWS), steam generator tube rupture (SGTR), turbine load mismatch, etc. Note that Section 4 below provides a more complete discussion of plant design and licensing basis events that can serve as a population of candidate sequences for which initial demonstration analyses can be performed.

2. Identify new phenomena which may need to be considered as a result of adoption of the new technologies. For example, for the more advanced ATF concepts under current consideration, it is possible that the ATF design is sufficiently robust that failure of core structural components may occur before the fuel fails. These phenomena need to be considered at this stage as they may require changes to the plant response and the accident analyses that are performed.

3. Perform best estimate plus uncertainty calculations for the analyzed sequences using fully coupled physics models by simulating the core/fuel/cladding and plant/system interactions in order to determine plant integrity (fuel, primary system, and containment) for the candidate accident tolerant fuel/enhanced resilient nuclear power plant systems.

4. Conduct a detailed probabilistic risk assessment by performing scenario-specific accident analyses to assess the impact on applicable plant risk metrics (e.g. CDF and LERF). The analyses will reflect the plant responses including the stochastic behavior of applicable systems, structures, components, and human actions. The evaluation will investigate risk analysis perturbations, including potential changes in system success criteria, human actions, and component performance. These perturbations will be characterized according to their risk impact in order to identify and assess beneficial plant changes.

5. Evaluate results to determine key assumptions, insights, data, and uncertainties that are important to support risk informed decision making. Once the possible changes in risk and safety margins are identified, the IREM can be used in high value risk-informed decision making applications, both in operational and regulatory applications.

The operational applications include: enhanced fuel performance and core design efficiency through increased enrichment, burnup extension, fuel cycle length extension and load following (also known as flexible plant operations); risk informed surveillance test intervals; risk informed technical specification completion times; risk informed emergency planning zone, and a10CFR 50.69 alternative treatment considerations to better understand potential changes that are possible for the specific systems, structures, and components of interest. Application of 10CFR 50.69 allows plant equipment to be recategorized based on their safety designation (i.e., safety-related or non-safety-related) and their risk significance (i.e., risk-significant or non-risk-significant). For example, with the application of ATF and Enhanced Resilient Plant Systems, some safety-related equipment may be able to be recategorized as safety-related but non-risk-significant (RISC-3) due to the reduction of the risk significance of these equipment with the increased coping time and increased availability of FLEX mitigating equipment/strategies. This recategorization implies potential cost savings on the procurement, maintenance, surveillance, and administration of plant equipment designated as safety-related.

The regulatory applications include support for: justification for continued operation evaluations, limiting conditions for operation, and component design bases inspection processes. The risk significance reduction of the current plant equipment could also benefit plant owners and operators in complying with the U.S. Nuclear Regulatory Commission Reactor Oversight Process (ROP) including Significance Determination Process (SDP) and Mitigating Systems Performance Index (MSPI). All of the aforementioned risk-informed applications can be translated to direct economic benefits with the continuation of plant operation and reduction of operating, oversight, maintenance, and administration costs.

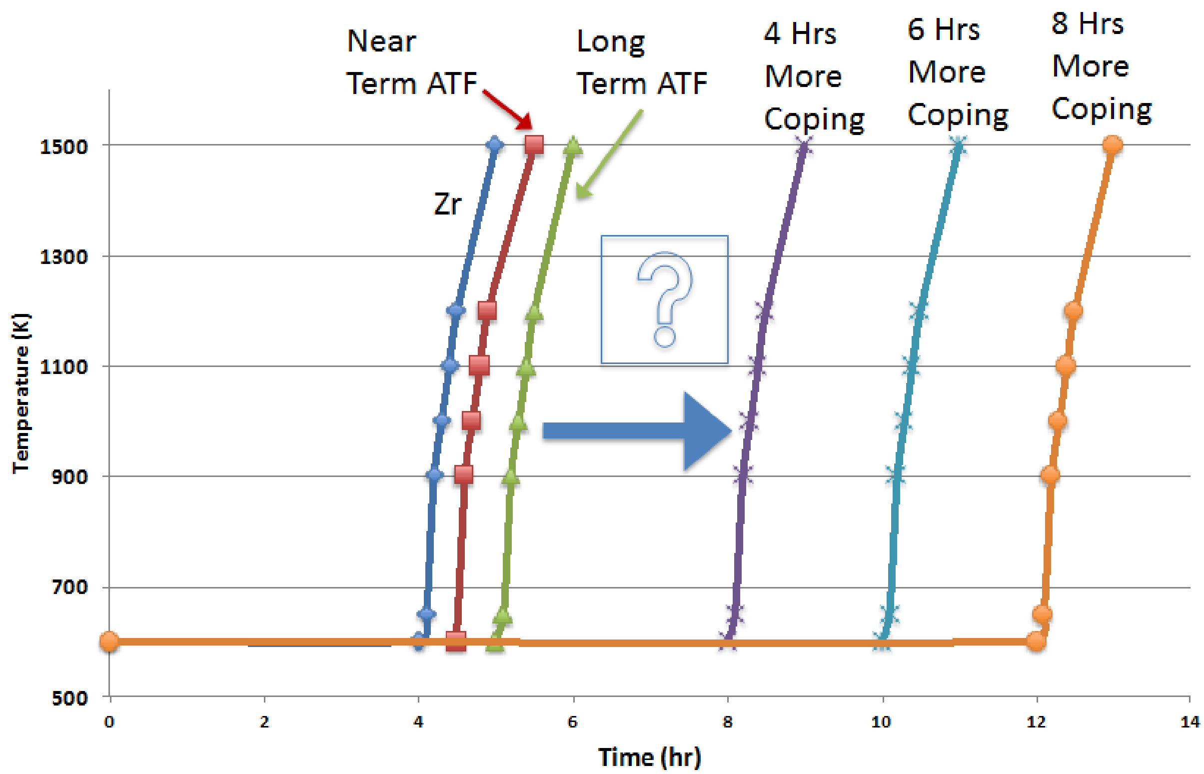

Figure 2. Illustration of Increasing Coping Time for Enhanced Resilient Plant Systems.

Increasing the coping time during plant transient and accident conditions is an essential aspect of enhanced resilient plant systems and warrants further discussions here. Previous studies indicate that the coping time gain with ATF for the most extreme events (i.e. a short-term station blackout (STSBO) event) is on the order of 30 minutes to two hours depending on the initial conditions and specific ATF characteristics. For this type of event, ATF, by itself, will only provide incremental improvements on the estimated plant risk metrics (CDF and LERF). In order to achieve significant improvements (reductions) in CDF, much longer increases in coping time (e.g., 4, 6, 8 hours) would be needed (see Figure 2). In this research plan, risk-informed systematic studies on various configurations of enhanced resilient plant systems will be performed to identify the opportunities for enhancements on plant components/systems including optimal utilization of the FLEX equipment to gain increases on coping time (additional to the coping time gain due to ATF alone). The approach to study coping time increase is described as follows:

Starting from a PRA model for a given initiating event the accident sequences that have high contributions to the overall risk for that accident (risk significant accident sequences) will be identified. These risk significant accident sequences will be further studied to identify potential enhancements (including design changes / modifications on the risk significant components/systems, enhancements in mitigation strategies, etc.) such that plant resiliency can be enhanced. The best estimate plus uncertainty plant simulation models (e.g. RELAP5-3D models) will be built based upon the risk significant accident sequences. The BEPU simulations will be carried out to study the impact of the proposed enhancements of the risk significant components/systems in combination with ATF on Coping Time and safety margins. The outcomes of the analyses (e.g. observed extensions in Coping Times) will be fed back into the PRA model to calculate the CDF and LERF reduction. The results from the PRA/BEPU calculations will be used in the cost and benefit tradeoff studies to evaluate an optimal cost-effective strategy to enhance NPP's safety and operational economic performance.

3. Industry Collaboration

The research described in this plan is a proposed collaboration between the LWRS Program, the Electric Power Research Institute (EPRI), and JENSEN HUGHES (JH). The research is being coordinated via ongoing interactions between the three organizations to develop integrated research plans which will maximize the effective allocation of resources and support industry objectives to begin core reload batch size deployment of "near-term" ATF concepts (coated claddings and doped fuel concepts) by 2025. The collaboration will provide a dedicated mechanism to develop the RISA suite of methods and tools with direct input from NPP owner / operators through their membership in EPRI and to apply them at the operating plants.

Even though the R&D will be conducted collaboratively, each organization will have different focuses as illustrated in Figure 3 in terms of advancing the technology readiness level (TRL) of ATF. The LWRS Program will focus on the technology development and demonstration (TRL 5 & 6, shaded in green). LWRS has the role of performing early stage high impact research to improve the resiliency of NPPs, such as developing integrated risk evaluation models to quantify the resiliency enhancements and to predict performance of the plants' response with regards to various initiating events for AOOs, DBAs and BDBAs. Industry organizations, such as fuel vendors, plant owner/operators, EPRI and JENSEN HUGHES will focus on technology deployment and business development (TRL 7, 8, & 9). They have the responsibility to evaluate the performance of ATF (ATF Valuation 2.0) using industry tools such as CAFTA and MAAP, and develop the business case for the industry to adopt ATF. This distinction clearly indicates the need for collaboration to provide a clear and efficient path for technology maturation leading to industry deployment of ATF concepts. This collaborative approach will provide a well-structured and defined path that will permit the development, demonstration, licensing and deployment of ATF with emphasis on achieving DOE and industry objectives. This structure permits direct industry involvement during the development and initial demonstrations of the methods and tools; thus providing a mechanism to ensure they will meet industry needs. It also provides a mechanism for industry feedback to the developers via direct participation in first of a kind applications of the methods and tools as they are used by industry. This collaborative approach is essential for advancing this technology in a timely and cost effective manner and meet industry's needs to deploy commercial accident tolerant fuel reloads by mid-2020.

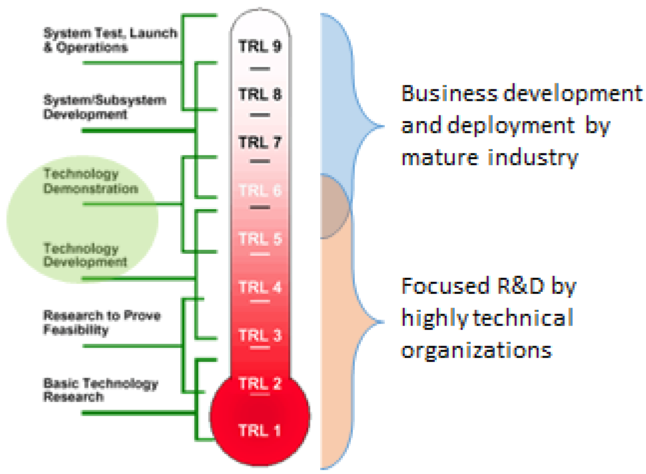

Figure 3. Illustration of Technology Readiness Level.

The following provides a breakdown of each of the collaborator's specific activities.

- INL will develop and validate the methods and tools that will be used in the conduct of RISA. Specifically, INL will integrate the existing and high fidelity modeling and simulation tools being developed by DOE's modeling and simulation programs into an easy to use multi-physics framework with uncertainty quantification and sensitivity analysis capability built-in. The multi-physics best estimate plus uncertainty framework will be integrated with the PRA tools to yield the integrated risk evaluation model. INL will apply these tools to provide integrated analyses of all technical and economic aspects of ATF and Enhanced Resilient Plant Systems, including fuel performance and plant response to normal operational, off-normal, and accident (including severe accident) conditions.

- EPRI will provide technical leadership for conducting analyses for NPP owner / operators that assess the benefits of ATF. As an independent research and development organization, EPRI provides guidance on technical requirements that advanced fuels will need to meet to allow them to be commercially viable for use in operating NPPs. Within this perspective, EPRI will serve as the primary interface between the NPP owner / operators (i.e. EPRI members) and other key ATF stakeholders including US DOE, US NRC, and the fuel vendors that are developing advanced fuel concepts (including ATF concepts). As part of this collaboration, EPRI is leading efforts to conduct a gap analysis (PIRT-like effort) to identify material property and modeling needs for the ATF concepts under development. EPRI will conduct independent research that corroborates results provided by the other stakeholders. Through its access to its members, EPRI also will sponsor relevant analyses at operational NPPs to assess and demonstrate the benefits of ATF deployment.

- JENSEN HUGHES will conduct technical analyses to evaluate the performance and benefits associated with the development and deployment of advanced nuclear fuels. JENSEN HUGHES will provide integrated analysis capabilities in all technical and economic assessments of ATF using current industry tools. Such assessments will include fuel performance and plant response to normal operational, off-normal, and accident (including severe accident) conditions. JENSEN HUGHES also will assess the impact of ATF on plant probabilistic risk assessments (PRA) results to quantify the safety benefits provided by the advanced fuels. Additionally, JENSEN HUGHES will support industry assessments of the economic benefits that may be obtained from the deployment of advanced fuel (including ATF) concepts. Finally, JENSEN HUGHES 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.

Each of the partners will work collaboratively to prioritize the research tasks necessary to support the efficient and cost effective development, testing, licensing, and deployment of ATF at operational NPPs at the earliest practicable date (with particular emphasis on support of industry objectives to begin core reload batch size deployment of “near-term” ATF concepts by 2025). Additionally, EPRI, INL and JENSEN HUGHES will work collaboratively with other ATF stakeholders in their ATF related research and development activities.

4. Research Activities

This plan of work is to assess the safety, risk, and economic impacts of accident tolerant fuel concepts/Enhanced Resilient Plant Systems. The plan will focus resources on analyses of the anticipated performance of ATF concepts during the full spectrum of plant transients and accidents including selected abnormal operational occurrences (AOOs), design basis accidents (DBAs) and beyond design basis accidents (BDBAs). Because it is highly desired to transition to ATF designs in a much shorter timeframe than has historically been accomplished, this research plan will build upon analyses that have been conducted up to the present time. See for example references [5 – 8].

A requirement for the licensing and deployment of ATF will be an assessment of ATF performance under postulated transient and accident conditions. The applicable postulated events that will require these analyses are defined in Section 15.0 of the US NRC Standard Review Plan (SRP – NUREG 0800) [9]. For commercial LWRs in the United States, the following DBA events are required to be analyzed:

- Major rupture of a pipe containing reactor coolant up to and including the double-ended rupture of the largest pipe in the reactor coolant pressure boundary (RCPB). This accident typically is referred to as a large break loss of coolant accident (LB-LOCA).

- Major rupture of a secondary system pipe up to and including a double-ended pipe rupture.

- Ejection of a PWR control rod assembly or dropping of a BWR control blade (i.e. a reactivity insertion accident – RIA).

- Single reactor coolant pump (RCP) locked rotor in a PWR or seizure of one reactor recirculation pump in a BWR.

These events constitute the spectrum of design basis accidents required for analysis in the licensing of a NPP in the United States. Because the DBAs listed above represent some of the most extreme conditions that a 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 ATF.

Although the performance of ATF during LB-LOCA events was previously evaluated as described in references [5 – 8], various attributes (e.g. ATF material properties, etc.) were uncertain, so a number of simplifying assumptions were employed. Additionally, the uncertainties in these data and applications were not systematically assessed. As a result, additional assessments of the LB-LOCA event for both BWRs and PWRs will be conducted in this research project. The following DBA events also will be included in the analyses performed as part of this research:

- RIA event (PWR control rod assembly ejection and BWR control rod drop).

- PWR locked rotor and BWR recirculation pump shaft seizure.

From the viewpoint of the impact on nuclear fuel (i.e. likelihood of an event resulting in core damage), the rupture of a major secondary system pipe is less challenging than a LB-LOCA. As a result, this accident will not be assessed in the course of this research. For longer-term ATF concepts that replace the current UO_2 fuel matrix (e.g. use of U_3Si_2 fuel) or replace the cladding with a completely new material (e.g. ceramic cladding such as SiC), the effect of an accident that occurs during refueling operations (when primary containment is not operational) also should be investigated (see NUREG 0800 Section 15.7.4 [10]). However, since these concepts are not the primary focus of current industry priorities (which are on rapid licensing and deployment of short-term ATF concepts such as coated claddings and doped UO_2 fuel), this accident will not be evaluated as part of this research.

In addition to the DBAs indicated above, Section 15.0 of the SRP provides a listing of plant AOOs which are required to be evaluated. These events span the full range of conditions for which LWR fuel must be evaluated for the purposes of licensing the fuel for use. Because ATF is in the early stages of development and maturation, only the analyses of the most significant of these AOOs would be a prudent use of resources at the current time. Thus, to achieve the objectives previously stated and to elicit the efficient use of resources, a subset of the events listed in Section 15.0 of the SRP will be analyzed as part of the research described in this plan. In the development of this plan, each of the following events was considered and a decision was made regarding whether it should be analyzed during this portion of ATF development. The decision and basis for each are provided below.

- Inadvertent control rod or rod group withdrawal

  Not Included: Most significant reactivity insertion event is addressed by RIA (PWR control rod assembly ejection and BWR control rod drop). Analysis of this event is not needed for initial ATF benefits assessments.

- Loss / interruption of core coolant flow (excluding PWR RCP locked rotor)

  Not Included: Most significant loss of flow event is addressed by locked rotor (PWR) and reactor recirculation pump trips (BWR). Analysis of this event is not needed for initial ATF benefits assessments.

- Inadvertent moderator cooldown

  Not Included: Most significant reactivity insertion event is addressed by RIA (PWR control rod assembly ejection and BWR control rod drop). Analysis of this event is not needed for initial ATF benefits assessments.

- Inadvertent chemical shim – PWR only

  Not Included: Most significant reactivity insertion event is addressed by RIA (PWR control rod assembly ejection and BWR control rod drop). This transient is slow to develop, with time for mitigating actions. Analysis of this event is not needed for initial ATF benefits assessments.

- Depressurization by spurious active element operation (e.g. relief valve):

  Not Included: Event impact is more significant for the reactor pressure vessel (RPV) than for the fuel. Analysis of this event is not needed for initial ATF benefits assessments.

- Reactor coolant blowdown through a safety relief valve (SRV)

  Not Included: Event impact is more significant for the RPV than for the fuel. Analysis of this event is not needed for initial ATF benefits assessments.

- Loss of normal feedwater

  Included: This AOO previously was included in analyses of events for PWRs (part of EPRI study [5] and Westinghouse evaluations [8]) through transition to feed and bleed cooling. It will be analyzed in more detail to include combined effects of ATF, FLEX, and passive cooling strategies. Additionally, BWR analyses (failure of feedwater controller initiating event) will be performed.

- Loss of condenser cooling

  Not Included: Primary impact of loss of condenser cooling is reduced primary system heat removal, and eventual turbine trip and reactor scram. The event is less severe than loss of feedwater event. Analysis of this event is not needed for initial ATF benefits assessments.

- Steam generator tube leaks or rupture (SGTR) – PWR only

  Included: This AOO is significant as a containment bypass event for PWRs. It will be analyzed as part of this research effort.

- Reactor – turbine load mismatch (including load rejection and turbine trip events)

  Included: This AOO previously was included in analysis of events for BWR turbine trip without bypass (part of EPRI study [5]). It will be analyzed in more detail to include combined effects of ATF, FLEX, and passive cooling strategies. Also will perform analyses for PWR as part of this research effort.

- Control rod drop (inadvertent addition of neutron absorber) – PWR only

  Not Included: Event impact is to skew reactor flux shape; if extreme, it can result in reactor trip. Analysis of this event is not needed for initial ATF benefits assessments.

- Single operator error

  Not Included: Most significant effects of single operator error are less severe than other AOO / DBA events. Analysis of this event is not needed for initial ATF benefits assessments.

- Single failure of core component

  Not Included: Most significant effects of failure of single core component are limited in spatial extent (i.e. effects limited to neighboring fuel bundles) or are less severe than other AOO / DBA events. Analysis of this event is not needed for initial ATF benefits assessments.

- Single electrical system failure

  Not Included: Most significant effects of single electrical system failure are less severe than other AOO / DBA events. Analysis of this event is not needed for initial ATF benefits assessments.

- Small RCS system leak (small line break or crack in large pipe)

  Not Included: Most severe effects of small RCS leak are covered by LB-LOCA DBA. Smaller leaks take a longer time to develop, providing time for operators to take mitigation actions per emergency operating procedures (EOPs). Analysis of this event is not needed for initial ATF benefits assessments.

- Minor secondary system break – PWR only

  Not Included: This event is less severe than loss of feedwater event. Analysis of this event is not needed for initial ATF benefits assessments.

- Loss of off-site power (LOOP)

  Included: This AOO was previously included in analyses in various studies as a station blackout (SBO) BDBA (see [5 – 8]). It will be analyzed in more detail to include combined effects of ATF, FLEX, and passive cooling strategies.

- Improper fuel assembly position

  Not Included: The impact of this event depends on details of core cycle design and is not needed for initial scoping assessments. Analysis of this event is not needed for initial ATF benefits assessments.

- Inadvertent RCS blowdown – BWR only

  Not Included: Event impact is more significant for the RPV than for the fuel. Analysis of this event is not needed for initial ATF benefits assessments.

- Loss of feedwater heating

  Not Included: The primary impact of loss of condenser cooling is reduced primary system heat removal, and eventual turbine trip and reactor scram. This event is less severe than loss of feedwater event. Analysis of this event is not needed for initial ATF benefits assessments.

- Trip of all reactor recirculation pumps (natural circulation) – BWR only

  Included: This AOO results in natural flow reactor operation and potential for significant oscillations in reactor power. This AOO will be analyzed as part of this research effort. This event also should include oscillations that occur without the occurrence of a reactor scram.

- Inadvertent pump start in hot recirculation loop – BWR only

  Not Included: This transient results in forced water injection to the core with potential positive reactivity insertion (cold stratified water from vessel bottom head and sweeping of voids). This event is less severe than startup of idle cold recirculation loop. The most significant reactivity insertion event is addressed by control rod drop. Analysis of this event is not needed for initial ATF benefits assessments.

- Startup of idle recirculation pump in cold loop – BWR only

  Not Included: This transient results in forced water injection to the core with potential positive reactivity insertion (cold stratified water from vessel bottom head and sweeping of voids). The most significant reactivity insertion event is addressed by control rod drop. Analysis of this event is not needed for initial ATF benefits assessments.

- Condenser tube leak – BWR only

  Not Included: This AOO is a potential containment bypass event for BWRs but is less severe than SGTR event for PWRs. Analysis of this event is not needed for initial ATF benefits assessments.

- Reactor overpressure with delayed scram

  Not Included: The event impact is more significant for the RPV than for the fuel. Analysis of this event is not needed for initial ATF benefits assessments.

In addition to the design basis events described in the SRP, beyond design basis accident (BDBA) events also are of importance for the assessment of potential benefits of ATF. The most significant of these events has already been the subject of extensive analyses [5 – 8] and included station blackout (SBO) events (both short-term (STSBO) and long-term (LTSBO) for both BWR and PWR NPPs). Additionally, loss of all feedwater events (including normal feedwater and both motor and turbine driven auxiliary feedwater) events for PWR NPPs have been evaluated in these previous studies.

In addition to the assessment of the impact of ATF on key safety related events described above, this research will assess potential impacts (benefits) on high value operational applications. The following specific economic enhancements have been targeted by the industry as providing high value for which ATF should be evaluated as an enabling technology:

- flexible plant operations (load follow capability),
- extended operating cycles (in particular extend PWR cycles to 24 months),
- enhanced fuel enrichments (greater than current 5 w/o limit),
- extended discharge burnup (greater than current 62,000 MWd/MTU limit).

It is important to consider that these economic benefits assessments are interrelated and likely will require several iterations. For example, enhanced fuel enrichments likely will be needed to support extended operating cycles [11].

It should be noted that the application of RISA to achieve enhancements to plant safety and economics involves a systems analysis and optimization approach. As a result, the most cost-effective way to achieve the desired benefits likely will require enhancements across multiple dimensions in addition to development of advanced nuclear fuels. Therefore, this research program also will investigate the application of RISA to high value systems level applications. Because these applications may have critical time dependencies, this research will include investigations into approaches to apply dynamical methods (e.g., dynamic PRA approaches) that address the fundamental issues which have limited the practical application of these technologies in commercial NPP decision-making.

Finally, to achieve additional economic benefits from an enhanced resilient NPP, additional risk assessment capabilities will be required to evaluate the potential safety benefits from such enhancements. A critical element of these enhancements will involve a comprehensive evaluation of the uncertainties in the engineering (deterministic) assessments and subsequent integration of these uncertainties into plant risk assessment (PRA) models and risk management programs. One such important limitation that has been identified in recent evaluations of ATF concepts [5] is the current state of practice associated with human reliability analysis (HRA).

5. DESCRIPTION OF COMPUTER CODES

Both existing and advanced analysis tools will be utilized in the application of RISA to the key transient and accident sequences identified previously. Due to the high costs associated with the qualification and regulatory acceptance of analytical tools, it is anticipated that the licensing of ATF concepts 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 ATF concepts (material properties, material response models, fuel performance models, etc.), these existing tools will need to be informed and enhanced to support ATF licensing and deployment. The codes that have been identified for use in the execution of this research plan are detailed below.

5.1 Core Design and Analysis Tools

5.1.1 VERA-CS

VERA-CS [12] includes coupled neutronics, thermal-hydraulics, and fuel temperature components with an isotopic depletion capability. The neutronics capability employed is based on MPACT [13], 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 [14]. The isotopic depletion is performed using the ORIGEN code system.

5.1.1.1 MPACT

MPACT [13] 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, are 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 for 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 “2D/1D" 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 5-10 cm thick, and the planar solution is obtained for each plane using 2D MOC. The axial solution is obtained for each pin, and the planar and axial problems are coupled through a transverse leakage. The use of a lower order 1-D solution, which is most often the nodal expansion method (NEM) 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 3D MOC solution can be performed, 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 on the order of fifty (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 NEM generated axial coupling coefficients.

In the simulation of depletion, MPACT can call the ORIGEN code, which is included in the SCALE 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 this study.

5.1.1.2 COBRA-TF

COBRA-TF (Coolant Boiling in Rod Arrays – Two Fluid) [14] is a transient subchannel code based on two-fluid formulation that separates the conservation equations of mass, energy, and momentum to three fields of vapor, 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 and finite-difference numerical scheme, using closure equations to account for inter-phase mass and heat transfer and drag, mechanical losses, inter-channel mixing, and fluid 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 T/H or heat transfer solutions. It allows for full 3D LWR core modeling and has been used extensively for LWR Loss-Of-Coolant Accident (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 several decades by several organizations. CTF is being further improved as part of the VERA multi-physics software package, including:

- Improvements to user-friendliness of the code through creation of a PWR 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,

- 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 and Yao-Hochreiter-Leech grid-heat-transfer enhancement model, and Tong factor for the W-3 CHF correlation,
- Addition of consistent set of steam tables from IAPWS-97 standard,
- Application of 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, and 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 software is used for visualizing three-dimensional data resulting from large full core models and calculations.

5.2 Fuels Performance Tools

The following codes currently are in wide use for analysis of the performance of nuclear fuel by the US commercial industry.

5.2.1 FRAPCON/FRAPTRAN

FRAPCON/FRATRAN is a suite of codes developed by Pacific Northwest National Laboratory (PNNL) for the US NRC for the purposes of performing fuel performance analyses under steady state (FRAPCON) and transient (FRAPTRAN) conditions. FRAPCON [15] is a computer code that calculates 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 the code 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 [16]) is a Fortran language 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 a) heat conduction, b) heat transfer from cladding to coolant, c) elastic-plastic fuel and cladding deformation, d) cladding oxidation, e) fission gas release, and f) fuel rod gas pressure.

5.2.2 FALCON

The Fuel Analysis and Licensing Code—New (FALCON) is a state-of-the-art LWR fuel performance analysis and modeling code [17]. The code was developed by EPRI and has been validated to high fuel burnup conditions. It is based on a robust 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.

5.2.3 BISON

BISON [18] is a finite element-based nuclear fuel performance code applicable to a variety of fuel forms including light water reactor fuel rods, TRISO particle fuel, and metallic rod and plate fuel. It is an advanced fuel performance code being developed at INL and offers distinctive advantages over FRAPCON/FRAPTRAN such as 3D simulation capability, etc. BISON solves the fully-coupled equations of thermomechanics and species diffusion, for either 1D spherical, 2D axisymmetric or 3D geometries. Fuel models are included to describe temperature and burnup dependent thermal properties, fission product swelling, densification, thermal and irradiation creep, fracture, and fission gas production and release. Plasticity, irradiation growth, and thermal and irradiation creep models are implemented for clad materials. Models 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 fully-coupled multiscale fuel performance capability. BISON is based on the MOOSE framework and can therefore efficiently solve problems using standard workstations or very large high-performance computers. BISON is currently being validated against a wide variety of integral light water reactor fuel rod experiments.

5.3 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 analysis of AOO / DBA / BDBA events at commercial nuclear power plants (NPPs) operating in the United States. Additional summary descriptions of these codes are available in reference [19]. Reference [19] provides basic descriptions of the code capabilities, computational structure, and available documentation. The descriptions in [19] also detail the range of applicability for each code as well as the limitations and precautions relevant to its use.

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

5.3.1 RELAP5-3D

The RELAP5-3D [20] code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. Specific applications of the code have included simulations of transients in light water reactor (LWR) systems such as loss of coolant, anticipated transients without scram (ATWS), and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip. RELAP5-3D, the latest in the series of RELAP5 codes, is a highly generic code that, in addition to calculating the behavior of the reactor coolant system during a transient, can be used for simulation of 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 loss-of-coolant accidents (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 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.

5.3.2 TRACE

The TRAC/RELAP Advanced Computational Engine (TRACE) [21] is a modernized 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 currently is 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, the TRACE best-estimate thermal-hydraulics code system has been coupled with the Purdue Advanced Reactor Core Simulator (PARCS). The coupling of TRACE and PARCS takes into account the interaction of the plant dynamic thermal-hydraulic performance and the neutron kinetics for the reactor core.

5.3.3 MAAP

The Modular Accident Analysis Program (MAAP) [22] is a computational code developed by the Electric Power Research Institute (EPRI). As an EPRI developed code it is only available to EPRI members; however all US operated NPPs (as well as a large number of international NPPs) are EPRI members which utilize the MAAP code for the conduct of severe accident analyses. The code simulates the response of light water reactors (LWRs) during severe accidents. 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:

- prediction of the timing of key events (e.g., core uncovery, core damage, core relocation to the lower plenum, and vessel failure),
- evaluation of the influence of mitigation systems and operator actions,
- prediction of the magnitude and timing of fission product releases, and
- evaluation of 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 core damage frequency (CDF) and large early release frequency (LERF). MAAP is an integral systems analysis code that treats the full spectrum of important phenomena that could occur during an LWR accident.

5.3.4 MELCOR

The Methods for Estimation of Leakages and Consequences of Releases (MELCOR) [23] is a computational code developed by the Sandia National Laboratory (SNL) for the US Nuclear Regulatory Commission (NRC), US Department of Energy (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 also simulates the response of LWRs during severe accidents. 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, and
- 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.

5.3.5 RELAP-7

The RELAP-7 [24] (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 the INL's modern scientific software development framework MOOSE (Multi-Physics Object Oriented Simulation Environment). 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. The end result will be 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. RELAP-7 uses modern numerical methods, which allow implicit time integration, second-order schemes in both time and space, and strongly coupled multi-physics.

5.4 Risk Assessment Tools

The following codes represent the current suite of mature as well as advanced tools still being developed to perform probabilistic risk assessments (PRAs) of commercial nuclear power plants (NPPs) operating in the United States.

5.4.1 SAPHIRE

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) [25] is a software application developed for performing a complete probabilistic risk assessment (PRA) using a personal computer (PC) running the Microsoft Windows operating system. SAPHIRE is developed by the Idaho National Laboratory (INL) for the U.S. Nuclear Regulatory Commission (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 nuclear power plant PRAs, SAPHIRE can be used to model a plant's response to initiating 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 could 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).

5.4.2 CAFTA

The Computer Aided Fault Tree Analysis System (CAFTA) [26] is a computer program developed by EPRI used 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 (R&R) Workstation Suite of risk assessment tools to permit rapid and comprehensive risk assessments. Since CAFTA was developed by EPRI it is 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. 10CFR50 Appendix B and ISO 9001 quality assurance requirements.

5.4.3 EMRALD

EMRALD [27] is a dynamic Probabilistic Risk Analysis (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, 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.

5.4.4 RAVEN

RAVEN [28] 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 used to perform plant safety analyses. The provided Application Programming Interfaces (APIs) 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 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.

5.5 Integration Tools

5.5.1 LOTUS

LOTUS [29] is a multi-physics best estimate plus uncertainty (MP-BEPU) analysis framework being developed at INL. It established the automation interfaces among the five disciplines depicted in Figure 4 such that uncertainties can be propagated consistently in multi-physics simulations. These five 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 system analysis, 3) 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, 4) Uncertainty Quantification and Risk Assessment which focuses on establishing the interfaces to enable combined deterministic and probabilistic analysis, and 5) Core Design Optimization which focuses on developing a core design optimization tool that can perform in-core and out-of-core design optimization.

LOTUS integrates the existing computer codes as well as the advanced computer codes still being developed under various DOE programs to provide feedback and guide development of advanced tools. Regardless the specific codes used to model the physics involved, 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 beside the small pool of experts within the specific organization or even groups within the organization that develops such 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 carries a heavy cost of bookkeeping and regulatory review.

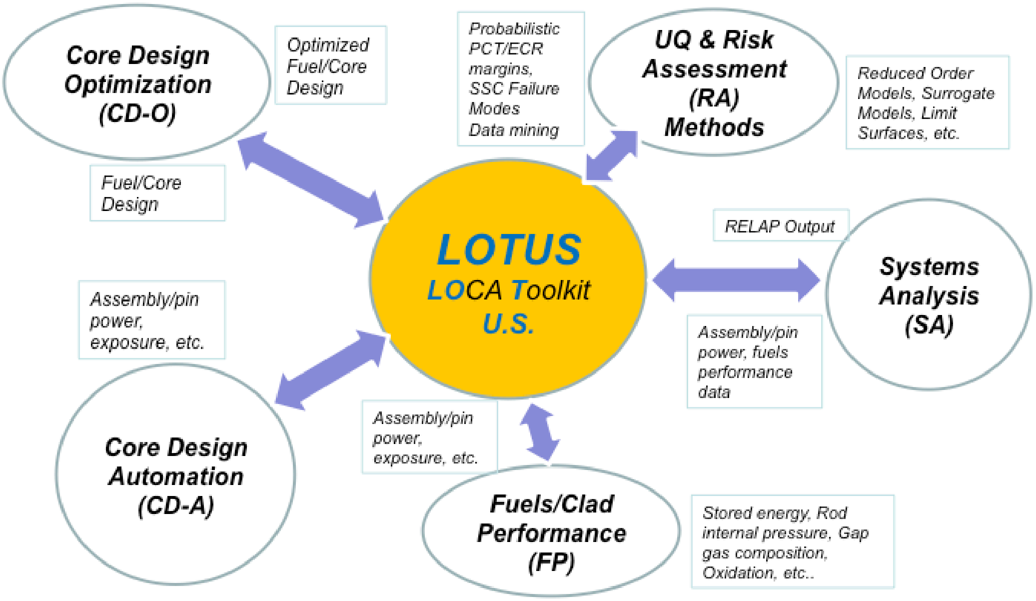

Figure 4. Schematic Illustration of LOTUS.

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 on managing the data stream at a system level. LOTUS is essentially a workflow engine with capability to drive physics simulators, 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 use within their established codes and methods. Therefore, it could potentially become the engine for license-grade methodologies. In other words, it is possible that LOTUS technology could be advanced in the future to a level of fidelity and maturity that it could be used for some licensing or regulatory situations.

6. PROJECT SCHEDULE

This project plan is intended to be conducted in collaboration with work being performed as part of broader industry efforts to develop, mature, license, and deploy ATF and Enhanced Resilient Plant Systems. In particular, the project activities and schedule are developed to be performed collaboratively with industry efforts being led by EPRI with specific attention to conducting the identified research and development activities in a manner that efficiently and cost-effectively utilizes resources. There are different ATF concepts being developed and studied, both in fuel and cladding materials that have different timeframes for licensing and deployment. The different ATF concepts can be categorically grouped into near-term and long-term concepts, as shown in Table 1, with a full reload of near-term ATF in NPPs in 2022 to 2025 timeframe and long-term ATF to be deployed at a later time. To coordinate with the industry to accelerate the development, licensing and deployment of ATF, the Enhanced Resilient Plant Systems Project will be developed in two phases:

6.1 Phase I – Near Term ATF Concepts Centric

In this phase, the work will focus on the near term concepts of ATF as well as the optimization of FLEX equipment and development / possible deployment of new passive cooling systems. The following combinations of ATF concepts will be evaluated in this phase,

- UO_2 fuel / Zircaloy clad (reference case)
- UO_2 fuel / FeCrAl clad
- UO_2 fuel / Cr-coated Zircaloy clad
- Cr-doped UO_2 fuel / Cr-coated Zircaloy clad

Since the NPPs have purchased the FLEX equipment and developed processes and procedures for their use to enhance plants' resiliency to cope with a BDBA, how to optimize the utilization and take credit of the FLEX equipment, especially in the human reliability area, will be studied in this project in the context of near-term ATF development to provide the quantification of the enhanced resiliency of NPPs with the introduction of these new technologies. In parallel, augmented cooling systems such as extending the reactor core isolation cooling (RCIC) system operating band, etc. or installation of new passive cooling systems, will be studied to remove decay heat from the reactor and to improve its safety. It should be noted that incorporation of new passive cooling systems may require plant modifications or retrofitting of plant equipment. Because the design and verification testing of these systems is at a relatively early stage, this activity will be carried over into the Phase II of this project.

6.2 Phase II – Long Term ATF Concepts Centric

In conjunction with the near term ATF concepts, for a period of about three years, the advanced phase of this project will be executed (FY2021-23). The duration and timeline associated with this phase is, in part, dependent on the execution and lessons learned from the near term ATF concepts evaluation phase, and availability and maturity of the long term ATF concepts in development today. The combinations of long term ATF concepts are the following:

- U_3Si_2 fuel / Zircaloy clad (not an ATF concept but will serve as a comparison case for U_3Si_2 fuel)
- UO_2 fuel / SiC clad
- U_3Si_2 fuel / SiC clad
- Metallic fuel
- Metallic fuel with Cr coating

The impact of FLEX and the incorporation of passive cooling systems also will be evaluated in this phase in combination of long term ATF concepts.

Table 1. Near-Term and Long-Term ATF Concepts

	Cladding		Fuel	
Near-term	Coated Cladding	FeCrAl	Doped UO_2	
Long-term	SiC		U_3Si_2	Metallic Fuel

6.3 Project Scope, Approach and Schedule

The scope, approach and schedule of this project are summarized in this section.

6.3.1 Project Scope

The following is a summary of the accident scenarios to be analyzed for the proposed plant changes.

I. Operational Enhancements
   a. Flexible plant operations (load follow capability)
   b. Increased fuel enrichment (greater than current 5 w/o limit)
   c. Enhanced fuel discharge burnup (greater than current 62 GWd/MTU limit)
   d. Extended operating cycle length (in particular extend PWR cycles to 24 months)

II. Anticipated Operational Occurrences
   a. Loss of Feedwater
   b. SGTR (PWR)
   c. Turbine load rejection without bypass
   d. Trip of all Reactor Recirculation Pumps (BWR)

III. Design Basis Accidents
   a. LB-LOCA
   b. RIA
      i. PWR control rod assembly ejection
      ii. BWR control rod drop
   c. Loss of Flow
      i. PWR locked rotor
      ii. BWR recirculation pump shaft seizure

IV. Beyond Design Basis Accidents
   a. Assess integrated benefits of ATF + other strategies
      i. FLEX
      ii. Passive cooling systems
   b. Events
      i. LOOP / SBO events
      ii. Loss of feedwater with transition to feed and bleed cooling (PWR)

6.3.2 Project Schedule Outline

Only the project schedule for the evaluation of near term ATF concepts is developed in this document (see Table 2). The project schedule for the evaluation of the long term concepts of ATF will be developed in the future. It should be noted that this schedule reflects current industry objectives and priorities for the licensing and deployment of the near term ATF concepts (coated cladding / doped fuel pellets). This schedule is anticipated to evolve as additional information is obtained and interactions between industry, DOE, and NRC occur.

Table 2. Timeline for the Evaluation of Near-Term ATF Concepts

	FY18				FY19				FY20			
	Q1	Q2	Q3	Q4	Q1	Q2	Q3	Q4	Q1	Q2	Q3	Q4
IREM Development & Optimization												
BDBA												
SBO + FLEX												
SBO+FLEX+Passive Cooling												
DBA												
LB-LOCA												
PWR Locked Rotor												
BWR Recirculation Pump Shaft Seizure												
PWR Rod Ejection (RIA)												
BWR Rod Drop												
Fuel Handling Accident for Extended BU and Enhanced Enrichment												
AOO												
Loss of Feedwater (PWR)												
Loss of Feedwater (BWR)												
Steam Generator Tube Rupture												
Turbine Load Rejection w/o Bypass (PWR)												
Turbine Load Rejection w/o Bypass (BWR)												
Inadvertent RCS Blowdown (PWR)												
Inadvertent RCS Blowdown (BWR)												

BWR Recirculation Pumps Trip												
Normal Operations												
Extended Operating Cycles												
Burnup Extension												
Enhanced Enrichment												
Load Follow												
FLEX Operations & Credit												
Human Reliability Analysis on FLEX												
FLEX Scenario Analysis & Optimization												
FLEX Industry Application												
Industry Cross-Walk												
Enhanced/Passive Cooling Systems												

7. ANTICIPATED OUTCOMES

The intrinsic value of successful R&D in this area is expected to be significant. The integrated risk evaluation approach developed in this project has the potential to accelerate the development and deployment of ATF and Enhanced Resilient Plant Systems in order to simultaneously enhance the safety and reduce the operating costs of NPPs. The integrated risk evaluation approach will allow a comprehensive, integrated, and risk-informed evaluation of plant components/systems that in combination with ATF would achieve desired Coping Time increase (i.e. additional 4, 6, and 8 hours of coping time) and plant risk reduction in CDF & LERF to demonstrate Enhanced Plant Resiliency. Demonstrated enhanced plant resiliency would allow the safety, economic, and regulatory benefits to be investigated.

Once the integrated evaluation approach achieves its objectives, it can potentially outweigh some of the costs associated with potential plant modifications that would be used to implement ATF and Enhanced Resilient Plant Systems, therefore keeping the US LWR fleet competitive with other sources of energy. A more informed analysis with respect to actual margins and risks available in an operating plant can potentially reduce extensive (and costly) iterations between licensees and regulators when dealing with rule compliance issues. Ultimately more information will yield a higher degree of safety and improved economics.

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