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Revision 1

Accelerated Materials Deployment in Advanced Nuclear Power Plants

Proposed Risk Management Framework

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INL ART Program

Accelerated Materials Deployment in Advanced Nuclear Power Plants

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EXECUTIVE SUMMARY

This report will begin development of a maximally efficient process for licensing and deploying new materials in advanced non-light-water reactors (ANLWRs). New materials used in some new plants are seen as possibly introducing risk because our understanding of their behavior in the conditions generated by some novel plant designs is less complete than our understanding of material behavior with long use histories in existing designs. An approved code, standard, or code case to support use of these new materials in these novel design safety cases may not exist for designers, license applicants and regulators to use as part of a licensing determination. This creates the potential for an extremely long licensing process, especially for new designs using new materials. The present strategy is to develop a process that shows how to manage risks proactively to permit timely licensing decisions.

This report outlines gaps in current codes to support deployment and use of novel materials and begins the development of a risk management framework focused on the subject materials issues based on risk-informed in-service surveillance practices that compensate for current limitations in our state of knowledge. This development will enable licensing and deployment of the subject materials, conditional on the establishment of a suitable surveillance process. While there are clear limitations in our knowledge of certain material issues that may arise in advanced designs, in-service surveillance is always done to compensate for a lack of knowledge even in current-generation plants. What is different about the surveillance program discussed here is that the issues are newer, and the relevant experience base is less complete, so the surveillance contemplated in this report may need to measure new things or measure them more often than has been traditional for surveillance coupons.

The risk management framework herein applies American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to establish the structure of a protocol for carrying out the necessary surveillance. These documents are generic, and they do not tell us how often or what to surveille, or what to measure, but rather give guidelines on how to determine these details given certain technical inputs. The Department of Energy (DOE)-funded INL Regulatory Development (R&D) Program will provide the companion supporting technical basis for a representative materials surveillance technology. When completed and validated, the technical basis can be used by owners or operators and the United States (U.S.) Nuclear Regulatory Commission (NRC) to implement a materials degradation management program for ANLWRs.

Salient points of discussion, positive potential outcomes, and potential concerns are also outlined. These aspects intend to inform future work on a proposed technical process for adoption by the industry and endorsement by the NRC to allow developers to propose a risk-informed, conservative approach for the use of materials where operating experience, data, and codes and standards may not exist for use of novel material in an operating reactor environment.

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ACRONYMS

AEC	Atomic Energy Commission
ANLWR	advanced non-light-water reactors
APA	accident precursor analysis
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
FOAK	first-of-a-kind
FY	fiscal year
LOCA	loss-of-coolant accident
LWR	light-water reactor
MARVEL	Microreactor Applications Research, and Validation and Evaluation Project
MSR	molten salt reactor
NASA	National Aeronautics and Space Administration
NRC	Nuclear Regulatory Commission
PRA	Probabilistic Risk Assessment
R&D	research and development
RIM	Reliability Integrity Management
SSC	structures, systems, and components
UK	United Kingdom
US	United States

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Accelerated Materials Deployment in Advanced Nuclear Power Plants

1. PROJECT DESCRIPTION

1.1 BACKGROUND

Components in Advanced Non-Light Water Reactors (ANLWRs) are subject to extreme environments with high temperatures, neutron radiation, and corrosive coolant, and the components are under severe cyclic stresses for long periods of time. Under these extreme conditions, materials properties of reactor components degrade continuously. To ascertain materials performance and to protect against premature failures, observational data capturing, reflecting these changes, are required to support the design and safe operation of ANLWRs.

As it is not practical to generate test data to cover the full operating lives of ANLWRs, shorter term, single-effect materials data are extrapolated to very long design lifetimes to develop design parameters and allowable stresses for incorporation in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) [1]. Although design margins are built into these extrapolation schemes based on available data and engineering judgment, the resulting design procedures could be either overly restrictive, limiting the design space, or potentially non-conservative, as degradation mechanisms could change at longer times and under prototypical operating conditions compared to accelerated test conditions. This presents a significant challenge for the licensing and deployment of existing or new structural materials in ANLWRs.

This report initiates development of a framework for applying materials surveillance technology that, when completed and validated, can be used by owner/operators to implement a materials degradation management program for ANLWRs that would also address associated NRC requirements and implementing guidance. In this report, reference is occasionally made to MSR (molten salt reactor) technology. References to MSR technology must be regarded as examples since the report makes no essential use of MSR-unique features. The framework herein is believed to be technology-neutral and generally applicable to ANLWRs. Additionally, such a technology could be leveraged to reduce part of the upfront materials data requirements from ongoing long-term materials testing so that early action on license application could be undertaken by NRC in parallel with the continuation of long-term data collections. This could accelerate the schedule for a first-of-a-kind ANLWR deployment or a nth-of-a-kind new materials insertion for established ANLWR designs.

1.2 SCOPE AND OBJECTIVES

The scope of this project is to produce a guidance document for ANLWR industry stakeholders to review and endorse that will present regulatory options to leverage the materials surveillance technology being developed by MSR research funded under the Department of Energy's (DOE) Regulatory Development R&D Program [2].

The broader long-term goal of this work is an acceptable framework that can be tailored based on specific developer technologies. This framework should (i) develop and implement a materials degradation management program that provides early warnings on structural degradation and on impending structural failure of safety significant components, and (ii) develop a strategy for the ANLWR industry to balance and potentially reduce part of the upfront new materials data requirements from ongoing -long-term materials testing. The out-year goal of this effort is an industry-endorsed regulatory proposal to the United States (U.S.) NRC so early licensing action can be undertaken.

The outcome of this fiscal year (FY) 2022 report is to support the establishment of a defined and predictable regulatory framework by providing the overarching principles necessary. The follow-on

FY-23 work will support development of a guidance document that delineates the framework for a safety case development and NRC-endorsement of a developer's nuclear technology and associated novel materials and environment. This work addresses early deployment when an ASME Code case may partially exist, or potentially not exist, for a novel material in a known environment, a material used in design that may have a code case that is not specific for the operating environment, or a combination of both.

2. THE CODE CASE – NOVEL MATERIALS AND ENVIRONMENTS

2.1 CURRENT PRACTICES

The existing fleet of U.S. nuclear plant licensees utilizes the allowable materials specified by ASME BPVC Section III [3] in conjunction with approved ASME Code Cases endorsed by the NRC under 10 CFR Part 50.55(a). ASME BPVC Section III provides the rules for the construction of nuclear facility components. Materials allowed for use in ASME BPVC Section III, Div. 1, within the prescribed temperature limits and design conditions, are listed in ASME BPVC Section II, Part D [4].

The slate of advanced reactor designs being developed utilize high temperature gas, molten salts, and structural graphite. Materials subject to these environments may have additional design criteria that is not addressed in ASME BPVC Section II, Part D. ASME BPVC Section III, Div. 5, High Temperature Reactors, specifies modified and additional material requirements to that of Division 1 for use in Division 5.

Advanced reactor designs may seek to utilize materials not currently listed as acceptable in applicable divisions of ASME Section III. Material application challenges include:

- Materials currently approved for use by ASME Section III but used in new applications. (Outside accepted temperature ranges, structural material used for pressure boundary components, etc.)
- Materials currently approved for use by ASME Section III but fabricated in a new manner (Additive manufacturing).
- Materials which are not currently approved for use by ASME Section III [3].

The standard process by which a new material, or existing material used in a new application, can be formally approved for use is described in ASME BPVC Section II, Part D, Mandatory Appendix 5, *Guidelines on the Approval of New Materials Under the ASME Boiler and Pressure Vessel Code* [5]. Prior to being approved for use in a section of the Code, use of a new material may be permitted in the form of a Code Case that is adopted by the ASME BPVC committee and approved by the regulatory authority that has jurisdiction which, in this case, is NRC. A Code Case establishes at least the condition of use and necessary requirements linked to these conditions. It is the stated policy of the ASME BPVC Committee to admit, in this way, material for which full experience on all working parameters has not yet been acquired.

2.2 IDENTIFIED GAPS

For some advanced reactor designs, the attainment of a formal Code Case is not practical within the constraints of completely defining material interaction and response characteristics for the new application prior to receiving design approval. An accelerated material deployment process, endorsed by the jurisdiction, is needed to support ANLWR designs. Areas of consideration for accelerated deployment that are consistent with ASME BPVC material approval include:

- Chemical composition
- Metallurgical structure and heat treatment
- Mechanical properties
- Time-independent properties

- Time-dependent properties
- Low-temperature properties
- Toughness data
- Stress-strain curves
- Fatigue data
- Physical properties
- Requirements for welds, weldments, and weldability
- Long-term properties stability.

For ANLWR design, the application of new materials may identify a need for conditional acceptability when the above-listed properties cannot feasibly be determined. For example, basic physical and mechanical properties may be known for a given material, but the long-term effects of irradiation, fluid contact, or other degradation mechanisms, may need adequate design margins to allow surveillance during operation to inform adjustments to assumptions about material behavior.

3. A RISK-INFORMED PERFORMANCE-BASED APPROACH TO RELIABILITY INTEGRITY MANAGEMENT FOR ACCELERATED MATERIALS DEPLOYMENT

3.1 A RISK-INFORMED, PERFORMANCE-BASED APPROACH

The ASME Reliability Integrity Management (RIM) process [6] is a high-level framework combining science, engineering, and risk management. The RIM Program addresses the entire life cycle for all types of nuclear power plants, it requires a combination of monitoring, examination, tests, operation, and maintenance requirements that ensures each Structure, System, and Component (SSC) meets plant risk and reliability goals that are selected for the RIM Program. This section gives a brief overview of the RIM process, with a view to calling out a few topics where additional commentary seems warranted in later subsections. Based on a review of the RIM process, topics that could be developed more fully next year are identified.

3.2 OVERVIEW OF STEPS IN THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS RELIABILITY INTEGRITY MANAGEMENT PROCESS

Article RIM-2 of ASME BPVC Section XI, Division 2 [6], describes the primary steps for implementing a RIM process. Key steps of RIM-2 are as follows:

(1) RIM Program scope definition
(2) Degradation mechanism assessment
(3) Plant and structures, systems, and components (SSC) reliability target allocation originating from the probabilistic risk assessment (PRA)
(4) Identification and evaluation of RIM strategies
(5) Evaluation of uncertainties
(6) RIM Program implementation
(7) Performance monitoring and RIM Program updates.

These steps constitute a framework for formulating a compliant program. The details of the steps are necessarily application-specific, but even at this high level, certain features are noteworthy.

<u>RIM 2.2 RIM Program Scope Definition</u>

This step determines the scope of SSCs to be addressed in the program. As written, the RIM program includes “SSCs whose failure could adversely affect plant safety and reliability.” This wording would appear to include not only pipes and vessels containing working materials, but also components that play a purely structural role.

RIM 2.3. Degradation Mechanism Assessment

Because reliability of an SSC is influenced by the material type and operating conditions, a key step in applying the RIM process is identifying the potential degradation mechanisms an SSC is subject to. Mandatory Appendix VII [6] is meant to provide screening criteria for assessing SSC susceptibility to particular degradation mechanisms applicable to each reactor type. The section for MSRs is still under development which necessitates incorporating the domain of the experimental effort that parallels the present program.

RIM 2.4 Plant and SSC Reliability Target Allocation

RIM 2.4 describes a process that has several key steps. Together, the steps culminate in a designation of:

- Plant-level risk and reliability targets that comport with plant-level “regulatory limits” and plant operator goals
- SSC-level targets that are consistent with the plant-level targets, as determined using the plant PRA.

For a design with any complexity at all, there may be many ways of specifying SSC-level targets that nominally satisfy plant-level targets. These may vary greatly in feasibility and cost. One cost dimension is that an ambitious reliability target for a particular SSC may implicitly require a large amount of surveillance; in such a case, it might be desirable either to consider a different allocation for the design, or modify the design so that the risk is less sensitive to failures of the subject SSC. Notably, two priorities are cited in the RIM writeup: safety and plant availability. Along with other technical information (e.g., the degradation mechanisms), these targets will imply the scope, character, and frequency of surveillance. It is also possible that the developers have other priorities to be addressed, so there may be reasons to surveille a first-of-a-kind (FOAK) plant more often than the code requires. This will be retained as an option, always assuming the surveillance protocols emerging from the RIM process are at least as stringent as safety and availability would require.

The first step in RIM-2.4 [6] calls for identification of plant-level risk and reliability targets for RIM. These targets are meant to be derived from regulatory limits on risks, frequencies, and radiological consequences of licensing basis events defined in the PRA. Plant-level goals may be related to plant availability. For a particular plant type, this is actually a very important direction-setting exercise.

The latter step in RIM-2.4 is remarkable in that it invokes the PRA and calls for allocation of reliability performance over SSCs. The ASME Code links to the PRA Standard for ANLWR Nuclear Power Plants [7]. The allocation (the assignment of reliability targets to SSCs) is required to propagate through the PRA in such a way as to yield values of top-level metrics that satisfy plant goals, potentially including plant availability goals. More ambitious allocations will lead to more demanding surveillance, as provided in the text box below.

Failure Rate, Failure Probability, and Surveillance Interval

For readers not conversant with topics like "risk-informed in-service testing," the following extremely simple example is offered. Suppose that a particular SSC is presumed to have a constant standby failure rate of λ, and the unreliability target for that SSC is a standby failure probability less than q. Immediately after a test, we are sure that the SSC is good (if it is not, we repair or replace it, and then it is good), but as time passes, some mechanism might operate to cause component failure. How often should we test? If we consider a test interval T, we find that within that interval, q is, on average, ½ λ*T, a simple result that relates T to the unreliability target and the presumed constant failure rate. If we need a smaller q, we must implement a shorter T.*

In general, smaller q implies smaller T, but the simplicity of the ½ λ*T formula is an artifact of the constant-failure-rate assumption. In real-world cases, a new component may be very unlikely to fail until some wearout (or other sort of degradation) has occurred; and thereafter, that component may have a high failure rate. If we don't know where the component is in that evolution, or even if we just have substantial uncertainty in λ, we may have to make a conservative choice of T. Executing surveillance on a FOAK plant to understand failure rate and time dependence may well be worthwhile from a fleet point of view, even if we do not believe that we need it in order to legitimize operation of a single plant.*

In the above example, the essential simplicity of being able to test and immediately determine component state is not universally applicable. In a more general case, the component may not be simply "good" or "bad;" we may need to determine how degraded an SSC is, and how it is degraded, and what this state information implies about SSC reliability. This can be much more demanding than simply figuring out whether the current state is "good" or "bad."

There are decisions to be made here.

- While safety is the dominant licensing theme in the discussion, data collection for broader purposes may be necessary. This argues for more and better surveillance than might be required to protect the public. The need could be handled indirectly by calling for stringent plant availability goals (requiring more surveillance), but the purpose of such goals is different. There must be a unified position on the priority to be given to the number and types of instruments, coupons, etc., beyond what NRC might have required.
- Since the SSC allocations are required to fit together to achieve consistency with top-level metrics, taking more credit in one place may mean that less credit is needed elsewhere. This makes allocation an important investment decision since it would be a trade study in and of itself.

RIM-2.5 Identification and Evaluation of RIM Strategies

This step, too, is a compound step. There is a range of factors that affect RIM strategies. To account for these factors, a strategy or combination of strategies is chosen that is "necessary and sufficient to achieve and maintain SSC reliability consistent with" the targets. Finally, in 2.5.2 [6], RIM strategy effectiveness is evaluated.

RIM does not specifically prescribe what RIM Strategy is best for an SSC or group of SSCs. The use of an expert panel" is factored into two different aspects of the process that integrate current technologies and capabilities. The first expert panel is specifically focused on monitoring and non-destructive examination capabilities to determine the best approach to identifying degradation or flaws in the material using the most appropriate examination or inspection technique available. The panel then ensures that personnel and equipment are qualified and capable of finding such degradation. A second expert panel with broader experience gives a final evaluation of the chosen strategy to ensure that it achieves the desired reliability given the identified degradation mechanisms, potential failure modes, and importance of the function of the SSC(s).

A substantial component of the broader community of practice [8] recognizes that "reliability" is something that cannot be proven, and it is not possible to go from a set of RIM Strategies to formal proof of a resulting reliability. Rather, if reliability assurance is needed, there must be an *assurance case.* This topic is discussed later in subsection 4.3, Safety Case. For now, there is a disconnect between 2.5.1 (quoted above) and 2.5.2, which acknowledges that the strategies cannot be perfect, but it must be understood how effective they are.

RIM-2.6. Evaluation of Uncertainties

This step does not explicitly require for evaluation of uncertainties. The step identifies additional strategies over and above those developed in RIM-2.5 to provide additional assurance that the reliability targets will be achieved and maintained. This is another indication that one cannot go from a given set of strategies to a guaranteed level of reliability.

Adding strategies to a list that was already nominally adequate should, of course, add margin through earlier identification of degradation.

RIM-2.7. RIM Program Implementation

This step comprises numerous substeps having to do with implementing (and documenting) the strategies developed above to assure compliance with the allocated targets.

One additional substep suggests itself. Some approaches to risk management embark on well-intentioned monitoring activities that are undertaken without explicit consideration of what the organizational response will be to adverse observations. The Davis-Besse example [9], discussed later in section 4.5, Precursor Analysis, offers an illustration. The RIM program for MSR would benefit conceptually from a clear articulation of the boundary between acceptable observed performance and unacceptable performance *a priori*, and NRC will likely want a commitment to adhere to the implications of violating that boundary.

The NRC has long had a conceptual basis for such a substep: 10 CFR 50.59 [10]. This has evolved over the years, but the underlying principle stands: facility operation is licensed based on an understanding of how the plant will behave inside a certain region of state space; if that turns out to be wrong, or if the plant goes outside that region, then the situation is either clearly adverse, or at best unanalyzed, and operation should cease until reanalysis shows that the plant is safe. This premise is further discussed in section 5.1, Enhancements Needed for Technology-Specific Application of RIM.

RIM-2.7 actually tells us what to do if a flaw greater than a previously agreed-to size is observed. But it does not provide analogous information covering all potential issues in MSRs. Some thought will need to be given to formulating and committing to an actual operational protocol that anticipates possible deviations from nominal expectations. The Davis-Besse example discussed in section 4.5, Precursor Analysis, suggests that it is important to investigate anomalies, not just obvious near misses. Unfortunately, in the case of Davis-Besse, some of the SSCs involved in the anomaly were not considered important, so the task of investigating anomalies is not as straightforward as one might hope. But the task is arguably important, especially in a new technology.

Summary

RIM provides a high-level framework, but each step entails significant technical work. Some RIM topics will need to be fleshed out next year for present purposes.

3.3 APPLICATION TO NEW MATERIALS

By definition, a convincing case cannot be made that it is appropriate to deploy new materials in safety-critical areas with minimal surveillance. If that case could be made for a particular material, that material would not be "new." This discussion addresses what it takes to make the case that accelerated deployment is acceptable, based in part on proactive surveillance. At a high level, the argument looks like this:

SSC reliability targets are established

- Potential degradation mechanisms are known
- Engineering techniques have been applied to promote reliability despite known degradation mechanisms
- Surveillance intervals have been determined to ensure the reliability targets are satisfied
- Ample surveillance has been planned to identify problems before they occur
- Protocols have been established for organizational response to observations of significantly more degradation than expected.

NRC is in the final stages of reviewing RIM for use in the development of a reactor's Inservice Inspection Program. The to-be-approved RIM framework does not specifically target new materials, but it does provide a high-level discussion of the information identified in the list above that can be used to provide in-service monitoring, inspection, and surveillance of a new material. However, more detail is needed for present purposes. Within the RIM framework, what can be done to help formulate an RIM application appropriately to make the strongest possible case to NRC for approval? Before that question can be answered, the following issues must be addressed:

- A substantial amount of work must be done to understand the behavior of candidate materials in enough detail to support execution of the RIM steps. That work is the subject of a parallel effort in materials [2].
- It is useful to recognize that RIM is an example of an important emerging trend. Until recently, it was customary to presume that the main licensing safety case was grounded in the historical Atomic Energy Commission (AEC) and NRC design-basis thinking, enhanced by (but not determined by) insights from risk analysis. The RIM process takes a step beyond that and is informed in detail by a combination of risk analysis insights and fundamental engineering analysis of what it really takes to understand component degradation and prevent component failures.
- Moreover, RIM is about achieving reliability, which is a metric whose value cannot be proved in the classical sense of that term. Reliability must be argued, and in particularly important applications, it must be argued in a sufficiently formal way to support NRC review. In short, an argument needs a "case." The case needs to argue that the proposed surveillance plan has sufficient margin to justify

NRC acceptance despite residual uncertainties in materials behavior. The "case" concept is discussed in Section 4.

- The allocation step represents a potentially significant investment decision. For a design with appreciable complexity, allocation would be a significant direction-setting undertaking. Currently the details of the topic of "allocation" are beyond the scope of RIM, but allocation will need to be completed.
- In deployment of a relatively novel technology, an allowance has to be made for completeness issues. However hard we work on hazard analysis, we are well advised to formulate and assiduously implement a program of anomaly precursor analysis, in the hope of picking up on symptoms of previously missed problems. This topic is also discussed in Section 4.

4. METHODS AND TOOLS FOR APPLICATION WITHIN RIM

4.1 INTRODUCTION

The ASME Code discussion of RIM mandates a high-level framework applicable to a wide range of risk management situations. The present section offers historical perspective on related matters that, in a more narrowly focused treatment, might serve to support "may" or "should" statements related to RIM for novel specific materials applications in novel technologies.

If there were no uncertainty regarding current component states, or current levels of material degradation, surveillance of these situations would be unnecessary. But some surveillance is done in all plants because residual uncertainty remains. The need for extra surveillance of novel materials in novel technologies is a matter of degree: more uncertainty => more (and perhaps different) surveillance.

4.2 TIME-LIMITED AGING ANALYSIS

Time-limited aging analysis [11] is important in license renewal. The present subsection offers an analogy between the state of knowledge obtained after license renewal of an understood technology, and initial licensing of a novel technology with significant associated uncertainties.

Figure 1 below notionally compares two situations. In the top portion of the figure, "Generation II Plant," the uncertainty is essentially understood during the license term of a current operating plant, but after license renewal (after decades of operation), uncertainties regarding the behavior of decades-old components increases. In the lower half of the figure, "FOAK Plant," where the technology is novel and the materials are novel, the uncertainties set in much earlier.

In both cases, to compensate for uncertainties, extra work is necessary to assure that component degradation is not occurring to the point where safety is threatened. At a sufficiently high level, managing the risks in the two situations is somewhat analogous. There may be lessons learned from aging management of Generation II plants—or even regulatory precedents—that could be harnessed for application in the present context.

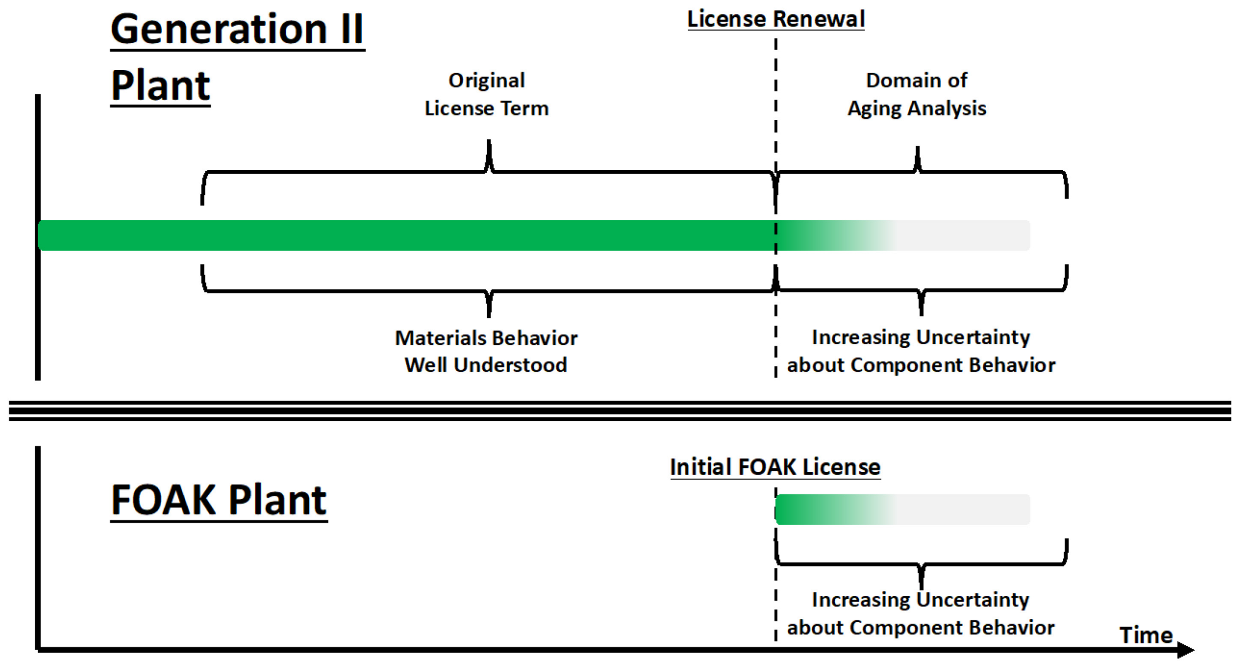

Figure 1. Time-dependent uncertainty in a Generation II Plant compared to a FOAK Plant.

4.3 SAFETY CASE/ASSURANCE CASE

After decades of successfully licensing Light-Water Reactor nuclear plants, both industry and the NRC have gained significant experience; in particular, the NRC knows how to formulate regulatory requirements (especially prescriptive requirements) whose satisfaction will be deemed to provide reasonable assurance of adequate protection [12]. However, following such a path for novel technologies would be extremely challenging. Much needs to be learned about technologies that are substantially different from the technologies applied in currently operating plants. Given the number and variety of new technologies that must be examined, it is impractical for NRC to take the time to develop prescriptive requirements for novel plants analogous to those that apply to operating plants. In the current era, a significantly greater decision velocity in licensing is needed in order to deal appropriately with new technologies.

The Nuclear Energy Institute has proposed a "safety case" approach to licensing basis development for non-LWRs, which has been endorsed by the NRC, and has received a great deal of attention [12,13]. Other countries have been applying safety-case thinking for many years, but this has only recently begun to occur in the US, and so far mainly in non-nuclear industries. In that sense, among others, the Licensing Modernization Project is groundbreaking.

Within a pure safety-case approach, the applicant is expected to formulate the argument (the "case") that a proposed facility will be "safe" as a result of a particular body of facility attributes and engineering practices. Within such an approach, the case provides evidence and argument showing that the facility is safe. Compared to traditional licensing practice, a safety-case approach is essentially technology-neutral.

Dealing comprehensively with the licensing problem for novel technologies is beyond the scope of this report. However, since prescriptive regulatory requirements do not yet exist for certain materials challenges, the *case* needs to be made that safety can be achieved despite uncertainties regarding materials, by compensating for these uncertainties with increased surveillance and associated modeling and analysis. The following section discusses the "case" concept in more detail, in light of the possibility that certain associated practices will be useful in the present context.

4.4 WHAT IS A "SAFETY CASE" APPROACH?

The "safety case" topic is introduced for the following reasons:

- Accelerated deployment of new materials in a novel technology is not covered in detail by NRC requirements. Therefore, a case must be made that these materials can be deployed safely despite uncertainties, whether or not this is explicitly called a "safety case."
- "Reliability" is an attribute that cannot be proven in the usual sense (e.g., we can prove that a given flowrate is adequate, but not that it will *always* be achieved) and the "safety case" must therefore be explored. Arguing reliability to approval authorities using the "safety case" idea has received a great deal of attention worldwide, including the development of consensus standards for "cases," and the present accelerated materials deployment framework can benefit from that work.

A RIM-based application to NRC will be a narrowly scoped safety case, whether or not it is described as a safety case.

The following is a definition of "safety case" [15]:

> *A structured argument, supported by a body of evidence that provides a compelling, comprehensible and valid case that a system is safe for a given application in a given operating environment.*

The "case" idea calls for an explicit coherent safety argument, formulated by an applicant who is taking responsibility for the claim. This is distinct from falling back on a "compliance" argument; the case needs to provide a structured, technical argument that the fundamental plant safety objectives are met. For novel technologies, an adequate body of regulatory engineering requirements does not exist to fall back on. As highlighted earlier, the RIM process calls for applicants to develop their own engineering protocols; justifying the adequacy of these protocols is the essence of the case.

The ASME Code sections discussed earlier do not discuss the "case" concept. But it seems clear that the burden is squarely on the applicant to make the case that:

- The SSCs that need to be in the RIM program have been identified.
- Relevant degradation mechanisms have been identified.
- A sensible reliability allocation has been developed for application to the population of SSCs in the RIM program. The allocation:
  - Is feasible;
  - Satisfies top-level safety objectives.
- Strategies expected to provide the needed level of reliability (including surveillance strategies) have been identified.
- Protocols have been established committing to particular organizational responses to deviant surveillance results.

4.5 ANOMALY PRECURSOR ANALYSIS

Licensing a truly novel reactor design would be a challenge if the only path forward required that despite the design's novelty, the behavior of the design and its associated risks are completely understood. A licensing path that addresses this dilemma for novel designs is provided in 10 CFR 50.43 e [16]:

> *(e) Applications for a design certification, combined license, manufacturing license, operating license or standard design approval that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997. Or use simplified, inherent, passive, or other innovative means to accomplish their safety functions will be approved only if:*
>
> *(1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;*
>
> *(ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and*
>
> *(iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or*
>
> *(2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If a prototype plant is used to comply with the testing requirements, then the NRC may impose additional requirements on siting, safety features, or operational conditions for the prototype plant to protect the public and the plant staff from the possible consequences of accidents during the testing period [16].*

10 CFR 50.43 e provides paths forward for licensing plants that "differ significantly from light-water reactor designs that were licensed before 1997," which describes the present context. The above excerpt clearly contemplates improving the state of knowledge through experience, which is how the present application of RIM is intended to operate. But this entails learning a great deal from operating experience [17].

Precursor analysis is a way of learning from operating experience to improve the facility models that we use for risk management: that is, to make those models more complete and more accurate, so we will make better risk management decisions. Informally, a "precursor" is an anomaly or event that is not necessarily in itself a real accident, but signals previously unappreciated potential for a real accident, by manifesting model incompleteness (unappreciated accident-promoting phenomena) or pointing out an underestimate of a significant event likelihood. In the present context, seeing degradation occur more rapidly than had been expected qualifies as an "anomaly" that needs to be understood.

The general idea is illustrated in Figure 2 below [18], showing a continuous-improvement loop in which models are systematically improved as a result of gathering and analyzing operating experience and reflecting new knowledge in our models. NRC has engaged in a form of precursor analysis for many years [19].

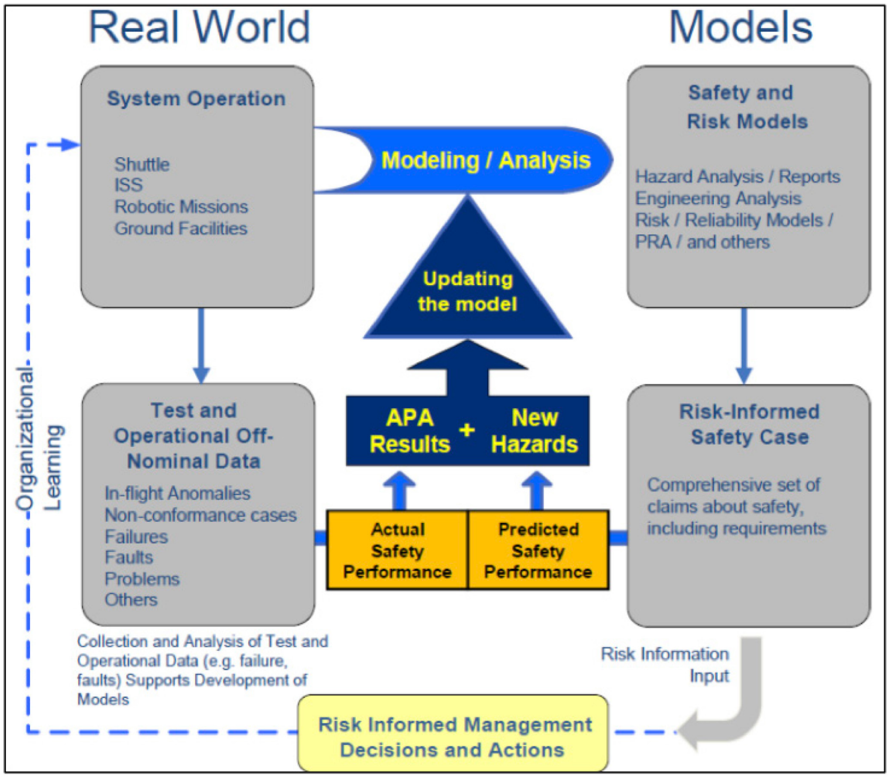

Figure 2. Real world vs. models information flow in the APA [Accident Precursor Analysis] Context (Figure 2–3 of [18]) (Public use permitted).

The U.S. Congress requires NRC to report significant precursors in operating reactors. The idea is that the report to Congress reflects the performance of NRC and of the operating fleet assuming an increase in the incidence of precursors may imply something is amiss. However, the process advocated here is different in emphasis and detail. The process was developed by the National Aeronautics and Space Administration (NASA) and does not limit itself to "events" because it also considers anomalies. Moreover, NASA's reason for doing the analysis is model improvement, not reporting quantitative results to Congress.

The process developed by NASA [18] is illustrated in Figure 3 below. This process was partially driven by the need to consider a potentially large number of anomalies, and it is correspondingly important to screen out anomalies that have little or no significance before detailed analysis. This may sound contradictory: before analysis, how does the analyst know what to screen out? Before screening an anomaly out, the analyst needs to understand what caused it. The observed anomaly may be just a symptom. It is the cause of that symptom whose risk significance matters to the model improvement effort, so the cause needs to be understood, along with how well the current model addresses it; its potential risk significance needs to be judged in that light before the screening decision is made.

This is an essential difference between the NASA's precursor analysis method and NRC's method: in the NASA method, precursor analysis is essentially a kind of hazard analysis, cued by an actual anomaly. The method is as applicable to different technologies as the hazard analysis itself. Focusing on the causes of anomalies is important for many precursors, such as the observations made at Davis-Besse leading up to its vessel-head wastage event [19]. In the NRC method, the conditional probability of core damage is quantified, based on what was observed to have occurred, and this was the basis of their report to Congress.

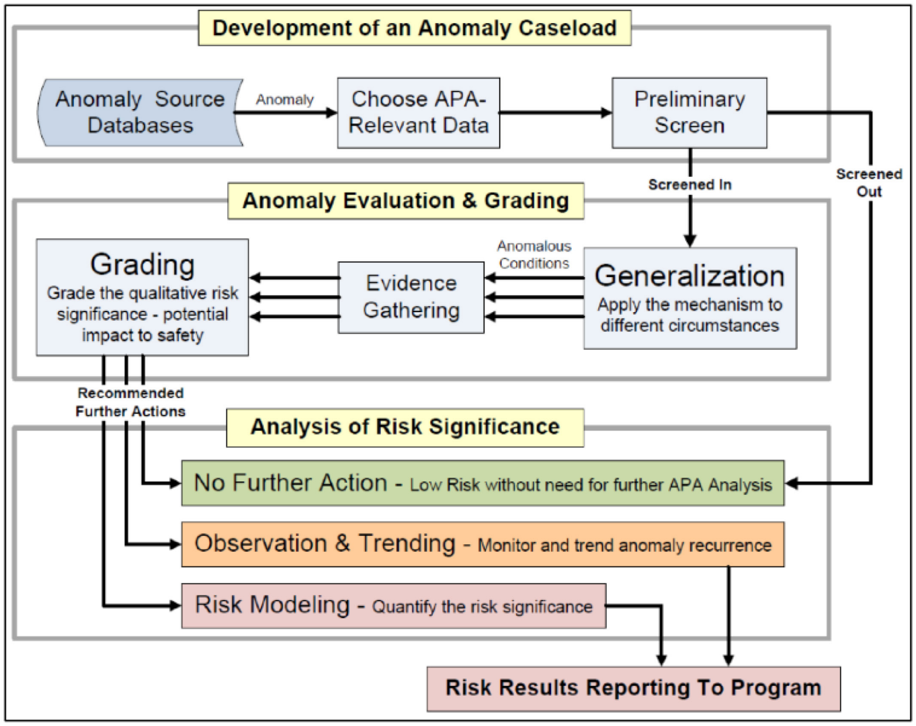

Figure 3. NASA Accident Precursor Analysis Process Overview Diagram (Figure 2–5 of [17]; public use permitted).

An important element of Figure 3 is the box labeled "Generalization." In this step, the identified cause (the "mechanism") is examined to see whether that mechanism could have led to effects different from (more severe than) the observed effects. For example, if a particular valve behaved sluggishly due to a contaminant in the regulated fluid, it could be of interest to see whether that effect could have been more severe, or whether the same cause could have operated in a different system, or in multiple systems at once.

In this process, model improvement is the goal. If the goal is merely to construct an indicator of fleet performance based on precursors, screening based on the conditional probability of an accident may suffice, rather than attending to the underlying cause of the observation and its risk significance. If an observation is surprising, it is likely that the observer's model of the world is incomplete [20]. From this point of view, the degree to which an event is "surprising" can help to inform a decision on whether to proceed to detailed analysis of a given anomaly.

The present emphasis on the topic of new designs is not meant to suggest that current precursor analysis practice is completely satisfactory for the operating fleet of commercial reactors. Current practice has demonstrated value, but it has also demonstrated certain imperfections.

Operating experience provides a cautionary tale related to this step. In March 2002, the reactor vessel head at Davis-Besse was found to have been seriously degraded as a result of an external attack of boric acid [9]. It was known that boric acid would attack the carbon steel used in vessels, but it was not appreciated that boric acid could reach the *outside* of the vessel head. (The inside of the head is clad with stainless steel, which is resistant to boric acid attack.) At the time, there was concern about cracking of the penetration nozzles in the vessel head, which would release a certain amount of coolant containing boric acid, but the concern was what might be called a "nozzle loss-of-coolant accident" (LOCA);

external vessel head erosion was not foreseen. It was believed that given leakage due to cracking, the leaked reactor coolant leakage would flash, leaving behind dry boric acid residue (which would not have been a problem), rather than wet boric acid attacking the head. Unfortunately, the analysis of the behavior of the leaked coolant was incorrect.

Incorrect analysis was not the only problem. The plant had been experiencing anomalous phenomena in containment and accepted the anomalies as insignificant, rather than research their underlying causes. These phenomena turned out to have been related to the effects of boric acid wastage of the vessel head, and allowing them to continue uncorrected represented (among other things) a lost opportunity to become aware of what could have become an even larger problem. The plant staff were criticized for lack of a questioning attitude. The relevance for the present discussion is that a questioning attitude is an essential element of the risk management activity we are discussing here.

In the event, the wastage was discovered before a substantial LOCA occurred, but many observers were taken aback by how close the plant came to experiencing that event. Part of the NRC response to the event was development and promulgation of "NRR Office Instruction LIC-504, 'Integrated Risk-Informed Decision making Process for Emergent Issues.'" [20] Both the office instruction and the course devote substantial attention to fostering a questioning attitude and avoiding the modeling pitfalls that operated at Davis-Besse. Accordingly, in the materials surveillance program that is implemented as part of RIM, it will be necessary to be vigilant about anomalies: to be on the lookout for them and understand them, even if they do not look like immediate threats. [21]

5. SUMMARY: PATH FORWARD TO REGULATORY GUIDANCE

5.1 ENHANCEMENTS NEEDED FOR TECHNOLOGY-SPECIFIC APPLICATION OF RIM

Even in existing plants, residual uncertainties call for a range of surveillance activities. Novel materials in novel technologies pose greater uncertainties, and therefore call for more and perhaps different surveillance, but the difference between relatively well-known situations and novel-technology situations is largely one of degree: prudent risk management calls for surveillance in both cases.

The RIM program provides a modern, high-level, risk-informed framework for Reliability Integrity Management that covers both ends of the spectrum (known technology and novel technology). It is exceptional in calling for allocation of reliability targets based on risk analysis.

However, the RIM program is specified at a very high level, and more specifics must be added for application to any given technology. In FY 2023, we will consider the following:

1. For novel technologies, the community's understanding of safety issues is, on average, less complete than the community' knowledge of safety issues for `Generation II plants, and therefore, extra care is warranted in the hazard analysis to identify degradation mechanisms.
2. In principle, addressing uncertainty by performing more surveillance is a familiar concept, but novel materials in novel technologies pose a more significant risk management challenge. Moreover, it may be desirable to make the argument to NRC more formally than usual, perhaps through a focused safety case. A key element of a safety case is an applicant's commitment to dedicated analysis of operating experience of a sort that is warranted for novel materials in novel technologies.

In order to adequately support an approach to risk management, materials research must communicate the following to the risk management team:

- Researchers know what to measure and what to trend in order to determine whether MSR components are OK, how much margin remains, how rapidly materials are degrading, etc. The risk management team must know how often to measure, and what to measure, in order to provide the needed ongoing assurance that reliability targets are being met and will continue to be met (at least for a while).

- We know that those measurements are practical in an operating plant and that there is a sufficient estimate of how many surveillance coupons to install and when to pull them out.
- Measurement uncertainties are understood.

Based on the above, the operator will propose a safety case (or technical equivalent) including certain assumptions regarding SSC performance and lifetime and will commit to a method and frequency of surveillance that are adequate to provide ongoing assurance that those assumptions remain valid. If surveillance shows that they are not valid, then unless there are design features to compensate for SSC degradation, the operator will need to provide further analysis. So, in addition to the above, the risk management team needs a protocol saying that if certain specified parameter thresholds are ever crossed, or certain trends exceed prior expectations, then the plant is in an unanalyzed condition that requires additional risk management steps. In this scenario, unknown unknowns may be present, and anomalies that are not understood must be investigated. These conditions are implied by the safety case, which is implied, in part, by Regulatory Development R&D Program input.

In other words: any departure from the final safety analysis report (FSAR) must be analyzed to determine the effect on the facility.

Developments like the above are needed to make the case to the regulator to allow MSR operation despite current uncertainties, unless a commitment is made to engineer a confinement good enough (or siting remote enough) that public safety is not threatened even if an accident occurs.

6. INDUSTRY ENGAGEMENT

As part of the development of the Accelerated Materials Qualification and Deployment effort discussed in this paper, key inputs on the viability and utility of the effort reside with industry stakeholders and the regulator. Prior to the present undertaking, initial discussions were held with the NRC and industry stakeholders by Regulatory Development Staff at INL to understand initial interest and determine the value of the alternative process being considered. Initial discussions provided feedback that there was significant interest by both industry and the regulator to provide a potential pathway to early deployment of reactor technologies that utilize novel materials and designs. These novel designs and materials may not be fully approved and endorsed, but with proper validation of performance and lifetime through monitoring and testing, can be licensed with reasonable assurance of safety. Significant value was seen in having a technically sound methodology that is risk-informed and invokes performance-based attributes to establish the initial safety case justification without entering the 10 CFR 50.12 exemptions process.

Significant insights into the framework for surveillance testing of materials will be developed from regulatory development research performed as part of the INL MSR research funded through the DOE Regulatory Development Program. This program is researching and developing test specimen surveillance articles that can be leveraged by developers to establish baseline and in situ monitoring of novel materials or existing known materials in novel environments. This MSR-centric work, coupled with other potential test cases from DOE-driven reactor research and deployment, may be useful in the framework development and example case insight. Additional insight and stakeholder discussions will stem from these examples and knowledge obtained. One such example may be available through the DOE novel reactor Microreactor Applications Research, and Validation and Evaluation Project (MARVEL) design and deployment considerations. MARVEL will utilize uranium zirconium hydride high assay low enrichment fuel with a sodium-potassium eutectic (NaK) coolant. Numerous reactivity control, reflector, piping, and structural components may provide opportunities for data studies to inform research processes. Draft guidance will be discussed with NRC and industry stakeholders and adjusted throughout development in FY-23 to provide a higher likelihood of acceptance.

Upon completion of the initial FY 2022 work researching the code gaps and providing the theoretical risk-informed framework for consideration contained in this report, industry and NRC subject matter experts will be engaged for review and collaboration through targeted NRC and stakeholder meetings, and discussions with the working groups responsible for ASME BPVC Section XI Division 2 – RIM [6]. Insights gained from these discussions will be incorporated into the development of the FY-23 guidance document approach.

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