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Development of the Expanded Materials Degradation Assessment for Prioritizing Research in Support of Subsequent License Renewal

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Introduction and Motivation

Nuclear reactors present a very harsh environment for components service. Components within a reactor core must tolerate high-temperature water, stress, vibration, and an intense neutron field. Degradation of materials in this environment can lead to challenges in required performance. Materials degradation phenomena within a nuclear power plant are very complex. There are many different types of materials that make up different components: over 25 different metal alloys can be found within the primary and secondary systems, not to mention the containment vessel and internal concrete structures, instrumentation and controls, and other support systems. When this diverse set of materials is placed in the complex and harsh environments of an operating reactor and is coupled with varying types of loadings (from changes in reactor power levels or internal/external events), degradation over an extended life is indeed quite complicated. Routine surveillance and component replacement can mitigate these factors, although failures can still occur. However, while all components can, in theory, be replaced, it may not be practical or economically favorable. Therefore, understanding, controlling, and mitigating materials degradation processes are key priorities for extending reactor operating lifetimes.

According to the provisions in Title 10 of the Code of Federal Regulations, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," plant operating licenses may be renewed for periods of 20 years after the initial 40-year licensing term. The licensee must provide reasonable assurance that plant activities will be conducted within the plant's current licensing basis during the extended operating period. An integral part of ensuring that the nuclear power plant can continue to operate safely is demonstrating that the effects of aging-related degradation on systems, structures, and components are well understood and can be adequately managed. The majority of U.S. nuclear power plants have received a first license renewal to operate for up to 60 years, and over a third of the plants have entered the extended operating period. At present, industry is considering the feasibility of pursuing subsequent license renewal to operate from 60 to 80 years. While applications for subsequent license renewal may not be prepared for several years, the U.S. Nuclear Regulatory Commission (NRC), U.S. Department of Energy (DOE), and the nuclear industry all have an interest in proactively identifying issues that may affect the ability of nuclear power plants to operate for up to 80 years.

Given the importance of materials, issues for a subsequent license renewal and the extreme complexity and diversity of the issues, it is imperative that an objective approach be used for prioritizing research needs. To address this need, NRC and DOE (through the Light Water Reactor Sustainability [LWRS] Program) collaborated on a systematic assessment that builds upon previous work documented in the NRC report, "Expert Panel Report on Proactive Materials Degradation Assessment" (NUREG/CR-6923), referred to as the PMDA report. In this study, NRC conducted a comprehensive evaluation of potential aging-related degradation modes for core internal components, as well as primary, secondary, and some tertiary piping systems, considering operation up to 40 years. The PMDA report has been a very valuable resource, supporting NRC staff evaluations of licensees' aging management programs and allowing for prioritization of research needs.

The Expanded Materials Degradation Assessment (EMDA) (NUREG/CR-7153, "Expanded Materials Degradation

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Assessment") builds on the PMDA and includes potential degradation scenarios for operation up to 80 years. While the PMDA mainly addresses primary system and some secondary system components, the EMDA covers a broader range of component groups, including (1) core internals and piping systems, (2) reactor pressure vessels (RPVs), (3) concrete and civil structures, and (4) cables and cable systems. To conduct the assessment and prepare the EMDA report, an expert panel for each of the four component groups was assembled. The panels had 8 to 10 members, including representatives from NRC, DOE national laboratories, industry, independent consultants, and international organizations. Each panel was responsible for preparing a technical background volume and conducting a Phenomena Identification Ranking Technique (PIRT) scoring assessment. The technical background chapters in each volume summarize the current state of knowledge concerning degradation of the component group and highlight technical issues deemed to be the most important for subsequent license renewal. In addition, the key gaps identified for each material system are presented.

The discussion presented in this article only covers the results at the highest level. A detailed analysis is described in the complete set of documents:

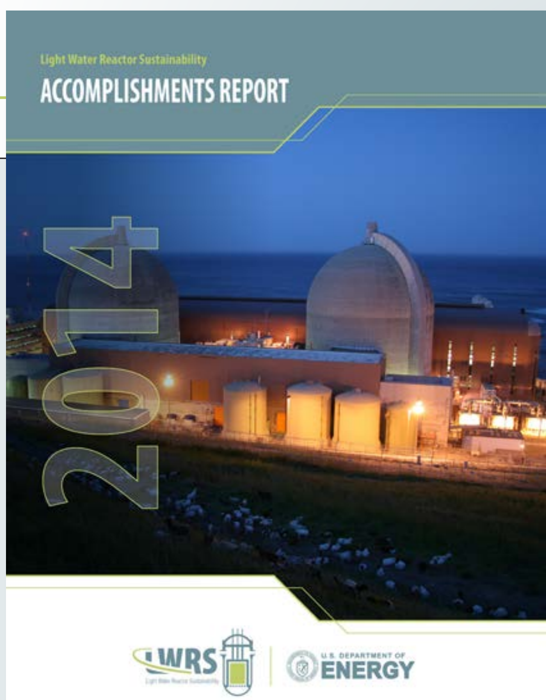
- Volume 1, Expanded Materials Degradation Assessment (EMDA): Executive Summary of EMDA Process and Results
- Volume 2, Expanded Materials Degradation Assessment (EMDA): Aging of Core Internals and Piping Systems
- Volume 3, Expanded Materials Degradation Assessment (EMDA): Aging of Reactor Pressure Vessels
- Volume 4, Expanded Materials Degradation Assessment (EMDA): Aging of Concrete and Civil Structures
- Volume 5, Expanded Materials Degradation Assessment (EMDA): Aging of Cables and Cable Systems.

These documents can be found at:

http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7153/.

Organization and Process

The expert elicitation process used for the EMDA is based on the same PIRT process that was employed for previous assessments. This process has been used in many industries for ranking and prioritizing any number of issues. The PIRT process provides a systematic means of obtaining information from experts and involves generating lists (tables) of phenomena where "phenomena" can refer to a particular reactor condition, a physical or engineering approximation, a reactor component or parameter, or anything else that might influence some relevant figure-of-merit that is related to reactor safety. The process usually involves ranking of these phenomena using a series of scoring criteria.



2014 Light Water Reactor Sustainability Accomplishments Report

The 2014 LWRS Program Accomplishments Report covers selected highlights from the three research pathways in the LWRS Program: Materials Aging and Degradation, Risk-Informed Safety Margin Characterization, and Advanced Instrumentation, Information, and Control Systems Technologies, as well as a look-ahead at planned activities for 2015. If you have any questions about the information in the report, or about the LWRS Program, please contact Richard A. Reister (Federal Program Manager), Kathryn A. McCarthy (Technical Integration Office Director), or the respective research pathway leader (noted on pages 26 and 27 of the report), or visit the LWRS Program website (www.inl.gov/lwrs).

The results of the scoring can be assembled to lead to a quantitative ranking of issues or needs.

Each PIRT application has been unique in some respect and the current project is unique in its application. The current PIRT can be described in terms of several key steps. These are described for the generic process below, although each panel made minor adjustments based on the needs of that material system and the operational environment and expected interactions. These adjustments are also described below.

As previously noted, expert panels were assembled to evaluate each of the four main component groups. To ensure a diverse set of background and expertise, each panel was assembled to ideally include the following:

- At least one member from regulatory bodies, including NRC
- At least two members representing industry (e.g., Electric Power Research Institute and vendors)
- At least one member from DOE national laboratories
- At least one member from academia
- At least two members from outside the United States.

Members from non-nuclear fields were also selected for the concrete and civil structure panel. NRC and DOE cooperatively selected and assembled the various panels.

The EMDA report volume for each component group consists of a technical background assessment to summarize the current state of knowledge concerning the relevant degradation scenarios, as well as the PIRT scoring and analysis. Ideally, the technical background assessments provide the context and rationale for which scenarios were scored and how they were ranked. For the core internals and piping systems volume, the existing PMDA report was used as a starting point for identifying important degradation scenarios and additional discussion focused on the potential changes that might be experienced during subsequent operating periods. For the other volumes, given that there was no pre-existing PIRT, the latest technical literature was reviewed and experts used their judgment to identify the important degradation scenarios. Generally, one panel member was assigned to write each chapter of the technical background assessment, which could focus on a particular material or degradation mode, after which the chapters were peer reviewed by the entire panel. Subsequent discussion amongst the entire panel was also used to identify key themes, and revisions to the technical background assessments were made accordingly. These assessments are listed as the opening chapters of each volume in the EMDA. It is important to note that these background assessments are not intended to be all-encompassing primers on particular degradation modes or material systems. Detailed information and background assessments exist in other publications and it was beyond the scope of this project to reproduce them. Rather, the presented discussions intended to introduce the subject and context for evaluation of key modes of degradation for subsequent operating periods. The reader is referred to the publications listed in the background chapters of each volume for more in-depth technical information.

Based on the input from the technical background volume, the panels then developed a PIRT matrix with a list of degradation scenarios to score. A degradation scenario generally encompasses a particular material, system, component, or subcomponent (depending on the categorization scheme devised by the panel); the environmental condition to which that material is exposed; and the degradation mode that material may experience based on laboratory and operational data. It was recognized that these data do not exist for reactor operational periods beyond 40 years, thus posing a considerable challenge for the expert panels to extrapolate reactor operation for greater than 60 years. Some materials are used in different components and experience different environments or may experience multiple degradation modes in a single location. Each material, environment, and degradation mode was scored as a distinct scenario. The number of degradation scenarios varied widely by component group, from less than 50 for the cables and cable systems to over a 1,000 for the core internals and piping systems.

After the scoring matrix was developed, panelists independently scored the degradation scenarios in three categories that were originally used in the PMDA report: (1) susceptibility, (2) confidence, and (3) knowledge. The susceptibility score rated the likelihood that degradation will occur on a scale from 0 (not considered to be an issue) to 3 (demonstrated, compelling evidence for occurrence or multiple plant observations). The knowledge score rated the expert's current belief of how adequately the relevant dependencies have been quantified through laboratory studies and/or operating experience on a scale from 1 (poor understanding, little and/or low-confidence data) to 3 (extensive, consistent data covering all dependencies relevant to the component). Finally, the confidence score measured the expert's personal confidence in his or her judgment of susceptibility on a scale from 1 (low) to 3 (high).

After completion of scoring and identification of "outliers," the panels were reassembled for discussion of the scoring. In most panels, this was done in a face-to-face meeting, but this was not required in all cases. During this discussion, each degradation mode and related scoring was discussed, with the "outliers" being of highest priority.

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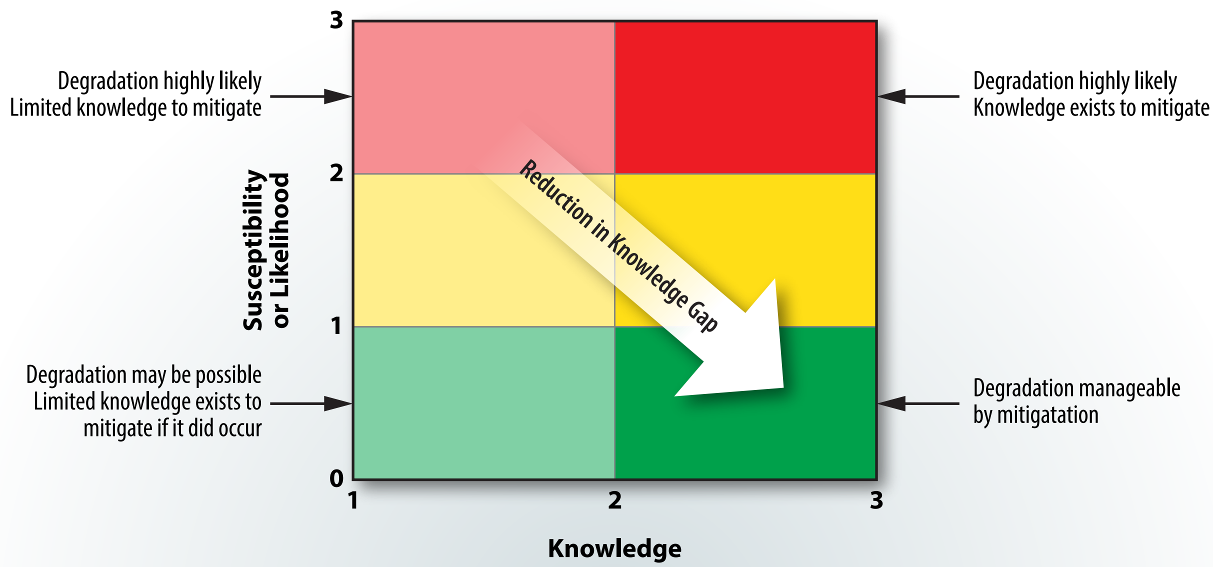

Figure 1. Schematic illustrating the combinations of susceptibility and knowledge scores, suggesting various life management responses.

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In these discussions, the scoring panelist presented rationale for any scores that differed from the average. The objective was not to develop a consensus score or force conformity among the panelists. The primary goal of this discussion was to foster debate and exchange differing points of view. This debate and discussion among panelists was an important part of the process to ensure all points of view were considered, including consideration of any new information on the subject area that was not previously considered and accounted for in the final scoring.

After compiling any changes in scoring following this debate, the PIRT scoring was tabulated to determine the relative needs and priorities. In this process, the average susceptibility and average knowledge scores were plotted versus each other on a simple plot. An example plot of knowledge versus susceptibility is shown in Figure 1. The left side of the plot with the lighter shading is indicative of low knowledge, while the darker shading on the right side of the plot is indicative of high knowledge. The labeled areas in the corners of the plot indicate the high knowledge, low susceptibility; high knowledge, high susceptibility; and low knowledge, high susceptibility areas discussed above. Moving from upper right to lower left can be accomplished via additional research and development to understand and predict key forms of degradation. The different domains of these plots highlight the key areas of concern, including the following:

- Low knowledge, high susceptibility degradation modes are indicated by the pink shading in Figure 1 and represent modes of degradation that could be detrimental to service with high susceptibility scores (greater than 2) and low knowledge scores (less than 2). These scores indicate gaps in understanding for degradation modes that have been demonstrated in service.
- Low knowledge and moderate susceptibility also indicate gaps in knowledge, although with lower consequences. These scoring regions are useful in identifying potential knowledge gaps and areas requiring further research into mechanisms and underlying causes to predict occurrence.
- High knowledge, high susceptibility degradation modes are shown in red in Figure 1 and represent areas that could be detrimental to service with high susceptibility scores (greater than 2) and high knowledge scores (greater than 2). These modes of degradation are well understood and have likely been observed in service. While there may be some mechanistic understanding of the underlying causes, re-confirmation for extended service and research into mitigation or detection technologies may be warranted.

- High knowledge, low susceptibility degradation modes (dark green in Figure 1) are those that are relatively well understood and of low consequence to service with low susceptibility scores (less than 1) and high knowledge scores (greater than 2). These modes of degradation are adequately understood and may have been observed in service. Mitigation and maintenance can currently manage this form of degradation. Research on these modes of degradation is a lower priority.

Of course, other combinations of knowledge and susceptibility are possible and fit between the cases listed above in terms of priority.

Finally, the results of the PIRT scoring were compared to the background technical chapters to ensure all important modes of degradation and points were captured. Revisions were then made to the supporting chapters and analysis to ensure adequate discussion of key topics, outcomes, and underlying causes. Thus, the technical basis information for conducting the PIRT assessment and the results of the PIRT were re-iterated to ensure that coverage and consistency is maintained in the various PIRT subject areas.

Given the diversity of the materials and systems considered by each panel, some minor variations in the process described above were implemented by each panel. These changes and their motivation are listed specifically in subsequent sections of the applicable volume and in the appropriate material system volume.

Results

For the reactor core and primary systems, several key issues have been identified. Thermo-mechanical considerations such as aging and fatigue must be examined. Irradiation-induced processes must also be considered for higher fluences, particularly the influence of radiation-induced segregation, swelling, and/or precipitation on embrittlement. Corrosion takes many forms within the reactor core and piping systems, although irradiation-assisted stress corrosion cracking and primary water stress corrosion cracking are of high interest in extended life scenarios. Research in these areas can build on other ongoing programs in the light water reactor industry, as well as other reactor materials programs (such as fusion and fast reactors) to help resolve these issues for extended light water reactor life. In the secondary systems, corrosion is extremely complex. Understanding the various modes of corrosion and identifying mitigation strategies is an important step for long-term service.

For RPVs, a number of significant issues have been identified for future research. Relatively sparse or nonexistent data at high fluences and for long radiation exposure (duration) create large uncertainties for embrittlement predictions. The use of test reactors at high fluxes to obtain high fluence data is not the most direct representation of the low flux conditions in RPVs. Irradiation-induced phase transformations have been observed at very high fluences for both commercial and model alloys and, particularly, in high nickel weldments. These transformations may not lead to observable changes in performance until very late in service and, as such, are commonly called "late-blooming phases." Additional experimental data are needed in the high fluence regime, where they are expected to accurately assess their potential in second license renewal periods. Other discussed issues include specific needs regarding application of the fracture toughness master curve, data on long-term thermal aging, attenuation of embrittlement through the RPV wall, and development of an embrittlement trend curve, based on fracture toughness measurements.

Concrete structures may also suffer undesirable changes in properties with time, including adverse performance of its cement paste matrix or aggregate constituents under environmental influences (e.g., physical or chemical attack). Changes to embedded steel reinforcement, as well as its interaction with concrete, can also be detrimental to concrete's service life. Aging effects can be exacerbated if improper concrete specifications were used at the time of construction. A number of areas of research would help assess the long-term integrity of the reactor's concrete structures.

Cable and cable insulation systems play an important role in the safety and operation of a nuclear power plant. Degradation of polymer insulation due to the combined effects of mechanical stress, elevated temperature, irradiation, and high humidity environments (or complete submergence) has been observed, although there may be knowledge gaps for reactor long-term operation.

Summary

The EMDA activity required a considerable effort from LWRS Program research personnel and NRC staff and an enormous amount of input from the expert panelists. Their input has been analyzed to assess the level of knowledge and susceptibility for thousands of different material/environment/degradation combinations. This approach has helped identify the most important potential knowledge gaps for subsequent license renewals. This has provided tremendous value to the LWRS Program in prioritizing research needs.

Multiscale Approach to Reactor Pressure Vessel Integrity Assessment

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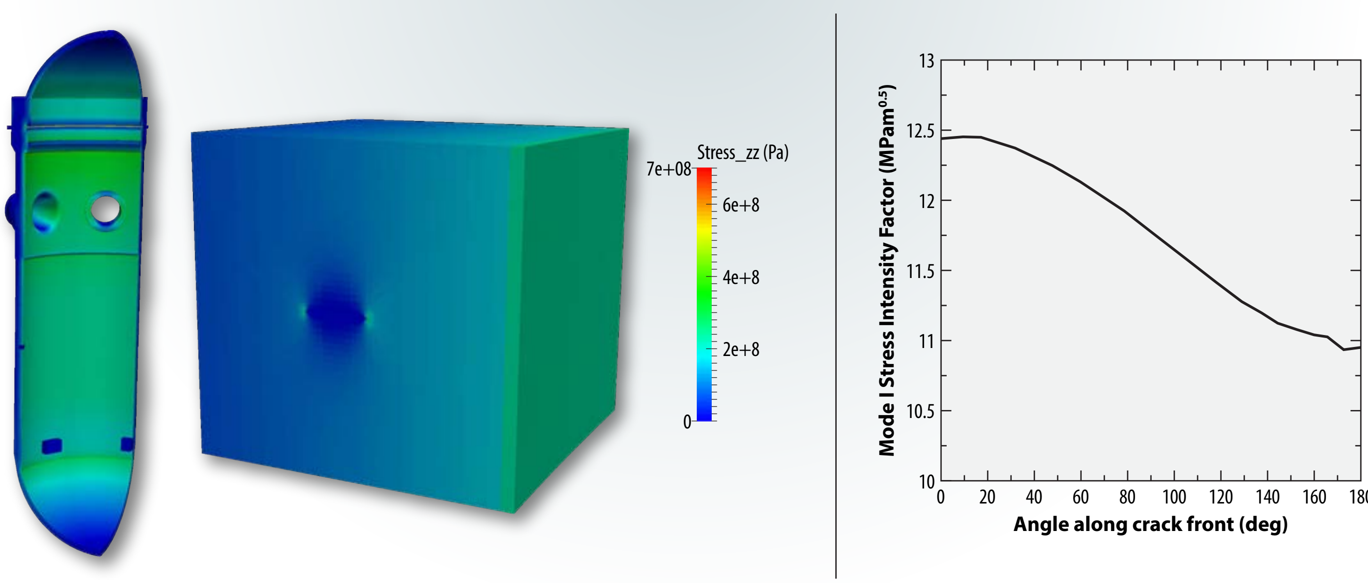
Risk-Informed Safety Margin Characterization Pathway

Introduction

The effects of age-related degradation mechanisms on critical systems, structures, and components will play a key role in decisions related to the long-term operation of nuclear power plants, including license renewal decisions. The Grizzly simulation code is being developed under the Risk-Informed Safety Margin Characterization (RISMC) Pathway in collaboration with the Materials Aging and Degradation Pathway. Grizzly is intended to provide a multiphysics, multiscale computational platform to simulate both the aging processes and the effects of aging on the ability of these components to safely perform their intended tasks. Ultimately, Grizzly will address a wide range of aging issues in key components, including the RPV, core internals, containments, piping, concrete structures, and cables. The RPV has been chosen as the first application for Grizzly because of its critical role in ensuring safe operation of nuclear power plants, and because of the extreme difficulty involved in replacing or repairing the RPV. Significant developments have been made toward the goal of developing a tool that can assess RPV performance during long-term operation.

RPVs must maintain integrity during both normal and off-normal operating conditions. One such off-normal operating condition of concern involves pressurized thermal shock loading events, in which an RPV is rapidly

Figure 2. Global model of a reactor pressure vessel under off-normal conditions (left) and submodel of material in the vicinity of a postulated embedded circular crack (middle) used to calculate mechanical response and stress intensity factor (right) along the front of that crack.



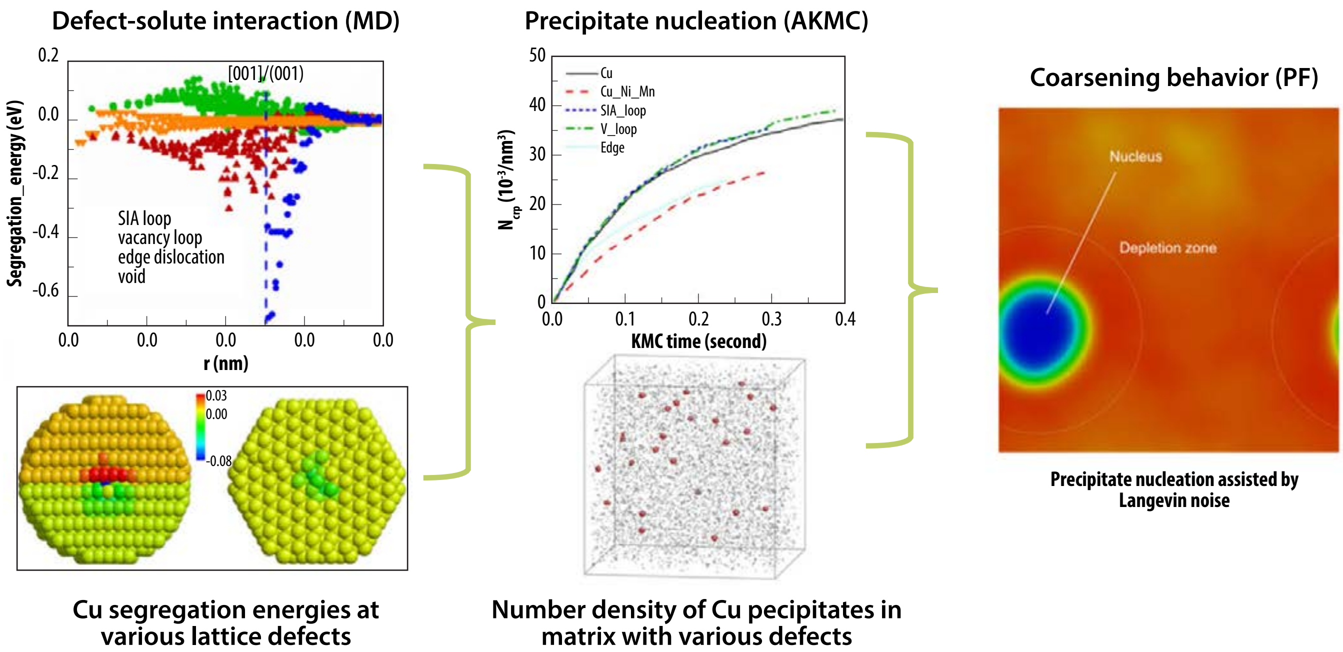


Figure 3. Schematic of the multiscale modeling approach for solute precipitation in reactor pressure vessel steels.

cooled and depressurized, and then potentially rapidly repressurized. If an RPV subjected to such loading was to contain a large pre-existing crack, the crack could potentially propagate through the wall of the RPV (Odette and Lucas 2001, Williams et al. 2012). This is more likely to occur if the material is in a brittle state, which occurs at lower temperatures and due to aging mechanisms resulting from long-term exposure to irradiation and elevated temperatures. Because of this, ensuring that RPVs will not fail due to pressurized thermal shock is an important consideration for life extension beyond 60 years.

The susceptibility of aged RPVs to fracture is a function both of the material aging and of the loading conditions imposed on the RPV. Capabilities are being developed to model both the aging processes and response to loading. A variety of modeling techniques, summarized here, are used to simulate response to aging mechanisms and loading at all applicable scales.

Thermomechanical Response and Fracture Modeling

To determine whether crack growth is likely to occur in an RPV, a modeling tool to address this issue must be able to capture the global thermal and mechanical response of the RPV under off-normal conditions, as well as calculate the stress intensity along the fronts of flaws modeled three-dimensionally (3-D). Grizzly is built on Idaho National Laboratory's Multiphysics Object-Oriented Simulation Environment (MOOSE) framework, which provides the ability to solve tightly coupled multiphysics problems using the finite element method. Grizzly uses this capability to solve for the response of the RPV when subjected to thermal and pressure boundary conditions that represent a pressurized thermal shock scenario.

Detailed 3-D models of the material surrounding postulated flaws have been developed and analyzed with Grizzly, which can interpolate solutions from global models of the entire structure to define boundary conditions for submodels of flaw regions. Figure 2 demonstrates this capability, showing the stress response of a global 3-D RPV model subjected to pressurized thermal shock loading conditions, and the corresponding stress field calculated in a submodel with an embedded circular flaw aligned in a plane normal to the axis of rotation of the RPV. Domain integral techniques are used to calculate the stress intensity along the 3-D crack front.

Grizzly provides the ability to calculate the extent of material embrittlement, which is manifested as a shift in the ductile-to-brittle transition temperature, using the physically based correlations for RPV steel provided by the EONY model (Eason et al. 2013), as well as the fracture toughness of the embrittled material at any position in the model at the current temperature using the fracture master curve methodology (Wallin 1984, ASTM 2014). This can be used for a deterministic assessment of whether crack growth would occur.

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Multiscale Modeling of Material Embrittlement in Reactor Pressure Vessel Steels

To determine whether unstable crack growth is likely to occur in an embrittled RPV, it is critical to have a reliable and fast-running model to predict the extent of embrittlement. The correlations of the EONY model are based on extensive experimental data, are straightforward to evaluate, and provide a valuable tool for engineering assessments of RPV integrity. While the EONY model is based on extensive experimental data, that data only covers behavior observed during the lifetime of the current reactor fleet. Because different underlying degradation mechanisms may affect the material during longer-term irradiation, that model cannot be confidently applied to RPVs subjected to longer irradiation times. For this reason, efforts are underway to develop models of the microstructural mechanisms leading to embrittlement. This improved understanding of the microstructure evolution and its effect on engineering properties will be used to develop engineering models to predict the embrittlement of RPV steel subjected to longer irradiation times. These models will be used to inform decisions on long-term operation.

A bottom-up approach is taken here to capture the effect of irradiation and thermal aging on RPV fracture toughness. A series of models are used to capture the microstructural evolution. The information gained from these models is used in crystal plasticity models to capture the hardening effect of the microstructural evolution. This is, in turn, used as input for a fracture model to capture the ductile-to-brittle transition behavior of the material, which is the quantity of interest for engineering analysis. A brief summary of these models is provided here.

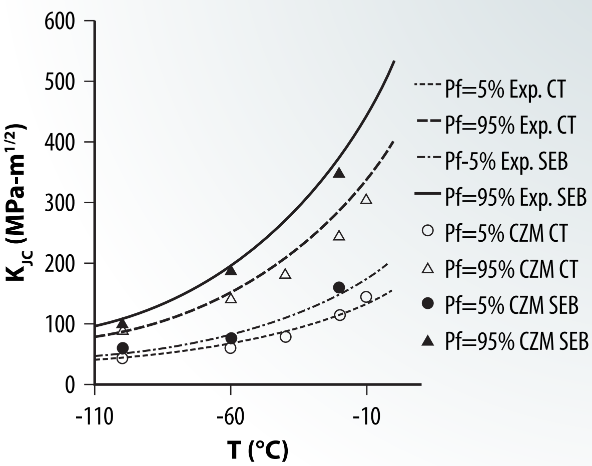
Microstructural Evolution Modeling

From the microstructural point of view, the hardening and embrittlement of RPV steels come from two factors: solute precipitation and radiation damage accumulation (Odette and Lucas 2001). During long-term service at high temperature, the alloying elements precipitate into small clusters such as Cu-rich-precipitates, which are a primary concern for RPV embrittlement. At the same time, neutron irradiation produces lattice defects, which agglomerate into matrix features, including voids and prismatic loops. Both precipitates and matrix features may impede dislocation motion, causing hardening and, thus, embrittlement. The approaches under development in the Grizzly project to model microstructural evolution of RPV steels are summarized here.

Both solute precipitation and radiation damage accumulation involve multiple time and spatial scales. Precipitate sizes are on the order of a nanometer and modeling precipitation requires methods that span time scales ranging from approximately 10-14 seconds (for precipitate nucleation) to decades (for precipitate coarsening).

In this project, a multiscale modeling approach is being developed using molecular dynamics (MD), atomic kinetic

Figure 4. (a) Comparison of fracture toughness from experiments fitted with the master-curve and simulations using the cohesive zone model (CZM) developed in this work for compact tension (CT) and single edge bending (SEB) specimens. (b) Final configuration of a cracked compact tension specimen modeled with the cohesive zone model.



(a)

(b)

Monte-Carlo (AKMC), and phase field (PF) techniques to model precipitate nucleation and coarsening over these time scales (Figure 3). Molecular dynamics simulations are used to identify how solutes interact with other solutes and with defects in the matrix. The atomic kinetic Monte-Carlo model captures nucleation and early stage-growth of precipitates, based on input from the molecular dynamics simulations. Phase field simulations represent the long-term coarsening of precipitates, using results from the atomic kinetic Monte Carlo model to define initial conditions.

A similar multiscale approach is also under development for radiation damage accumulation. Molecular dynamics have been used to better understand the fundamental aspects of defect formation on very short time scales (Zhang et al. 2015). Work is also underway to develop rate theory models to represent long-term radiation damage.

The microstructural evolution is of interest because it results in hardening and embrittlement of the material. Efforts are underway to develop crystal plasticity models to evaluate hardening induced by lattice defects and precipitation.

Modeling of Ductile-to-Brittle Transition of Fracture Toughness

The ultimate goal of this material modeling effort is to determine the fracture toughness of the steel to enable engineering assessments of susceptibility of an RPV to fracture. The fracture toughness of RPV steels reduces significantly with decreasing temperature due to a transition from the ductile–to-brittle mechanism of the fracture. Limited ductile damage precedes unstable cleavage failure in this regime. To determine the fracture toughness at a given temperature, it is critical to characterize this ductile-to-brittle transition.

To characterize the fracture toughness as a function of parameters that can be obtained from microstructure modeling, a unified cohesive fracture model that captures both stable and unstable crack propagation has been developed (Chakraborty and Biner 2014). This model is based on the response of a damage law for a material unit cell and provides a probabilistic description of fracture toughness as a function of temperature. Figure 4(a) shows that there is good agreement between the fracture toughness predicted by this model and experimental data fitted to the fracture master curves; Figure 4(b) shows a fracture specimen modeled using this cohesive zone model in its deformed state. This model is being extended to consider irradiated and sub-sized specimens.

Summary

A general capability to assess the susceptibility of RPVs to fracture is being developed in the Grizzly code. By necessity, modeling the aging phenomena and the response of aged components to loading involves multiple scales and multiple modeling techniques.

The intent is to incrementally develop a complete bottom-up approach to calculate fracture toughness and use that in engineering assessments of RPV integrity. As developments are made to improve the lower-length scale models, they will be used to make incremental refinements to models used at the engineering scale.

Significant progress has already been made in many areas of this modeling effort and the ability of Grizzly to perform a basic deterministic engineering fracture assessment has been demonstrated. During the current fiscal year, extensive work is under way across multiple length scales. The engineering fracture capabilities of Grizzly will be extended to handle a wider variety of conditions and verified against known solutions. Models to represent microstructural damage and precipitation will be further developed, as will methods to transfer data between some of these models. Finally, crystal plasticity models will be developed to calculate hardening due to microstructure evolution. These developments will form the foundation for a bottom-up characterization of the evolution of the engineering properties of irradiated RPV steel and for engineering fracture assessment of RPVs.

References

ASTM E1921-14a, 2014, "Standard Test Method for Determination of Reference Temperature, To, for Ferritic Steels in the Transition Range," ASTM International, West Conshohocken, PA.

Chakraborty, P. and S. Bulent Biner, 2014, "A unified cohesive zone approach to model the ductile to brittle transition of fracture toughness in reactor pressure vessel steels," *Engineering Fracture Mechanics*, 131:194-209.

Eason, E., G. Odette, R. Nanstad, and T. Yamamoto, 2013, "A physically-based correlation of irradiation-induced transition temperature shifts for RPV steels," *Journal of Nuclear Materials*, 433(1-3):240–254.

Odette, G. R. and G. E. Lucas, 2001, "Embrittlement of nuclear reactor pressure vessels," *JOM*, 53(7): 18–22.

Wallin, K., 1984, "The scatter in K_{IC} results," *Engineering Fracture Mechanics*, 19(6):1085–1093.

Williams, P., T. Dickson, and S. Yin, 2012, "Fracture Analysis of Vessels – Oak Ridge, FAVOR, v12.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations," ORNL/TM-2012/567, USNRC Adams number ML13008A015, Oak Ridge National Laboratory, Oak Ridge, TN.

Zhang, Y. F., X. M. Bai, M. R. Tonks, and S. B. Biner, 2015, "Formation of prismatic loops from C15 Laves phase interstitial clusters in body-centered-cubic iron," *Scripta Materialia* 98:5-8.

Value of the Limit Surface for Safety Applications



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Risk-Informed Safety Margin Characterization Pathway

From Classical to Simulation-Based Probabilistic Risk Assessment

One of the common misconceptions about probabilistic risk assessment (PRA) is that the only outcome produced is a probability value for a certain outcome (e.g., core damage or containment breach). Even though such a value has precise use from a regulatory point of view, much more information is actually being generated and this extra information is potentially extremely valuable.

An example of event-tree and fault-tree combinations to determine the risk associated to an initiating event (such a seismic event) is shown in Figure 5. The accident progression is described in the event tree, while system failure models (i.e., reactor scram and reactor cooling systems) are modeled using fault trees.

The final outcome of this analysis is not only a probability value for core damage; it generates the sequences of events (i.e., cut sets or the route through a tree between an initiating event and an end state such as "OK" or "core damage CD") at the fault tree level that lead to core damage. A cut set generated from Figure 5 would be as follows (the red path in Figure 5):

Seismic initiating event + Successful reactor scram (via Back-up) + Failed reactor cooling due to failure of a pump

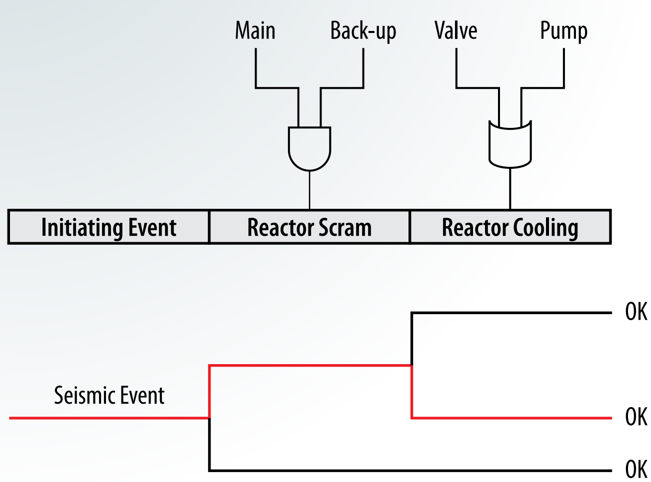
The analyst has the possibility of exploring the sequence of events and ranking them in terms of probability of occurrence, as well as ranking the system and components that are more safety relevant.

Note that in the evaluation of core damage probabilities, the actual simulation of accident evolution is not performed. In addition, the impact of timing and sequencing of events is only considered in an "averaged" fashion. In order to provide a more complete assessment of such events and their implications to plant safety, a series of PRA methodologies that employ system simulator codes were developed and, consequently, the "dynamic" branch of PRA was born. The common ground for all dynamic PRA methodologies is to simulate the actual accident scenarios using simulation-based codes, while a simulation driver changes (either deterministically or stochastically) system configuration (e.g., activation/deactivation of systems/components) throughout the simulation. Dynamic PRA methodologies perform a large number of simulation runs until a completeness of the limit surface (or other metric of interest) is reached. In addition, the analysts are able to identify the impacts of the timing/sequencing of events on safety parameters such as maximum clad temperature or containment pressure.

Obtaining the Limit Surface

One of early issues explored in the RISMC Pathway was: "Is there a metric to measure the impacts of power uprates and life extension?" The RISMC Pathway's answer to that question was the limit surface produced by combining probabilistic and mechanistic modeling approaches. Once the limit surface is obtained, the safety margins can be determined because

Figure 5. Example of event-tree fault-tree architecture.



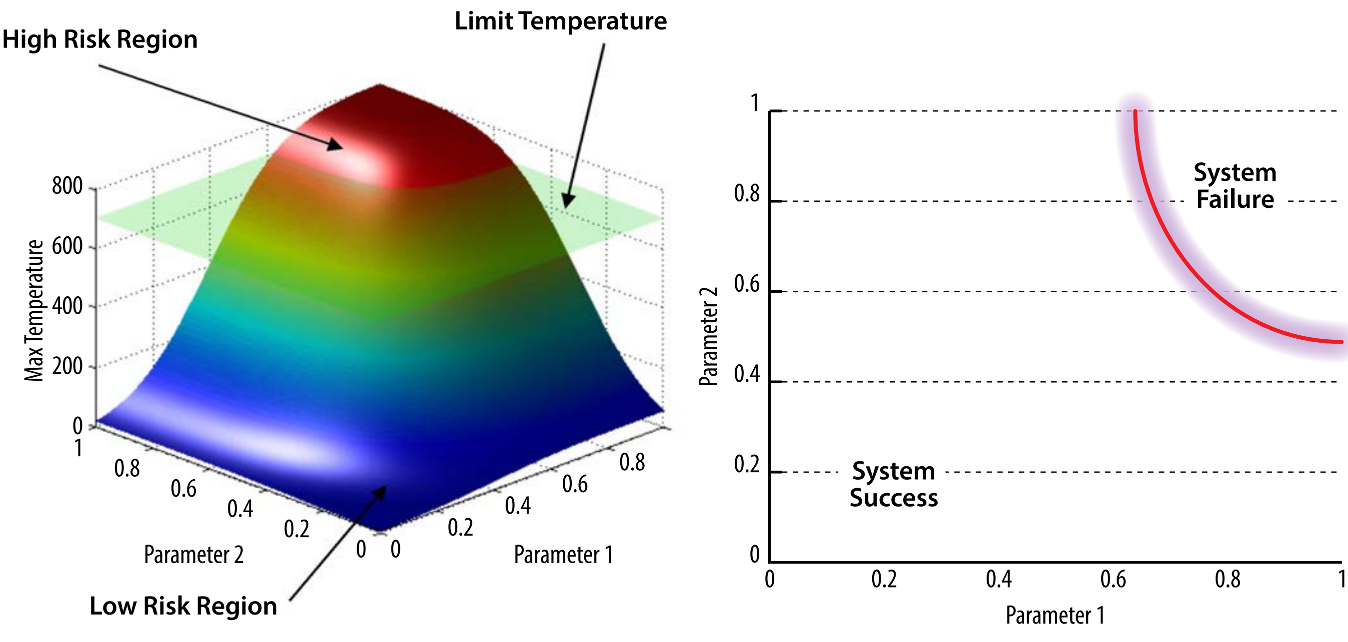


Figure 6. Response surface (left) and limit surface (right).

they represent the "distance" between the normal plant statuses during abnormal conditions and the limit surface.

The following two steps are required to determine the limit surface:

1. Determine how the safety figure of merit (such as maximum clad temperature) changes when the uncertain parameters vary (to obtain a response surface).
2. Find the regions in the space of the uncertain parameters where clad temperature overcomes its limits.

Figure 6 shows a graphical example of limit surface determination in the presence of two uncertain parameters. A series of simulation runs are performed from each pair of values randomly sampled for both parameters. Using a mix of interpolation and regression algorithms, Step 1 is determined from the maximum clad temperature values for each simulation run. In Figure 6 (left), the response surface is pictured with a gradient from blue to red.

The limit surface is determined by intersecting the response surface with the clad limit temperature as shown in Figure 6 (left). This intersection is also shown in Figure 6 (right) and is viewed from top to bottom in Figure 6 (left). In other words, the limit surface provides a way to determine when and how we might exceed key safety thresholds such as a peak clad temperature.

From a safety point of view, the limit surface is analogous to the cut set described earlier for event-tree fault-tree structures. They both represent the minimal condition to reach a failure scenario. However, the limit surface contains a much greater amount of information content because they are simulation based and actual plant dynamics (e.g., timing, operational rules, thermal-hydraulics, and core neutronics) are considered; it is the evolution of the "cut set approach" in a simulation-based PRA framework. Note that metrics, such as core damage probability, still can be obtained from the dynamic approach.

At this point in the dynamic simulation process, the user has the possibility to evaluate the impacts of power uprates and life extension by comparing the limit surfaces for different scenarios or for potential modifications to the plant. As an example for the case of power uprates, the impact on safety margins is determined by evaluating the limit surface for the nominal case (i.e., 100% power) and the extended power (i.e., 120%) for several accident scenarios (e.g., loss–of-coolant accidents or station blackout), results of which are shown in Figure 7. Figure 7 was produced for a boiling water reactor during a station blackout-initiating event. This evaluation is performed not only in terms of failure probability, but also in terms of the reduction of system recovery timing. An example is shown in Figure 7 for a station blackout when two parameters are sampled: (1) time of failure of diesel generators after a reactor scram and (2) AC power recovery time. Two limit surfaces are plotted: one for 100% and the other for 120% power levels. Figure 7 shows the failure region expands when reactor power is increased, as well as the risk-informed analysis on reduction of recovery times. If the diesel generators fail at 10,000 seconds and the power level is at 100%, the operators

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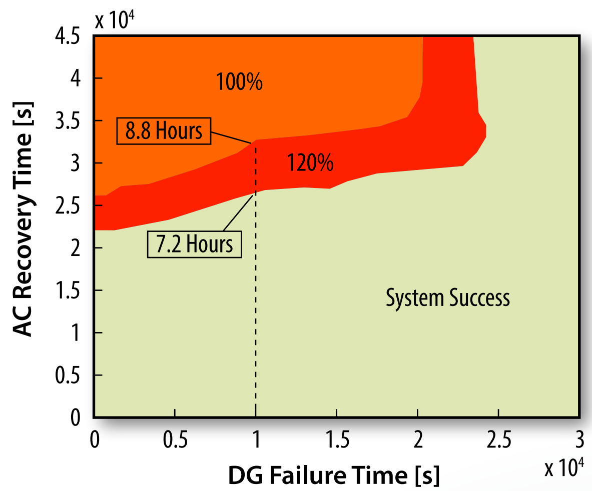

Figure 7. Limit surface for the boiling water reactor station blackout test case for two different power levels: 100 and 120%.

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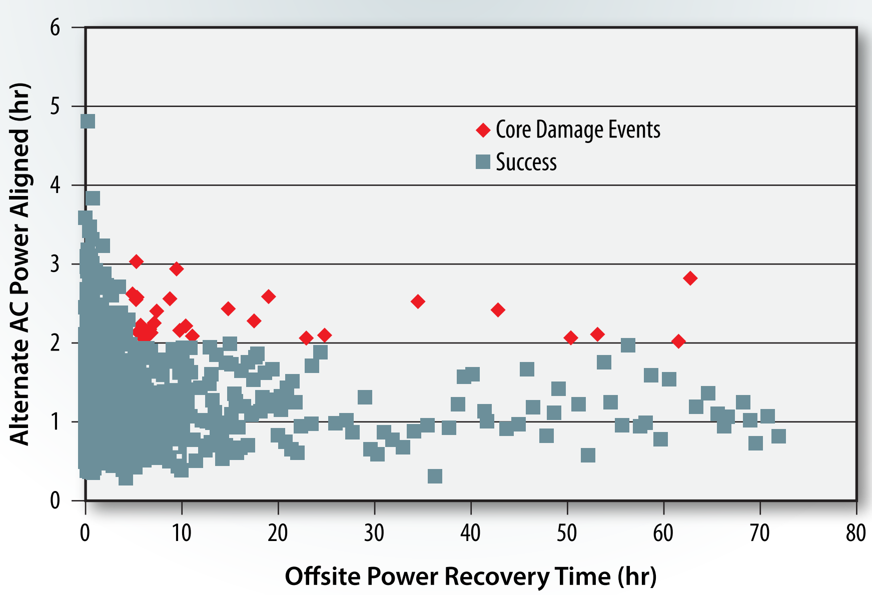
have 8.8 hours to recover AC power before reaching core damage. When power is increased to 120%, recovery time is decreased to 7.2 hours. Consequently, from a decision-making point of view, a power uprate can be compensated for by investing in more effective AC power recovery strategies (e.g., FLEX systems).

The RISMC analysis performed by the Electric Power Research Institute (EPRI 2013) has also demonstrated the value of the limit surface approach. In their work, they used the RISMC approach for analysis of the impact of a power uprate on plant safety for selected transient and accident sequences. These initial applications were conducted to demonstrate the feasibility and practicality of using the RISMC approach to analyze the safety impact of the uprate at both a pressurized and boiling water reactor. A consistent and repeatable process was developed and applied to identify those key parameters that would be analyzed. Distributions were constructed to represent the uncertainties associated with each of the key parameters. These distributions were sampled using a Latin Hypercube Sampling technique to generate sets of sample cases that were used in the physics simulation runs using the Modular Accident Analysis Program code. Simulation results were evaluated to determine the changes to safety margins, which would occur due to the uprated power conditions. The results obtained were then compared to those for the current nominal full power. The results obtained indicate, as expected, that safety margins may be reduced with increases in plant power level. However, for most power uprate levels, these safety margin reductions were found to be small. An example of the type of limit surface that can be produced as a function of the simulation sampling is shown in Figure 8.

Exploring the Response Surface

To this point, we have shown the safety value of the limit surface. However, the response surface also has informational content that can be explored. Earlier, we described how the response surface for safety-related variables, such as

Figure 8. Limit surface for the Electric Power Research Institute's boiling water reactor station blackout analysis (EPRI 2013).



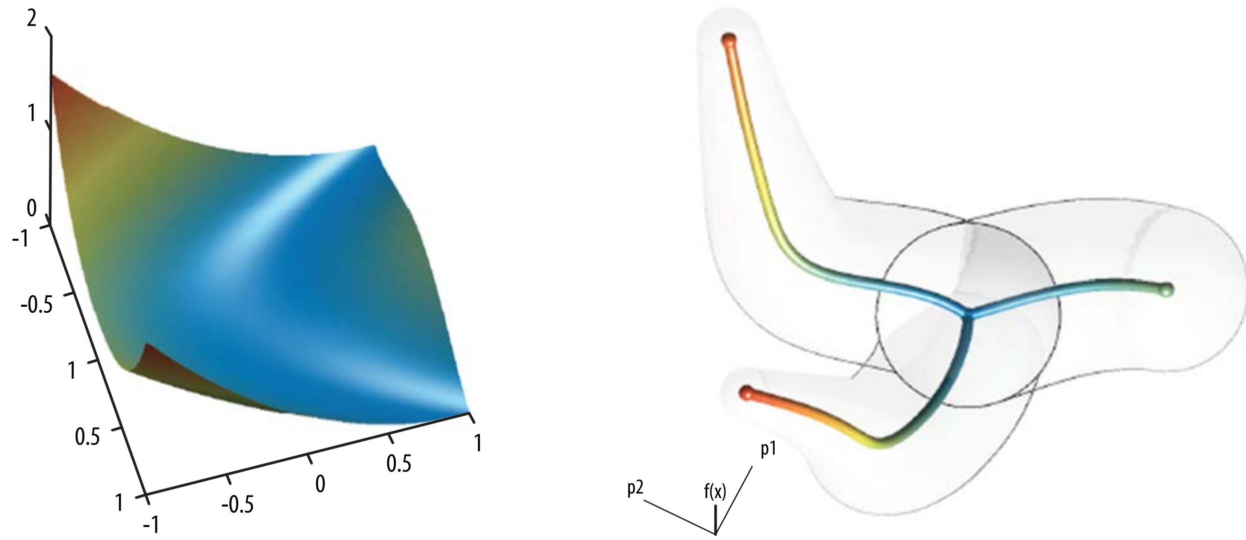


Figure 9. Example of topological visualization of a 3-D surface.

maximum clad temperature, is affected by a set of uncertain parameters. From a safety point of view, we are interested in understanding how we can remain in a safe condition (i.e., low maximum clad temperature) and avoid an unsafe condition (i.e., high maximum clad temperature).

In our safety analysis, because the number of uncertain parameters is high, the visualization of these high-dimensional surfaces can be challenging. In collaboration with the University of Utah, we have developed a tool that can visualize high-dimensional data. This tool topologically decomposes the response surface by doing the following:

1. Looking for local maxima and minima of the response surface
2. Identifying how they are connected to each other. The visualization of the maxima/minima connections is performed in a 3-D way.

An example of the topological surface is shown in Figure 9 for a 3-D surface (shown on left) and its corresponding visualization (shown on the right). Three local maximas and a local minima are identified; their connections are shown as branches.

From a safety point of view, the user can use this tool to identify under which conditions it is possible to move from a safe (i.e., a local minima) to an unsafe condition (i.e., a local maxima) by exploring the topological space contained in the visualization results.

Conclusions

The RISMC Pathway provides a systematic approach to the characterization of safety margins, leading to the support of margins management options (the proposed alternatives that work to control margin changes due to age-related effects or plant modifications). The research and development products of this pathway will provide vital input to the owner and regulator to support decision making for nuclear power plant operations now and for extended lifetimes.

RISMC uses a probability-margin approach to quantify impacts in order to understand and avoid conservatisms (where appropriate) and to treat uncertainties directly. An example of the types of results that are calculated for a station blackout analysis was described in terms of outcomes represented by a limit surface. The calculation to determine the limit surface uses a blended approach of probabilistic and mechanistic calculations and, as such, is a type of evolution of traditional cut set-based PRA calculations. While the limit surface can produce standard metrics (such as core damage frequencies), they also contain other valuable information specific to why, how, and when things fail. This information provides a starting point for potential modifications to the plant or procedures in the form of risk-informed margins management.

References

EPRI, 2013, "Pilot Application of Risk Informed Safety Margins to Support Nuclear Plant Long-Term Operation Decisions Impacts on Safety Margins of Extended Power Uprates for BWR Station Blackout Events," Technical Report 3002000573.

Using Computer Analysis Tools to Investigate the Fukushima Daiichi Event – MAAP/MELCOR Comparison

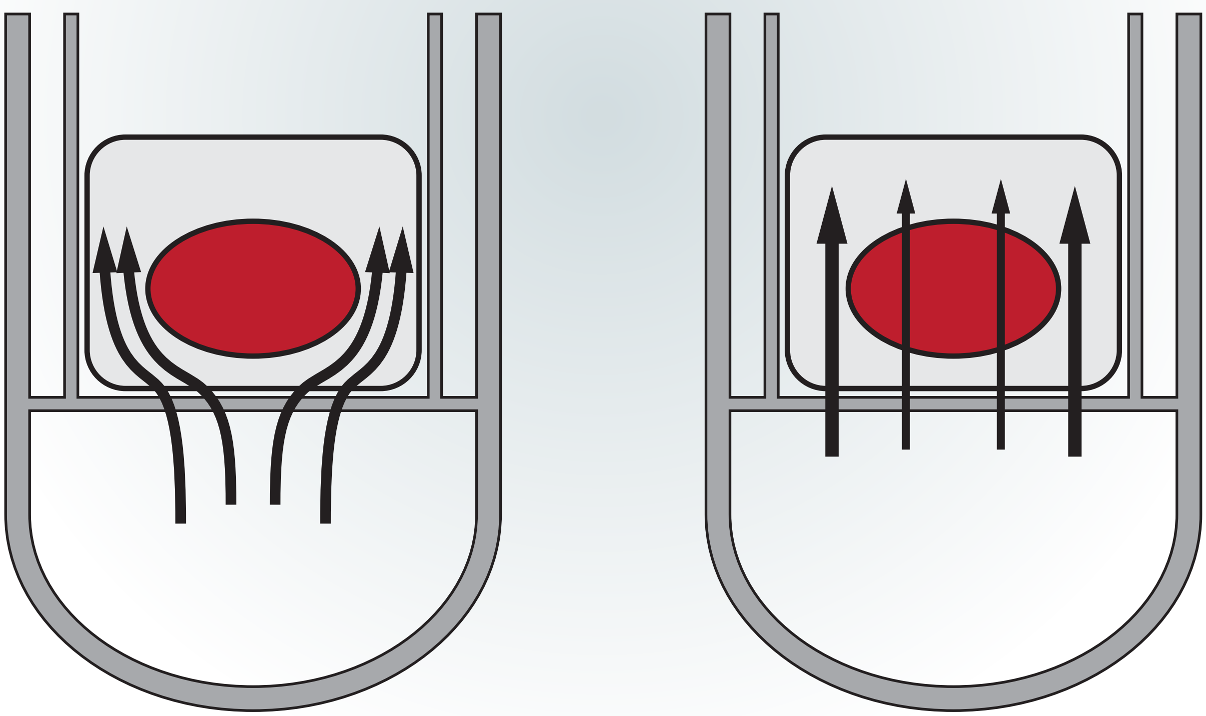
Michael Corradini
Reactor Safety Technologies Pathway

The Tohoku earthquake and tsunami in Japan in 2011 were the major events that led to the accident at the Fukushima Daiichi Nuclear Power Plant site. To aid in understanding the state of the reactors at Fukushima Daiichi for decommissioning and inspection, as well as to better understand the accident evolution, the DOE Office of Nuclear Energy investigated the Fukushima Daiichi event using computer analysis tools. The industry's severe accident simulation computer model, Modular Accident Analysis Program (MAAP), and NRC's severe accident computer model, Methods of Estimation of Leakages and Consequences of Releases (MELCOR), were used to evaluate and cross-compare the accident evolution for the Fukushima Daiichi Unit 1 reactor plant. The report (Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk Phase 1 Study) can be found on the Electric Power Research Institute's website. The following characteristics of the accident progression were compared to better understand the most significant differences observed between the MAAP and MELCOR simulations:

- RPV and fuel/debris temperature response up to the time of vessel lower head failure.
- History of the core debris mass and its temperature that may relocate into containment.
- Hydrogen generated during this in-vessel degradation process.

During the course of accident progression, core overheating and loss of fuel rod geometry took place

Figure 10. Illustration of different flow geometries (MAAP left and MELCOR right) through a degraded reactor core, where the MAAP model predicts the formation of a blockage causing steam flow to bypass the degraded core materials in contrast to the MELCOR model where the blockage is porous and allows steam flow to pass through the degraded core materials.



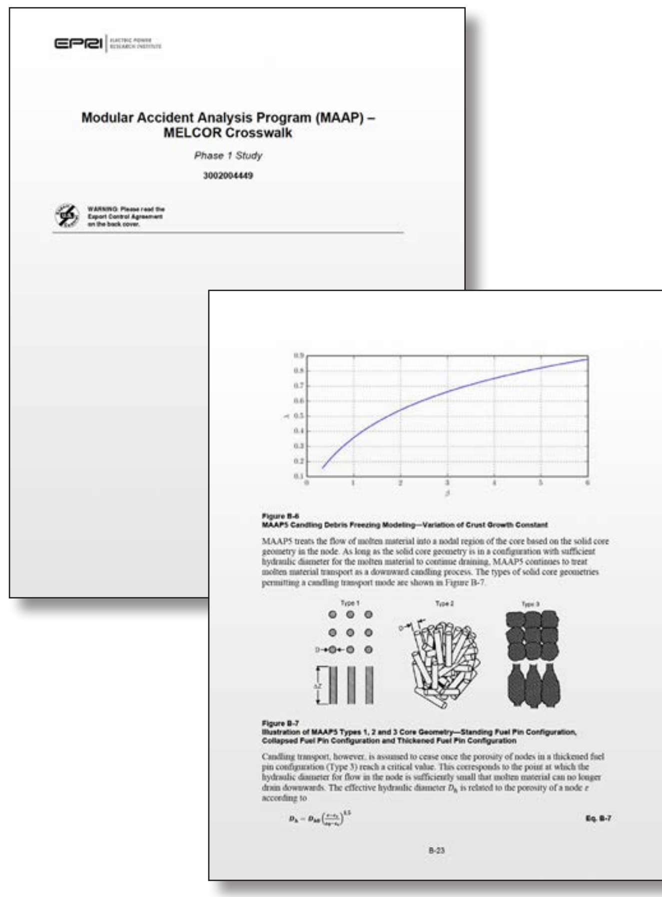
due to the boil-off of water in the reactor vessel. Both MELCOR and MAAP codes predicted very similar core heat-up trends and hydrogen generation up to the point that loss of fuel rod geometry occurred. Hydrogen is produced through a chemical reaction between steam and the fuel rod zirconium-metal cladding. However, as each code projected ongoing core degradation, the similarities began to diverge with MAAP, predicting development of a large in-core molten fuel region encapsulated within an outer refrozen crust of solidified fuel core materials. This pool/crust configuration limited additional hydrogen generation because of the reduced surface area of the geometry. Eventually, this configuration failed through the crust sidewall, spilled downward as liquefied core materials, and flowed into the reactor vessel lower plenum region. In contrast, MELCOR predicted a more gradual downward slumping of melted core materials that maintained some open flow area, allowing steam ingress and greater hydrogen generation. This downward slumping corium was partially molten in contrast to the MAAP-generated molten pool and eventually melted through the lower core plate that supports the reactor core, dropping into the reactor vessel lower plenum region. The different core flow geometries predicted by MAAP and MELCOR are illustrated in Figure 10.

These divergences in melt progression behavior lead to higher core melt temperatures and lower hydrogen generation in the MAAP predictions, and conversely lower corium (partially molten) temperatures with higher hydrogen generation in the MELCOR predictions. Because of these differences in core melt progression treatment in MELCOR and MAAP, higher core exit gas temperatures are predicted by MELCOR compared to MAAP. MAAP predicts higher temperatures in core materials exiting the reactor vessel on failure of the vessel lower plenum wall compared with MELCOR. These differences subsequently affect timing of the lower plenum vessel wall failure and lead to resultant differences in thermally induced failure potential for steam lines (i.e., core exit gas temperature) or severity of ex-vessel melt-structural interactions (i.e., core melt temperatures). This range of results is being considered as the Severe Accident Management Guidelines are being reviewed and revised. In addition, expected Fukushima Daiichi inspection activities will inform these accident simulations to help improve severe accident models.

The objective of this research and development activity in the Reactor Safety Technologies Pathway is to improve understanding of and reduce uncertainty in severe accident progression, phenomenology, and outcomes using existing industry and government analytical codes (an auxiliary benefit can be improvements in the models being used) and to use the insights from this improved understanding of the accident to aid in improving severe accident management guidelines for the current light water reactor fleet.

References

"Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk Phase 1 Study," 3002004449, Technical Update, November 2014, EPRI Project Manager: R. Wachowiak.



Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk

Phase 1 Study

3002004449

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Figure B-6
MAAP5 Candling Debris Freezing Modeling—Variation of Crust Growth Constant

MAAP5 treats the flow of molten material into a nodal region of the core based on the solid core geometry in the node. As long as the solid core geometry is in a configuration with sufficient hydraulic diameter for the molten material to continue draining, MAAP5 continues to treat molten material transport as a downward candling process. The types of solid core geometries permitting a candling transport mode are shown in Figure B-7.

Figure B-7
Illustration of MAAP5 Types 1, 2 and 3 Core Geometry—Standing Fuel Pin Configuration, Collapsed Fuel Pin Configuration and Thickened Fuel Pin Configuration

Candling transport, however, is assumed to cease once the porosity of nodes in a thickened fuel pin configuration (Type 3) reach a critical value. This corresponds to the point at which the hydraulic diameter for flow in the node is sufficiently small that molten material can no longer drain downwards. The effective hydraulic diameter D_h is related to the porosity of a node ε according to

Eq. B-7

B-23

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- ***RELAP-7 User's Guide***
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- ***Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk Phase 1 Study***
  https://lwrs.inl.gov/Reactor%20Safety%20Technologies/Crosswalk-phase1-final.pdf

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