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Prioritization and Implementation Plan for Collaborative Case Study on RPV Steels During Extended Service

February 2010

Prepared by

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Light Water Reactor Sustainability

Prioritization and Implementation Plan for Collaborative Case Study on RPV Steels During Extended Service

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Light Water Reactor Sustainability
Materials Aging and Degradation Pathway

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EXECUTIVE SUMMARY

The LWRS R/D program is designed to support and inform decisions for reactor service beyond the initial relicense period of 60 years. Within the LWRS R/D program, the Materials Aging and Degradation Pathway is charged with providing improved understanding and prediction of materials aging issues. The Risk-Informed Safety Margin Characterization Pathway (RISMC) is designed to provide a new framework for the development and implementation of safety and operating margins. Clearly, the missions and objectives of these two pathways provide opportunities for collaboration and coordination. The understanding of materials aging issues provides fundamental input to RISMC tasks and models. Conversely, output from RISMC research can drive materials testing plans and prioritizations.

A collaborative research effort between the two pathways has been proposed and initiated. A detailed case study will be used as a demonstration of the impact of emerging data on modern safety-limit calculations and the utility of using risk-informed tools to optimize experimental materials test matrices. Several possible case-study candidates were considered in the early portions of the LWRS program, including stress-corrosion cracking, concrete structures, and reactor pressure vessel embrittlement.

Pressurized Thermal Shock (PTS) of reactor pressure vessel (RPV) steels was selected as a specific topic for this case study. PTS can occur under some accident scenarios where cold water is introduced into a reactor pressure vessel that subsequently repressurizes. The recent development of a risk-informed safety margin for PTS of RPV steels is of high value, allowing for near term accomplishments and implementation in this program. The detailed and recent database also provides a solid foundation for the efficient development of new tools. Furthermore, the impact of data relevant to extended operations on PTS limits can be readily evaluated and used to drive further testing needs.

A preliminary implementation plan has also been completed. This plan will be finalized during the rest of FY2010 as future budgets and programmatic organization are resolved.

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1. INTRODUCTION

Nuclear power currently provides a significant fraction of the United States' non-carbon emitting power generation. In future years, nuclear power must continue to generate a significant portion of the nation's electricity to meet the growing electricity demand, clean energy goals, and ensure energy independence. New reactors will be an essential part of the expansion of nuclear power. However, given limits on new builds imposed by economics and industrial capacity, the extended service of the existing fleet will also be required.

Ensuring public safety and environmental protection is a prerequisite to all nuclear power plant operating and licensing decisions at all stages of reactor life. This includes the original license period of 40 years, the first license extension to 60 years, and certainly for any consideration of life beyond 60 years. For extended operating periods, it must be shown that adequate aging management programs are present or planned and that appropriate safety margins exist throughout the subsequent license renewal periods. Materials degradation can impact reactor reliability, availability, and potentially, safe operation. Components within a reactor must tolerate the harsh environment of high temperature water, stress, vibration, and/or an intense neutron field. Degradation of materials in this environment can lead to reduced performance, and in some cases, sudden failure. Clearly, understanding materials degradation and accounting for the effects of a reactor environment in operating and regulatory limits is essential.

The Light Water Reactor Sustainability (LWRS) Program is designed to support the long-term operation (LTO) of existing domestic nuclear power generation with targeted collaborative research programs into areas beyond current short-term optimization opportunities [1]. Within the LWRS program, two pathways have been initiated to perform research essential to informing relicensing decisions [1]. The Materials Aging and Degradation Pathway is designed to help develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants and to provide data and methods to assess performance of systems, structures, and components essential to safe and sustained operation. The Risk-Informed Safety Margins Characterization Pathway (RISMC) seeks to merge fundamental scientific understanding of critical phenomenological conditions and deterministic predictions of nuclear power plant performance with risk-informed characterization tools. This will provide an integrated characterization of public safety margins in an optimization of nuclear safety, plant performance, and long-term asset management.

Clearly, these two pathways have many synergies in goals and outcomes. The data and mechanisms generated in the Materials Pathway may feed into and mold efforts within the RISMC Pathway. In addition, insights from the characterization tools developed in RISMC tasks may inform materials testing needs and experiments. To demonstrate this potentially powerful interaction, a joint case study has been proposed and initiated.

This document describes the initial planning for a coordinated study between the Materials and the RISMC Pathways. A brief description of each Pathway is presented along with a more detailed description of the needs and requirements of this collaborative task. A list of criteria for any case-study candidate are then listed, along with the rationale for choosing pressurized thermal shock as the prime candidate an inter-pathway collaboration. A proposed timeline and organization of future interactions on this subject area is also presented.

2. PATHWAY AND COLLABORATION BACKGROUND

The LWRS R/D program is designed to support and inform potential extended service decisions. As noted in Ref. [1], there are many research requirements ensure safe, economic, and reliable operation of the existing LWR fleet beyond 60 years of life. The Materials Aging and Degradation Pathway is charged with providing improved understanding and prediction of materials aging issues, while the Risk-Informed Safety Margin Characterization Pathway is designed to provide a new framework for the development and implementation of safety and operating margins. Clearly, there are opportunities for collaboration and coordination between these two pathways. In this section, a brief description of each pathway will be given. Then, the motivation for collaboration between these pathways will be presented along with a description of the criteria for selecting and initial area of combined research.

2.1 MATERIALS AGING AND DEGRADATION PATHWAY

Nuclear reactors present a very harsh and complex environment for components service. Containment components such as the reactor vessel and containment structure must withstand high temperatures, complex environmental conditions, and high-stress over long periods, while remaining capable of withstanding high pressures, shocks, and impacts in the event of catastrophic failures. Components within a reactor core must tolerate high temperature water, stress, vibration, and an intense neutron field. Degradation of materials in these environments can lead to reduced performance, and in some cases, unexpected failure. As shown in Figure 1, there are many different types of materials within the reactor itself: over 25 different metal alloys can be found within the primary and secondary systems, not to mention the materials that make up the concrete containment vessel, instrumentation and control, and other support facilities.

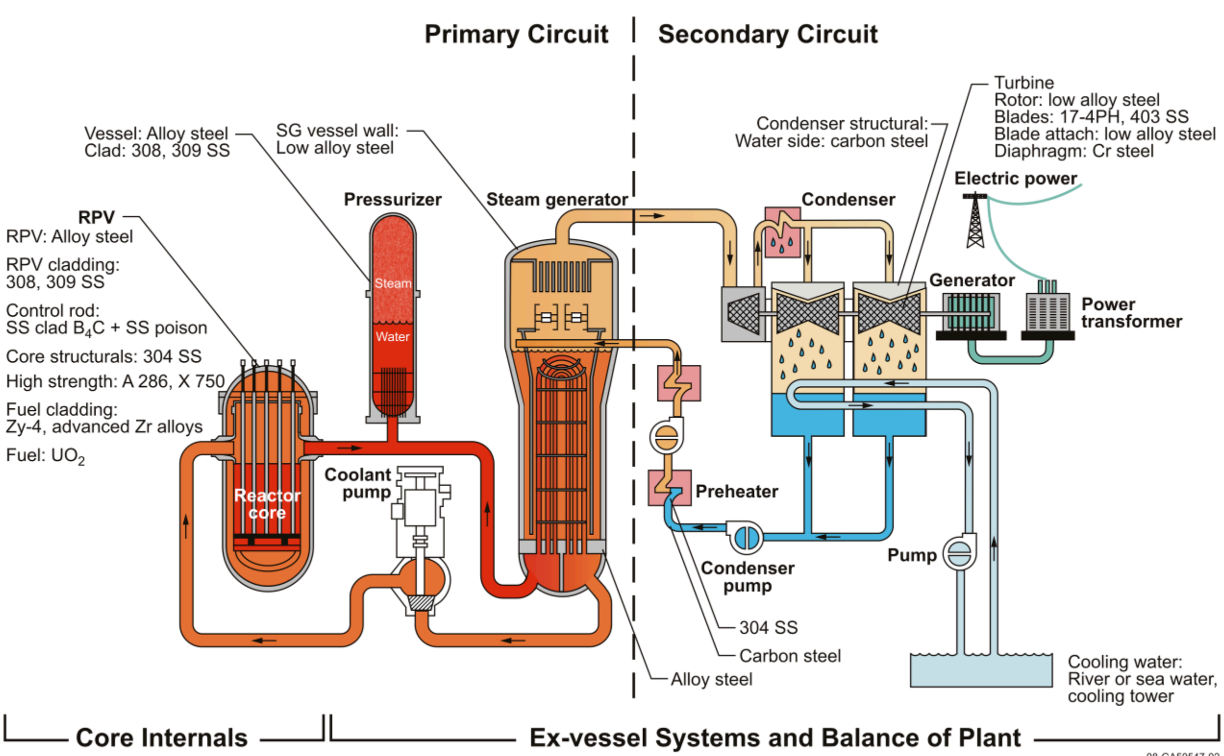

Figure 1. LWR reactor materials.

Clearly, materials aging and degradation will impact reactor reliability, availability, and, potentially, safe operation. Understanding and predicting degradation of materials exposed to this aggressive environment for an extended time is a key and significant challenge. With reactor life extensions beyond 60 years, many components must tolerate the reactor environment for a previously unanticipated service life, increasing exposure to neutrons, time under stress, time under temperature, and/or time exposed to an aggressive environment. Any one of these factors may increase susceptibility for components and introduce new degradation modes. Understanding, controlling, and mitigating materials degradation processes are thus key priorities for safe and efficient reactor operation and safe life extensions.

The strategic objectives of the Nuclear Materials Aging and Degradation R&D pathway are to develop the scientific basis for understanding and predicting long-term environmental degradation behavior of materials in nuclear power plants and to provide data and methods to assess performance of SSCs essential to safe and sustained nuclear plant operations. The research activities within this pathway have been organized into four areas: (1) reactor metals, (2) concrete, (3) cable aging, and (4) buried piping. Unlike consumable materials such as fuel and other components that are periodically replaced during plant life, these SSCs were installed with the intention that they would operate for the safe life of the nuclear power plant. As the nuclear power plant licensees seek approval for extended operation, the way in which these materials age beyond 60 years will need to be evaluated and their capabilities reassessed in order to ensure that they maintain the required design functions safely and economically. A more detailed description of the key forms of degradation in nuclear power plants is presented in Ref. [2]. Each of these research areas and specific research tasks are described in Ref. [1].

One key area of research is related to the reactor pressure vessel (RPV). The RPV is the largest single component at a nuclear power plant site and forms the major pressure boundary for an LWR. The pressure vessel for a boiling water reactor (BWR) is shown in Figure 2. Current regulations require RPV steels to maintain conservative margins of fracture toughness so that postulated flaws do not threaten the integrity of the RPV during either normal operation and maintenance cycles or under accident transients, like pressurized thermal shock.

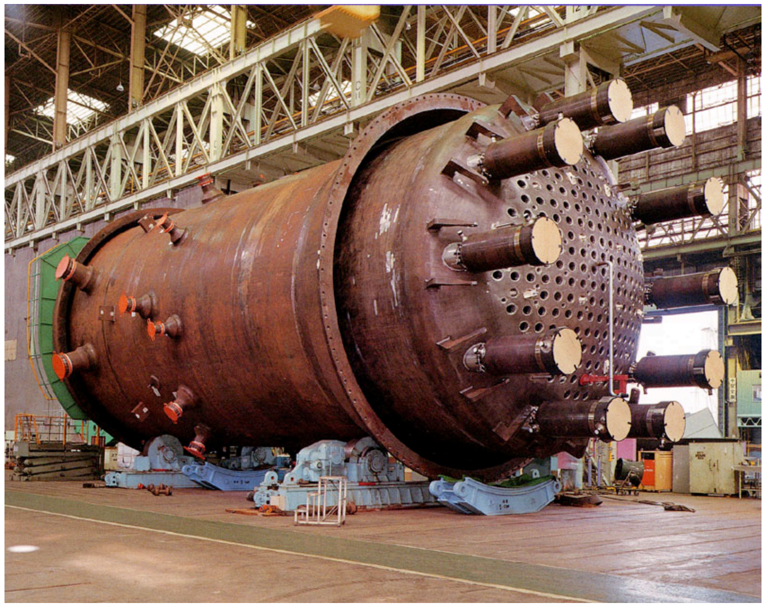

Figure 2. Boiling water reactor reactor pressure vessel.

The last few decades have seen remarkable progress in developing a mechanistic understanding of irradiation embrittlement of the reactor pressure vessel. There are still significant technical issues that need to be addressed to reduce the uncertainties in regulatory application, particularly for possible service beyond 60 years. Within the Materials Pathway, a high-priority research task on examining irradiation-induced hardening at very high neutron fluences has been initiated. However, despite a solid experimental foundation, our present understanding of radiation damage is not fully quantitative and does not address all potentially significant variables and issues. The combination of irradiation experiments with modeling and microstructural studies will provide an essential understanding to aging evaluations of RPVs.

2.2 RISK-INFORMED SAFETY MARGIN CHARACTERIZATION PATHWAY

As noted in [1], the concept of safety margins as a cornerstone in nuclear reactor design emerged during the early days of nuclear power as a part of the defense-in-depth approach to ensuring nuclear safety. Defined as the minimum distance between the system's "loading" and "capacity," safety margin is expressed in terms of safety-significant parameters (e.g., fuel cladding temperature, containment pressure) and determined for a range of anticipated system operating conditions. Traditionally, in nuclear power plant design and licensing, availability of safety margins must be demonstrated for a prescribed set of design-basis accidents (DBA). Due to the limited knowledge, large (i.e., conservatively specified) safety margins are applied to compensate for approximations used in (the phenomenological, deterministic) models and associated computer codes which estimate the "loads" and the "capacity" in the reactor systems during the complex postulated transient and accident sequences.

With respect to methodology for integrated safety assessments, quantification and utilization of plant safety margins and their regulatory implications have received increased attention during recent years, paving way to the formulation of Risk-Informed Safety Margin Characterization (RISMC) as an R&D area. A comprehensive review of the state of the art and discussion of open issues related to RISMC can be found in the CSNI Safety Margin Action Plan (SMAP) group report [3]. Beyond the still-open formidable questions on a RISMC framework, it is widely recognized that the success of a risk-informed approach requires enhanced simulation tools (computer codes) to enable system analysis with high fidelity and treatment of uncertainties, which can be significant, for example, in non-DBA and beyond-DBA situations. These challenges will increase as plant operational life is extended.

Figure 3 depicts elements of RISMC in the context of the LWRS Program. In the spirit of defense-in-depth, margin is considered to be significant to the degree that it exceeds uncertainties and variabilities associated with a given comparison between "load" and "capacity." This idea applies to success of active functions as well as passive SSC integrity, which is instrumental to the characterization, mechanistic understanding, prediction, and monitoring of the plant aging and degradation behaviors and their impact on plant life extension decision making.

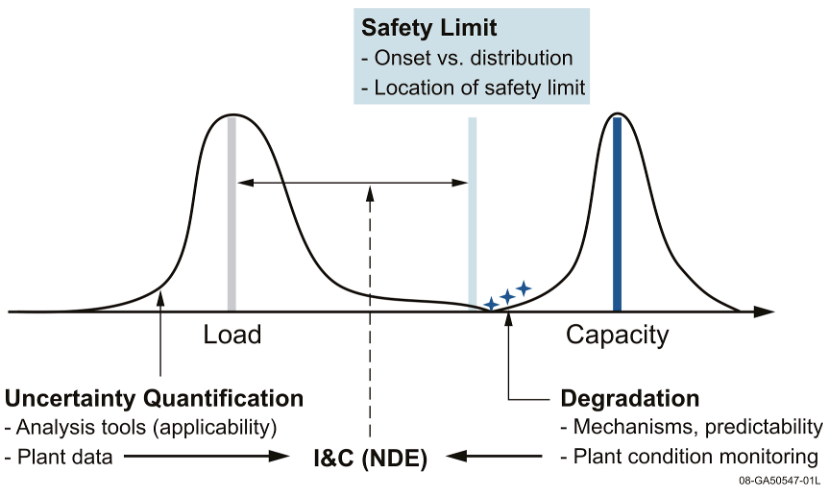

Figure 3. Elements of the RISMC model for LWRS Program.

The strategic objectives of the RISMC R&D pathway are to bring together risk-informed, performance-based methodologies with fundamental scientific understanding of critical phenomenological conditions and deterministic predictions of nuclear plant performance, leading to an integrated characterization of public safety margins in an optimization of nuclear safety, plant performance, and long-term asset management.

The RISMC R&D pathway scope is shown in Figure 4. The guiding principle is to focus on developing knowledge/capability to facilitate enhanced decision-making and improved regulatory/public acceptance of long-term plant operation. Furthermore, the RISMC R&D pathway is envisioned as a mechanism to provide an integrating science-based framework to enable *effective visualization and efficient implementation* of advances achieved in the other LWRS Program pathways. The RISMC pathway provides, on the front end, a means to risk-inform the R&D efforts performed in the other LWRS Program pathways; and on the back end, a mechanism for the effective implementation of advances achieved in them.

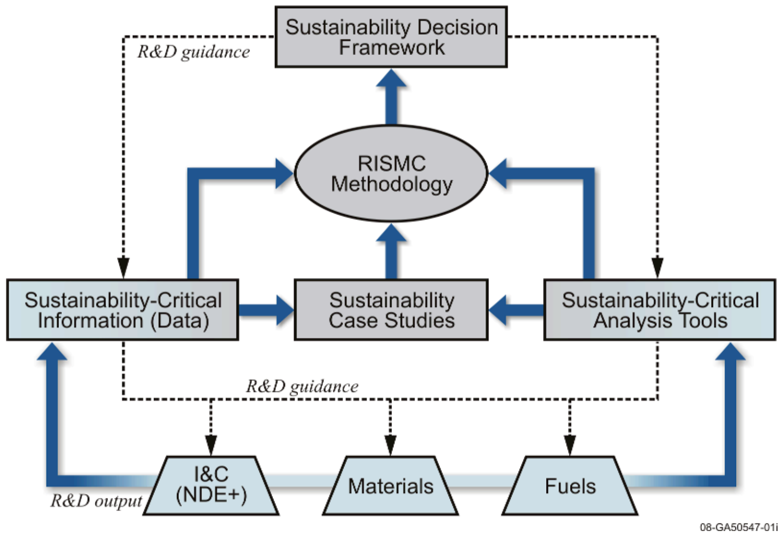

Figure 4. R&D strategy of RISMC for LWR sustainability.

The centerpiece of the strategy is the development of the "RISMC Methodology" to support the overall LWR "Sustainability Decision (Framework)." The approach taken to this development (at least in the initial stages of the LWRS Program) is through "(Sustainability) Case Studies," which will examine the potential benefits and issues of RISMC in situations where existing decision processes may not be sufficient to address long-term sustainability needs and objectives.

2.3 INTER-PATHWAY COLLABORATIONS

The missions and objectives of the Materials Aging and Degradation and Risk-Informed Safety Margin Characterization Pathways provide opportunities for collaboration and coordination. The understanding of materials aging issues provides fundamental input to RISMC tasks and models. Conversely, output from RISMC research can drive materials testing plans and prioritizations.

Experimental data and predictive capabilities obtained from the Materials Pathway will be essential in informing licensing decisions. Experimental results are also a requirement for establishing a relicensing case and operating limits. In addition, the outputs from the Materials Pathway can be used to define mechanisms of materials degradation as inputs for RISMC activities and simulations. Further, with simulations in hand, these datasets are invaluable for the validation and verification of any modeling or simulation tool. However, an entirely empirical approach is not viable due to the time, expense, and difficulty associated with the number of experiments required for so many materials and variables.

Tools developed in the RISMC pathway provide the opportunity to interpolate and extrapolate within and between datasets, extending the utility of experimental results. As noted above, "the RISMC pathway provides a means to risk-inform the R&D efforts performed in the other LWRS Program pathways; and on the back end, a mechanism for the effective implementation of advances achieved in them." Of particular interest here is that outputs from the RISMC pathway can be used to inform the Materials Aging and Degradation R&D pathway of issues in which the current (or anticipated future) safety margin is identified to be a significant potential limitation to long-term nuclear power plant operation. With this information, the Materials Aging and Degradation R&D pathway can then take appropriate actions such as targeted testing or reprioritization of tasks are taken to efficiently inform licensing decisions.

The opportunities and value of interactions and collaboration between the Materials and RISMC pathways have been discussed since the initial formulation of the program in August 2008. Both pathways have provided limited funding in FY10 to identify and initiate a case study. This first interaction between pathways is intended provide a demonstration of the valuable insights into licensing concerns for key components as new data is generated. Interpretation of experimental data and operation limits will also help determine the most pressing data needs. The following sections describe the framework for this collaboration, including selection of a specific topic for the case study and a proposed implementation plan for the rest of FY2010 and beyond.

3. CASE STUDY CRITERIA AND SELECTION

As noted above, a collaborative effort between the Materials Aging and Degradation and Risk-Informed Safety Margin Characterization Pathways has been initiated. This first case study is intended to be a demonstration of the impact of emerging data on modern safety-limit calculations and the utility of using advanced simulation tools to optimize experimental materials test matrices. In this section, the criteria for selection of this first case study are presented, along with several options that were considered. The specific topic selected for further study is also discussed.

3.1. SELECTION CRITERIA FOR CASE STUDY

There are several key criteria that must be satisfied for a potential case study topic to be relevant and meaningful to both the Materials Aging and Degradation Pathway and the Risk-Informed Safety Margin Characterization Pathway. Some are key technical considerations while others are more practical. Specific criteria considered included:

1. **Safety relevant to extended operations:** a case-study topic must be relevant to the safety case for life extension of a nuclear reactor. Topics such as cable aging, while technically interesting, are not primarily safety-related and therefore not as relevant for a detailed case study at this time.

2. **Recent experience and modeling:** potential topics for consideration would also, ideally, include areas where there is recent research into characterization or modeling a degradation phenomenon. Areas with recent analysis and simulation related to safety limits are of the highest value. Furthermore, current research topics are beneficial as the original experts involved may be available to participate in this program.

3. **Leverages existing research tasks:** with limited funding in the early years of LWRS and many topics requiring research, the resources available for a case-study may be restricted. Case studies that can be complementary to ongoing or planned research tasks in either pathway are preferred.

4. **Sufficient existing data:** an existing database is beneficial as it allows for research to begin in the RISMC effort immediately without waiting for data to be generated in the Materials pathway tasks. Further, a database for current operations provides a valuable benchmark for any tools developed, speeding tool development.

5. **Existing modeling experience:** As above, if a simulation or safety margin tool exists, it can be used in the near term to evaluate testing needs for extended service. This information can steer testing programs in the early stage of the program. Existing data and simulation tools also indicate that the dominant material degradation mechanisms are well known and predictable, more readily allowing for identification of new mechanisms associated with extended service.

Clearly, fulfilling all criteria may be difficult, although possible. During the earliest stages of the LWRS program, leaders of both pathways proposed several potential case studies.

3.2. CASE STUDY CANDIDATES

Early in the LWRS program, the need for collaboration between the Materials and RISMC pathways was identified. Several potential areas for a case study were discussed as early as August 2008. These included stress-corrosion cracking, concrete structures, and reactor pressure vessel embrittlement. All three areas of research are the subject of ongoing research in the Materials Aging and Degradation Pathway and may be key for extended service operations (thus satisfying criteria 3 and 1 above).

> **Stress-corrosion cracking:** Within a nuclear reactor system, a number of different corrosion mechanisms are at work. When combined with stress, stress-corrosion cracking (SCC) can and does occur in numerous alloys and locations within the reactor. Extended service will increase exposure to coolant, stress, and neutrons further increasing susceptibility to SCC. SCC is an area of active research in many organizations worldwide, including the LWRS program. SCC of different materials and components also receives considerable regulatory interest. However, despite over thirty years of research, the fundamental mechanisms of SCC are relatively unknown. Further, there is no existing model or tool for the prediction of SCC, making an immediate impact on a case study difficult.
>
> **Concrete structures:** Concrete structures are a key component within a nuclear power plant, providing support to all other components and providing a final containment boundary. As a result, the degradation of concrete structures may be a key safety concern during extended service. Research into concrete structure degradation has been initiated within LWRS and other programs around the world. However, concrete structures may not be useful for a case study. While of high value from a safety point-of-view, there is very little existing surveillance data (which is not specifically required by regulators) making it difficult to establish adequate modeling tools.
>
> **Reactor pressure vessels:** Reactor pressure vessels are critical safety-related components in nuclear power plants. Repairing or replacing the pressure vessel is not practical, yet its mechanical integrity must be conservatively demonstrated for up to 80 years of operation. During operation, neutrons from the nuclear core impinge on the vessel wall, reducing its strength and ductility and diminishing its ability to withstand flaws that might be present, such as those that may have been introduced during fabrication. Research is ongoing in LWRS in the area of RPV embrittlement for extended service, building on decades of research designed to establish safe operating limits. As a result, there is a considerable experimental database and multiple models and tools for predicting performance and establishing safety limits under a variety of normal and off-normal operating conditions.

All three areas of research were initially discussed. However, the area of RPV steels and embrittlement was quickly determined to be the most viable and desirable for collaboration, being the only topic satisfying all the criteria listed above. This active area of research has been the recent subject of research for both the current and extended operating periods.

One specific area of RPV research has been specifically suggested as a case study. Pressure vessel embrittlement is of particular concern in PWRs due to pressurized thermal shock (PTS). PTS can occur under some accident scenarios where cold water is introduced into a reactor pressure vessel that subsequently repressurizes. The cold water causes the vessel to cool rapidly, resulting in large thermal stresses, which could initiate cracks that could propagate during repressurization in the embrittled vessel material, possibly to the point of breaching the vessel wall.

Although such a failure has never happened, the U.S. Nuclear Regulatory Commission (NRC) issued a rule in the mid-1980s (10 CFR part 50.61, the "PTS Rule") that limits the amount of embrittlement before additional evaluations or corrective actions are required. Operators of older nuclear plants found it extremely difficult to adequately demonstrate reactor pressure vessel integrity using the 1970s-vintage analytical assumptions embodied in the original PTS rule. One plant, Yankee Rowe, shut down pre-maturely in 1992 due, in part, to the high cost and difficulty of demonstrating pressure vessel integrity under postulated PTS conditions. Several other plants, including Palisades, Beaver Valley and Kewaunee, also faced premature shutdown since they were expected to exceed the PTS regulatory limit before the end of the license period.

More detailed technical descriptions of PTS and the "PTS Rule" are in the section below.

3.3. PRESSURIZED THERMAL SHOCK FOR PRESSURIZED WATER REACTORS

During the operation of a nuclear power plant, the reactor pressure vessel walls are exposed to neutron radiation, which, for radiation-sensitive steel, may result in localized embrittlement of the vessel steel and weld metals in the area of the reactor core. Certain abnormal severe transients can cause a rapid depressurization in the RPV, which may cause the water level in the core to drop. In such an event, automatic systems and operators must provide makeup water in the primary system to prevent overheating of the fuel in the core. However, the makeup water is much colder than that held in the primary system and the impingement of cold water (~ room temperature) on the hot RPV wall (~ 288°C) will produce significant thermal stresses in the thick steel wall of the RPV. If an embrittled RPV has an existing flaw of critical size close to the RPV inner surface and the vessel is repressurized, the combined stresses from the thermal shock and the pressure may cause the flaw to propagate very rapidly through the vessel, resulting in a through-wall crack and challenging the integrity of the RPV. Such through-wall cracking of the RPV could precipitate core damage or, in rare cases, a large early release of radioactive material to the environment. Fortunately, the coincident occurrence of critical-size flaws, embrittled vessel steel and/or weld metal, and a severe PTS transient is a very low-probability event. In fact, only a few currently operating PWRs are projected to closely approach the current statutory limit on the level of embrittlement during their planned 40-y operational life.

As set forth in 10 CFR 50.61, the PTS Rule requires licensees to monitor the embrittlement of their RPVs using a reactor vessel material surveillance program qualified under Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements." The surveillance results are then used together with the formulae and tables in 10 CFR 50.61 to estimate the fracture toughness transition temperature (RT_{NDT}) of the steels in the vessel's beltline and how those transition temperatures increase as a result of irradiation damage throughout the operational life of the vessel. For licensing purposes, 10 CFR 50.61 provides instructions on how to use these estimates of the effect of irradiation damage to estimate the value of RT_{NDT} that will occur at end of license (EOL), a value called RT_{PTS}. 10 CFR 50.61 also provides "screening limits" (maximum values of RT_{NDT} permitted during the plant's operational life) of +270°F (132°C) for axial welds, plates, and forgings, and +300°F (149°C) for circumferential welds. These screening limits correspond to a limit of 5×10^{-6} events/year on the annual probability of developing a through-wall crack [RG 1.154]. Should RT_{PTS} exceed these screening limits, 10 CFR 50.61 requires the licensee to either take actions to keep RT_{PTS} below the screening limit (by implementing "reasonably practicable" flux reductions to reduce the embrittlement rate, or by recovering the fracture toughness of the vessel by annealing [RG 1.162]), or perform plant-specific analyses to demonstrate that operating the plant beyond the 10 CFR 50.61

screening limit does not pose an undue risk to the public [RG 1.154]. While no currently operating PWR has an RT_{PTS} value that exceeds the 10 CFR 50.61screening limit before EOL (40 y), several plants are close to the limit (3 are within 2°F, while 10 are within 20°F). Those plants are likely to exceed the screening limit during the 20-year license renewal period that is currently being sought by many operators.

The state of knowledge and data limitations in the early 1980s necessitated conservative treatment of several key parameters and models used in the probabilistic calculations that provided the technical basis for the current PTS Rule. Information and data obtained since that time indicate the high likelihood that the current 10 CFR 50.61 PTS screening limits are unnecessarily conservative. Consequently, the NRC staff believed that reexamining the technical basis for these screening limits, based on a modern understanding of all the factors that influence PTS, would most likely provide strong justification for substantially relaxing these limits.

For these reasons, the NRC undertook a new PTS study with the objective of developing the technical basis to support a risk-informed revision of the PTS Rule and the associated PTS screening limit. Three main models, taken together, allow for an estimation of the annual frequency of through-wall cracking in an RPV: (1) probabilistic risk assessment (PRA) event sequence analysis, (2) thermal-hydraulic (TH) analysis, and (3) probabilistic fracture mechanics (PFM) analysis. Some of the major factors contributing to a relaxation of the existing conservatisms are, (1) removal of a significant conservative bias in the fracture toughness model, (2) recognition of a spatial variation in neutron fluence, (3) placement of the presumed flaws embedded in the vessel wall rather on the surface, and (4) improvements in the thermal hydraulics code Following collection of the most recent data available related the many variables required for each of the above models, some of the key findings are noted:

- *the degree of PTS challenge is low for currently anticipated lifetimes and operating conditions.*

- *axial flaws, and the toughness properties that can be associated with such flaws, control nearly all the through-wall cracking frequency (TWCF).*

- *because the severity of the most significant transients in the dominant transient classes is controlled by factors that are common to PWRs in general, the TWCF results can be used with confidence to develop revised PTS screening criteria that apply to the entire fleet of operating PWRs.*

- *the current guidance provided by Regulatory Guide 1.174 [RG 1.174] for large early release is appropriately applied to setting an acceptable annual TWCF limit of $1x10^{-6}$ events/year.*

Given the results of the PTS reevaluation, operating PWRs do not closely approach the $1x10^{-6}$/year limit for 40-y EOL; the differences of between the plant-specific values and the proposed screening limits range from 70 to 290°F (39 to 161°C). Moreover, these differences are reduced only by 10–20°F (5.5 to 11°C) at end of license extension (EOLE, defined as 60 operating years or 48 EFPY). Additionally, no forged plant is anywhere close to the limit of $1x10^{-6}$ events per year at either EOL or EOLE. This separation of operating plants from the screening limit contrasts markedly with the current situation, where the most embrittled plants are within 1°F (0.5°C) of the screening limit set forth in 10 CFR 50.61.

The major result of the PTS reevaluation project described above is the recent revision to 10 CFR 50, with the addition of section 50.61a, "Alternate fracture toughness requirements for protection against pressurized thermal shock events." This sections states "The requirements of this section may be implemented as an alternative to the requirements of 10 CFR 50.61."

As noted above, the recent reevaluation of PTS was quite extensive in scope and involved twenty-five institutions from around the world. Three different methodologies were studied and combined, including a probabilistic risk-assessment, thermal hydraulic analysis, and probabilistic fracture mechanics. A detailed summary of the organization of this research and the results are beyond the scope of this report and are readily found in Ref. [4].

The development of a risk-informed safety margin for PTS of RPV steels is of very high value to this proposed case study, allowing for near term accomplishments and implementation. The detailed and recent database also provides a solid foundation for the efficient development of new tools. Furthermore, the impact of data relevant to extended operations on PTS limits can be readily evaluated and used to drive further testing needs. In the next section, an implementation plan for PTS is presented.

4. CASE STUDY IMPLEMENTATION PLAN AND SCHEDULE

The PTS area of research appears to be an ideal topic for an inter-pathway collaboration given the depth of recent data and analysis, recent development of a risk-informed safety analysis, and ongoing research in the Materials and Aging Degradation pathway on RPV hardening under extended operation conditions. In the coming months and years, a detailed case study will be implemented in both pathways as more research funds become available. In the following sections, a more detailed implementation plan is proposed. (Note: this plan is considered a proposal given current uncertainty in future funding levels and organization of Office of Nuclear Energy research programs. This plan will be finalized during the remainder of FY10 as funding levels and organization are confirmed.)

Several key steps will be required for successful implementation of this case study into PTS of RPV steels. These are listed below along with estimated completion dates.

Milestone: Initiate research on RPV embrittlement for extended service conditions
Responsible Pathway: Materials Aging and Degradation
Due Date: 6/1/2010
Status: Complete
Description: Experimental research must be initiated to provide new data to the case study. Research tasks were initiated in FY09 with the first data taken during that same year. Current research has focused on hardening trends and microstructure characterization in RPV steels exposed to high fluences [1]. All of this data can be utilized in this case study.

Milestone: Finalize implementation plan for case study
Responsible Pathway: Materials Aging and Degradation and Risk-Informed Safety Margin Characterization Pathways
Due Date: 8/31/2010
Status: In progress
Description: As budgets are developed and research reorganized over the coming months, this implementation plan will be finalized. Milestones will be finalized based on current and future funding profiles, expert availability, and incorporation of any additional needs.

Milestone: Establish mechanism for data transfer
Responsible Pathway: Materials Aging and Degradation Pathway
Due Date: FY2011
Status: In progress
Description: A critical requirement in a successful collaboration is the transfer of data from experimental tasks to the RISMC efforts. An efficient mechanism must be identified and implemented. Data reports, dedicated collaboration meetings, and continuous data-transfer are all options. Specific options for transfer and frequency of data transfer will be discussed between pathways and finalized.

Milestone: Acquire models/codes for risk-informed safety margin characterization of PTS
Responsible Pathway: Risk-Informed Safety Margin Characterization Pathway
Due Date: FY2011
Status: Future task

Description: The recent reassessment of the PTS rule by an international group of experts has provided key tools that will be valuable in evaluating PTS for extended service as part of this case study. An important first step will be acquiring the risk-informed safety case developed in the 1999 reassessment and supporting model codes. In addition, access to the original database will provide a valuable baseline. This can initially be accomplished by engaging the leading experts from Ref. [4].

Milestone: Finalize report on modeling requirements on R7 simulation tools for PTS
Responsible Pathway: Risk-Informed Safety Margin Characterization Pathway
Due Date: FY2011
Status: Future task
Description: Based on the evaluation of existing tools, characterization and data, the requirements for a risk-informed characterization for extended service will be evaluated. The needs for R7 simulation tools will be compiled and prioritized.

Milestone: Issue guidance to Materials Pathway on revised data needs
Responsible Pathway: Materials Aging and Degradation and Risk-Informed Safety Margin Characterization Pathways
Due Date: FY2012
Status: Future task
Description: Using the R7 simulation tools, a set of data needs prioritized to reduce safety margin uncertainty will be issued to the Materials pathway. This informed guidance to the testing program is an important demonstration of the utility of future collaborations.

Milestone: Complete assessment of R7-enabled case study of extended service PTS
Responsible Pathway: Risk-Informed Safety Margin Characterization Pathway
Due Date: FY2013
Status: Future task
Description: A final report detailing the incorporation of new experimental data into the R7-enabled case study on PTS under extended service conditions will be completed.

Successful completion of these items will provide a valuable demonstration of both the RISMC tools and methodology and the experimental testing program under the Materials Pathway. However, successful completion will require sufficient resources to support research in both pathways. The LWRS Program Plan [1] lists the estimated funding requirements for over for FY09-FY13. Those estimates are shown below in Table 1.

Table 1: Five-year budget profile for PTS case-study (in $k)

	FY09	FY10	FY11	FY12	FY13
Materials Aging and Degradation	-	50	150	200	200
Risk-Informed Safety Margin Characterization	-	50	150	200	200
TOTAL	-	$100k	$300k	$400k	$400k

5. SUMMARY

The LWRS R/D program is designed to support and inform extended service relicensing decisions. The Materials Aging and Degradation Pathway is charged with providing improved understanding and prediction of materials aging issues, while the Risk-Informed Safety Margin Characterization Pathway is designed to provide a new framework for the development and implementation of safety and operating margins. Clearly, the missions and objectives of the Materials Aging and Degradation and Risk-Informed Safety Margin Characterization Pathways provide opportunities for collaboration and coordination. The understanding of materials aging issues provides fundamental input to RISMC tasks and models. Conversely, output from RISMC research can drive materials testing plans and prioritizations.

An initial collaboration between the two pathways has been initiated. An initial case study is intended to be a demonstration of the impact of emerging data on modern safety-limit calculations and the utility of using advanced simulation tools to optimize experimental materials test matrices. Several possible case-study candidates were considered in the early portions of the LWRS program, including stress-corrosion cracking, concrete structures, and reactor pressure vessel embrittlement.

Pressurized Thermal Shock (PTS) was selected as a specific topic for this case study. PTS can occur under some accident scenarios where cold water is introduced into a reactor pressure vessel that subsequently repressurizes. The development of a risk-informed safety margin for PTS of RPV steels is of very high value to this proposed case study, allowing for near term accomplishments and implementation. The detailed and recent database also provides a solid foundation for the efficient development of new tools. Furthermore, the impact of data relevant to extended operations on PTS limits can be readily evaluated and used to drive further testing needs.

A preliminary implementation plan has also been completed. This plan will be finalized during the rest of FY2010 as budgets and programmatic organization are resolved.

6. REFERENCES

1. Light Water Reactor Sustainability Research and Development Program Plan, INL Document, INL/MIS-08-14918, September 2009.

2. J.T. Busby, R.K. Nanstad, R. E. Stoller, Z. Feng, and D.J Naus, " Materials Degradation in Light Water Reactors: Life After 60," ORNL Report, ORNL/TM-2008/170.

3. Task Group on Safety Margins Action Plan (SMAP), Nuclear Energy Agency Committee on the Safety of Nuclear Installations Report, NEA/CSNI/R(2007)9, 2007.

4. Comparison Report of RPV Pressurized Thermal Shock, Nuclear Energy Agency Committee on the Safety of Nuclear Installations Report, NEA/CSNI/R(99)3, 1999.

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